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(3) the air space above the suppression pool; and

(4) the reactor building which is structurally integrated with the concrete primary containment structure.

A secondary containment which surrounds the primary containment permits monitoring and treating all potential radioactive leakage from the primary containment. Treatment consists of HEPA and activated charcoal filtration.

1.2.2.15.2 Containment Internal Structures

The containment internal structures are summarized in Subsection 6.2.1.1.2.

1.2.2.15.3 Reactor Pressure Vessel Pedestal

The reactor pressure vessel pedestal is a prefabricated cylindrical steel structure filled with concrete which supports the RPV and is maintained below design temperature by cooling. The pedestal provides drywell connecting vents which lead to the horizontal vent pipes to the suppression pool.

1.2.2.15.4 Standby Gas Treatment System

The standby gas treatment system (SGTS) minimizes exfiltration of contaminated air from the secondary containment to the environment following an accident or abnormal condition which could result in abnormally high airborne radiation in the reactor building. Because the fuel storage area is also in the secondary containment it also can be exhausted to the SGTS.

All safety-related components of the SGTS are operable during loss of offsite power.

1.2.2.15.5 PCV Pressure and Leak Testing Facility

The PCV pressure and leak testing facility is summarized in Subsection 9.1.5.2.8 Special Servicing Room/Areas.

1.2.2.15.6 Atmospheric Control System

The atmospheric control system is summarized in Subsection 6.2.5.2.1.

1.2.2.15.7 Drywell Cooling System

The drywell cooling system is summarized in Subsection 9.4.9.2.

1.2.2.15.8 Flammability Control System

An atmospheric control system is designed to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during plant shutdown for refueling or maintenance. A recombiner system is provided to control the concentration of oxygen produced by radiolysis in the primary containment.

1.2.2.15.9 Suppression Pool Temperature Monitoring System

The suppression pool temperature monitoring system is summarized in Subsection 7.6.1.7.1.

1.2.2.16 Structures and Servicing Systems

1.2.2.16.1 Foundation Work

The analytical design and evaluation methods for the containment and reactor building walls, slabs and foundation mat and foundation soil are summarized in Subsection 3.8.1.4.1.1.

1.2.2.16.2 Turbine Pedestal

The description for the turbine pedestal is the same as that for foundation work in Subsection 3.8.1.4.1.1.

1.2.2.16.3 Cranes and Hoists

The crane and hoist are summarized in Subsection 9.1.4.2.2.1.

1.2.2.16.4 Elevator

The controlled elevators service the reactor building radiation controlled zones from the basemat to the refueling floor. Two additional clean elevators service all elevations of the clean zone.

1.2.2.16.5 Heating, Ventilating and Air Conditioning

The plant environmental control systems control temperature, pressure, humidity, and airborne contamination to ensure the integrity of plant equipment, provide acceptable working conditions for plant personnel, limit offsite releases of airborne contaminants.

The following environmental systems are provided:

(1) the control room air conditioning system consisting of supply, recirculation/exhaust and makeup air cleanup units to ensure the habitability of the control room under normal and abnormal conditions of plant operation;

(2) the reactor building secondary containment HVAC system maintains a negative pressure in the secondary containment under normal and abnormal operating conditions thereby isolating the environs from potential leak sources. This system removes heat generated during normal plant operation, shutdown, and refueling periods;

(3) the drywell cooling system to remove heat from the drywell generated during normal plant operations including startup, reactor scrams, hot standby, shutdown, and refueling periods;

(4) the power block pressure control supply and exhaust system to distribute air so that a negative pressure is maintained in the emergency core cooling equipment rooms, thereby isolating the potential airborne contamination in these rooms;

(5) the electrical equipment supply and exhaust system to pressurize the electrical rooms allowing exfiltration of air to the battery rooms for exhaust to the outside atmosphere;

(6) the power block exhaust system to maintain the refueling floor at a negative pressure with respect to the outside atmosphere to prevent the potential release of airborne contamination;

(7) the diesel generator area air exhaust system to provide cooling during operation of the diesel generators. A tempered air supply system controls the thermal environment when the diesel generators are not operating; and

(8) coolers in the steam tunnel and ECCS rooms to remove heat generated during operation of the equipment in these rooms.

1.2.2.16.6 Fire Protection System

The fire protection system is designed to provide an adequate supply of water or chemicals to points throughout the plant where fire protection is required. Diversified fire-alarm and fire-suppression types are selected to suit the particular areas or hazards being protected. Chemical fire fighting systems are also provided as additions to or in lieu of the water fire fighting systems. Appropriate instrumentation and controls are provided for the proper operation of the fire detection, annunciation and fire fighting systems.

1.2.2.16.7 Floor Leakage Detection System

The drainage system is also used to detect abnormal leakage in safety related equipment rooms and the fuel transfer area.

1.2.2.16.8 Vacuum Sweep System

A portable, submersible-type, underwater vacuum cleaner is provided to assist in removing crud and miscellaneous particulate matter from the pool floors or reactor vessel. The pump and the filter unit are completely submersible for extended periods. The filter "package" is capable of being remotely changed, and the filters will fit into a standard shipping container for offsite burial.

1.2.2.16.9 Decontamination System

The decontamination system provides areas, equipment and services to support low radiation level decontamination activities. The services may include electrical power, service air, demineralized water, condensate water, radioactive and nonradioactive drains, HVAC and portable shielding.

1.2.2.16.10 Reactor Building

The reactor building includes the containment, drywell, and major portions of the nuclear steam supply system, steam tunnel, refueling area, diesel generators, essential power, non-essential power, emergency core cooling systems, HVAC and supporting systems;

1.2.2.16.11 Turbine Building

The turbine building houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.

Table 1.9-1

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ITEM NO.	SUBJECT	SUBSECTION
1.1	Standard review plan sections for remainder of plant identified as "Interface" in Table 1.8-19	1.8.4
1.2	Applicability of regulatory guides for remainder of plant indentified as "Interface" in Table 1.8-20	1.8.4
1.3	Applicability of Experience Information for remainder of plant identified as "Interface" in Table 1.8-22	1.8.4
1.4	Emergency procedures and emergency procedures training program	1A.3.1
1.5	Procedures for removing safety-related systems from service	1A.3.2
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1A.2.13 Containment Design-Dedicated Penetration [II.E.4.1]

NRC Position

For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.

Response

A flammability control system (FCS-T49) is provided to control the concentration of oxygen in the primary containment. The FCS utilizes two permanently installed recombiners located in secondary containment. The FCS is operable in the event of a single active failure. The FCS is described in Subsection 6.2.5.

1A.2.14 Containment Design-Isolation Dependability [II.E.4.2]

NRC Position

(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan Subsection 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be non-essential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

(3) All nonessential systems shall be automatically isolated by the containment isolation signal.

(4) The design of control systems for automatic containment isolation valves shall be such that reseting the isolation signal will not result in the

automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

(5) The containment setpoint pressure that initiates containment isolation for non-essential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, Item II.6.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

(7) Containment purge and vent isolation valves must close on a high radiation signal.

Response

(1) The isolation provisions described in the Standard Review Plan, Subsection 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation) were reviewed in conjunction with the ABWR Standard Plant design. It was determined that the ABWR Standard Plan is designed in accordance with these recommendations of the SRP.

(2) This request appears to be directed primarily toward operating plants. However, the classification of structures, systems and components for the ABWR Standard Plant design is addressed in Section 3.2 of this SSAR. The basis for classification is also presented in Section 3.2. The ESF system, with remote manual valves with leakage detection outside containment are delinated in Tables 6.2-7. The ABWR Standard Plant fully conforms with the NRC position so far as it relates to the new equipment supplier.

(3) All non-essential systems comply with the NRC position to automatically isolate by the containment isolation signals, and by redundant safety grade isolation valves.

(4) Control systems for automatic containment isolation valves are designed in accordance with this position for the ABWR Standard Plant Design.

(5) The ABWR Standard Plant design is consistant with this position.

(6) All ABWR containment purge valves meet the criteria provided in BTP CSB 6-4. The main 22" purge valves are fail-closed and are maintained closed through power operation as defined in the plant technical specifications. All purge and vent valves are remote pneumatically operated, fail closed and receive containment isolation signals. Certain vent valves can be opened manually in the presents of an isolation signal, to permit venting through the SGTS.

(7) In the ABWR design, the containment purge and vent isolation valves will be automatically isolated on high radiation levels detected in the reactor building HVAC air exhaust or in the fuel handling area air exhaust.

1A.2.15 Additional Accident-Monitoring Instrumentation [II.F.1(1)]

NRC Position

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

(1) Noble gas effluent monitors with an upper range capacity of 10^5 Ci/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA)) concentrations to a maximum of 10^5 Ci/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by onsite laboratory analysis.

In-containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and -5 psig for all containments.

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for BWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for BWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

Response

GE believes the requirements of Regulatory Guide 1.97, Revision 3, incorporate the above requirements. Section 7.5 compares the ABWR design against this Regulatory Guide.

1A.2.16 Identification of and Recovery From Conditions Leading to Inadequate Core Cooling [II.F.2]

NRC Position

Licensees shall provide a description of any additional instrumentation controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Response

The direct water level instrumentation provided in the ABWR design is capable of detecting conditions indicative of inadequate core cooling.

The ABWR has two sets of four wide range reactor water level sensing units (eight total) which are used in two separate two out of four logics which initiate ECCS and other safety functions. Each set of four sensors are used in two separate two out of four logics which initiate ECCS operation. Four separate sets of sensing lines, one from each quadrant of the reactor pressure vessel, supply the pressure to the eight sensors for reliability. This ABWR arrangement of reactor water level sensing exceeds or is at least equal to the redundancy and reliability of the BWR reator water level measurement systems reviewed in BWR Owners Group Report SLI-821, July 1982. The conclusions reached in SLI-821 and companion report SLI-8218, December 1982, also apply to the ABWR. These conclusions meet the NRC staff expectation given in paragraph 4.4.7 of GESSAR II SER (Safety Evaluation Report NUREG-0979, April 1983) regarding NUREG-0737, Item II F.2.

Based on the above information, the existing highly redundant direct water level instrumentation already provides an unambiguous, easy to interpret indication of inadequate core cooling and there are no plans to include core-exit thermocouples in the ABWR design.

TABLE 2.0-1

ENVELOPE OF ABWR STANDARD PLANT SITE DESIGN PARAMETERS

Maximum Ground Water Level:
61.0 cm below grade

Maximum Flood (or Tsunami) Level:[3]
30.5 cm below grade

Precipitation (for Roof Design):
- Maximum rainfall rate: 49.3 cm/hr[8]
- Maximum snow load: 0.024 kg/cm^2

Design Temperatures:
- Ambient
  1% Exceedance Values
  - Maximum: 37.8°C dry bulb/25°C wet bulb (coincident), 26.6°C wet bulb (non-coincident)
  - Minimum: -23.3°C

  0% Exceedance Values (Historical limit)
  - Maximum 46.1°C dry bulb/26.7°C wet bulb (coincident), 27.2°C wet bulb (non-coincident)
  - Minimum: -40°C

Extreme Wind: Basic Wind Speed:
177 km/hr[1]/209 km/hr[2]

Tornado:[4]

- Maximum tornado wind speed:	483 km/hr
- Maximum Rotational Speed:	386 km/hr
- Translational velocity:	97 km/hr
- Radius:	45.7 m
- Maximum pressure drop:	0.141 kg/cm^2
- Rate of pressure drop:	0.0846 kg/cm^2/sec
- Missile Spectra:	Per SRP 3.5.1.4 Spectrum I

Soil Properties:

- Minimum Static Bearing Capacity:	7.32 kg/cm^2
- Minimum Shear Wave Velocity:	305 m/sec[9]
- Liquefaction Potential:	None at plant site resulting from SSE[7]

Seismology:
- SSE Peak Ground Acceleration: 0.30g[5]
- SSE Response Spectra: per Reg.Guide 1.60
- SSE Time History: Envelope SSE Response Spectra

---

(1) *50-year recurrence interval; value to be utilized for design of non-safety-related structures only.*

(2) *100-year recurrence interval; value to be utilized for design for safety-related structures only.*

(3) *Probable maximum flood level (PMF), as defined in ANSI/ANS-2.8, "Determining Design Basis Flooding at Power Reactor Sites."*

(4) *10,000,000-year tornado recurrence interval.*

(5) *Free-field, at plant grade elevation.*

(6) *Deleted*

(7) *See item 3 in Section 3A.1 for additional information.*

(8) *Maximum value for 1 hour over 2.6 km^2 probable maximum precipitation (PMP) with ratio of 5 minutes to 1 hour PMP of 0.32 as found in National Weather Source Publication HMR No. 52. Maximum short term rate: 15.7 cm/5 min.*

(9) *This is the minimum shear wave velocity at low strains after the soil property uncertainties have been applied.*

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2.3 COL LICENSE INFORMATION

2.3.1 Envelope of Standard Plant Design Parameters

2.3.1.1 Non-Seismic Design Parameters

Compliance with the envelope of ABWR Standard Plant site non-seismic design parameters of Table 2.0-1 shall be demonstrated for design bases events. (See Subsection 2.2.1)

2.3.1.2 Seismic Design Parameters

To confirm the seismic design adequacy of the standard plant, the COL applicants shall demonstrate that the eight (8) site-dependent conditions specified in Section 3A.1 are satisfied. In meeting these eight conditions, the compliance with the site envelope parameters shown in Table 2.0-1 for soil properties and seismology is also established.

If there is any deviation of the eight site-dependent conditions, a site specific evaluation is required. The type of evaluation will vary depending on the deviation. If the deviation is for condition 1 (peak ground acceleration), 2 (ground response spectra), or 6 (shear wave velocity), a site specific SSE soil-structure interaction analysis (SSI) is required. The calculated site unique responses are compared to the site-envelope responses defined in Section 3G.4 to confirm the seismic design adequacy of the standard plant according to the following procedures and acceptance criteria.

The Seismic Category I structures including the RPV and its internal components that are included in the SSI analysis model:

(1) Design adequacy is established if maximum structural responses in terms of force, moment, or acceleration are bounded by the Section 3G.4 responses (or the actual seismic loads considered in design if applicable) at key locations.

(2) If not, calculate resulting SSE stresses. Design adequacy is confirmed if combined stresses due to SSE and other appropriate loads are within design code allowable limits.

For Seismic Category I equipment and piping whose seismic input is in the form of floor response spectra:

(1) Design adequacy is established if floor response spectra are bounded by Section 3G.4 spectra (or the actual spectra considered in design if applicable) at key locations. The site unique response spectra used for comparison need not be broadened since uncertainties in the structural frequencies have been accounted for in the smooth broadened site envelope spectra.

(2) If not, examine whether the deviations are at major resonant frequencies of the component under consideration. If not, design adequacy is confirmed. Otherwise, perform analysis and/or testing to demonstrate that the acceptance criteria given in design specifications are met.

If the soil properties of the site vary very abruptly with depth (site-dependent condition 7), a site specific SSE SSI analysis is required. The evaluation procedures and acceptance criteria specified above are applicable.

If the soil bearing capacity at the site is not adequate to accommodate the standard plant design loads (site-dependent condition 8), the foundation material shall be removed and replaced with better material to achieve the required bearing capacity. Alternatively, the applicant referencing the ABWR design may perform a site specific analysis to demonstrate that the site has an adequate bearing capacity against the site unique loads.

The site-dependent conditions 3 (liquefaction potential) and 4 (fault displacement potential) require site specific investigation.

A site specific evaluation is required if the embedment depths of Seismic Category I buildings deviate from those from the standard plant design (site-dependent condition 5). The evaluation procedure and acceptance criteria are the same as those defined above for the site specific SSE SSI analysis.

2.3.2 Standard Review Plan Site Characteristics

Identification and description of all differences from SRP Section II Acceptance Criteria for site characteristics (as augmented by Table 2.1-1) shall be provided. Where such differences exist, the evaluation shall discuss how the alternate site characteristic is acceptable. In addition, the COL applicant will provide/address the following:

2.3.2.1 Site Location and Description

COL applicants will provide site-specific information to site location, including political subdivisions, natural and man-made features, population, highways, railways, waterways, and other significant features of the area.

2.3.2.2 Exclusion Area Authority and Control

COL applicants will provide site-specific information related to activities that may be permitted within the designated exclusion area.

2.3.2.3 Population Distribution

COL applicants will provide population data for the site environs.

2.3.2.4 Identification of Potential Hazards in Site Vicinity

COL applicants will provide information with respect to industrial, military, and transportation facilities and routes to establish the presence and magnitude of potential external hazards.

2.3.2.5 Evaluation of Potential Accidents

COL applicants will identify potential accident situations in the vicinity of the plant and the bases for which these potential accidents were or were not accommodated in the design.

2.3.2.6 External Impact Hazards

COL applicants will provide a review and evaluation of the effects on the protection criteria of some external impact hazards, such as general aviation or nearby explosions.

2.3.2.7 Local Meteorology

COL applicants will provide local meteorology for NRC review

2.3.2.8 Onsite Meteorological Measurements Program

COL applicants will provide the onsite meteorological measurements programs.

2.3.2.9 Short-Term Dispersion Estimates for Accidental Atmospheric Releases

COL applicants will provide site-specific short-term dispersion estimates for NRC review to ensure that the envelope values (Tables 15.6-3, 15.6-7, 15.6-13, 15.6-14 and 15.6-18) of relative concentrations are not exceeded.

2.3.2.10 Long-Term Diffusion Estimates

COL applicants will provide annual average atmospheric dispersion values for routine releases for NRC review.

2.3.2.11 Hydrologic Description

COL applicants will provide a detailed description of all major hydrologic features on or in the vicinity of the site. They will also provide a specific description of the site and all safety-related elevations, structures, exterior accesses, equipment, and systems from the standpoint of hydrology considerations.

2.3.2.12 Floods

COL applicants will provide site-specific information related to historical flooding and the potential flooding at the plant site, including flood history, flood design considerations, and effects of local intense precipitation.

2.3.3.13 Probable Maximum Flood on Streams and Rivers

COL applicants will provide site-specific information related to determining design-basis flooding at po[illegible] reactor sites and the extent of flood protection [illegible]quired for those safety-related systems, structures, and components.

2.3.2.14 Ice Effects

COL applicants will demonstrate that safety-related facilities and water supply are not affected by ice flooding or blockage.

2.3.2.15 Cooling Water Channels and Reservoirs

COL applicants will provide the basis for the hydraulic design of channels and reservoirs used to transport and impound plant cooling and for protection of safety-related structures.

2.3.2.16 Channel Division

COL applicants will provide site-specific information related to channel diversion.

2.3.2.17 Flooding Protection Requirements

COL applicants will provide site-specific information related to flooding protection requirements.

2.3.2.18 Cooling Water Supply

COL applicants will identify natural events that may reduce or limit the available cooling water supply and ensure that an adequate water supply will exist to operate or shutdown the plant as required.

2.3.2.19 Accidental Release of Liquid Effluents in Ground and Surface Waters

COL applicants will provide information on the ability of the surface water environment to disperse, dilute, or concentrate accidental releases. Effects of these releases on existing and known future use of surface water/resources shall also be provided.

2.3.2.20 Technical Specifications and Emergency Operation Requirement

COL applicants will establish the technical specifications and emergency procedures required to implement flood protection for safety-related facilities and provide assurance of an adequate water supply to shutdown and cool the reactor.

2.3.2.21 Basic Geological and Seismic Information

COL applicants will provide site-specific information related to regional and site physiography, geomorphology, stratigraphy, lithology and tectonics.

2.3.2.22 Vibratory Ground Motion

COL applicants will develop site-specific geological, seismological, and geotechnical data and will submit these data to the NRC for review. These data should be comparable to the design basis assumptions regarding the SSE, including the verification of the ground motion response spectra.

2.3.2.23 Surface Faulting

COL applicants will develop site-specific geological data to ensure that no potential exists for surface faulting at the site.

2.3.2.24 Stability of Subsurface Material and Foundation

COL applicants will develop and submit to the NRC site-specific geotechnical data to demonstrate that they are comparable to the design assumptions concerning the soil-deposit depths, the soil profile and properties, and the ground water level. Particular attention should be paid to the assumptions for the depth of embedment in the case of rock and the three cases of soil-deposit depths for which fixed values of depths are assumed. COL applicants will demonstrate that the envelope of structural response with fixed soil depth will cover completely the cases for which the soil deposit depths and properties are different from those assumed in the SSAR.

2.3.2.25 Site and Facilities

COL applicants will provide a detailed description of the site conditions and geologic features and demonstrate the site characteristics are enveloped by the 0.3g peak horizontal ground acceleration for the SSE. The description will include site topographical features and the location of various Seismic Category I structures and appurtenances (pipelines, channels, etc.) with respect to the source of normal and emergency cooling water.

2.3.2.26 Field Investigations

The type, quantity, extent, and purpose of all field exploration will be provided by COL applicants. Logs of all borings and test pits should be provided. Results of geophysical surveys should be presented in tables and profiles. Records of field plate load tests, field permeability tests, and other special field tests (e.g., bore-hold extensometer or pressuremeter tests) should be given.

2.3.2.27 Laboratory Investigations

The number and type of laboratory tests and the location of samples should be provided by the COL applicant in tabular form. The results of laboratory tests on disturbed and undisturbed soil and rock samples obtained from field investigations should also be provided.

2.3.2.28 Subsurface Conditions

COL applicants will investigate and define the subsurface conditions and provide the engineering classifications and descriptions of soil and rock supporting the foundations. The information should include the history of soil deposition and erosion, past and present goundwater levels, glacial or other preloading influences, rock weathering, and any rock or soil characteristics that may present a hazard to plant safety. Profiles through the Seismic Category I structures will be provided that show generalized subsurface features beneath these structures.

2.3.2.29 Excavation and Backfilling for Foundation Construction

COL applicants will provide site specific thickness and properties of soil (if any) between the base of the foundation and the underlying rock. The configuration, along with detailed longitudinal sections and cross sections of other safety-related structures of the plant, including the ultimate heat sink and Seismic Category I buried pipes and electrical ducts, should be provided. COL applicants will provide data concerning the extent (horizontally and vertically) of all Seismic Category I excavations, fills, and slopes. The locations, elevations, and grades for excavated slopes should be described and shown on plot plans and typical cross-sections. COL applicants' submittals should discuss, as appropriate, excavating and dewatering methods, excavation depths below grade, field inspection and testing of excavations, protection of foundation excavations from deterioration during construction, and the foundation dental fill work. The sources, quantities, and static and dynamic engineering properties of borrow materials will be described. The compaction requirements; results of test fills, and fill properties, such as moisture content, density, permeability, compressibility, and gradation should be provided.

2.3.2.30 Groundwater Level

COL applicants will analyze the goundwater condition for the specific site and demonstrate its comparability with the ABWR design assumptions. This demonstration should include the effect of the actual groundwater level on such site geotechnical properties as total and effective unit weights, cohesion and angle of internal friction, and dynamic soil properties used in dynamic response analysis.

2.3.2.31 Liquefaction Potential

COL applicants must demonstrate that no liquefaction potential exists for soils under and around all Seismic Category I structures, including Category I buried pipelines and electrical ducts. COL applicants will justify the selection of the soil properties used in the liquefaction potential evaluation (e.g., laboratory tests, field tests, and published data), the magnitude and duration of the earthquake and the number of cycles of earthquakes.

2.3.2.32 Response of Soil and Rock to Dynamic Loading

COL applicants must establish and document site-specific geotechnical properties to demonstrate their comparability with the conditions used for the seismic design envelope described in Appendix 3A.

2.3.2.33 Maximum Soil Bearing Pressure

COL applicants will provide the site-specific maximum soil pressure along with supporting calculations and will compare them with allowable values.

2.3.2.34 Earth Pressures

COL applicants will provide a discussion and evaluation of static and dynamic lateral earth pressures and hydrostatic groundwater pressures acting on plant facilities to the extent necessary to demonstrate that these pressures meet the design bases for the ABWR and to address all facilities outside the ABWR scope.

2.3.2.35 Soil Properties for Seismic Analysis of Buried Pipes

COL applicants will provide and justify the soil properties used fro the seismic analysis of seismic Category I buried pipes and electrical conduits.

2.3.2.36 Static and Dynamic Stability of Facilities

COL applicants will analyze all safety-related facilities to the extent necessary to demonstrate that their stability meets the ABWR design bases and to address all plant facilities outside the ABWR scope. These analyses may include foundation rebound, settlement, differential settlement, and bearing capacity that will be addressed for design loads of fills and plant facilities. Assumptions made in stability analyses will be confirmed by as-built data.

2.3.2.37 Subsurface Instrumentation

Instrumentation, if any, proposed for the surveillance of the performance of the foundations for safety-related structures will be described by COL applicants. The type, location, and purpose of each instrument and significant details of installation methods will be provided. For example, the location and the installation procedures for permanent benchmarks and markers required for monitoring the settlement of Category I structures should be described. In the case of safety-related water-control structures (such as dams, slopes, canals), the details of installing instrumentation such as piezometers, slope indicators, and settlement plates should be described. A schedule for installing and reading all instruments and for interpreting the data will be presented. Limiting values for continued safety should be identified.

2.3.2.38 Stability of Slopes

COL applicants will provide information about the static and dynamic stability of all soil and rock slopes, the failure of which could adversely affect the safety of the plant. The staff will evaluate the stability of all slopes at the site, using the state-of-the-art procedures available at the time of application.

2.3.2.39 Embankments and Dams

COL applicants should provide information about the static and dynamic stability of all embankments and dams that impound water required for safe operation (and shutdown) of the ABWR for review by the NRC if embankments and dams are used.

2.3.3 CRAC 2 Computer Code Calculations

Compliance with acceptance criteria, data input and analysis of Subsection 2.2.2 for the determination of ABWR site acceptability for severe accidents shall be demonstrated.

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TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
K1 Radwaste System						
1. Drain piping including supports and valves - radioactive	N	ALL (except RZ,X)	D	E	--	(p)
2. Drain piping including supports and valves - nonradioactive	N	ALL	D	E	---	(p)
3. Piping and valves - containment isolation	2	C,SC	B	B	I	
4. Piping including supports and valves forming part of containment boundary	N	C,SC	B	B	I	
5. Pressure vessels including supports	N	W	---	E	---	(p)
6. Atmospheric tanks including supports	N	C,SC,H, T,W	---	E	---	(p)
7. 0-15 PSIG Tanks and supports	N	W	---	E	---	(p)
8. Heat exchangers and supports	N	C,SC,W	---	E	---	(p)
9. Piping including supports and valves	N	C,SC,H, T,W	---	E	---	(p)
10. Other mechanical and electrical modules	N	ALL	---	E	E	(p)
11. ECCS equipment room sump backflow protection check valves	N	SC	C	B	I	
N1 Turbine Main Steam System						
1. Deleted (See B2.5)						

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
N1 Turbine Main Steam System (Continued)						
2. Branch line of MSL including supports between the second isolation valve and the turbine stop valve from branch point at MSL to and including the first valve in the branch line	N	SC,T	B	B	---	(r)
N2 Condensate, Feedwater and Condensate Air Extraction System						
1. Main feedwater line (MFL) including supports from second isolation valve branch lines and components beyond and including outboard shutoff valves	N	SC	B	B	I	
2. Feedwater system components beyond outboard shutoff valve	N	T	D	E	---	
N3 Heater, Drain and Vent System	N	T	---	E	---	
N4 Condensate Purification System	N	T	---	E	---	
N5 Condensate Filter Facility	N	T	---	E	---	
N6 Condensate Demineralizer	N	T	---	E	---	
N7 Main Turbine	N	T	---	E	---	
N8 Turbine Control System						
1. Turbine stop valve, turbine bypass valves, and the main steam leads from the turbine stop valve up to the turbine casing	N	T	D	E	---	(l)(n)(o) (r)

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
P3 Reactor Building Cooling Water System (Continued)						
3. Electrical modules with with safety-related function	3	SC,C,X	---	B	I	
4. Cable with safety-related function	3	SC,C,X	---	B	I	
5. Other mechanical and electrical modules	N	SC,C,X,M	---	E	---	
P4 Turbine Building Cooling Water System	N	T	D	E	---	
P5 HVAC Normal Cooling Water System	N	C,SC,RZ, T,X	---	E	---	
P6 HVAC Emergency Cooling Water System						
1. Chillers, pumps, valves, and piping including supports	3	SC,X	C	B	I	
2. Piping including supports and valves forming part of containment boundary	2	C,SC	B	B	I	
3. Electrical modules and cable with safety-related function	3	SC,X	---	B	I	
4. Other mechanical and electrical modules	N	C,SC,RZ, T,X	---	E	---	
P7 Oxygen Injection System	N	T	---	E	---	
P8 Ultimate Heat Sink	3	O	C	B	I	
P9 Reactor Service Water System						
1. Safety-related piping including supports, piping and valves	3	U,O,X	C	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
P9 Reactor Service Water System (Continued)						
2. Electrical modules and cables with safety-related function	3	U,O,X	---	B	I	
3. Other non-safety related mechanical and electrical modules	N	U,O,X	---	E	---	
P10 Turbine Service Water System						
1. Non-safety related piping including supports, piping and valves	N	P,O,T	---	E	---	260.4
2. Electrical modules and cables with non-safety related function	N	P,O,T	---	E	---	
P11 Station Service Air System						
1. Containment isolation including supports, valves and piping	2	C	B	B	I	210.20
2. Other non-safety related mechanical and electrical components	N	SC,RZ, X,T,H, W,C	---	E	---	260.4
P12 Instrument Air Service						
1. Containment isolation including supports, valves and piping	2	C	B	B	I	210.20
2. Other non-safety related mechanical and electrical components	N	SC,RZ, X,T,H, W,C	---	E	---	260.4
P13 High Pressure Nitrogen Gas Supply Systems						
1. Containment isolation including supports, valves and piping	2	C	B	B	I	

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SECTION 3.3

TABLES

3.3 WIND AND TORNADO LOADINGS

ABWR Standard Plant structures which are Seismic Category I are designed for tornado and extreme wind phenomena.

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

Seismic Category I structures are designed to withstand a design wind velocity of 130 mph at an elevation of 33 feet above grade with a recurrence interval of 100 years. See Subsection 3.3.3.1 for interface requirement.

3.3.1.2 Determination of Applied Forces

The design wind velocity is converted to velocity pressure in accordance with Reference 1 using the formula:

$$q_z = 0.00256\, K_z\, (IV)^2$$

where K_z = the velocity pressure exposure coefficient which depends upon the type of exposure and height (z) above ground per Table 6 of Reference 1.

I = the importance factor which depends on the type of exposure; appropriate values of I are listed in Table 3.3-1,

V = design wind velocity of 130 mph, and

q_z = velocity pressure in psf

The velocity pressure (q_z) distribution with height for exposure types C and D of Reference 1 are given in Table 3.3-2.

The design wind pressures and forces for buildings, components and cladding, and other structures at various heights above the ground are obtained, in accordance with Table 4 of Reference 1 by multiplying the velocity pressure by the appropriate pressure coefficients and gust factors. Gust factors are in accordance with Table 8 of Reference 1. Appropriate pressure coefficients are in accordance with Figures 2, 3a, 3b, 4, and Tables 9 and 11 through 16 of Reference 1. Reference 2 is used to obtain the effective wind pressures for cases which Reference 1 does not cover. Since the Seismic Category I structures are not slender or flexible, vortex-shedding analysis is not required and the above wind loading is applied as a static load.

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

The design basis tornado is described by the following parameters:

(1) A maximum tornado wind speed of 300 mph at a radius of 150 feet from the center of the tornado;

(2) A maximum translational velocity of 60 mph;

(3) A maximum tangential velocity of 240 mph, based on the translational velocity of 60 mph;

(4) A maximum atmospheric pressure drop of 2.00 psi with a rate of the pressure change of 1.2 psi per second; and

(5) The spectrum of tornado-generated missiles and their pertinent characteristics as given in Subsection 3.5.1.4.

See Subsection 3.3.3.2 for COL license information.

3.3.2.2 Determination of Forces on Structures

The procedures of transforming the tornado loading into effective loads and the distribution across the structures are in accordance with Reference 4. The procedure for transforming the tornado-generated missile impact into an effective or equivalent static load on structures is given in Subsection 3.5.3.1. The loading combinations of the individual tornado loading components and the load factors are in accordance with Reference 4.

The reactor building and control building are not vented structures. The exposed exterior roofs and walls of these structures are designed for the 2.00 psi pressure drop. Tornado dampers are provided on all air intake and exhaust openings. These dampers are designed to withstand a negative 1.46 psi pressure.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

All safety-related system and components are protected within tornado-resistant structures.

See Subsection 3.3.3.3 for COL license information.

3.3.3 COL License Information

3.3.3.1 Site-Specific Design Basis Wind

The site-specific design basis wind shall not exceed the design basis wind given in Table 2.0-1 (See Subsection 2.2.1).

3.3.3.2 Site-Specific Design Basis Tornado

The site-specific design basis tornado shall not exceed the design basis tornado given in Table 2.0-1 (See Subsection 2.2.1).

3.3.3.3 Effect of non-Seismic Category I Structures, and Components not Designed for Tornado Loads

The COL applicant will ensure that the collapse of non-seismic Category I structures, such as cooling towers or stacks outside the scope of the ABWR Standard Plant, will not endanger seismic Category I structures and that site-dependent effects of blast loads will be less than those of design tornado pressures (see Subsection 3.3.2.3).

3.3.4 References

1. ANSI Standard A58.1, *Minimum Design Loads for Buildings and Other Structures*, Committee A. 58.1, American National Standards Institute.

2. ASCE Paper No. 3269, *Wind Forces on Structures*, Transactions of the American Society of Civil Engineers, Vol. 126, Part II.

3. Deleted

4. Bechtel Topical Report BC-TOP-3-A, Revision 3, *Tornado and Extreme Wind Design Criteria for Nuclear Power Plants*.

3.4.1.1.2.1.6 Evaluation of Floor 600 (3F)

Flooding events at this floor level may involve fuel oil as well as water. Those divisional rooms associated with the emergency diesel generator fuel tank and cooling system, have the potential of leakage from the fuel storage tanks. These rooms must accommodate leakage of 11.4 cubic meter (3000 gallons) for each division. Twenty cm (8 inches) sills on entry to these areas successfully contain all the volume in the tanks. Leakage from these tanks will also be monitored through safety grade level indication and alarm equipment so that protracted leakage as well as gross leakage can be identified. The rooms are protected by CO_2 firefighting system. Water flooding may occur from the cooling system at about .15 cubic meter/minutes (41 gpm). If undetected for several hours water may begin cascading down the nearest stairwell but is prevented from entering other division areas by raised sills.

In the SGTS areas, the room cooling equipment may cause flooding at a rate .15 cubic meter/minute (41 gpm). Raised sills prevent intrusion of water into rooms of another division. Flooding may also occur from manual firefighting in equipment maintenance areas or from leakage from the standby liquid control tanks. Maximum tank leak rate will be .1 cubic meter/minute (25 gpm) so that a response to tank level alarms within 10 minutes will limit loss to one cubic meter (or 250 gallons). Large floor areas permit spread of water at limited depth.

3.4.1.1.2.1.7 Evaluation of Floor 700 (M4F)

Flooding in the FMCRD panel rooms may occur from firefighting activities at an input rate of .57 cubic meters/minute (150 gpm). Since these activities are manually controlled, any excessive depth of water will be noted and action taken to mitigate water intrusion to other areas.

Flooding on this level may also occur from room cooling systems or from firefighting efforts. Cooling system failures in air supply, exhaust or filter rooms may allow flooding at the rate of .3 cubic meter/minute (80 gpm) which will flow out into adjacent corridor areas. If undetected for 10 minutes, the approximate 3 cubic meter (800 gallons) released may create a depth of a few millimeters over the available floor area; a very limited amount of water will cascade down the stairwells. Divisional areas encompassing the three emergency electric supply fans and the RIP A exhaust will include raised sills to preclude water intrusion although water depth will be slight. Equipment pedestals will also minimize flooding impact on all equipment.

Firefighting activities in this area would cause water inflow of .57 cubic meter/minute (150 gpm) under controlled conditions and expected water intrusion is no more than that above.

3.4.1.1.2.1.8 Evaluation of Floor 800 (4F)

Flooding on this floor can be caused by rupture of the RCW surge tanks A, B & C piping. However, each tank and its associated piping is located in a separate compartment which can be sealed off in the event of accidental flooding. The use of raised sills on entry ways will contain the seepage to the flooded area. Also, the use of pedestals for equipment installation of the RIP supply and exhaused fans and for the DG-C exhaust fans will guard against flooding this equipment.

Flooding in the main reactor hall may occur from reactor service operations, but will be drained into service pools. Firefighting water expended into this area would occur at a maximum rate of .57 cubic meter/minute (150 gpm) but will spread over the large service area available. Minor amounts of water may find the way to stairwells, but would not impede operations.

3.4.1.1.2.1.9 Flooding Summary Evaluation

Floor-by-floor analysis of potential pipe failure generated flooding events in the reactor building shows the following:

(1) Where extensive flooding may occur in a division rated compartment, propagation to other divisions is prevented by watertight doors or sealed hatches. Flooding in one division is limited to that division and flood water cannot propagate to other divisions.

(2) Leakage of water from large circulating water lines, such as reactor building cooling water lines may flood rooms and corridors, but through sump alarms and leakage detection systems the control room is alerted and can control flooding by system isolation. Divisional areas are protected by watertight doors, or where only limited water depth can occur, by raised sills with pedestal mounted equipment within the protected rooms.

(3) Limited flooding that may occur from manual firefighting or from lines and tanks having limited inventory is restrained from entering division areas by raised sills and elevation differences.

Therefore, within the reactor building, internal flooding events as postulated will not prevent the safe shutdown of the reactor.

3.4.1.1.2.2 Evaluation of Control Building Flooding Events

The control building is a seven story building. It houses in separate areas, the control room proper, control and instrument cabinets with power supplies, closed cooling water pumps and heat exchangers, mechanical equipment (HVAC and chillers) necessary for building occupation and environmental control for computer and control equipment, and the steam tunnel.

The only high energy lines in the control building are the mainsteam lines and feedwater lines which pass through the steam tunnel connecting the reactor building to the turbine building. There are no openings into the control building from the steam tunnel. The tunnel is sealed at the reactor building end and open at the turbine building end. It consists of reinforced concrete with 2 meter thick walls. Any break in a mainsteam or a feedwater line will flood the steam tunnel with steam. The rate of blowdown will cause most of the steam to vent out of the tunnel into the turbine building. Water or steam cannot enter the control building. See Section 3.6.1.3.2.3 for a description of the subcompartment pressurization analysis performed for the steam tunnel.

Moderate energy water services in the control building comprise 28-inch service water lines, 18-inch cooling water lines, 6-inch cooling water lines to the chiller condenser, 6-inch fire protection lines, and 6-inch chilled water heater lines. Smaller lines supply drinking water, sanitary water and makeup for the chilled water system. Areas with water pipe routed through are supplied with floor drains and curbs to route leakage to the basement floor so that control or computer equipment is not subjected to water. In those areas where water infusion cannot be tolerated, the access sills are raised.

Maximum flooding may occur from leakage in a 28-inch service water line at a maximum rate of 12.0 cubic meters/minute (3150 gpm). Early detection by alarm to control room personnel will limit the extent of flooding which will also be mitigated by drainage to exterior of the building. The expected release of a service water leak is limited to line volume plus operator response time times leakage rate. The assumed operator response time is 30 minutes to close isolation valves and turn off the pump in the affected service water division. Water will be contained inside a division of closed cooling water equipment rooms in the bottom level of the control building. A maximum of 5.0 meters of water in a divisional room is expected. Water tight doors will confine the water to a division.

The failure of a cooling water line in the mechanical rooms of the turbine building may result in a leak of 0.6 cubic meter/minute (160 gpm). Early detection by control room personnel will limit the extent of flooding. Total release from the chilled water system will be limited to line inventory and surge tank volume, spillage of more than 6 cubic meters (1500 gallons) is unlikely. Elevation differences and separation of the mechanical functions from the remainder of the control building prevent propagation of the water to the control area.

maximum of 2.15 meters of water in a divisional room is expected. Water tight doors will confine the water to a division.

The failure of a cooling water line in the mechanical rooms of the turbine building may result in a leak of 0.6 cubic meter/minute (160 gpm). Early detection by control room personnel will limit the extent of flooding. Total release from the chilled water system will be limited to line inventory and surge tank volume, spillage of more than 6 cubic meters (1500 gallons) is unlikely. Elevation differences and separation of the mechanical functions from the remainder of the control building prevent propagation of the water to the control area.

Flooding events that may result from the failure of the fire fighting systems within the control building do not inhibit plant safety. There are no sprinkler systems in the control building. Hose and standpipes are located in the corridors. Service equipment rooms may build up limited water levels from either service water, cooling water, or chilled water leaks, but elevation differences and raised sills prevent intrusion of water into control areas. Control room responses to those various levels of flooding may extend from system isolation and correction to reduction of plant load or shutdown, but control room capability is not compromised by any of the postulated flooding events.

3.4.1.1.2.3 Evaluation of Radwaste Building Flooding Event

The radwaste building is a reinforced concrete structure designed as Seismic Category I, consisting of a substructure 13.8 meters below grade and a super-structure 16 meters above grade. This building does not contain safety-related equipment and is not contiguous with other plant structures except through a pipe tunnel. In case of a flood, the building substructure serves as a large sump which can collect and hold any leakage within the building. Also, the medium and large radwaste tanks are housed in sealed compartments which are designed to contain any spillage or leakage from tanks that may rupture. The piping that transfers the liquid waste from the other buildings traverses through a sealed water-tight tunnel to the radwaste building at an elevation of -3,500 mm, which is 3 meters above the radwaste building basement slab. This tunnel connects to the turbine and reactor buildings at the same elevation.

The structural design of this building is such that no internal flooding is expected or will occur under the worst case conditions from those tanks that are isolated by the Seismic Category I compartments.

Flooding from other sources within the building such as internal radwaste and non-radwaste piping, plant drains, small tanks, and pumps is not expected to cause the water level to rise more than 1 meter above the flood depth of 3 meters to reach the tunnel and spread radioactive liquid waste to other buildings that house safety-related systems.

Therefore, it can be concluded from the above analysis that there is no uncontrolled leakage path of radioactive liquid from the radwaste building under the conditions of worst-case internal flooding.

3.4.1.1.2.4 Evaluation of Service Building Flooding Events

The service building is a non-seismic concrete structure consisting of four floors, two above and two below grade. It serves as the main security entrance to the plant and provides the controlled access tunnels to the control building, the turbine building, and the reactor building. This building does not house any safety-related equipment.

The connecting access tunnels to other buildings are below plant grade as indicated in Table 3.4-1. These passage ways are water tight to prevent seepage into the tunnels. Also, the controlled access chambers employ curbs and closed doors at both ends of the tunnel that guard against water leakage into structures that house safety-related equipment.

The only plant piping that run through this building are those needed for fire protection, water services, HVAC heaters and chillers, and for draining the sumps. This building has floor drains and two sump pumps (HCW & HSD) for collecting and transferring the liquid waste. Under worst-case conditions, flooding from line ruptures is unlikely and can be contained from spreading to the structures that house safety-related equipment.

3.4.1.1.2.5 Evaluation of Turbine Building Flooding Events

Circulating water system and turbine building service water system are the only systems large enough to fill the condenser pit; therefore, only these two systems can flood into adjacent buildings.

A failure in either of these systems will result in the total flooding of the turbine building up to grade. Water is prevented from crossing to other buildings by two means. The first is a normally closed alarmed door in the connecting passage between the turbine building and service building. The second is that the radwaste tunnel will be sealed at all ends to prevent water from either entering the tunnel or leaving the tunnel. A large hydrostatic head is prevented by a large non-water-tight truck door at grade to provide a release point for any water.

Because of the large size of the circulating water system, a leak will fill the condenser pit quickly. Monitors were added in the condenser pit of the turbine building to provide leak detection and an automatic means to shutdown the circulating water system in the event of flooding in the turbine building (see Subsection 10.4.5.2.3 and 10.4.5.6).

3.4.1.2 Permanent Dewatering System

There is no permanent dewatering system provided for in the flood design.

3.4.2 Analytical and Test Procedures

Since the design flood elevation is one foot below the finished plant grade, there is no dynamic force due to flood. The lateral hydrostatic pressure on the structures due to the design flood water level, as well as ground water and soil pressures, are calculated.

Structures, systems, and components in the ABWR Standard Nuclear Island designed and analyzed for the maximum hydrostatic and hydrodynamic forces in accordance with the loads and load combinations indicated in Subsection 3.8.4.3 and 3.8.5.3 using well established methods based on the general principles of engineering mechanics. All Seismic Category I structures are in stable condition due to either moment or uplift forces which result from the proper load combinations including the design basis flood.

3.4.3 COL License Information

3.4.3.1 Flood Elevation

The design basis flood elevation for the ABWR Standard Plant structures is one foot below grade.

3.4.3.2 Ground Water Elevation

The design basis ground water elevation for the ABWR Standard Plant structures is two feet below grade.

3.4.3.3 Flood Protection Requirements for Other Structures

The COL applicant will demonstrate, for the structures outside the scope of the ABWR Standard Plant, that they meet the requirements of GDC 2 and the guidance of RG 1.102. (See Subsection 3.4.1.1.2)

3.4.4 References

1. Crane Co., *Flow of Fluids Through Valves, Fittings, and Pipe,* Technical Paper No. 410, 1973.

2. ANSI/ANS 56.11, Standard, *Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants.*

3. Regulatory Guide 1.59, Rev. 2 *Design Basis Floods for Nuclear Power Plants.*

Table 3.4-1

STRUCTURES, PENETRATIONS, AND ACCESS OPENINGS DESIGNED FOR FLOOD PROTECTION

Structure	Reactor Building	Service Building	Control Building	Radwaste Building	Turbine Building
Design Flood Level (mm)	11,700	11,700	11,700	11,700	11,700
Reference Plant Grade (mm)	12,000	12,000	12,000	12,000	12,000
Base Slab (mm)	-8,200	-2150 & 3500	-8,200	-1,500	5,300
Actual Plant Grade (mm)	12,000	12,000	12,000	12,000	12,000
Building Height (mm	49,700	22,200	22,200	28,000	54,300
Penetrations Below Design Flood Level	Refer to Table 6.2-9	None	Refer to Table 6.2-9 for RCW lines	None	None
Access Openings Below Design Flood Level	Access way from S/B @ 4,800mm TMSL	Main Entrance @ grade level	Area Access from S/B @ -2150mm, HX. Area Access from S/B @ 12,050mm	Pipe Tunnel from R/B&T/B @1500mm Note 3	Area Access from S/B @ 7,900mm Tunnel from RWB @8,800mm

Notes:

1. Water tight doors (bulkhead type) are provided at all reactor and control building access ways that are below grade.

2. Water tight penetrations will be provided for all reactor, radwaste building and control building penetrations that are below grade.

3. The lines that run through the radwaste building tunnel are not exposed to outside ground flooding.

4. Penetrations below design flood level will be sealed against any hydrostatic head resulting from a moderate energy pipe failure in the tunnel or connecting building.

Table 3.4-2

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SECTION 3.5

CONTENTS

SECTION 3.5

CONTENTS (Continued)

TABLES

3.5.1.1 Internally Generated Missiles (Outside Containment)

These missiles are considered to be those missiles resulting internally from plant equipment failures within the ABWR Standard Plant but outside containment.

3.5.1.1.1 Rotating Equipment

3.5.1.1.1.1 Missile Characterization

Equipment within the general categories of pumps, fans, blowers, diesel generators, compressors, and turbines and, in particular, components in systems normally functioning during power reactor operation, has been examined for any possible source of credible and significant missiles.

3.5.1.1.1.2 RCIC Steam Turbine

The RCIC steam turbine driving the pump is not a credible source of missiles. It is provided with mechanical overspeed protection as well as automatic governing; very extensive industrial and nuclear experience with this model of turbine has never resulted in a missile which penetrated the turbine casing.

3.5.1.1.1.3 Main Steam Turbine

Acceptance criteria 1 of SRP Section 3.5.1.3 considers a plant with a favorable turbine generator placement and orientation and adhering to the guidelines of Regulatory Guide 1.115 adequately protected against turbine missile hazards. Further, this criterion specifies that exclusions of safety-related structures, systems or components from low trajectory turbine missile strike zones constitutes adequate protection against low trajectory turbine missiles. The turbine generator placement and orientation of the ABWR Standard Plant meets the guidelines of Regulatory Guide 1.115 as illustrated in Figure 3.5-2.

In addition, the COL applicant shall:

(1) Submit for NRC approval, within three years of obtaining an operating license, a turbine system maintenance program including probability calculations of turbine missile generation based on the NRC approved methodology (such as Reference 10), or

(2) Volumetrically inspect all low pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff.

(3) Meet the minimum requirement for the probability of turbine missile generation given in Table 3.5-1.

See Subsection 3.5.4.6 for COL license information.

3.5.1.1.1.4 Other Missile Analysis

No remaining credible missiles meet the significance criteria of having a probability (P_4) greater than 10^{-7} per year for rotating or pressurized equipment, because either:

(1) The equipment design and manufacturing criteria mentioned previously result in (P_1) being less than 10^{-7} per year; or

(2) Sufficient physical separation (barriers and/or distance) of safety-related and redundant equipment exists so that the combined probability ($P_1 \times P_2$) is less than 10^{-7} per year.

These conclusions are arrived at by noting that pumps, fans, and the like are AC powered. Their speed is governed by the frequency of the AC power supply. Since the AC power supply frequency variation is limited to a narrow range, it is not likely they will attain an overspeed condition. At rated speed, if a piece such as a fan blade breaks off, it will not penetrate the casing. The issue of missile generation in rotating machinery is a general safety problem which is not limited to nuclear applications. The designers and manufacturers of these equipment consider this factor as a requirement in their design. Industrial experience and studies conducted on system components indicate that the probability of a missile escaping the casing is very low. GE has also conducted a study on potential missile generation from electrical machines (motors, exciters, generators), flexible couplings and fluid drives. One example where missile generation is significant is in fluid drives where the rotating part and housing diameters are big and the relative thickness of the housing is small, Reference 1. Based on the results from this study it was concluded that the potential of a missile being generated and leaving the equipment housing is negligibly small.

3.5.1.1.2 Pressurized Components

3.5.1.1.2.1 Missile Characterization

Potential missiles which could result from the failure of pressurized components are analyzed in this subsection. These potential missiles may be categorized as contained fluid-energy missiles or stored strain-energy (elastic) missiles. These potential missiles have been conservatively evaluated against the design criteria in Subsection 3.5.1.

Examples of potential contained fluid-energy missiles are valve bonnets, valve stems, and retaining bolts. Valve bonnets are considered jet-propelled missiles and have been analyzed as such. Valve stems have been analyzed as piston-type missiles, while retaining bolts are examples of stored strain-energy missiles.

3.5.1.1.2.2 Missile Analyses

Pressurized components outside the containment capable of producing missiles have been reviewed. Although piping failures could result welded to the wall of the pipe. An analysis of a postulated failure of this weld has been performed. The following expression relates the missile displacement and velocity following the postulated failure:

$$\frac{y}{(W/A)} = \upsilon_\infty \left[\ell n \left(\frac{1}{1 - V/u_\infty} \right) - \frac{V}{u_\infty} \right]$$

where

y	=	distance traveled by the missile from the break (ft)
W	=	missile weight (lb)
A	=	frontal area of missile (ft^2)
u_∞	=	asymptotic velocity of jet (ft/sec)
υ_∞	=	asymptotic specific volume of jet (ft^3/lb)
V	=	velocity of missile (ft/sec)

Inherently, the water and steam velocities are equal (i.e., a unity velocity ratio) in a saturated water blowdown. The jet asymptotic velocity (u_∞) and the jet asymptotic specific volume are determined by the methods described by Reference 2. The corresponding velocity-displacement relationships for missiles resulting from saturated water and saturated steam blowdowns are presented in Figure 3.5-1. The ordinate is the missile velocity, V, and the abscissa is the displacement parameter, Y*, given by

$$Y^* = \frac{y}{(W/A)}$$

Included in Figure 3.5-1 is the influence of different values of the friction parameter, f*, defined by

$$f^* = \left(\frac{\bar{f}l}{D}\right)_P \left(\frac{A_E}{A_P}\right)^2$$

where

$\left(\frac{\bar{f}l}{D}\right)_P$	=	equivalent loss coefficient between the broken pressurized component and fluid reservoir, dimensionless;
A_E	=	area of break, ft^2; and
Ap	=	area of pressurized component between break and fluid reservoir, ft^2 (assumes $A_P \geq A_E$).

As illustrated in Figure 3.5-1, the effect of friction on the velocity-displacement relationship is reasonably small. It can be conservatively assumed that the most extreme friction condition persists with f* = 100 for the case of saturated water blowdown and f* = 0 for the case of saturated steam blowdown.

A typical thermowell weights about 2 lb. Based on ejection by steam at 1050 psig, the ejection velocity could reach 200 ft/ sec which is not sufficient to inflict significant damage to critical systems. (P_4) is therefore less than 10^{-7} per year.

(5) **Retaining Bolts** - Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are of no concern as potential missiles.

(6) **Blowout Panels** - Blowout panels are hinged to prevent them from becoming missiles. Guard rails for personnel protection have been provided where required by the swing pattern. Thus by design, (P_2) is less than 10^{-7} per year.

3.5.1.1.3 Missile Barriers and Loadings

Certain cases of rotating and pressurized components generating missiles described in Subsection 3.5.1.1.2 give credit for potential missle-consequence mitigation by structural walls and slabs. These walls and slabs are designed to withstand internal missile effects; the applicable seismic category and quality group classification are listed in Section 3.2. Penetration of the structural walls by internally generated missiles is not considered credible.

For local shields and barriers see the response to Question 410.9.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Internal missiles are those resulting from plant equipment failures within the containment. Potential missile sources from both rotating equipment and pressurized components are considered.

3.5.1.2.1 Rotating Equipment

By an analysis similar to that in Subsection 3.5.1.1.1, it is concluded that no items of rotating equipment inside the containment have the capability of becoming potential missiles. All reactor internal pumps are incapable of achieving an overspeed condition and the motors and impellers are incapable of escaping the casing and the reactor vessel wall, respectively.

3.5.1.2.2 Pressurized Components

Identification of potential missiles and their consequences outside containment are specified in Subsection 3.5.1.1.2. The same conclusions are drawn for pressurized components inside of contaiment. For example, the ADS accumulators are moderate energy vessels and are therefore not considered a credible missile source. One additional item is fine motion control rod drives (FMCRD) under the reactor vessel. The FMCRD mechanisms are not credible missiles. The FMCRD housings are designed (Section 4.6) to prevent any significant nuclear transient in the event of a drive housing break.

3.5.1.2.3 Missile Barriers and Loadings

Credit is taken in some cases of rotating and pressurized components generating missiles for missile-consequence mitigation by structural walls and slabs. Penetration for the containment walls, floors and slabs by potential missiles is not considered credible. However, credible secondary missiles, e.g., concrete fragments, may be formed following impact of primary missiles. See Subsection 3.5.4.4 for COL license information requirements.

3.5.1.2.4 Evaluation of Potential Gravitational Missiles Inside Containment

Gravitational missiles inside the containment have been considered as follows:

Seismic Category I systems, components, and structures are not potential gravitational missile sources.

Non-Seismic Category I items and systems inside containment are considered as Follows:

(1) Cable Tray

All cable trays for both Class 1E and non-Class 1E circuits are seismically supported whether or not a hazard potential is evident.

(2) Conduit and Non-Safety Pipe

Non-Class 1E conduit is seismically sup- ported if it is identified as a potential hazard to safety-related equipment. All Nuclear Island non-safety related piping that is identified as a potential hazard is seismically analyzed per Subsection 3.7.3.13.

(3) Equipment for Maintenance

All other equipment, such as hoists, that is required during maintenance will either be removed prior to operation, moved to a location where it is not a potential hazard to safety-related equipment, or seismically restrained to prevent it from becoming a missile. See Subsection 3.5.4.7 for COL license information.

3.5.1.3 Turbine Missiles

See Subsection 3.5.1.1.1.3.

generated from other natural phenomena. The design basis tornado for the ABWR Standard Plant is the maximum tornado windspeed corresponding to a probability of 10E-7 per year (300 mph). The other characteristics of this tornado, summerized in Subsection 3.3.2.1. The design basis tornado missiles are per SRP 3.5.1.4, Spectrum I.

Using the design basis tornado and missile spectrum as defined above with the design of the Seismic Category I buildings, compliance with all of the positions of Regulatory Guide 1.117, "Tornado Design Classification," Positions C.1 and C.2 is assured.

The SGTS charcoal absorber beds are housed in the tornado resistant reactor building and therefore are protected from the design basis tornado missiles. The offgas system charcoal absorber beds are located deep within the turbine building and it is considered very unlikely that these beds could be ruptured as a result of a design basis tornado missile. These features assure compliance with Position C.3 of Regulatory Guide 1.117.

See Subsections 3.5.4.2 and 3.5.4.5 for COL license information requirements.

3.5.1.5 Site Proximity Missiles Except Aircraft

External missiles other than those generated by tornados are not considered as a design basis (i.e. $\leq 10^{-7}$ per year).

3.5.1.6 Aircraft Hazards

Aircraft hazards are not a design basis event for the Nuclear Island (i.e. $\leq 10^{-7}$ per year). See Subsection 3.5.4.3 for COL license information requirements.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

The sources of external missiles which could affect the safety of the plant are identified in Subsection 3.5.1. Certain items in the plant are required to safely shut down the reactor and maintain it in a safe condition assuming an additional single failure. These items, whether they be structures, systems, or components, must therefore all be protected from externally generated missiles.

These items are the safety-related items listed in Table 3.2-1. Appropriate safety classes and equipment locations are given in this table. All of the safety-related systems listed are located in buildings which are designed as tornado resistant. Since the tornado missiles are the design basis missiles, the systems, structures, and components listed are considered to be adequately protected. Provisions are made to protect the charcoal delay tanks against tornado missiles.

See Subsections 3.5.4.1 and 3.5.4.8 for COL license information requirements.

3.5.3 Barrier Design Procedures

The procedures by which structures and barriers are designed to resist the missiles described in Subsection 3.5.1 are presented in this section. The following procedures are in accordance with Section 3.5.3 of NUREG-0800 (Standard Review Plan).

3.5.3.1 Local Damage Prediction

The prediction of local damage in the impact area depends on the basic material of construction of the structure or barrier (i.e., concrete or steel). The corresponding procedures are presented separately. Composite barriers are not utilized in the ABWR Standard Plant for missile protection.

3.5.3.1.1 Concrete Structures and Barriers

The modified Petry formula (Reference 3) is applied analytically for missile penetration in concrete. To prevent perforation, a minimum concrete thickness of 2.2 times the penetration thickness determined for an infinitely thick concrete slab is employed. In the event that spalling or scabbing is unacceptable, a minimum concrete thickness of 3 times the penetration thickness determined for an infinitely thick concrete slab is provided. These design procedures have been substantiated by full-scale impact tets in which reinforced concrete panels (12 to 24 inches thick, 3000-psi design strength) were impacted by poles, pipes, and rods simulating tornado-borne debris (Reference 4).

3.5.3.1.2 Steel Structure and Barriers

The Stanford equation (Reference 5) is applied for steel structures and barriers.

3.5.3.2 Overall Damage Prediction

The overall response of a structure or barrier to missile impact depends largely upon the location of impact (e.g., near mid-span or near a support), dynamic properties of the steucture/barrier and missile, and on the kinetic energy of the missile. In general, it has been assumed that the impact is plastic with all of the initial momentum of the missile transferred to the structure or barrier and only a portion of the kenetic energy absorbed as strain energy within the structure or barrier.

After demonstrating that the missile does not perforate the structure or barrier, an equivalent static load concentrated at the impact area is determined. The structural response to this load, in conjunction with other appropriate design loads, is evaluated using an analysis procedure similar to that in Reference 6 for rigid missiles, and the procedure in Reference 7 for deformable missiles.

3.5.4 COL License Information

3.5.4.1 Protection of Ultimate Heat Sink

Compliance with Regulatory Guide 1.27 as related to the ultimate heat sink and connecting conduits being capable of withstanding the effects of externally generated missiles shall be demonstrated (See Subsection 3.5.2).

3.5.4.2 Missiles Generated by Other Natural Phenomena

The COL applicant shall identify missiles generated by other site-specific natural phenomena that may be more limiting than those considered in the ABWR design and shall provide protection for the structures, systems and components against such missiles. The COL applicant will provide this information to the NRC. (See Subsection 3.5.1.4)

3.5.4.3 Site Proximity Missiles and Aircraft Hazards.

Analyses shall be provided that demonstrate that the probability of site proximity missiles (including aircraft) impacting the ABWR Standard Plant and causing consequences greater than 10CFR Part 100 exposure guidelines is $\leq 10^{-7}$ per year (See Subsection 3.5.1.6).

3.5.4.4 Secondary Missiles Inside Containment

Protection against the secondary missiles inside containment described in Subsection 3.5.1.2.3 shall be demonstrated.

3.5.4.5 Impact of Failure of Out of ABWR Standard Plant Scope Non Safety-Related Structures, Systems, and Components Due to a Design Basis Tornado

An evaluation of all out of ABWR Standard Plant scope non safety-related structures, systems, and components (not housed in a tornado structure) whose failure due to a design basis tornado missile that could adversely impact the safety function of a safety-related systems and components will be provided to the NRC by the COL applicant. (See Subsection 3.5.1.4).

3.5.4.6 Turbine System Maintenance Program

A turbine system maintenance program including probability calculations of turbine missile generation meeting the minimum requirement for the probability of missile generation shall be provided to the NRC (See Subsection 3.5.1.1.3).

3.5.4.7 Maintenance Equipment Missile Prevention Inside Containment

The COL applicant will provide procedures to ensure that all equipment inside containment, such as hoists, that is required during maintenance will either be removed prior to operation, moved to a location where it is not a potential hazard to safety-related equipment, or seismically restrained to prevent it from becoming a missile. (See Subsection 3.5.1.2.4(3)).

3.5.4.8 Failure of Structures, Systems and Components Outside ABWR Standard Plant Scope

Any failure of structures, systems and components outside ABWR Standard Plant scope which may result in external missile generation shall not prevent safety-related structures, systems and components from performing their intended safety function. The COL applicant will provide an evaluation of the adequacy of these designs for external missile protection for NRC review. (See Subsection 3.5.2)

3.5.5 References

1. K. Karim-Panahi et al, *Recirculation MG Set Missile Generation Study*, PED-18-0389, March 1989 (Proprietary).

2. F. J. Moody, *Prediction of Blowdown Thrust and Jet Forces,* ASME Publication 69-HT-31, August 1969.

3. A. Amirikan, *Design of Protective Structures,* Bureau of Yards and Docks, Publica

ion No. NAVDOCKS P-51, Department of the Navy, Washington, D.C., August 1960.

4. A. E. Stephenson, Full-Scale Tornado-Missile Impact Tests, EPRI NP-440, July 1977, prepared for Electric Power Research Institute by Sandia Laboratories.

5. W. B. Cottrell and A. W. Savolainen, *U. S. Reactor Containment Technology*, ORNL-NSIC-5, Vol. 1, chapter 6, Oak Ridge National Laboratory.

6. R. A. Williamson and R. R. Alvy, *Impact Effect of Fragments Striking Structural Elements,* Holmes and Narver, Inc., Revised November 1973.

7. J. D. Riera, *On the Stress Analysis of Structures Subjected to Aircraft Impact Forces,* Nuclear Engineering and Design, North Holland Publishing Co., Vol. 8, 1968.

8. Deleted

9. *River Bend Station Updated Safety Analysis Report,* Docket No. 50-458, Volume 6, pgs. 3.5-4 and 3.5-5, August 1987.

10 NUREG-1048, Safety Evaluation Report Related to the Operation of Hope Creek Generating Station, Supplement No. 6, July 1986.

Strain rate effects and other material property variations have been considered in the design of the pipe whip restraints. The material properties utilized in the design have included one or more of the following methods:

(1) Code minimum or specification yield and ultimate strength values for the affected components and structures are used for both the dynamic and steady-state events;

(2) Not more than a 10% increase in minimum code or specification strength values is used when designing components or structures for the dynamic event, and code minimum or specification yield and ultimate strength values are used for the steady-state loads:

(3) Representative or actual test data values are used in the design of components and structures including justifiably elevated strain rate-affected stress limits in excess of 10%; or

(4) Representative or actual test data are used for any affected component(s) and the minimum code or specification values are used for the structures for the dynamic and the steady-state events.

3.6.2.4 Guard Pipe Assembly Design

The ABWR primary containment does not require guard pipes.

3.6.2.5 Material to be Supplied for the Operating License Review

See Subsection 3.6.4.1 for COL license information requirements

3.6.3 Leak-Before-Break Evaluation Procedures

Per Regulatory Guide 1.70, the safety analysis Section 3.6 has traditionally addressed the protection measures against dynamic effects associated with the non-mechanistic or postulated ruptures of piping. The dynamic effects are defined in introduction to Section 3.6. Three forms of piping failure (full flow area circumferential and longitudinal breaks, and throughwall leakage crack) are postulated in accordance with Subsection 3.6.2 and Branch Technical Position MEB 3-1 of NUREG - 0800 (Standard Review Plan) for their dynamic as well as environmental effects.

However, in accordance with the modified General Design Criterion 4 (GDC-4), effective November 27, 1987, (Reference 1), the mechanistic leak-before-break (LBB) approach, justified by appropriate fracture mechanics techniques, is recognized as an acceptable procedure under certain conditions to exclude design against the dynamic effects from postulation of breaks in high energy piping. The LBB approach is not used to exclude postulation of cracks and associated effects as required by Subsections 3.6.2.1.5 and 3.6.2.1.6.2. It is anticipated, as mentioned in Subsection 3.6.4.2, that a COL applicant will apply to the NRC for approval of LBB qualification of selected piping. These approved piping, referred to in this SSAR as the LBB-qualified piping, will be excluded from pipe breaks, which are required to be postulated by Subsections 3.6.1 and 3.6.2, for design against their potential dynamic effects.

The following subsections describe (1) certain design bases where the LBB approach is not recognized by the NRC as applicable for exclusion of pipe breaks, and (2) certain conditions which limit the LBB applicability. Appendix 3E provides guidelines for LBB applications describing in detail the following necessary elements of an LBB report to be submitted by a COL applicant for NRC approval: fracture mechanics methods, leak rate prediction methods, leak detection capabilities and typical special considerations for LBB applicability. Also included in Appendix 3E is a list of candidate piping systems for LBB qualification. The LBB application approach described in this subsection and Appendix 3E is consistent with that documented in Draft SRP 3.6.3 (Reference 4) and NUREG-1061 (Reference 5). (See Subsection 3.6.4.2 for COL license information requirements.)

3.6.3.1 Scope of LBB Applicability

The LBB approach is not used to replace existing regulations or criteria pertaining to the design bases of emergency core cooling system (Section 6.3), containment system (Section 6.2) or environmental qualification (Section 3.11). However, consistent with modified GDC-4, the design bases dynamic qualification of mechanical and electrical equipment (Section 3.10) may exclude the dynamic load or vibration effects resulting from postulation of breaks in the LBB-qualified piping. This is also reflected in a note to Table 3.9-2 for ASME components. The LBB-qualified piping may not be excluded from the design bases for environmental qualification unless the regulation permits it at the time of LBB qualification. For clarification, it is noted that the LBB approach is not used to relax the design requirements of the primary containment system that includes the primary containment vessel (PCV), vent systems (vertical flow channels and horizontal vent discharges), drywell zones, suppression chamber (wetwell), vacuum breakers, PCV penetrations, and drywell head.

3.6.3.2 Conditions for LBB Applicability

The LBB approach is not applicable to piping systems where operating experience has indicated particular susceptibility to failure from the effects of intergrannular stress corrosion cracking (IGSCC), water hammer, thermal fatigue, or erosion. Necessary preventive or mitigation measures are used and necessary analyses are performed, as discussed below, to avoid concerns for these effects. Other concerns, such as creep, brittle cleavage type failure, pote ..al indirect source of pipe failure, and deviation of as-built piping configuration, are also addressed.

(1) Degradation by erosion, erosion/corrosion and erosion/cavitation due to unfavorable flow conditions and water chemistry is examined. The evaluation is based on the industry experience and guidelines. Additionally, fabrication wall thinning of elbows and other fittings is considered in the purchase specification to assure that the code minimum wall requirements are met. These evaluations demonstrate that these mechanisms are not potential sources of pipe rupture

(2) The ABWR plant design involves operation below 700°F in ferritic steel piping and below 800°F in austenitic steel piping. This assures that creep and creep-fatigue are not potential sources of pipe rupture.

(3) The design also assures that the piping material is not susceptible to brittle cleavage-type failure over the full range of system operating temperatures (that is, the material is on the upper shelf).

(4) The ABWR plant design specifies use of austenitic stainless steel piping made of material (e.g., nuclear grade or low carbon type) that is recognized as resistant to IGSCC. The material major high energy piping in the primary and secondary containments is carbon steel or ferritic steel, except for the austenitic stainless reactor water cleanup piping in the primary containment.

(5) A systems evaluation of potential water hammer is made to assure that pipe rupture due to this mechanism is unlikely. Water hammer is a generic term including various unanticipated high frequency hydrodynamic events such as steam hammer and water slugging. To demonstrate that water hammer is not a significant contributor to pipe rupture, reliance on historical frequency of water hammer events in specific piping systems coupled with a review of operating procedures and conditions is used for this evaluation. The ABWR design includes features such as vacuum breakers and jockey pumps coupled with improved operational procedures to reduce or eliminate the potential or water hammer identified by past

experience. Certain anticipated water hammer events, such as a closure of a valve, are accounted for in the Code design and analysis of the piping.

(6) The systems evaluation also addresses a potential for fatigue cracking or failure from thermal and mechanical induced fatigue. Based on past experience, the piping design avoids potential for significant mixing of high- and low- temperature fluids or mechanical vibration. The startup and preoperational monitoring assures avoidance of detrimental mechanical vibration.

(7) Based on experience and studies by Lawrence Livermore Laboratory, potential indirect sources of indirect pipe rupture are remote causes of pipe rupture. Compliance with the snubber surveillance requirements of the technical specifications assures that snubber failure rates are acceptably low.

(8) Initial LBB evaluation is based on the design configuration and stress levels that are acceptably higher than those identified by the initial analysis. This evaluation is reconciled when the as-built configuration is documented and the Code stress evaluation is reconciled. It is assured that the as-built configuration does not deviate significantly from the design configuration to invalidate the initial LBB evaluation, or a new evaluation coupled with necessary configuration modifications is made to assure applicability of the LBB procedure.

3.6.4 COL License Information

3.6.4.1 Details of Pipe Break Analysis Results and Protection Methods

The following shall be provided by the COL applicant (See Subsection 3.6.2.5):

(1) A summary of the dynamic analyses applicable to high-energy piping systems in accordance with Subsection 3.6.2.5 of Regulatory Guide 1.70. This shall include:

(a) Sketches of applicable piping systems showing the location, size and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers.

(b) A summary of the data developed to select postulated break locations including calculated stress intensities, cumulative usage factors and stress ranges as delineated in BTP MEB 3-1.

(2) For failure in the moderate-energy piping systems listed in Table 3.6-6, descriptions showing how safety-related systems are protected from the resulting jets, flooding and other adverse environmental effects.

(3) Identification of protective measures provided against the effects of postulated pipe failures for protection of each of the systems listed in Tables 3.6-1 and 3.6-2.

(4) The details of how the MSIV functional capability is protected against the effects of postulated pipe failures.

(5) Typical examples, if any, where protection for safety-related systems and components against the dynamic effects of pipe failures include their enclosure in suitably designed structures or compartments (including any additional drainage system or equipment environmental qualification needs).

(6) The details of how the feedwater line check and feedwater isolation valves functional capabilities are protected against the effects of postulated pipe failures.

3.6.4.2 Leak-Before-Break Analysis Report

As required by Reference 1, an 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 accordance with the guidelines presented in Appendix 3E and Submitted by the COL applicant to the NRC for approval. (See Subsection 3.6.3).

3.6.5 References

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.

The frequency range used in generating the response spectra from synthetic histories is 0.2 to 33 Hz. The frequency range intervals used in generating those spectra is the same as given in Table 3.7.1-1 of SRP Section 3.7.1.

The coherence function for the three earthquake acceleration time history components H1, H2, and V are generated to check the statistical independence among them. The coherence function for H1 and H2 is given in Figure 3.7-21; for H1 and V in Figure 3.7-22; and for H2 and V in Figure 3.7-23. All values within the frequency range between 0 to 50 Hz are calculated at a frequency increment of 0.1 Hz. The small values of these coherence functions indicate that the three components are sufficiently statistically independent.

To assess the energy content of the synthetic time history, the power spectral density functions (PSDFs) are generated from the two horizontal components H1 and H2. The PSDFs are computed at a frequency increment of 0.024 Hz, and are smoothed using the average method as recommended in Revision 2 of Reference 3.

The stationary duration used in the calculation is taken to be 22 seconds which is the total duration of the synthetic time history. The calculated PSDFs for the H1 and H2 time histories normalized to 0.15g peak ground acceleration are shown in Figures 3.7-24 and 3.7-25, respectively, for frequencies ranging from 0.3 to 24 Hz.

The target PSDFs and 80% of target PSDFs specified on revision 2 of Reference 3 are also plotted on these figures for comparison. As shown, PSDF of H1 and H2 time histories envelope the target PSDF with a wide margin in the specified frequency range of 0.3 to 24 Hz. This demonstrates that the two synthetic time histories have sufficient energy content.

3.7.1.3 Critical Damping Values

The damping values for OBE and SSE analyses are presented in Table 3.7-1 for various structures and components. They are in compliance with Regulatory Guides 1.61 and 1.84

For seismic system evaluation of the SSE, the larger SSE damping values shown in Table 3.7-1 are not used. The SSE loads are obtained by doubling the OBE loads that result from the OBE Seismic System analysis based on the lower OBE damping values (see Subsection 3.7.1.2).

For analysis and evaluation of seismic subsystems (piping, components and equipment), the floor response spectra are obtained from the OBE time-history response of the seismic system, that supports the subsystems. The floor response spectra are computed (see Subsection 3.7.2.5) for damping values that are applicable to the subsystems under OBE as well as SSE; and further the OBE spectra are doubled to obtain the SSE floor response spectra for input to the SSE analysis in design of the subsystems.

3.7.1.4 Supporting Media for Seismic Category I Structures

The following ABWR Standard Plant Seismic Category I structures have concrete mat foundations supported on soil, rock or compacted backfill. The maximum value of the embedment depth below plant grade to the bottom of the base mat is given below for each structure.

(1) Reactor Building (including the enclosed primary containment vessel and reactor pedestal) - 25.7 m (84 ft, 4 in.).

(2) Control Building - 12.2 m (40 ft).

(3) Service Building - Surface founded.

All of the above buildings have independent foundations. In all cases the maximum value of embedment is used for the dynamic analysis to determine seismic soil-structure interaction effects. The foundation support materials withstand the pressures imposed by appropriate loading combinations without failure. The total structural height of each building is described in Subsection 3.8.2 through 3.8.4. For details of the structural foundations refer to Subsection 3.8.5. The ABWR Standard Plant is designed for a range of soil conditions given in Appendix 3A.

3.7.1.4.1 Soil-Structure Interaction

When a structure is supported on a flexible foundation, the soil-structure interaction is taken into account by coupling the structural model with the soil medium. The finite-element representation is used for a broad range of supporting medium conditions. A different representation based on the continuum impedance approach is also used for selected site conditions. Detailed methodology and results of the soil-structure interaction analysis are provided in Appendices 3A and 3G, respectively.

3.7.2 Seismic System Analysis

This subsection applies to the design of Seismic Category I structures and the reactor pressure vessel (RPV). Subsection 3.7.3 applies to all Seismic Category I piping systems and equipment.

3.7.2.1 Seismic Analysis Methods

Analysis of Seismic Category I structures and the RPV is accomplished using the response spectrum or time-history approach. The time-history approach is made either in the time domain or in the frequency domain.

Either approach utilizes the natural period, mode shapes, and appropriate damping factors of the particular system toward the solution of the equations of dynamic equilibrium. The time-history approach may alternately utilize the direct integration method of solution. When the structural response is computed directly from the coupled structure-soil system, the time-history approach solved in the frequency domain is used. The frequency domain analysis method is described in Appendix 3A.

3.7.2.1.1 The Equations of Dynamic Equilibrium for Base Support Excitation

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped-mass, distributed-stiffness system are expressed in a matrix form as:

Not used (3.7-1)

$$[M]\{\ddot{u}(t)\} + [c]\{\dot{u}(t)\} + [K]\{u(t)\} = \{P(t)\} \qquad (3.7\text{-}2)$$

where

- $\{u(t)\}$ = time-dependent displacement vector of non-support points relative to the supports $(u_t(t) = u(t) + u_s(t))$
- $\{u(t)\}$ = time-dependent velocity vector of non-support points relative to the supports
- $\{u(t)\}$ = time-dependent acceleration vector of non-support points relative to the supports
- $[M]$ = mass matrix
- $[C]$ = damping matrix
- $[K]$ = stiffness matrix
- $\{P(t)\}$ = time-dependent inertia force vector $(-[M]\{u_s(t)\}$ acting at non-support points

The manner in which a distributed-mass, distributed-stiffness system is idealized into a lumped-mass, distributed-stiffness system of Seismic Category I structures and the RPV is

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acceleration considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted accelerations have been determined using the same conservatism contained in the horizontal and vertical accelerations used for rigid valves. The adjusted acceleration is then used in the static analysis and the valve operability is assured by the methods outlined in Steps (2) through (4), using the modified acceleration input. Alternatively, the valve including the actuator and all other accessories is qualified by shake table test.

Valves which are safety-related but can be classified as not having an overhanging structure, such as check valves and pressure-relief valves, are considered as follows:

3.9.3.2.5.1.2.1 Active Check Valves

Due to the particular simple characteristics of the check valves, the active check valves are qualified by a combination of the following tests and analysis:

(1) Stress analysis including the dynamic loads where applicable;

(2) in-shop hydrostatic tests;

(3) in-shop seat leakage test; and

(4) periodic in-situ valve exercising and inspection to assure the functional capability of the valve.

3.9.3.2.5.1.2.2 Active Pressure-Relief Valves

The active pressure-relief valves (RVs) are qualified by the following procedures. These valves are subjected to test and analysis similar to check valves, stress analyses including the dynamic loads, in-shop hydrostatic seat leakage, and performance tests. In addition to these tests, periodic in-situ valve inspection, as applicable, and periodic valve removal, refurbishment, performance testing, and reinstallation are performed to assure the functional capability of the valve. Tests of the RV under dynamic loading conditions demonstrate that valve actuation can occur during application of the loads. The tests include pressurizing the valve inlet with nitrogen and subjecting the valve to accelerations equal to or greater than the dynamic event (SSE plus other RBV) loads.

3.9.3.2.5.1.3 Qualification of Electrical and Instrumentation Components Controlling Valve Actuation

A practical problem arises in attempting to describe tests for devices (relays, motors, sensors, etc.) as well as for complex assemblies such as control panels. It is reasonable to assume that a device, as an integral part of an assembly, can be subjected to dynamic loads tests while in an operating condition and its performance monitored during the test. However, in the case of complex panels, such a test is not always practical. In such a situation, the following alternate approach is recommended.

The individual devices are tested separately in an operating condition and the test levels recorded as the qualification levels of the devices. The panel, with similar devices installed but inoperative, is vibration tested to determine if the panel response accelerations as measured by accelerometers installed at the device attachment locations are less than the levels at which the devices were qualified. Note that the purpose of installing the nonoperating devices is to assure that the panel has the structural characteristics it will have when in use. If the acceleration levels at the device locations are found to be less than the levels to which the device is qualified, then the total assembly is considered qualified. Otherwise, either the panel is redesigned to reduce the acceleration level to the device locations and retested, or the devices is requalified to the higher levels.

3.9.3.2.5.2 Documentation

All of the preceding requirements (Subsection 3.9.3.2.5.1) are satisfied to demonstrate that functionality is assured for active valves. The documentation is prepared in a format that clearly shows that each consideration has been properly evaluated and tests have been validated by a designated quality assurance representative. The analysis is included as a part of the certified stress report for the assembly.

3.9.3.3 Design and Installation of Pressure Relief Devices

3.9.3.3.1 Main Steam Safety/Relief Valves

SRV lift in a main steam (MS) piping system results in a transient that produces momentary unbalanced forces acting on the MS and SRV discharge piping system for the period from opening of the SRV until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the main steam and discharge piping following the relatively rapid opening of the SRV cause this piping to vibrate.

The analysis of the MS and discharge piping transient due to SRV discharge consists of a stepwise time-history solution of the fluid flow equation to generate a time history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum set pressure specified in the steam system specification and the value of ASME Code flow rating increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves in a MS line is assumed in the analysis because simultaneous discharge is considered to induce maximum stress in the piping. Reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum-change, and fluid-friction terms.

The method of analysis applied to determine response of the MS piping system including the SRV discharge line, to relief valve operation is time-history integration. The forces are applied at locations on the piping system where fluid flow changes direction thus causing momentary reactions. The resulting loads on the SRV, the main steamline, and the discharge piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. In accordance with Tables 3.9-1 and 3.9-2, the Code stress limits for service levels corresponding to load combination classification as normal, upset, emergency, and faulted are applied to the main steam and discharge pipe.

3.9.3.3.2 Other Safety/Relief Valves

An SRV is identified as a pressure relief valve or vacuum breaker. SRVs in the reactor components and subsystems are described and identified in Subsection 5.4.13.

The operability assurance program discussed in Subsection 3.9.3.2.5 applies to safety/relief valves. The qualification of active relief valves is specifically outlined in Subsection 3.9.3.2.5.1.2.2.

ABWR safety/relief valves (safety valves with auxiliary actuating devices and pilot operated valves) are designed and manufactured in accordance with the ASME Code, Section III, Division 1 requirements. Specific rules for pressure relieving devices are as specified in Article NB-7000, and NB-3500 (pilot operated and power actuated pressure relief valves).

The design of ABWR SRVs incorporates SRV opening and pipe reaction load considerations required by ASME III, Appendix O, and including the additional criteria of SRP, Section 3.9.3, Paragraph II.2 and those identified under Subsection NB-3658 for pressure and structural integrity. Safety/relief valve operability is demonstrated either by dynamic testing or analysis of similarly tested valves or a combination of both in compliance with the requirements of SRP Subsection 3.9.3.

3.9.3.3.3 Rupture Disks

There are no rupture disks in the ABWR plant design, that must function during and after a dynamic event (SSE including other RBV loads).

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)

3.9.8 References

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

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

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

November 1977. Also NEDO-24057-P, Amendment 1, December 1978, and NEDE-2-P 24057 Amendment 2, June 1979.

4. *General Electric Company, Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K,* NEDE-20566P, Proprietary Document, November 1975.

5. *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,* NUREG-0619.

6. *General Electric Environmental Qualification Program,* NEDE-24326-1-P, Proprietary Document, January 1983.

7. Deleted

8. *Generic Criteria for High Frequency Cutoff of BWR Equipment,* NEDO-25250, Proprietary Document, January 1980.

Table 3.9-7

FATIGUE LIMIT
FOR SAFETY CLASS REACTOR INTERNAL STRUCTURES ONLY

Summation of fatigue damage usage following Miner hypotheses(1):

Cumulative Damage in Fatigue	Limit for Service Levels A&B (Normal and Upset Conditions)
Design fatigue cycle usage from analysis using the method of the ASME Code	≤ 1.0

NOTE

(1) Miner, M.A., *Cumulative Damage in Fatigue,* Journal of Applied Mechanics, Vol. 12, ASME, Vol. 67, pp A159-A164, September 1945.

Table 3.9-8

IN-SERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

This table responds to NRC Questions 210.47, 210.48 and 210.49 regrading provisions for inservice testing of safety-related pumps and valves within the scope of the ABWR Standard Plant in accordance with the ASME Code. The information is presented separately for each system for the MPL numbers given below.*

3.11 ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED MECHANICAL AND ELECTRICAL EQUIPMENT

This section defines the environmental conditions with respect to limiting design conditions for all the safety-related mechanical and electrical equipment, and documents the qualification methods and procedures employed to demonstrate the capability of this equipment to perform safety-related functions when exposed to the environmental conditions in their respective locations. The safety-related equipment within the scope of this section are defined in Subsection 3.11.1. Dynamic qualification is addressed in Sections 3.9 and 3.10 for Seismic Category I mechanical and electrical equipment, respectively.

Limiting design conditions include the following:

(1) Normal Operating Conditions - planned, purposeful, unrestricted reactor operating modes including startup, power range, hot standby (condenser available), shutdown, and refueling modes;

(2) Abnormal Operating Conditions - any deviation from normal conditions anticipated to occur often enough that the design should include a capability to withstand the conditions without operational impairment;

(3) Test Conditions - planned testing including pre-operational tests;

(4) Accident Conditions - a single event not reasonably expected during the course of plant operation that has been hypothesized for analysis purposes or postulated from unlikely but possible situations or that has the potential to cause a release of radioactive material (a reactor coolant pressure boundary rupture may qualify as an accident; a fuel cladding defect does not); and

(5) Post-Accident Conditions - during the length of time the equipment must perform its safety-related function and must remain in a safe mode after the safety-related function is performed.

3.11.1 Equipment Identification and Environmental Conditions

Safety related electrical equipment within the scope of this section includes all three categories of 10CFR50.49(b) (Reference 1). Safety-related mechanical equipment (e.g., pumps, motor-operated valves, safety-relief valves, and check valves) are as defined and identified in Section 3.2. Electrical and mechanical equipment safety classifications are further defined on the system design drawings.

Safety related equipment located in a harsh environment must perform its proper safety function during normal, abnormal, test, design basis accident and post accident environments as applicable. A list of all safety-related electrical and mechanical equipment that is located in a harsh environment area will be included in the Environmental Qualification Document (EQD) to be prepared as mentioned in Subsection 3.11.6.1. The COL applicant referencing the ABWR design will provide a list of impacted non-safety-related control systems and the design features for preventing the potential adverse consequences identified in IE Information Notice 79-22, Qualification of Control Systems. The COL applicant will also address issues related to equipment wetting and flooding above the flood level identified in IE Information Notice 89-63, Possible Submergence of Electrical Circuits Located Above the Flood Level Because of Water Intrusion and Lack of Drainage, as required in Subsection 3.11.6.

Environmental conditions for the zones where safety-related equipment is located are calculated for normal, abnormal, test, accident and post-accident conditions and are documented in Appendix 3I, Equipment Qualification Environmental Design Criteria (EQEDC). Environmental conditions are tabulated by zones, contained in the referenced building arrangements. Typical equipment in the noted zones is shown in the referenced system P&ID and IED design drawings.

Occurences of anticipated abnormal operating conditions are similar to test conditions and their significant environments are comparable.

Environmental parameters include temperature, pressure, relative humidity, and neutron dose rate and integrated dose. Radiation dose for gamma and beta data for both normal and accident conditions will be provided by the COL applicant in accordance with the requirements in Subsection 12.2.3.1. The radiation requirements are site specific documentation owing to the need to model specific equipment which is applicant determined. The HVAC detailed modeling and the evolving considerations in the area of accident source terms are expected to generate significantly differing radiation requirments Where applicable, these parameters are given in terms of a time-based profile.

The magnitude and 60-year frequency of occurrence of significant deviations from normal plant environments in the zones have insignificant effects on equipment total thermal normal aging or accident aging. Abnormal conditions are overshadowed by the normal or accident conditions in the Appendix 3I tables.

Margin is defined as the difference between the most severe specified service conditions of the plant and the conditions used for qualification. Margins shall be included in the qualification parameters to account for normal variations in commercial production of equipment and reasonable errors in defining satisfactory performance. The environmental conditions shown in the Appendix 3I tables do not include margins.

Some mechanical and electrical equipment may be required by the design to perform an intended safety function between minutes of the occurrence of the event but less than 10 hours into the event. Such equipment shall be shown to remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident analysis unless a time margin of less than 1 hour can be justified. Such justification will include for each piece of equipment: (1) consideration of a spectrum of breaks; (2) the potential need for the equipment later in the event or during recovery operations; (3) detemination that failure of the equipment after performance of its safety function will not be detrimental to plant safety or mislead the operator; and (5) determination that the margin applied to the minimum operability time, when combined with other test margins, will account for the uncertainties accociated with the use of analytical techniques in the derivation of environmental parameters, the number of units tested, production tolerances, and test equipment inaccuracies

The environmental conditions shown in the Appendix 3I tables are upper-bound envelopes used to establish the environmental design and quali- fication bases of safety-related equipment. The upper bound envelopes indicate that the zone data reflects the worse case expected environment produced by a compendium of accident conditions. Estimated chemical environmental conditions are also reported is Appendix 3I.

3.11.2 Qualification Tests and Analyses

Safety-related electrical equipment that is located in a harsh environment is qualified by test or other methods as described in IEEE 323 and permitted by 10CFR50.49(f) (Reference 1). Equipment type test is the preferred method of qualification.

Safety-related mechanical equipment that is located in a harsh environment is qualified by analysis of materials data which are generally based on test and operating experience.

The qualification methodology is described in detail in the NRC approved licensing Topical Report on GE's environmental qualification program (Reference 2). This report also addresses compliance with the applicable portions of the General Design Criteria of 10CFR50, Appendix A, and the Quality Assurance Criteria of 10CFR50, Appendix B. Additionally, the report describes conformance to NUREG-0588 (Reference 3), and Regulatory Guides and IEEE Standards referenced in Section 3.11 of NUREG-0800 (Standard Review Plan).

Mild environment is that which, during or after a design basis event (DBE, as defined in Reference 2), would at no time be significantly more severe than that which exists during normal, test and abnormal events.

The COL applicant will require vendors of equipment located in a mild environment to submit a certificate of compliance certifying that the equipment has been qualified to assure its required safety- related function in its applicable environment. This equipment is qualified for dynamic loads as addressed in Sections 3.9 and 3.10. Further, a surveillance and maintenance program will be developed to ensure equipment operability during its designed life. (See Subsection 3.11.6).

3.11.3 Qualification Test Results

The results of qualification tests for safety-related equipment will be documented, maintained, and reported as mentioned in Subsection 3.11.6.

3.11.4 Loss of Heating, Ventilating, and Air Conditioning

To ensure that loss of heating, ventilating, and air conditioning (HVAC) system does not adversely affect the operability of safety-related controls and electrical equipment in buildings and areas served by safety-related HVAC systems, the HVAC systems serving these areas meet the single-failure criterion. Section 9.4 describes the safety-related HVAC systems including the detailed safety evaluations. The loss of ventilation calculations are based on maximum heat loads and consider operation of all operable equipment regardless of safety classification.

3.11.5 Estimated Chemical and Radiation Environment

3.11.5.1 Chemical Environment

Equipment located in the containment drywell and wetwell is potentially subject to water spray modes of the RHR system. In addition, equipment in the lower portions of the containment is potentially subject to submergence. The chemical composition and resulting pH to which safety-related equipment is exposed during normal operation and design basis accident conditions is reported in Appendix 3I.

Sampling stations are provided for periodic analysis of reactor water, refueling and fuel storage pool water, and suppression pool water to assure compliance with operational limits of the plant technical specifications.

3.11.5.2 Radiation Environment

Safety-related systems and components are designed to perform their safety-related function when exposed to the normal operational radiation levels and accident radiation levels.

Electronic equipment subject to radiation exposure in excess of 1000 R and mechanical equipment in excess of 10,000 R will be qualified in accordance with Reference 1.

The normal operational exposure is based on the radiation sources provided in Chapter 12.

Radiation sources associated with the DBA and developed in accordance with NUREG-0588 (Reference 3) are provided in Chapter 15.

Integrated doses associated with normal plant operation and the design basis accident condition for various plant compartments are described in Appendix 3I.

3.11.6 COL License Information

3.11.6.1 Environmental Qualification Document

The EQD shall be prepared summarizing the qualification results for all safety-related equipment. The EQD shall include the following:

(1) The test environmental parameters and the methodology used to qualify the equipment located in mild and harsh environments shall be identified.

(2) A summary of environmental conditions and qualified conditions for the safety-related equipment located in a harsh environment zone shall be presented in the system component evaluation work (SCEW) sheets as described in Table I-1 of GE's environmental qualification program (Reference 2). The SCEW sheets shall be compiled in the EQD.

(3) Equipment gamma and beta radiation dose data for both normal and accident conditions will be provided in accordance with the requirements of Subsection 12.2.3.1.

3.11.6.2 Environmental Qualification Records

The results of the qualification tests shall be recorded and maintained in an auditable file.

3.11.6.3 Surveillance, Maintenance and Experience Information

The COL applicant will require vendor equipment certificates of qualification compliance and will develop a surveillance and maintenance program in accordance with Subsection 3.11.2.

Non-safety-related control systems subjected to adverse environments will be evaluated for safety implications to safety-related protective functions, and equipment wetting and flooding above the flood level will be addressed in accordance with Subsection 3.11.1.

3.11.7 References

(1) Code of Federal Regulations, Title 10, Chapter I, Part 50, Paragraph 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plant.

(2) General Electric Environmental Qualification Program,NEDE-24326-1-P, Proprietary Document, January 1983.

(3) Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, NUREG-0588.

APPENDIX 3H
STRUCTURAL DETAILS

years in BWR applications. Extensive laboratory tests have demonstrated that XM-19 is a suitable material and that it is resistant to stress corrosion in a BWR environment.

4.5.3 COL License Information

4.5.3.1 CRD Inspection Program

The CRD inspection program shall include provisions to detect incipient defects before they could become serious enough to cause operating problems. The CRD nozzle and CRD bolting are included in the inservice inspection program. [See Table 5.2-8, System Number B11/B12] CRD bolting is available for inservice examinations during normally scheduled CRD maintenance. (See Subsection 4.5.1.2(1)).

unidentified leakage rate might be emitted from a single crack in the nuclear system process barrier.

An allowance for leakage that does not compromise barrier integrity and is not identifiable is established for normal plant operation.

The unidentified leakage rate limit is established at 3.785 liters/min to allow time for corrective action before the process barrier could be significantly compromised. This unidentified leakage rate is a small fraction of the calculated flow from a critical crack in a primary system pipe (Appendix 3E).

5.2.5.5.2 Margins of Safety

The margins of safety for a detectable flaw to reach critical size are presented in Subsection 5.2.5.5.3. Figure 5.2-8 shows general relationships between crack length, leak rate, stress, and linesize using mathematical models.

5.2.5.5.3 Criteria to Evaluate the Adequacy and Margin of Leak Detection System

For process lines that are normally open, there are at least two different methods of detecting abnormal leakage from each system comprising the nuclear system process barrier, located both inside the primary containment (drywell) and external to the drywell, in the reactor building the steam tunnel and the turbine building (Tables 5.2-6 and 5.2-7). The instrumentation is designed so it can be set to provide alarms at established leakage rate limits and isolate the affected system if necessary. The alarm points are determined analytically or based on measurements of appropriate parameters made during startup and preoperational tests.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly. The established limit is sufficiently low so that, even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier would be threatened.

The leak detection system will satisfactorily detect unidentified leakage of 3.785 liters/min within the drywell.

5.2.5.6 Differentiation Between Identified and Unidentified Leaks

Subsection 5.2.5.1 describes the leak detection methods utilized by the leak detection system. The ability of the leak detection system to differentiate between identified and unidentified leakage is discussed in Subsections 5.2.5.4 and 5.2.5.5.

5.2.5.7 Sensitivity and Operability Tests

Sensitivity, including sensitivity tests and response time of the leak detection system, and the criteria for shutdown if leakage limits are exceeded are covered in Subsections 5.2.5.1.1, 5.2.5.1.2, 5.2.5 2.1(1) and 7.3.1.1.2.

Testability of the LDS is contained in Subsection 7.3.1.1.2(10).

5.2.5.8 Testing and Calibration

Provisions for testing and calibration of the leak detection and isolation system are covered in Chapter 14.

5.2.5.9 Regulatory Guide 1.45: Compliance

These guidelines are prescribed to assure that leakage detection and collection systems provide maximum practical identification of leaks from the RCPB.

Leakage is separated into identified and unidentified categories and each is independently monitored, thus meeting Position C.1 requirements.

Leakage from unidentified sources from inside the drywell is collected into the floor drain sump and monitored with an accuracy better than

3.785 liters/min thus meeting Position C.2 requirements.

By monitoring (1) floor drain sump fillup and pumpout rate, (2) airborne particulates, and (3) air coolers condensate flow rate, Position C.3 is satisfied.

Monitoring of the reactor building cooling water heat exchanger coolant return lines for radiation due to leaks within the RHR, RIP and CUW and the fuel pool cooling system heat exchangers satisfies Position C.4. For system detail, see Subsection 7.6.1.2.

The floor drain sump monitoring, air particulates monitoring, and air cooler condensate monitoring are designed to detect leakage rates of 3.785 liters/min within one hour, thus meeting Position C.5 requirements.

The fission products monitoring subsystem is qualified for SSE. The containment floor drain sump monitor, air cooler, and condensate flow meter are qualified for OBE, thus meeting Position C.6 requirements.

Leak detection indicators and alarms are provided in the main control room. This satisfies Position C.7 requirements. Procedures and graphs will be provided by the COL applicant to plant operators for converting the various indicators to a common leakage equivalent, when necessary, thus satisfying the remainder of Position C.7 (See Subsection 5.2.6.1 for COL license information). The leakage detection system is equipped with provisions to permit testing for operability and calibration during the plant operation using the following methods:

(1) simulation of trip signal;

(2) comparing channel to channel of the same leak detection method (i.e., area temperature monitoring);

(3) operability checked by comparing one method versus another (i.e., sump fillup rate versus pumpout rate and particulate monitoring or air cooler condensate flow versus sump fillup rate); and

(4) continuous monitoring of floor drain sump level, and a source of water for calibration and testing is provided.

These satisfy Position C.8 requirements.

Limiting unidentified leakage to the 3.785 liters/min and identified to 95 liters/min satisfies Position C.9.

5.2.6 COL License Information

5.2.6.1 Conversion of Indications

Procedures and graphs will be provided by the COL applicant to operations for converting the various indicators into a common leakage equivalent (See Subsection 5.2.5.9).

5.2.6.2 Plant-Specific ISI/PSI

COL applicants will submit the complete plant- specific ISI/PSI program. Each applicant will submit or address the following:

(1) The PSI program should include reference to the edition and addenda of ASME Code Section XI that will be used for selecting of components for examinations, lists of the components subject to examination, a description of the components exempt from examination by the applicable code, and isometric drawings used for the examination.

(2) Submits plans for preservice examination of the reactor pressure vessel welds to address the degree of compliance with RG 1.150.

(3) Discusses the near-surface examination and resolution with regard to detecting service-induced flaws and the use of electronic gating as related to the volume of material near the surface that is not being examined. Discusses how the internal surfaces (e.g., inner radius of a pipe section and reactor vessel internals) will be examined.

(4) Submits an acceptable resolution of the information requested regarding the ISI/PSI program.

(5) Submits all relief requests, if needed, with a supporting technical justification.

Table 5.2-6

LDS CONTROL AND ISOLATION FUNCTION VS MONITORED PROCESS VARIABLES

LDS Control & Isolation Functions / Monitored Variables	Reactor Water Level Low	Turbine Inlet SL Press Low	Reactor Pressure High	MSL Flow Rate High	MSL Radiation High	MSL Tunnel Amb. Temp High	Turbine area Amb. Temp High	Main Condenser Vacuum Low	Drywell Pressure High	RHR Equip Area Temp High	RCIC Equip Area Temp High	RCIC SL Pressure Low	RCIC SL Flow Rate High	RCIC Vent Exhaust Press High	CUW Equip Area Temp High	CUW Differential Flow High	SLCS Pumps Running	LCW Drain Line Radiation High	HCW Drain Line Radiation High	R/B HVAC Exhaust Air Rad High	F/H Area Exhaust Air Rad High
MSIVs & MSL Drain Line Valves	L1.5	X		X	X	X	X	X													
CUW Process Lines Isolation	L2		X1			X									X	X	X				
RHR S/C PCV Valves	L3		X							X											
RCIC Steam Line Isolation											X	X	X	X							
ATIP Withdrawel	L3								X												
DW RAD Sampling Line Isolation	L2								X												
SPCU Process Line Isolation	L3								X												
DW LCW Sump Drain Line Isolation	L3								X									X			
DW HCW Sump Drain Line Isolation	L3								X										X		
RCW PCV Valves Isolation	L1								X												
HNCW PCV Valves Isolation	L1								X												
AC System P&V Valves Isolation	L3								X											X	X
FCS PCV Valves Isolation	L3								X												
R/B HVAC Air Ducts Isolation	L3								X											X	X
SGTS Initiation	L3								X											X	X

1- HEAD SPRAY VALVE ONLY

Table 5.2-7
LEAKAGE SOURCES VS MONITORED TRIP ALARMS

Leakage Source \ Monitored Plant Variable			Reactor Vessel Water Level Low	Drywell Pressure High	DW Floor Drain Sump High Flow	DW Equip Drain Sump High Flow	DW Fission Products Radiation High	Drywell Temperature High	SRV Discharge Line Temperature High	Vessel Head Flange Seal Pressure High	RB Eq/Fl Drain Sump High Flow	DW Air Cooler Condensate Flow High	MSL or RCIC Steamline Flow High	MSL Tunnel or TB Ambient Area Temp High	Equip Areas Ambient or Diff Temp High	CUW Diferential Flow High	MSL Tunnel Radiation High	Inter-System Leakage (Radiation) High
Main Steam Lines	I		X	X	X		X	X	X			X	X					
		O	X								X		X	X	X		X	
RCIC Steamline	I		X	X	X		X	X				X	X					
		O	X								X		X		X			
RCIC Water	I																	
		O									X				X			
RHR Water	I		X	X	X		X	X				X						
		O									X				X			X
HPCF Water	I		X	X	X		X	X				X						
		O									X				X			
CUW Water	I		X	X	X		X	X				X						
		O	X								X			X	X	X		X
Feedwater	I		X	X	X		X	X				X						
		O									X			X				
Recirc Pump Motor Casing	I			X	X		X	X				X						X
		O																
Reactor Vessel Head Seal	I					X				X								
		O																
Valve Stem Packing	I					X												
		O									X							
Miscellaneous Leaks	I				X			X										
		O									X							X

I = Inside Drywell Leakage
O = Outside Drywell Leakage
X Reactor coolant leakage in cooling water to RHR Hx, RIP Hx, CUW Non-regen Hx's or to FP cooling Hx.

Preparation of impact testing procedures, calibration of test equipment, and the retention of the records of these functions and test data comply with the requirements of the ASME Code, Section III. Personnel conducting impact testing are qualified by experience, training or qualification testing that demonstrates competence to perform tests in accordance with the testing procedure.

(4) Charpy-V Curves for the RPV Beltline (G-IIIA and G-IVA-1)

A full transverse Charpy-V curve is determined for all heats of base material and weld metal used in the core beltline region with a minimum of three (3) specimen tested at the actual T_{NDT}. The minimum upper-shelf energy level for base material and weld metal in the beltline region is 10.4 kg fm as required by G-IVA.1.

In regard to G-III A, it is understood that separate, unirradiated baseline specimens per ASTM E 185, Paragraph 6.3.1 will be used to determine the transition temperature curve of the core beltline base material, HAZ and weld metal.

(5) Bolting Material

All bolting material exceeding one inch diameter has a minimum of 6.4 kg fm charpy-V energy and 0.64 mm lateral expansion at the minimum bolt preload temperature of 21 °C.

(6) Alternative Procedures for the Calculation of Stress Intensity Factor (Appendix G-IV A)

Stress intensity factors are calculated by the methods of ASME, Section III, Appendix G. Discontinuity regions are evaluated as shell and head areas, as part of the detailed thermal and stress analyses in the vessel stress report. Considerations are given to membrane and bending stresses, as outlined in Paragraph G-2222. Equivalent margins of safety to those required for shells and heads are demonstrated using a 1/4 T postulated defect at all locations, with the exception of the main closure flange to the head and shell discontinuity locations. Additional instruction on operating limits is required for outside surface flaw sizes greater than 6.0 mm at the outside surface of the flange to shell joint based on analysis made for ABWR reactor vessels using the calculations methods shown in WRCB 175. It will be demonstrated, using a test mockup of these areas, that smaller defects can be detected by the ultrasonic inservice examinations procedures required at the adjacent weld joint.

(7) Fracture Toughness Margins in the Control of Reactivity (Appendix G-IV A).

ASME Code, Section III, Appendix G, was used in determining pressure/temperature limitations for all phases of plan operation.

5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with Reactor Vessel Material Surveillance Program Requirements

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and thermal environment.

Reactor vessel materials surveillance specimens are provided in accordance with requirements of ASTM E-185 and 10CRF 50, Appendix H. Materials for the program are selected to represent materials used in the reactor beltline region. Specimens are manufactured from a plate or forging actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld material, and the weld heat-affected zone material. The plate and weld are heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel. Each in-reactor surveillance capsule contains 36 Charpy V-notch and 6 tensile specimens. The capsule loading consists of 12 Charpy V Specimens each of base metal, weld metal, heat-affected zone material, and 3 tensile specimens each from base metal and weld metal. Weld metal specimens will be made from the same heat of weld wire and lot of flux (if applicable) and by the same welding practice as used for the beltline weld. A set of out-of-reactor baseline Charpy V-notch specimens, tensile specimens, and archive material are provided with the surveillance test specimens. Neutron dosimeters and temperature monitors will be located within the capsules as required by ASTM E 185.

Four surveillance capsules are provided. The predicted end of the adjusted reference temperature of the reactor vessel steel is less than 38°C.

The following proposed withdrawal schedule is extrapolated from ASTM E 185.

First capsule: After 6 effective full-power years
Second capsule: After 20 effective full-power years
Third capsule: With an exposure not to exceed the peak EOL fluence.
Fourth capsule: Schedule determined based on results of first three capsules per ASTM E 185, Paragraph 7.6.2.

See Subsection 5.3.4.2 for additional capsule requirements.

Fracture toughness testing of irradiated capsule specimens will be in accordance with requirements of ASTM E 185 as called out for by 10CFR50, Appendix H.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsections 4.1.4.5 and 4.3.2.8.

5.3.1.6.3 Predicted Irradiation Effects on Beltline Materials

Transition temperature changes and changes in upper-shelf energy shall be calculated in accordance with the rules of Regulatory Guide 1.99. Reference temperatures shall be established in accordance with 10CFR50, Appendix G, and NB-2330 of the ASME Code.

Since weld material chemistry and fracture toughness data are not available at this time, the limits in the purchase specification were used to estimate worst-case irradiation effects.

These estimates show that the adjusted reference temperature at end-of-life is less than 100 °F, and the end-of-life upper-shelf energy exceeds 50 ft-lb. (See response to Question 251.5 for the calculation and analysis associated with this estimate).

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment (Appendix H.II B (2))

Surveillance specimen capsules are located at two azimuths at a common elevation in the core beltline region. The capsule placement is designed to produce a lead factor of approximately 1.2 to 1.5. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding. Since reactor vessel specifications require that all low-alloy steel pressure vessel boundary materials be produced to fine-grain practice, underclad cracking is of no concern. The capsule holder brackets allow the removal and reinsertion of capsule holders. Although not code parts, these brackets are designed, fabricated, and analyzed to the requirements of ASME Code Section III. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel. See Subsection 5.3.4.2 for COL license information pertaining to materials and surveillance capsules.

In areas where brackets (such as the surveillance specimen holder brackets) are located, additional nondestructive examinations are performed on the vessel base metal and stainless steel weld-deposited cladding or weld-buildup pads during vessel manufacture. The base metal is ultrasonically examined by straight-beam techniques to a depth at least equal to the thickness of the bracket being joined. The area examined is the area of width equal to at least half the thickness of the part joined. The required stainless steel weld-deposited cladding is similarly examined. The full penetration welds are liquid-penetrant examined. Cladding thickness is required to be at least 1/8 inch. These requirements have been successfully applied to a variety of bracket designs the main condenser, and the feedwater system will supply the makeup water required to maintain reactor vessel inventory.

In the event the reactor vessel is isolated and the feedwater supply unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat. Upon reaching a predetermined low level, the RCIC system will be initiated automatically. The turbine-driven pump will supply demineralized make-up water from (1) the condensate storage tank (CST) to the reactor vessel and (2) the suppression pool. Seismically installed level instrumentation is provided for automatic transfer of the water source with manual override from CST to suppression pool on receipt of either a low CST water level or high suppression pool level signals (CST water is primary source). The turbine will be driven with a portion of the decay heat steam from the reactor vessel and will exhaust to the suppression pool. Suppression pool water is not usually demineralized and hence should only be used in the event all sources of demineralized water have been exhausted.

During RCIC operation, the suppression pool shall act as the heat sink for steam generated by reactor decay heat. This will result in a rise in pool water temperature. RHR heat exchangers are used to maintain pool water temperature within acceptable limits by cooling the pool water.

5.4.6.1.1 Residual Heat and Isolation

5.4.6.1.1.1 Residual Heat

The RCIC system shall initiate and discharge, within 30 seconds, a specified constant flow into the reactor vessel over a specified pressure range. The RCIC water discharge into the reactor vessel varies between a temperature of $4.5^{\circ}C$ up to and including a temperature of $77^{\circ}C$. The mixture of the cool RCIC water and the hot steam does the following:

(1) quenches steam,

(2) removes reactor residual heat, and

(3) replenishes reactor vessel inventory.

Redundantly the HPCF system performs a similar function, hence providing single failure protection. Both systems use different reliable electrical power sources which permit operation with either onsite or offsite power. Additionally, the RHR system performs a residual heat removal function.

5.4.6.1.1.2 Isolation

Isolation valve arrangements include the following:

(1) Two RCIC lines penetrate the reactor coolant pressure boundary. The first is the RCIC steamline which branches off one of the main steamlines between the reactor vessel and the main steam isolation valves. This line has two automatic motor-operated isolation valves, one is located inside and the other outside the drywell. An automatic motor-operated inboard RCIC isolation bypass valve is used. The isolation signals noted earlier close these valves.

(2) The RCIC pump discharge line is the other line that penetrates the reactor coolant pressure boundary, which directs flow into a feedwater line just outboard of the primary containment. This line has a testable check valve and an automatic motor-operated valve located outside primary containment.

(3) The RCIC turbine exhaust line also penetrates the containment. Containment penetration is located about a meter above the suppression pool maximum water level. A vacuum breaking line with two vacuum breakers in series runs in the suppression pool air space and connects to the RCIC turbine exhaust line inside the containment. Located outside the containment in the turbine exhaust line is a remote-manually controlled motor operated isolation valve.

(4) The RCIC pump suction line, minimum flow pump discharge line, and turbine exhaust line penetrate the containment and are sub-

merged in the suppression pool. The isolation valves for these lines are outside the containment and require automatic isolation operation, except for the turbine exhaust line which has remote manual operation.

The RCIC system design includes interfaces with redundnat leak detection devices, monitoring:

(1) a high pressure drop across a flow device in the steam supply line equivalent to 300 percent of the steady state steam flow at 83.8 kg/cm^{2}abs pressure;

(2) a high area temperature utilizing temperature switches as described in the leak detection system (high area temperature shall be alarmed in the control room);

(3) a low reactor pressure of 3.5 kg/cm^{2}g minimum; and

(4) a high pressure between the RCIC turbine exhaust rupture diaphragms.

These devices, activated by the redundant power supplies, automatically isolate the steam supply to the RCIC turbine and trip the turbine. HPCF provides redundancy for RCIC should RCIC become isolated.

5.4.6.1.2 Reliability, Operability, and Manual Operation

5.4.6.1.2.1 Reliability and Operability

The RCIC system (Table 3.2-1) is designed commensurate with the safety importance of the system and its equipment. Each component is individually tested to confirm compliance with system requirements. The system as a whole is tested during both the start-up and pre-operational phases of the plant to set a base mark for system reliability. To confirm that the system maintains this mark, functional and operability testing is performed at predetermined intervals throughout the life of the plant.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the suppression pool and discharging through a full flow test return line to the suppression pool. All components of the RCIC system are capable of individual functional testing during normal plant operation. System control provides automatic return from test to operating mode if system initiation is required, and the flow is automatically directed to the vessel. Also, see Subsection 5.4.6.2.4.

See Subsection 5.4.15.1 for COL license information.

5.4.6.1.2.2 Manual Operation

In addition to the automatic operational features, provisions are included for remote-manual startup, operation, and shutdown of the RCIC system provided initiation or shutdown signals do not exist.

5.4.6.1.3 Loss of Offsite Power

The RCIC system power is derived from a reliable source that is maintained by either onsite or offsite power.

5.4.6.1.4 Physical Damage

The system is designed to the requirements presented in Table 3.2-1 commensurate with the safety importance of the system and its equipment. The RCIC is physically located in a different quadrant of the reactor building and utilizes different divisional power and separate electrical routings than its redundant system as discussed in Subsections 5.4.6.1.1.1 and 5.4.6.2.4.

5.4.6.1.5 Environment

The system operates for the time intervals and the environmental conditions specified in Section 3.11.

5.4.6.2 System Design

5.4.6.2.1 General

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reactor pressure vessel to the main turbine and condenser system, penetrate the primary containment. The main steam drain lines connect the low points of the steam lines, penetrate the primary containment and are routed to the condenser hotwell. The RCIC turbine steamline connects to the main steam line in the upper drywell and penetrates the primary containment. For these lines isolation is provided by automatically actuated block valves, one inside and one just outside the containment.

6.2.4.3.2.1.2.2 RHR Shutdown Cooling Line

Three RHR shutdown cooling lines connect to the reactor vessel and penetrate the primary containment. Isolation is provided by two automatically actuated block valves, one inside and other outside the containment.

6.2.4.3.2.1.2.3 Reactor Water Cleanup System Suction Line

The RWCU takes its suction from the bottom head of the RPV and from the RHR "B" shutdown cooling suction line. The RWCU suction line is isolated by two automatic motor-operated gate valves on the inside and outside of the containment. Should a break occur in the RWCU system, the check valves would prevent backflow from the RPV and the isolation valves would prevent forward flow from the RPV.

RWCU pumps, heat exchangers and filter demineralizers are located outside the drywell.

6.2.4.3.2.1.3 Conclusion on Criterion 55

In order to assure protection against the consequences of accidents involving the release of radioactive material, pipes which form the reactor coolant pressure boundary have been shown to provide adequate isolation capabilities on a case-by-case basis. In all cases, a minimum of two barriers were shown to protect against the release of radioactive materials.

In addition to meeting the isolation requirements stated in Criterion 55, the pressure-retaining components which comprise the reactor coolant pressure boundary are designed to meet other appropriate requirements which minimize the probability or consequences of an accidental pipe rupture. The quality requirements for these components ensure that they are designed, fabricated, and tested to the highest quality standards of all reactor plant components. The classification of components which comprise the reactor coolant pressure boundary are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 1.

It is therefore concluded that the design of piping system which comprise the reactor coolant pressure boundary and penetrate containment satisfies Criterion 55.

6.2.4.3.2.2 Evaluation Against Criterion 56

Criterion 56 requires that lines which penetrate the containment and communicate with the containment interior must have two isolation valves; one inside the containment, and one outside, unless it can be demonstrated that the containment isolation provisions for a specific class of lines are acceptable on some other basis.

Although a word-for-word comparison with Criterion 56 in some cases is not practical, it is possible to demonstrate adequate isolation provisions on some other defined basis.

6.2.4.3.2.2.1 Influent Lines to Suppression Pool

Figure 6.2-38 identifies the isolation provisions in the influent lines to the suppression pool.

6.2.4.3.2.2.1.1 HPCF and RHR Test and Pump Minimum Flow Bypass Lines

The HPCF and RHR test and pump minimum flow bypass lines have isolation capabilities commensurate with the importance to safety of isolating these lines. Each line has a motor-operated valve located outside the containment. Containment isolation requirements are met on the basis that the lines are low-pressure lines constructed to the same quality standards commensurate with their importance to safety. Furthermore, the consequences of a break in 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

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 located as close as practical to the containment vessel. The piping to both valves is an extension of the containment boundary.

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.

oxygen concentration does not exceed 3.5 volume percent during normal operation. The actual oxygen concentration shall not exceed five volume percent during an accident when the hydrogen concentration is greater than four percent. The inert containment can be deinerted to allow safe personnel access without breathing apparatus in less than four hours.

Each penetration and pipe carrying nitrogen is sloped as necessary to prevent condensation collection and line blockage and shall be protected against entry of debris.

All pipe volumes where liquid or very cold nitrogen could be trapped between closed valves have relief valves. All relief valves exhaust outside the reactor building. Means are provided to add nitrogen to the nitrogen storage tank vapor space (to decrease tank pressure) and the liquid volume (to increase tank pressure). Tank level and pressure indication are provided at the tank. Means for startup full scale testing of the inerting and makeup portions of the system without nitrogen injection to the containment is provided. During startup, the test discharges shall be temporarily piped away from the control panel and storage and vaporization equipment to avoid excessive noise from the open discharge. Strainers are provided in the liquid portion of the makeup and inerting lines. Means are provided to feed the makeup circuit from either the liquid or vapor portion of the nitrogen storage. Pressure is automatically maintained in the nitrogen storage tank during nitrogen discharge by a circuit with another ambient heat exchanger fed by a pressure control valve. The inerting and makeup portions of the system do not rely on pumps to perform their function. Means are provided to manually vent the tank vapor space to control pressure. Means are provided to drain the storage tank. The vessel bottom is sloped or dished to facilitate this draining.

Pressure relief for the nitrogen storage tank is provided at 10 percent above the upper limit of the normal range of operating pressures. Rupture disks, set 20 percent above the upper limit but not higher than the design pressure of the vessel, are provided. Redundant pressure relief valves are provided so that protection is immediately available should a disk rupture and then be isolated. Penetrations through the nitrogen storage tank insulation are minimized to reduce heat gain. The length of piping through the insulation is maximized to the extent practicable to reduce heat gain.

The drywell and wetwell atmospheric oxygen concentration will be less than 3.5% by volume within 24 hours after thermal power is greater than 15% of plant rating. Twenty four hours of operation above 3.5% oxygen by volume and 15% power is allowed before a scheduled shutdown. All piping outside the outboard primary containment isolation valves carrying nitrogen are protected from overpressurization by relief valves ducted to the atmosphere.

6.2.5.2.3 Nitrogen Makeup

(1) The nitrogen makeup equipment is sized to maintain a positive pressure in the drywell and wetwell during the maximum drywell cool down rate not caused by spray actuation.

(2) Automatic addition of nitrogen is physically limited to less than the maximum drywell bleed capacity.

6.2.5.2.4 Drywell Bleed

Primary containment bleed capability is provided in accordance with Regulatory Guide 1.7 and as an aid in cleanup following an accident. During normal plant operation, the bleed line also functions, in conjunction with the nitrogen purge line to maintain primary containment pressure at about 0.75 psig and oxygen concentration below 3.5 percent by volume. This is accomplished by makeup of the required quantity of nitrogen into the primary containment through the makeup line or relieving pressure through the bleed line. The drywell bleed line is manually operable from the control room. Flow through the bleed line will be directed through either the SGTS or the secondary containment HVAC, and be monitored by the SGTS and SCHVAC flow and radiation instrumentation. All ACS primary containment isolation valves are automatically closed when high radiation is detected in the exhaust flow.

The drywell bleed line is located above an elevation which would be covered by post-LOCA flooding for unloading the fuel.

6.2.5.2.5 Pressure Control

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

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

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

6.2.5.2.6 Overpressure Protection

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

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

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

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

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

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

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

6.2.5.2.7 Recombiner

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

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

6.2.5.3 Design Evaluation

The ACS is designed to maintain the containment in an inert condition except for nitrogen makeup needed to maintain a positive containment pressure and prevent air (0_2) leakage from the secondary into the primary containment.

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

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

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conservative direction simultaneously. The results of this calculation for the limiting case are given in Figure 6.3-67 through 6.3-75 and Table 6.3-4. Since the ABWR results have large margins to the 10CFR50.46 licensing acceptance criteria, the ABWR licensing PCT can be based on the bounding PCT which is well below the 2200°F PCT limit.

6.3.3.8 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Section 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the criteria in Appendix 4B, given operation at or below the MAPLHGRs provided by the utility for each fuel bundle. See Subsection 6.3.6.

6.3.4 Tests and Inspections

6.3.4.1 ECCS Performance Tests

All systems of the ECCS are tested for their operational ECCS function during the preoperational and/or startup test program. Each component is tested for power source, range, direction of rotation, setpoint, limit switch setting, torque switch setting, etc. Each pump is tested for flow capacity for comparison with vendor data. (This test is also used to verify flow measuring capability). The flow tests involve the same suction and discharge source (i.e., suppression pool).

All logic elements are tested individually and then as a system to verify complete system response to emergency signals including the ability of valves to revert to the ECCS alignment from other positions.

Finally, the entire system is tested for response time and flow capacity taking suction from its normal source and delivering flow into the reactor vessel. This last series of tests is performed with power supplied from both offsite power and onsite emergency power.

See Chapter 14 for a thorough discussion of preoperational testing for these systems.

See Subsection 6.3.6.2 for COL license information regarding ECCS testing requirements.

6.3.4.2 Reliability Tests and Inspections

The average reliability of a standby (nonoperating) safety system is a function of the duration of the interval between periodic functional tests. The factors considered in determining the periodic test interval of the ECCS are: (1) the desired system availability (average reliability); (2) the number of redundant functional system success paths; (3) the failure rates of the individual components in the system; and (4) the schedule of periodic tests (simultaneous versus uniformly staggered versus randomly staggered).

All of the active components of the HPCF System, ADS, RHR and RCIC Systems are designed so that they may be tested during normal plant operation. Full flow test capability is provided by a test line back to the suction source. The full flow test is used to verify the capacity of each ECCS pump loop while the plant remains undisturbed in the power generation mode. In addition, each individual valve may be tested during normal plant operation.

All of the active components of the ADS System, except the safety/relief valves and their associated solenoid valves, are designed so that they may be tested during normal plant operation. The SRVs and associated solenoid valves are all tested during plant initial power ascension per Appendix A, Paragraph D.2.c of Regulatory Guide 1.68. SRVs are bench tested to establish lift settings.

Testing of the initiating instrumentation and controls portion of the ECCS is discussed in Subsection 7.3.1. The emergency power system, which supplies electrical power to the ECCS in the event that offsite power is unavailable, is tested as described in Subsection 8.3.1. The frequency of testing is specified in the Chapter 16 Technical Specifications. Visual inspections of all the ECCS components located outside the drywell can be made at any time during power operation. Components inside the drywell can be visually inspected only during periods of access to the drywell. When the reactor vessel is open, the spargers and other internals can be inspected.

6.3.4.2.1 HPCF Testing

The HPCF can be tested at full flow with suppression pool water at any time during plant operation except when a system initiation signal is present. If an initiation signal occurs while the HPCF is being tested, the system returns automatically to the operating mode. The motor-operated valve in the line to the condensate storage system is interlocked closed when the suction valve from the suppression pool is open.

A design flow functional test of the HPCF over the operating pressure and flow range is performed by pumping water from the suppression pool back through the full flow test return line to the suppression pool.

The suction valve from the condensate storage tank and the discharge valve to the reactor remain closed. These two valves are tested separately to ensure their operability.

The HPCF test conditions are tabulated on the HPCF process flow diagram (Figure 6.3-1).

6.3.4.2.2 ADS Testing

An ADS logic system functional test and simulated automatic operation of all ADS logic channels are to be performed at least once per plant operating interval between reactor refuelings. Instrumentation channels are demonstrated operable by the performance of a channel functional test and a trip unit calibration at least once per month and a transmitter calibration at least once per operating interval.

All SRVs, which include those used for ADS are bench tested to establish lift settings in compliance with ASME Code Section XI.

6.3.4.2.3 RHR Testing

The RHR pump and valves are tested periodically during reactor operation. With the injection valves closed and the return line open to the suppression pool, full flowing pump capability is demonstrated. The injection valve and the check valve are tested in a manner similar to that used for the HPCF valves. The system test conditions during reactor operation are shown on the RHR system process diagram (Figure 6.3-3).

6.3.4.2.4 RCIC Testing

The RCIC loop can be tested during reactor operation. To test the RCIC pump at rated flow, the test bypass line valve to the suppression pool and the pump suction valve from the suppression pool are opened and the pump is started using the turbine controls in the control room. Correct operation is determined by observing the instruments in the control room.

If an initiation signal occurs during the test, the RCIC system returns to the operating mode. The valves in the test bypass lines are closed automatically and the RCIC pump discharge valve is opened to assure flow is correctly routed to the vessel.

6.3.5 Instrumentation Requirements

Design details including redundancy and logic of the ECCS instrumentation are discussed in Section 7.3.

All instrumentation required for automatic and manual initiation of the HPCF, RCIC, RHR and ADS is discussed in Subsection 7.3.1 and is designed to meet the requirements of IEEE-279 and other applicable regulatory requirements. The HPCF, RCIC, RHR and ADS can be manually initiated from the control room.

The RCIC, HPCF, and RHR are automatically initiated on low reactor water level or high drywell pressure. The ADS is automatically actuated by sensed variables for reactor vessel low water level and drywell high pressure plus indication that at least one RHR or HPCF pump is operating. The HPCF, RCIC, and RHR automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic invitation signal. The RHR LPFL mode injection into the RPV begins when reactor pressure decreases to the RHR's pump discharge shutoff pressure.

HPCF injection begins as soon as the HPCF pump is up to speed and the injection valve is open, since the HPCF is capable of injection water into the RPV over a pressure range from 1177 psid to 100 psid or pressure difference between the vessel and drywell.

6.3.6 COL License Information

6.3.6.1 ECCS Performance Results

The exposure dependent MAPLHGR, peak cladding temperature, and oxidation fraction for each fuel bundle design based on the limiting break size will be provided by the COL applicant to the USNRC for information. (See Subsection 6.3.3).

6.3.6.2 ECCS Testing Requirements

In accordance with Technical Specification SR 3.5.1.7, the COL applicant will perform a test every refueling in which each ECCS subsystem is actuated through the emergency operating sequence. (See Subsection 6.3.4.1)

6.3.7 Reference

1. *General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K,* (NEDE-20566-P-A), September 1986.

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6.5 FISSION PRODUCTS REMOVAL AND CONTROL SYSTEMS

6.5.1 Engineered Safety Features Filter Systems

The filter systems required to perform safety-related functions following a design basis accident are:

(1) Standby gas treatment system (T22-SGTS).

(2) Control room portion of the HVAC system. (U41-HVAC)

The control room portion of the HVAC system is discussed in Section 6.4 and Subsection 9.4.1. The SGTS is discussed in this Subsection (6.5.1).

6.5.1.1 Design Basis

6.5.1.1.1 Power Generation Design Basis

The SGTS has the capability to filter the gaseous effluent from the primary containment or from the secondary containment when required to limit the discharge of radioactivity to the environment to meet 10CFR100 requirements.

6.5.1.1.2 Safety Design Basis

The SGTS is designed to accomplish the following:

(1) Maintain a negative pressure in the secondary containment, relative to the outdoor atmosphere, to control the release of fission products to the environment.

(2) Filter airborne radioactivity (halogen and air particulates) in the effluent to reduce offsite doses to within the limits specified in 10CFR100.

(3) Ensure that failure of any active component, assuming loss of offsite power, cannot impair the ability of the system to perform its safety function.

(4) Remain intact and functional in the event of a safe shutdown earthquake (SSE).

(5) Meet environmental qualification requirements established for system operation.

(6) Filter airborne radioactivity (halogens and particulates) in the effluent to reduce offsite doses during normal and upset operations to within the limits of 10CFR20.

6.5.1.2 System Design

6.5.1.2.1. General

The SGTS P&ID is provided as Figure 6.5-1.

6.5.1.2.2 Component Description

Table 6.5-1 provides a summary of the major SGTS components. The SGTS consists of two parallel and redundant filter trains. Suction is taken from above the refueling area or from the primary containment via the atmospheric control system (T31-ACS). The treated discharge goes to the main plant stack.

The SGTS consists of the following principal components:

(1) Two filter trains each consisting of a moisture seprator, an electric process heater, a prefilter, a high efficiency particulate air (HEPA) filter, a charcoal adsorber, a second HEPA filter, and space heaters.

(2) Two independent process fans located downstream of the each filter train and two independent cooling fans for the removal of decay heat from charcoal.

6.5.1.2.3 SGTS Operation

6.5.1.2.3.1 Automatic

Upon the receipt of a high primary containment pressure signal or a low reactor water level signal, or when high radioactivity is detected in the secondary containment or 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

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

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 powered from separate Class 1E electrical busses.

(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 event 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

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6.6 PRESERVICE AND INSERVICE INSPECTION AND TESTING OF CLASS 2 AND 3 COMPONENTS AND PIPING

This subsection describes the preservice and inservice inspection and system pressure test programs for Quality Groups B and C, i.e., ASME Code Class 2 and 3 items*, respectively. It describes those programs implementing the requirements of ASME B&PV Code, Section XI, Subsections IWB and IWC. The requirements for subsequent inservice inspection intervals are addressed in Subsection 5.3.3.7.

The development of the preservice and inservice inspection program plans will be the responsibility of the COL applicant and will be based on the ASME Code, Section XI, Edition and Addenda specified in accordance with 10CFR50, Section 50.55a. Responsibility for designing components for preservice and inservice inspection is the responsibility of the COL applicant. The COL applicant will be responsible for specifying the Edition of the ASME Code, Section XI, to be used, based on the procurement date of the component per 10CFR50, Section 50.55a. The ASME Code requirements discussed in this section are provided for information and are based on the 1989 Edition of the ASME Section XI. See Subsection 6.6.9.1 for COL license information requirements.

6.6.1 Class 2 and 3 System Boundaries

The Class 2 and 3 system boundaries for both preservice and inservice inspection programs and the system pressure test program includes applicable items within the 3 boundary and the 4 boundary on the piping and instrumentation drawings (P&IDs). Those items boundaries include all or part of the following:

(1) Main steam system
(2) Feedwater system
(3) Reactor core isolation cooling system
(4) High pressure core flooder system
(5) Standby liquid control system
(6) Residual heat removal system.
(7) Reactor water clean up system
(8) Control rod drive system
(9) Deleted
(10) Purified make up water system
(11) Atmospheric control system
(12) Deleted
(13) HVAC normal cooling water system
(14) Deleted
(15) Deleted
(16) Deleted
(17) Reactor building cooling water system
(18) Deleted
(19) Fuel pool cooling and clean-up system
(20) Reactor service water system

6.6.1.1 Class 2 System Boundary Description

Those portions of the systems listed in Subsection 6.6.1 within the Class 2 boundary, based on Regulatory Guide 1.26, Revision 3, for Quality Group B, are as follows:

(1) Portions of the reactor coolant pressure boundary as defined in Subsection 5.2.4.1.1, but which are excluded from the Class 1 boundary pursuant to Subsection 5.2.4.1.2.

(2) Systems or portions of systems important to safety that are designed for reactor shutdown or residual heat removal.

(3) Portions of the steam systems extending from the outermost containment isolation valve up to but not including the turbine stop and bypass valves and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

(4) Systems or portions of systems that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is normally closed or capable of automatic closure.

(5) Systems or portions of systems important to safety that are designed for (1) emergency core cooling, (2) post accident containment heat removal, or (3) post accident fission product removal.

Items (1) through (5) above describe the Class 2 boundary only and are not related to exemptions

* *Items as used in this Section are products constructed under a Certificate of Authorization (NCA-3120) and material (NCA-1220). See Section III, NCA-1000, footnote 2*

from inservice examinations under ASME Code, Section XI rules. The Class 2 components exempt from inservice examinations are described in ASME Code, Section XI, IWC-1220.

6.6.1.2 Class 3 System Boundary Description

Those portions of the systems listed in Subsection 6.6.1 within the Class 3 boundary, based on Regulatory Guide 1.26, Revision 3, for Quality Group C, are not part of the reactor coolant pressure boundary but are as follows:

(1) Cooling water systems or portions of cooling water systems important to safety that are designed for emergency core cooling, post-accident containment heat removal, post-accident containment atmosphere cleanup, or residual heat removal from the reactor and from the spent fuel storage pool (including

primary and secondary cooling systems). Portions of these systems that are required for their safety functions and that do not operate during any mode of normal operation and cannot be tested adequately, however, are included in Class 2.

(2) Cooling water and seal water systems or portions of these systems important to safety that are designed for functioning of components and systems important to safety.

(3) Systems or portions of systems that are connected to the rector coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves each of which is normally closed or capable of automatic closure.

(4) Systems, other than radioactive waste management systems, not covered by items a, b and c above, that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (ref. Regulatory Guides 1.3 and 1.4), that exceed 0.5 rem to the whole body or its equivalent to any part of the body.

Items (1) through (4) above describe the Class 3 boundary only and are not exemptions from inservice examinations under ASME Code, Section XI rules. The Class 3 components exempt from inservice examinations are described in the ASME Code, Section XI, IWD-1220.

6.6.2 Accessibility

All items within the Class 2 and 3 boundaries are designed to provide access for the examinations required by IWC-2500 and IWD-2500. Responsibility for designing components for accessibility for preservice and inservice inspection is the responsibility of the COL applicant. See Subsection 6.6.9.2 for COL license information requirements.

6.6.2.1 Class 2 RHR Heat Exchangers

The physical arrangement of the residual heat removal (RHR) heat exchangers shall, be conducive to the performance of the required ultrasonic and surface examinations. The RHR heat exchanger nozzle-to-shell welds will be 100% accessible for preservice inspection during fabrication but might have limited areas that will not be accessible from the outer surface for inservice examination techniques. However, the inservice inspection program for the RHR heat exchanger is the responsibility of the COL applicant and any inservice inspection program relief request will be reviewed by the NRC staff based on the Code Edition and Addenda in effect and inservice inspection techniques available at the time of COL application. Removable thermal insulation is provided or those welds and nozzles selected for frequent examination during the inservice inspection. Platforms and ladders are provided as necessary to facilitate examination.

6.6.2.2 Class 2 Piping, Pumps Valves and Supports

Physical arrangement of piping pumps and valves provide personnel access to each weld location for performance of ultrasonic and surface (magnetic particle or liquid penetrant) examinations and sufficient access to supports for performance of visual, VT-3, examination. Working platforms are provided in some areas to facilitate servicing of pumps and valves. Removable thermal insulation is provided on welds and components which require frequent access for examination or are located in high radiation areas. Welds are located to permit ultrasonic examination from at least one side, but where component geometries permit, access from both sides is provided.

Restrictions: For piping systems and portions of piping systems subject to volumetric and surface examination, the following piping designs are not used:

(1) Valve to valve
(2) Valve to reducer
(3) Valve to tee
(4) Elbow to elbow
(5) Elbow to tee
(6) Nozzle to elbow
(7) Reducer to elbow
(8) Tee to tee
(9) Pump to valve

Straight sections of pipe and spool pieces shall be added between fittings. The minimum length of the spool piece has been determined by using the formulate $L = 2T + 6$ inches, where L equals the length of the spool piece (not including weld preparation) and T equals the pipe wall thickness.

6.6.3 Examination Categories and Methods

6.6.3.1 Examination Categories

The examination category of each item is listed in Table 6.6-1 which is provided as an example for the preparation of preservice and inservice program plans. The items are listed by system and line number where applicable. Table 6.6-1 also states the method of examination for each item.

For preservice examination, all of the items selected for inservice examination shall be performed once in accordance with ASME Section XI, IWC-2200 and IWD-2200, with the exception of the examinations specifically excluded by ASME Section XI from preservice requirements, such as the visual VT-2 examinations for Category C-H, D-A, D-B and D-C.

6.6.3.2 Examination Methods

6.6.3.2.1 Visual Examination

Visual Examination Methods, VT-2 and VT-3, shall be conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations shall also meet the requirements of IWA-5240.

described in Section XI, IWC-2412. Except where deferral is permitted by Table IWC-2500-1, the percentages of examinations completed within each period of the interval shall correspond to Table IWC-2412-1. An example of the selection of Code Class 2 items and examinations to be conducted within the 10-year intervals are described in Table 6.6-1.

6.6.4.2 Class 3 Systems

The inservice inspection intervals for Class 3 systems will conform to Inspection Program B as described in Section XI, IWD-2412. Except where deferral is permitted by Table IWD-2500-1, the percentages of examinations completed within each period of the interval shall correspond to Table IWD-2412-1. An example of the selection of Code Class 3 items and examinations to be conducted within the 10-year intervals are described in Table 6.6-1.

6.6.5 Evaluation of Examination Results

Examination results will be evaluated in accordance with ASME Section XI, IWC-3000 for Class 2 components, with repairs based on the requirements of IWA-4000 and IWC-4000. Examination results will be evaluated in accordance with ASME Section XI, IWD-3000 for Class 3 components, with repairs based on the requirements of IWA-4000 and IWD-4000.

6.6.6 System Pressure Tests

6.6.6.1 System Inservice Test

As required by Section XI, IWC-2500 for category C-H and by IWD-2500 for categories D-A, D-B and D-C, a system inservice test shall be performed in accordance with IWC-5221 on Class 2 systems, and IWD-5221 on Class 3 systems, which are required to operate during normal operation. The system inservice test shall include all Class 2 or 3 components and piping within the pressure retaining boundary and shall be performed once during each inspection period as defined in Tables IWC-2412-1 and IWD-2412-1 for Program B. For the purposes of the system inservice test of Class 2 systems, the pressure retaining boundary is defined in Table IWC-2500-1, Category C-H, Note 7. For the purposes of the system inservice test for Class 3 systems, the system boundary is defined in Note 1 of Table IWD-2500-1, for categories D-A, D-B and D-C. The system inservice test shall include a VT-2 examination in accordance with IWA-5240, except that, where portions of a system are subject to system pressure tests associated with two different functions, the VT-2 examination shall only be performed during the test conducted at the higher of the test pressures. The system inservice test will be conducted at approximately the maximum operating pressure and temperature indicated in the applicable process flow diagram for the system as indicated in Table 1.7-1. The system hydrostatic test (Subsection 5.2.4.6.2), when performed is acceptable in lieu of the system inservice test.

6.6.6.2 System Functional Test

As required by Section XI, IWC-2500 for category C-H and by IWD-2500 for categories D-A, D-B and D-C, a system functional test shall be performed in accordance with IWC-5221 on Class 2 systems, and IWD-5221 on Class 3 systems, which are not required to operate during normal operation but for which a periodic system functional test is performed. The system functional test shall include all Class 2 or 3 components and piping within the pressure retaining boundary and shall be performed once during each inspection period as defined in Tables IWC-2412-1 and IWD-2412-1 for Program B. For the purposes of the system functional test of Class 2 systems, the pressure retaining boundary is defined in Table IWC-2500-1, Category C-H, Note 7. For the purposes of the system functional test for Class 3 systems, the system boundary is defined in Note 1 of Table IWD-2500-1, categories D-A, D-B and D-C. The system inservice test shall include a VT-2 examination in accordance with IWA-5240, except that, where portions of a system are subject to system pressure tests associated with two different functions, the VT-2 examination shall only be performed during the test conducted at the higher of the test pressures. The system functional test will be conducted at the nominal operating pressure and temperature indicated in the applicable process flow diagram for the functional test for each system as indicated in Table 1.7-1. The system hydrostatic test (Subsection 5.2.4.6.2), when performed is acceptable in lieu of the system inservice test.

6.6.6.3 Hydrostatic Pressure Tests

As required by Section XI, IWC-2500 for Category B-P, the hydrostatic pressure test shall be performed in accordance with ASME Section IWC-5222 on all Class 2 components and piping within the pressure retaining boundary once during each 10 year inspection interval. For purposes of the hydrostatic pressure test, the pressure retaining boundary is defined in Table IWB-2500-1, Category B-P, Note 1. The system hydrostatic test shall include a VT-2 examination in accordance with IWA-5240. For the purposes of determining the test pressure for the system hydrostatic test in accordance with IWB-5222 (a), the system design pressure as indicated on the applicable piping and instrumentation diagram for the system, as shown in Table 1.7-1, shall be used for P_{sv} in all cases.

6.6.7 Augmented Inservice Inspection

6.6.7.1 High Energy Piping

All high energy piping between the containment isolation valves are subject to the following additional inspection requirements:

All circumferential welds shall be 100 percent volumetrically examined each inspection interval as defined in Subsection 6.6.3.2.3. Further, accessibility, examination requirements and procedures shall be as discussed in Subsections 6.6.2, 6.6.3 and 6.6.5, respectively. Piping in these areas shall be seamless, thereby eliminating all longitudinal welds.

6.6.7.2 Erosion-Corrosion

Piping systems determined to be susceptible to single-phase erosion-corrosion shall be subject to a program of nondestructive examinations to verify the system structural integrity. The examination schedule and examination methods shall be determined in accordance with the NUMARC Program (or another equally effective program) as discussed in Generic Letter 89-08, and applicable rules of Section XI of the ASME Boiler and Pressure Vessel Code.

6.6.8 Code Exemptions

As provided in ASME Section XI, IWC-1220 and IWD-1220, certain portions of Class 2 and 3 systems are exempt from the volumetric and surface and visual examination requirements of IWC-2500 and IWD-2500. These portions of systems are specifically identified in Table 6.6-1

6.6.9 COL License Information

6.6.9.1 PSI and ISI Program Plans

The COL applicant will develop a PSI and ISI program plans as outlined in Section 6.6.

6.6.9.2 Access Requirements

The COL applicant will incorporate plans for NDE during design and construction in order to meet all access requirements of the regulations. (See Subsection 6.6.2)

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 and temperature are shown in Figure 6.7-1.

6.7.3 System Evaluation

Vessels, piping and fittings of the safety portion of the system are designed to Seismic

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.

(11) Testability

The SLCS is capable of being tested by manual initiation of actuated devices during normal operation. In the test mode, demineralized water is circulated in the SLCS loops rather than sodium pentaborate. During reactor shutdown, demineralized water may be injected into the reactor vessel for the injection test mode.

(12) Environmental Considerations

The environmental considerations for the instrument and control portions of the SLCS are the same as for the active mechanical components of the system (Section 3.11). The instrument and control portions of the SLCS are seismically qualified not to fail during, and to remain functional following, a safe shutdown earthquake (SSE) (see Section 3.10 for seismic qualification aspects).

(13) Operational Considerations

The control scheme for the SLCS can be found in the interlock block diagram (Figure 7.4-1). The SLCS is automatically initiated upon receiving an ATWS signal or can be manually initiated in the control room by inserting the key in the A or B keylocking switch and turning it to the "pump run" position. It will take between 50 and 150 minutes to complete the injection and for the storage tank level sensors to indicate that the storage tank is dry. When the injection is completed, the system automatically shutdown on low tank level or may be manually turned off by turning the keylocking switch counterclockwise to the STOP position.

(14) Reactor Operator Information

(a) The following items are located in the control room for operation information:

(1) Analog Indication

(i) Storage tank level and temperature;

(ii) System pressures;

(2) Status Lights

(i) Pump or storage tank outlet valve overload trip or power loss;

(ii) Position of injection line manual service valve;

(iii) Position of storage tank outlet valve and in-test status;

(iv) Position of test tank discharge manual service valve;

(v) SLCS manually out of service;

(vi) Pump auto trip.

(3) Annunciators

The SLCS annunciators indicate:

(i) Manual or automatic out-of-service condition of SLCS A and/or B due to:

- operation of manual out-of-service switch;

- storage tank outlet valve in test status; or

- overload trip or power loss in pump or storage tank outlet valve controls;

(ii) Standby liquid storage tank high or low temperature;

(iii) Standby liquid tank high or low level;

(iv) Standby liquid pump A (B) auto trip.

(b) The following items are located locally at the equipment for operator utilization:

(1) Analog Indication

(i) Storage tank level;

(ii) System pressures;

(iii) Storage tank temperature.

(2) Indicating lamps

(i) Pump status;

(ii) Storage tank operating heater status;

(iii) Storage tank mixing heater status.

(15) Setpoints

The SLCS has setpoints for the various instruments as follows:

(a) The high and low standby liquid temperature switch is set to activate the annunciator at temperatures outside the range allowed for correct chemical balance of the boron concentration.

(b) The high and low standby liquid storage tank level switch is set to activate the annunciator when the level is outside its allowable limits.

(c) The thermostatic controller and operating heater assure the temperature of the liquid is maintained within the range allowed for correct chemical balance of the boron concentration.

The technical specifications for the SLCS are in Chapter 16.

7.4.1.3 Reactor Shutdown Cooling Mode - Instrumentation and Controls

(1) Function

The shutdown cooling mode of the RHR system is used during the normal or emergency reactor shutdown and cooldown. The RHR system P&ID is Figure 5.4-10 and the RHR system IBD is Figure 7.3-4.

The initial phase of the shutdown cooling mode is accomplished following insertion of the control rods and steam blowdown to the main condenser which serves as the heat sink.

Reactor shutdown cooling has three independent loops. Each loop consists of pump, valves, heat exchanger, and instrumentation designed to provide decay heat removal capability for the core. This mode specifically accomplishes the following:

(a) Reactor Shutdown - removes enough residual heat (decay and sensible) from the reactor vessel water to cool it to 125°F within 20 hours after the control rods are inserted, then maintains or reduces this temperature so that the reactor can be refueled and serviced. This mode is manually activated with the reactor pressure below 135 psig, with all three shutdown cooling loops available.

(b) Safe Shutdown (Emergency Shutdown) - brings the reactor to a cold shutdown condition (< 212°F) within 36 hours after control rod insertion. This mode is manually activated with the reactor pressure below 135 psig, with two-out-of-three shutdown cooling loops available.

The RHRS mode can accomplish its design objective by a preferred means by directly extracting reactor vessel water from the vessel shutdown nozzle and routing it to a heat exchanger and back to the vessel. Cooling water is returned to the vessel via the feedwater line (Loop A) and via the core cooling injection nozzles (Loops B and C).

(2) Classification

Electrical components for the reactor shutdown cooling mode of the residual heat removal system are safety-related and are classified as Class 1E.

(3) Power Sources

This system utilizes normal plant power

9.1.5.5 Safety Evaluations

The cranes, hoists, and related lifting devices used for handling heavy loads either satisfy the single-failure guidelines of NUREG-0612, Subsection 5.1.6, including NUREG-0554 or evaluations are made to demonstrate compliance with the recommended guidelines of Section 5.1, including Subsection 5.1.4 and 5.1.5.

The equipment handling components over the fuel pool are designed to meet the single failure proof criteria to satisfy NUREG-0554. Redundant safety interlocks and limit switches are provided to prevent transporting heavy loads other than spent fuel by the refueling bridge crane over any spent fuel that is stored in the spent fuel storage pool.

A transportation routing study will be made of all planned heavy load handling moves to evaluate and minimize safety risks.

Safety evaluation of related light loads and refueling handling tasks in which heavy load equipment is also used are covered in Subsection 9.1.4.3.

9.1.5.6 Inspection and Testing

Heavy load handling equipment is subject to the strict controls of Quality Assurance (QA), incorporating the requirements of Federal Regulation 10CFR50, Appendix B. Components defined as essential to safety have an additional set of engineering specified "Quality Requirements" that identify safety-related features which require specific QA verification of compliance to drawing/specification requirements.

Prior to shipment, every lifting equipment component requiring inspection will be reviewed by QA for compliance and that the required records are available. Qualification load and performance testing, including nondestructive examination (NDE) and dimensional inspection on heavy load handling equipment will be performed prior to QA acceptance. Tests may include load capacity, safety overloads, life cycle, sequence of operations and functional areas.

When equipment is received at the site it will be inspected to ensure no damage has occurred during transit or storage. Prior to use and at periodic intervals each piece of equipment will be tested again to ensure the electrical and/or mechanical functions are operational including visual and, if required, NDE inspection.

Crane inspections and testing will comply with the requirements of ANSI B30.2 and NUREG-0612, Subsection 5.1.1(6).

9.1.5.7 Instrumentation Requirements

The majority of the heavy load handling equipment is manually operated and controlled by the operator's visual observations. This type of operation does not necessitate the need for a dynamic instrumentation system.

Load cells may be installed to provide automatic shutdown whenever threshold limits are exceeded for critical load handling operations to prevent overloading.

9.1.5.8 Operational Responsibilities

Critical heavy load handling in operation of the plant shall include the following documented program for safe administration and safe implementation of operations and control of heavy load handling systems:

(1) Heavy Load Handling System and Equipment Operating Procedures

(2) Heavy Load Handling Equipment Maintenance Procedures and/or Manuals

(3) Heavy Load Handling Equipment Inspection and Test Plans; NDE, Visual, etc.

(4) Heavy Load Handling Safe Load Paths and Routing Plans

(5) QA Program to Monitor and Assure Implementation and Compliance of Heavy Load Handling Operations and Controls

(6) Operator Qualifications, Training and Control Program

See Subsection 9.1.6.7 for COL license information.

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9.1.6 COL License Information

9.1.6.1 New Fuel Storage Racks Criticality Analysis

The COL applicant referencing the ABWR design shall provide the NRC confirmatory critically analysis as required by Subsection 9.1.1.1.1.

9.1.6.2 Dynamic and Impact Analyses of New Fuel Storage Racks

The COL applicant shall provide the NRC confirmatory dynamic and impact analyses of the new fuel storage racks. See Subsection 9.1.1.1.6.

9.1.6.3 Spent Fuel Storage Racks Criticality Analysis

The COL applicant shall provide the NRC confirmatory criticality analysis as required by Subsection 9.1.2.3.1.

9.1.6.4 Spent Fuel Racks Load Drop Analysis

The COL applicant shall provide the NRC confirmatory load drop analysis as required by Subsection 9.1.4.3.

9.1.6.5 New Fuel Inspection Stand Seismic Capability

The COL applicant will install the new fuel inspection stand firmly to the wall so that it does not fall into or dump personnel into the spent fuel pool during an SSE. (See Subsection 9.1.4.2.3.2.)

9.1.6.6 Overhead Load Handling System Information

The COL applicant shall provide a list of all cranes, hoists, and elevators and their lifting capacities including any limit and safety devices required for automatic and manual operation. In addition, for all such equipment, the COL applicant shall provide: (1) heavy load handling system operating and equipment maintenance procedures, (2) heavy load handling system and equipment maintenance procedures and/or manuals, (3) heavy load handling system and equipment inspection and test plans; NDE, visual, etc., (4) heavy load handling safe load paths and routing plans, (5) QA program to monitor and assure implementation and compliance of heavy load handling operations and controls, (6) operator qualifications, training and control program.

9.1.6.7 Spent Fuel Racks Structural Evaluation

The COL applicant will provide the NRC confirmatory structural evaluation of the spent fuel racks as outlined in Subsection 9.1.2.1.3.

9.1.6.8 Spent Fuel Racks Thermal-Hydraulic Analysis

The COL applicant will provide the NRC confirmatory thermal-hydraulic analysis that evaluates the rate of naturally circulated flow and the maximum rack water exit temperature as required by Subsection 9.1.2.1.4.

9.1.6.9 Spent Fuel Firewater Makeup Provedures and Training

The COL applicant will develop detailed procedures and operator training for providing firewater makeup to the spent fuel pool. (See Subsection 9.1.3.3).

9.1.7 References

1. *General Electric Standard Application for Reactor Fuel*, (NEDE-24011-P-A, latest approved revision).

(4) The MUWC system is not safety related.

(5) The condensate storage tank shall have a capacity of 2,110 m^3. This capacity was determined by the capacity required by the uses shown in Table 9.2-2.

(6) All tanks, piping and other equipment shall be made of corrosion-resistant materials.

(7) The HPCF and RCIC instrumentation, which initiates the automatic switchover of HPCF and RCIC suction from the CST header to the suppression pool, shall be designed to safety-grade requirements (including installation with necessary seismic support).

(8) The instrumentation is mounted in a safety grade standpipe located in the reactor building secondary containment. With no condensate flowing, the water level is the same in both the CST and the standpipe. A suitable correction will be made for the effect of flow upon water level in the standpipe.

(9) High water level shall be alarmed both in the radwaste building control room and in the main control room. (See Subsection 11.2.1.2.1)

9.2.9.2 System Description

The MUWC P&ID is shown in Figure 9.2-4. This system includes the following:

(1) A condensate storage tank (CST) is provided. The volume is shown in Table 9.2-3.

(2) The following pumps take suction from the CST:

(a) RCIC pumps

(b) CRD pumps

(c) HPCF pumps

(d) SPCU pumps

(e) MUWC transfer pumps (see Table 9.2-3) (three 550 gpm at 141 psi head)

(3) Water can be sent to the CST from the following sources:

(a) MUWP pumps

(b) CRD system

(c) radwaste disposal system

(d) condensate demineralizer system effluent (main condenser high level relief)

(4) Associated receiving and distribution piping valves, instruments, and controls shall be provided.

(5) Overflow and drain from the CST shall be sent to the radwaste system for treatment.

(6) Any outdoor piping shall be protected from freezing.

(7) All surfaces coming in contact with the condensate shall be made of corrosion-resistant materials.

(8) All of the pumps mentioned in (2) above shall be located at an elevation such that adequate suction head is present at all water levels in the CST.

(9) Instrumentation shall be provided to indicate CST water level in the main control room. High water level shall be alarmed both in the radwaste building control room and in the main control room. See Subsection 11.2.1.2.1.

(10) Potential flooding is discussed in Subsection 3.4. Potential flooding from lines within the reactor building and the control building are evaluated in Subsection 3.4.1.1.1.

9.2.9.3 Safety Evaluation

Operation of the MUWC system is not required to assure any of the following conditions:

(1) integrity of the reactor coolant pressure boundary;

(2) capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) ability to prevent or mitigate the consequences of events that could result in potential offsite exposures.

The MUWC system is not safety-related. However, the systems incorporate features that assure reliable operation over the full range of normal plant operations.

9.2.9.4 Tests and Inspections

The MUWC system is proved operable by its use during normal plant operation. Portions of the system normally closed to flow can be tested to ensure operability and the integrity of the system.

The air-operated isolation valves are capable of being tested to assure their operating integrity by manual actuation of a switch located in the control room and by observation of associated position indication lights.

Flow to the various systems is balanced by means of manual valves at the individual takeoff points. Divisional isolation valves are installed at the primary containment boundaries.

9.2.10 Makeup Water System (Purified) Distribution System

9.2.10.1 Design Bases

(1) The makeup water-purified (MUWP) distribution system shall provide makeup water purified for makeup to the reactor coolant system and plant auxiliary systems.

(2) The MUWP system shall provide purified water to the uses shown in Table 9.2-2.

(3) The MUWP system shall provide water of the quality shown in Table 9.2-2a. If these water quality requirements are not met, the water shall not be used in any safety-related system. The out-of-spec water shall be reprocessed or discharged.

(4) The MUWP system is not safety-related.

(5) All tanks, pumps, piping, and other equipment shall be made of corrosion-resistant materials.

(6) The system shall be designed to prevent any radioactive contamination of the purified water.

(7) The interfaces between the MUWP system and all safety-related systems are located in the control building or reator building which are Seismic Category I, tornado-missile resistant and flood protected structures. The interfaces with safety-related systems are safety-related valves which are part of the safety-related systems. The portions of the MUWP system, which upon their failure during a seismic event can adversly impact structures, systems, or components important to safety, shall be designed to assure their integrity under seismic loading resulting from a safe shutdown earthquake.

(8) Safey-related equipment located by portions of the MUWP system are in Seismic Category I structures and protected from all system impact.

9.2.10.2 System Description

410.50 The MUWP system P&ID is shown in Figure 9.2-5. This system includes the following:

(1) Any purified water storage tank shall be provided outdoors with adequate freeze protection and adequate diking and other means to control spill and leakage.

(2) Two MUWP forwarding pumps shall take suction from any purified water storage tanks. They shall have a capacity of 308 gpm and a discharge head of 114 psi.

(3) Distribution piping, valves, instruments and controls shall be provided.

(4) Any outdoor piping shall be protected from freezing.

(5) All surfaces coming in contact with the purified water shall be made of corrosion-resistant materials.

(6) All pumps shall be located at an elevation such that adequate suction head is present at all levels in a purified water storage tanks.

(7) Instruments shall be provided to indicate purified water storage tank level in the main control room.

(8) Continuous analyzers are located at the demineralized water makeup system and at any demineralized water storage tank. These are supplemented as needed by grab samples. Allowance is made in the water quality specifications for some pickup of carbon dioxide and air in any demineralized water storage tank. The pickup of corrision products should be minimal because the MUWP piping is stainless steel.

(9) Intrusions of radioactivity into the MUWP system from other potentially radioactive systems are prevented by one or more of the following:

(a) check valves in the MUWP lines

(b) air (or syphon) breaks in the MUWP lines

(c) the MUWP system lines are pressurized while the receiving system is at essentially atmospheric pressure.

(d) piping to the user is dead ended.

(10) There are no automatic valves in the MUWP system. During a LOCA, the safety-related systems are isolated from the MUWP system by automatic valves in the safety-related system.

9.2.10.3 Safety Evaluation

Operation of the MUWP system is not required to assure any of the following conditions:

(1) integrity of the reactor coolant pressure boundary;

(2) capability to shut down the reactor and maintain it in a safe shutdown condition; or

the RCW system shall be designed to Seismic Category I and the ASME Code, Section III, Class 3, Quality Assurance B, Quality Group C, IEEE-279 and IEEE-308 requirements.

(4) The RCW system shall be designed to limit leakage to the environment of radioactive contamination that may enter the RCW from the RHR System.

(5) Safety-related portions of the RCW system shall be protected from flooding, spraying, steam impingement, pipe whip, jet forces, missiles, fire, and the effect of failure of any non-Seismic Category I equipment, as required.

(6) The safety-related portion of the RCW system shall be designed to meet the foregoing design bases during a loss of preferred power (LOPP).

(7) The safety-related electric modules and safety-related cables for the RCW system are in the control building which are Seismic Category I, tornado-missile resistant and flood protected structures.

(8) Protection from being impacted adversely by missiles generated by any nonsafety-related components shall be provided as discussed in Subsection 3.5.1.

(9) Protection against high-energy and moderate-energy line failures will be provided in accordance with Section 3.6.

9.2.11.1.2 Power Generation Design Bases

The RCW system shall be designed to cool various plant auxiliaries as required during: (a) normal operation; (b) emergency shutdown; (C) normal shutdown; and (d) testing.

9.2.11.2 System Description

The RCW system distributes cooling water during various operating modes, during shutdown, and during post-LOCA operation. The system removes heat from plant auxiliaries and transfers it to the reactor service water system (Subsection 9.2.15). Figures 9.2-1a through 9.2-1i show the piping and instrumentation diagram. Design characteristics for RCW system components are given in Table 9.2-4d.

The RCW system serves the auxiliary equipment listed in Table 9.2-4a, b, and c.

The reactor decay heat at four hours after shutdown is approximately 126 million BTU/H. Each division of RCW has the design heat removal capability of 102 million BTU/H from the RHR system. If three divisions of RHR/RCW/RSW are used for heat removal, each division must remove one third of the decay heat or 42 million BTU/H. This means that each division will remove 102 minus 42 or 60 million BTU/H of sensible heat, primarily by cooling the reactor water. If only two divisions of RHR/RCW/RSW are used for heat removal, each division must remove one half of the decay heat or 63 million BTU/H. This means the sensible heat removal will be 102 minus 63 or 39 million BTU/H of sensible heat primarily from the reactor water. Of course, the decay heat will decrease with time.

The above analysis shows that there is sufficient heat removal capability to remove not only the decay heat but also sensible heat primarily from the reactor water. If a division of RHR/RCW/RSW is not available or if heat removal capability has been lost due to tube plugging in any of the heat exchangers, only the rate of heat removal will decrease, but, heat will still be removed.

Shutdown cooling times are discussed in Subsection 5.4.7.1.1.7.

The RCW system is designed to perform its required safe reactor shutdown cooling function following a postulated LOCA, assuming a single active failure in any mechanical or electrical system. In order to meet this requirement, the RCW system provides three complete trains, which are mechanically and electrically separated. In case of a failure which disables any of the three divisions, the other two division meet plant safe shutdown requirements, including a LOCA or a loss of offsite power, or both. Each RCW division is supplied electrical power from a different division of the ESF power system.

During normal operation, RCW cooling water flows through all the equipment shown in Table 9.2-4a, b, and c.

During all plant operating modes, an RCW water pump and two heat exchangers are normally operating in each division. Therefore, if a LOCA occurs, the RCW systems required to shut down the plant safely are already in operation. The second pump and the third heat exchanger in each division are put in service if a LOCA occurs.

The nonsafety-related parts of the RCW system are not required for safe shutdown and, hence, are not safety systems. Isolation valves separate the essential subsystems from the nonsafety-related subsystems during a LOCA, in order to assure the integrity and safety functions of the safety related parts of the system. Some nonsafety-related parts of the system are operated during all other modes, including the emergency shutdown following an LOPP, or LOCA as shown in Table 9.2-4a, b, and c.

Surge tank water level is monitored. A level switch detects leakage and isolates the non-essential subsystem, thus assuring continued operability of the safety-related services. Instruments, controls, and isolation valves are located in the safety-related part of the RCW system and designed to safety-grade requirements as stated in design basis (3) of Subsection 9.2.11.1.1.

A dedicated sump and sump pump are provided for each RCW division. Any system leakage or drainage may be collected, sampled and analyzed, and either returned to the RCW system or sent to the liquid radwaste system for treatment or to the HSD sample tank for discharge depending upon the radioactivity and impurities in the water.

9.2.11.3 Safety Evaluation

9.2.11.3.1 Failure Analysis

A system failure analysis of passive and active components of the RCW system is presented in Table 9.2-5. Any of the assumed failures of the RCW system are detected in the control room by variations of and/or alarms from the various system instruments and also from the leak detection system sensing leakage in the ECCS pump and heat exchanger areas.

TABLE 9.2-3

CAPACITY REQUIREMENTS FOR CONDENSATE STORAGE TANK

Function	Capacity Required
dead space-top of pool	7,900g (Note 1)
normal operation variation and receiving volume for plant startup return water	264,000g
minimum storage volume	66,000g
dead space-middle of pool	34,320g (Note 1)
water source for station blackout	150,480g (Note 2)
dead space-bottom of pool	34,320g (Note 1)
Total	557,020g

NOTE

(1) These values are based on a bottom area of 1,400 ft^2.

(2) Water for operation of RCIC is taken from the condensate storage tank and the suppression pool as described in the EPGs of Appendix 18A.

TABLE 9.2-4a

REACTOR BUILDING COOLING WATER DIVISION A

Operating Mode/ Components	Normal Operating Conditions		Shutdown at 4 hours		Shutdown at 20 hours		Hot Standby (no loss of AC)		Hot Standby (loss of AC)		Emergency (LOCA) (Suppression Pool at 97°C	
ESSENTIAL	Heat (Note 1)	Flow	Heat	Flow	Heat	Flow	Heat	Flow	Heat	Flow	Heat	Flow
Emergency Diesel Generator A	—	—	—	—	—	—	—	—	12.5	1,010	12.5	1,010
RHR Heat Exchanger A	—	—	102.4	5,280	32.8	5,280	—	—	24.0	5,280	84.7	5,280
FPC Heat Exchanger A	6.6	1,230	6.6	1,230	6.6	1,230	6.6	1,230	6.6	1,230	9.1	1,230
Others (essential) (Note 2)	3.1	640	3.4	640	3.6	640	3.2	640	3.9	640	4.0	640
NON-ESSENTIAL												
RWCU Heat Exchanger	19.1	700	—	700	—	700	19.1	700	19.7	700	—	—
Inside Drywell (Note 3)	5.7	1,410	5.7	1,410	5.7	1,410	5.7	1,410	3.2	1,410	—	—
Others (non-essential) (Note 4)	2.5	440	2.5	440	2.5	440	2.5	440	0.8	260	0.7	260
Total Load	37.0	4,420	120.6	9,700	51.2	9,700	37.1	4,420	70.7	10,530	111.0	8,420

NOTES:

(1) Heat x 10^6 Btu/h; flow x g/m, sums may not be equal due to rounding.

(2) HECW refrigerator, room coolers (FPC pump, RHR, RCIC, SGTS, CAMS), RHR motor and seal coolers.

(3) Drywell (A & C) and RIP coolers.

(4) Instruments and service air coolers; RWCU pump cooler, CRD pump oil, and RIP Mg sets.

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SECTION 9.3

CONTENTS (Continued)

Barriers have been considered to assure SLCS protection from pipe break (Section 3.6).

It should be noted that the SLCS is not required to provide a safety function during any postulated pipe break events. This system is only required under an extremely low probability event, where all of the control rods are assumed to be inoperable while the reactor is at normal full power operation. Therefore, the protection provided is considered over and above that required to meet the intent of ASB 3-1 and MEB 3-1.

This system is used in special plant capability demonstration events cited in Appendix A of Chapter 15; specifically, Events 54 and 56, which are extremely low probability nondesign basis postulated incidents. The analyses given there are to demonstrate additional plant safety considerations far beyond reasonable and conservative assumptions.

9.3.5.4 Testing and Inspection Requirements

Operational testing of the SLCS is performed in at least two parts to avoid inadvertently injecting boron into the reactor.

With the valves to the reactor and from the storage tank closed, and the valves to and from the test tank opened, condensate water in the test tank can be recirculated by locally starting either pump.

During a refueling or maintenance outage, the injection portion of the system can be functionally tested by valving the suction line to the test tank and actuating the system from the control room. System operation is indicated in the control room.

After functional tests, all the valves must be returned to their normal positions as indicated in Figure 9.3-1.

After closing a local locked-open valve to the reactor, leakage through the injection valves can be detected by opening valves at a test connection in the line between the drywell check valves. Position indicator lights in the control room indicate that the local valve is closed for test or open and ready for operation. Leakage from the reactor through the first check valve can be detected by opening the same test connection in the line between the check valves when the reactor is pressurized.

The test tank contains condensate water for approximately 3 minutes of pump operation. Condensate water from the makeup system or the condensate storage system is available for refilling or flushing the system.

Should the boron solution ever be injected into the reactor, either intentionally or inadvertently, then after making certain that the normal reactivity controls will keep the reactor subcritical, the boron is removed from the reactor coolant system by flushing for gross dilution followed by operating the reactor cleanup system. There is practically no effect on reactor operations when the boron concentration has been reduced below approximately 50 ppm.

The concentration of the sodium pentaborate in the solution tank is determined periodically by chemical analysis.

Electrical supplies and relief valves are also subjected to periodic testing.

The SLCS preoperational test is described in Subsection 14.2.12.

See Subsection 9.3.12.2 for COL license information pertaining to SLCS storage tank discharge valve reliability.

9.3.5.5 Instrumentation Requirements

The instrumentation and control system for the SLCS is designed to allow the injection of liquid poison into the reactor and the maintenance of the liquid poison solution well above the saturation temperature. A further discussion of the SLCS instrumentation may be found in Section 7.4.

9.3.10.3 Safety Evaluation

The oxygen injection system is not required to assure any of the following conditions.

(1) integrity of the reactor coolant pressure boundary;

(2) capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) ability to prevent or mitigate the consequences of events which could result in potential offsite exposures.

Consequently, the injection system itself is not safety-related. The high pressure oxygen storage bottles are located in an area in which large amounts of burnable materials are not present. Usual safe practices for handling high pressure gases are followed.

281.17 #### 9.3.10.4 Tests and Inspections

The oxygen injection system is proved operable by its use during normal operation. The system valves may be tested to ensure operability from the main control room.

9.3.10.5 Instrumentation Application

The oxygen storage bottles have pressure gages which will indicate to the operators when a new bottle is required. A flow element will indicate the oxygen gas flow rate at all times. The gas flow regulating valves will have position indication in the main control room.

The oxygen monitors are discussed in Subsection 9.3.2.

9.3.11 Zinc Injection System

9.3.11.1 Design Bases

281.15 Provisions are made to permit installation of a system for adding a zinc solution to the feedwater. Piping connections (Figure 10.4-7) for a bypass loop around the feedwater pumps and space (Figure 1.2-25) for the zinc addition equipment are provided. If experience shows it to be necessary, a zinc injection system may be added later in plant life. The amount of zinc in the reactor water will be less than 10 ppb during nornal opereation.

9.3.11.2 Safety Evaluation

The injection system is not necessary to ensure:

(1) the integrity of the reactor coolant pressure boundary;

(2) the capability to shut down the reactor; or

(3) the capability to prevent or mitigate the consequences of events which could result in potential offsite exposures.

9.3.11.3 Test and Inspections

The zinc injection system is proved operable after installation. Zinc injection would not be performed when the plant is in cold shutdown. During these periods, the system could have maintenance or testing performed.

9.3.11.4 Instrumentation

Instrumentation would be provided so that the injection of zinc solution would be stopped automatically if feedwater flow stops. The zinc injection rate would be manually adjusted based on zinc concentration data in the reactor water.

9.3.12 COL License Information

9.3.12.1 Radioactive Drain Transfer System Collection Piping

The COL applicant will provide equipment and floor drain piping P&IDs as a part of the radioactive drain transfer system. This piping will be provided with the following features:

(1) These piping systems shall be non-nuclear safety class and quality Group D with the exception of any containment penetrations or piping within the drywell which shall be Seismic Category I and quality Group B.

(2) The floor drain piping system shall be arranged with a separate piping system for each quadrant. The piping shall be arranged so that flooding or backflow in one quadrant cannot adversely affect the other quadrants.

(3) There shall be no interconnection between any portion of the radioactive drain transfer system and any non-radioactive waste system.

(4) Effluent from non-radioactive systems shall be monitored prior to discharge to assure that there are no unacceptable discharges.

See Subsection 9.3.8.2 for information concerning the remainder of the radioactive drain transfer system.

9.3.12.2 Storage Tank Discharge Valve Reliability

The COL applicant will confirm that the SLCS storage tank discharge valves will have adequate reliability requirements and that the valves be incorporated into the Operational Reliability Assurance Program. (See Subsection 9.3.5.4)

Table 9.4-3

HVAC FLOW RATES

(Response to Question 430.243)

Essential HVAC System	Flow Rates (cmh)
RB Electrical HVAC Division A	30,000
RB Electrical HVAC Division B	30,000
RB Electrical HVAC Division C	30,000
DG HVAC Division A	160,000
DG HVAC Division B	160,000
DG HVAC Division C	160,000
CB Electrical HVAC Division A	35,000
CB Electrical HVAC Division B	35,000
CB Electrical HVAC Division C	35,000
MCR HVAC Division B	80,000
MCR HVAC Division C	80,000
Non-Essential HVAC Systems	**Flow Rate (cmh)**
RB Secondary Containment HVAC	170,000
TB Ventilation System	341,500
RIP Panel Room HVAC Division A	57,500
RIP Panel Room HVAC Division B	57,500

Table 9.4-4

HVAC SYSTEM COMPONENT DESCRIPTIONS
(Response to Question 430.243)

ESSENTIAL EQUIPMENT LIST

Heating/Cooling Coils	Quantity	Cooling (Btu/hr)	Heating (Btu/h
RB Electrical HVAC Division A	1	640,0 00	No Coil Required
RB Electrical HVAC Division B	1	640,000	No Coil Required
RB Electrical HVAC Division C	1	640,000	No Coil Required
CB Electrical HVAC Division A	1	840,000	No Coil Required
CB Electrical HVAC Division B	1	840,000	No Coil Required
CB Electrical HVAC Division C	1	840,000	No Coil Required
MCR HVAC Division B	1	628,000	76,300
MCR HVAC Division C	1	628,000	76,300

Fans	Quantity	Capacity (cmh)	Rated Power (kw)
RB Electrical Div A Recirculation Fans	2 (1 on standby)	30,000	75
RB Electrical Div B Recirculation Fans	2 (1 on standby)	30,000	75
RB Electrical Div C Recirculation Fans	2 (1 on standby)	30,000	75
RB Electrical Div A Exhaust Fans	2 (1 on standby)	6,000	4
RB Electrical Div B Exhaust Fans	2 (1 on standby)	6,000	4
RB Electrical Div C Exhaust Fans	2 (1 on standby)	6,000	4
DG Div A Exhaust Fans	2	80,000	22
DG Div B Exhaust Fans	2	80,000	22
DG Div C Exhaust Fans	2	80,000	22
CB Electrical Div A Recirculation Fans	2 (1 on standby)	35,000	75
CB Electrical Div B Recirculation Fans	2 (1 on standby)	35,000	75
CB Electrical Div C Recirculation Fans	2 (1 on standby)	35,000	75
CB Electrical Div A Exhaust Fans	2 (1 on standby)	4,000	4
CB Electrical Div B Exhaust Fans	2 (1 on standby)	4,000	4
CB Electrical Div C Exhaust Fans	2 (1 on standby)	4,000	4
MCR Div A Recirculating Fans	2 (1 on standby)	80,000	22
MCR Div C Recirculating Fans	2 (1 on standby)	80,000	22
MCR Div A Exhaust Fans	2 (1 on standby)	5,000	3
MCR Div C Exhaust Fans	2 (1 on standby)	5,000	3

The cable for the maintenance communication facility is unshielded with a flame and heat resistance PVC sheath and cross-linked polyethylene insulation. The cables are routed in existing control voltage level cable trays where available. The wiring used for this system is color coded and the color of the sheath is black.

9.5.2.3 System Operation

The telephonic communication systems are designed to assist the plant personnel during preoperational, start-up, testing, maintenance and limited emergency conditions. The system provides easily accessible means of communications between various intraplant locations and simultaneous broadcasting in those locations.

The various equipment involved in system operation is designed to function in the environment where is located. The power supply for the system is derived from the dedicated batteries, thus providing a reliable source of power and the communication system for up to 10 hours in the event of a loss of plant power supply.

9.5.2.4 Safety Evaluation

The communication system has no safety-related function as discussed in Section 3.2. However, see Subsection 9.5.13.2 for COL license information pertaining to use of the system in emergencies.

9.5.2.5 Inspection and testing Requirements

The communication systems are conventional and have a history of successful operation. Routine use of parts of the system during normal operation ensures availability. Measurements or tests required to guard against long-term deterioration shall be performed on a periodic basis. See Subsection 9.5.13.3 for COL license information pertaining to communication equipment maintenance and testing procedures.

9.5.2.6 Portable and Fixed Emergency Communication Systems

The portable radio communication system, and the fixed emergency communication system (independent of the normal plant communication system) are out of ABWR standard plant design scope. The COL applicants design shall comply with the BTP CMEB 9.5-1, position C.5.g(3) and (4). The COL applicant will supplement this subsection accordingly as applicable. See Subsection 9.5.13.14 for COL license information.

9.5.4 Diesel-Generator Fuel Oil Storage and Transfer System

9.5.4.1 Design Bases

9.5.4.1.1 Safety Design Bases

(1) Each engine is supplied by a separate diesel-generator fuel oil storage and transfer system. All fuel oil transfer equipment is designed, fabricated and qualified to Seismic Category I requirements. Failure of any one component could result in loss of fuel supply to only one diesel-generator.

(2) Minimum onsite storage capacity of the system is sufficient for operating each diesel-generator for a minimum of seven days while supplying post-LOCA maximum load demands.

(3) Design and construction of the diesel-generator fuel oil storage and transfer syst·m up to the first connection on the engine skid conforms to the IEEE Criteria for Class 1E Power Systems for Nuclear Power Generating Stations (IEEE-308); and ASME Code, Section III, Class 3, Quality Group C. Miscellaneous equipment conforms to applicable standards of NEMA, DEMA, ASTM, IEEE, ANSI, API, NFPA. ANSI Standard N195 "Fuel Oil Systems for Standby Diesel Generators" is applied.

(4) The diesel-generator fuel oil storage and transfer is of Seismic Category I design. In addition, the system is protected from damage by flying debris carried by tornados and hurricanes, from external floods, and other environmental factors. The fill connection is located at grade elevation. The vent and sample connection are located a little above the grade elevation, and are capped and locked to prevent entry to moisture. Each vent is of fireproof goose necked line with fine mesh screen to prevent access.

Damage to these lines would have no adverse safety consequences since they are not part of the fuel path from the storage tank to the diesel. In addition each diesel has its own day tank, which is located inside the reactor building. This provides another level of protected fuel supply for each diesel generator. Also, there are three independent diesel generator systems, any one of which can safely shut down the plant. Missile damage of such lines for more than one division is highly unlikely because each division is located in separated areas of the plant.

(5) System components selected to be corrosion resistant.

(6) System design also considers positive protection from damage caused by turbine missiles.

9.5.4.1.2 Power Generation Design Bases

The diesel-generator systems are standby power supply systems. The diesel fuel oil storage and transfer systems are capable of supporting the instant start requirements of the diesel-generators.

9.5.4.2 System Description

Although specific suppliers may differ in the final design, a typical P&ID is provided as Figure 9.5-6 (See 9.5.13.5).

There are three diesel-generators, DG-A, DG-B and DG-C, each one housed in its separate area within the reactor building. The units are identical and are held in reserve to furnish standby AC power in case of an emergency.

The diesel-generators DG-A and DG-C are located north side of the reactor building, but are separated from each other. The diesel-generator DG-B is located in the south side of the reactor building.

The diesel-generator fuel oil system for each engine consists of a fuel oil day tank, fuel oil transfer pump, suction strainer, duplex filter, instrumentation and controls, and the necessary interconnecting pipe and fittings. A bleed line returns excess fuel oil from the day tank for recirculation to the yard storage tank. Day tank elevation is such that the fuel oil pump operates with flooded suction. The bottom of the day tank shall never be lower than the pump suction centerline.

Each diesel-generator set has its own day tank, which holds a capacity of fuel oil sufficient to operate its corresponding diesel-generator set for a minimum of eight hours at full load. Fuel oil is supplied by transfer pumps to each day tank from the yard storage system.

A set of transfer pumps may be operated with manual control switches; however, they are normally operated automatically by level switches on the day tanks. A "low" level switch starts the first transfer pump, a "low-low" level switch starts the standby transfer pump and a "high" level switch stops both pumps.

An engine-driven fuel oil pump supplies fuel from the day tank to the diesel engine fuel manifold. Fuel oil transfer system piping is ASME, Section III, Class 3, Seismic Category I. Connections for an optional motor driven fuel oil booster pump are also provided.

9.5.4.3 Safety Evaluation

The overall diesel-generator fuel oil storage and transfer system is designed so that failure of any one component may result in the loss of fuel supply to only one diesel-generator. The loss of one diesel-generator does not preclude adequate core cooling under accident conditions.

Day tank fuel oil feed to the fuel pump is by gravity. There are no powered components to fail. A suction strainer prevents foreign matter from entering the pump and causing malfunction. The system is safety related and all piping and components up to the engine skid connection are designed and constructed in accordance with the ASME Code Section III, Class 3, and Seismic Category I requirements.

The diesel-generator fuel oil storage and transfer system is designed to withstand the adverse loadings imposed by earthquakes, tornadoes and winds. Earthquake protection is provided by the Seismic Category I construction. Tornado and wind protection is provided by locating system components either underground or within the reactor building. All underground piping is covered with protective coating and wrapping to guard against corrosion. The system will be provided with a protection against external and internal corrosion. The buried portion of the tanks and piping will be provided with waterproof protective coating and an impressed current-type cathodic protection, to control the external corrosion of underground piping system. The impressed current-type cathodic protection system will be designed to prevent the ignition of combustible vapors or fuel oil present in the fuel oil system, in accordance with Regulatory Guide 1.137, Paragraph C.1.g.

All storage and day tanks are located at a sufficient distance away from the plant control room to preclude any danger to control room personnel or equipment resulting from an oil tank explosion and/or fire. The fuel oil day tank is located in a separate room with 3-hr fire rated concrete walls. The quality of the fuel oil used for diesel engine will be ensured per Appendix B, of ANSI N195. The fuel oil stored will meet the requirements of the ASTM D975 "standard specifi- cation for diesel fuel oils" and the require- ments of the diesel engine manufacturer. Fuel oil not meeting these requirements will be replaced within a one-week period.

9.5.4.4 Tests and Inspections

The diesel generator fuel oil storage and transfer system is designed to permit periodic testing and inspection.

Diesel generator fuel oil storage and transfer system operability is demonstrated during the regularly scheduled operational tests of the diesel generators. Test frequency is given in Chapter 16. Periodic testing of instruments, controls, sensors and alarms is necessary to assure reliable operation.

ASTM standard fuel sample tests are conducted at regular intervals to ensure compliance with fuel composition limits recommended by the diesel engine manufacturer. The "Standard Specification for Diesel Fuel Oils ANSI/ASTM D975" is the governing specification.

Fuel oil may be stored by a minimum of six months without deterioration.

Each fuel oil storage tank will be emptied and accumulated sediments be removed every 10 years to perform the ASME Section XI, Article IWD-2000 examination requirements.

Periodic surveillance of cathodic protection for underground piping system will be provided, not to exceed a 12 month interval, to make sure that adequate protection exists.

New fuel oil will be tested for specific gravity, cloud point and viscosity and visually inspected for appearance prior to addition to ensure that the limits of ASTM D975 are not exceeded. Analysis of other properties of the fuel oil will be completed within two weeks of the fuel transfer.

9.5.4.5 Instrumentation Application

Fuel supply level in the storage and day tanks is indicated both locally and in the main control room. Also, alarms on the local diesel-generator panel annunciate low level and high level in the day tanks. The setting of the low level alarm shall provide fuel at least 60 minutes of DG operation at 100 percent load with 10 percent margin between the alarm and the suction line inlet. A group repeat trouble alarm is also provided in the main control room. Level switches in the day tank signal automatic start and stop of the fuel oil transfer pump.

9.5.8 Diesel-Generator Combustion Air Intake and Exhaust System

9.5.8.1 Design Bases

All components of the diesel-generator combustion air intake and exhaust system shall be designed and qualified to Seismic Category I requirements. All piping shall be designed in accordance with ASME Code, Section III, Class 3, Quality Group C. Failure of the intake and exhaust system in any one diesel generator shall not compromise the readiness or operability of any other diesel generator. Except for the exhaust silencers, the system shall be housed in a Seismic Category I and tornado missile-protective structure. The system shall also be protected from flooding and the effects of pipe breaks.

The exhaust silencers for the diesel generators shall be seismically mounted and bolted down in the horizontal position such that the likelihood of their sustaining significant damage, or becoming missiles during a tornado or hurricane event is extremely remote. However, the probability of the silencers themselves being damaged due to externally generated missiles is acceptable. This is because the silencer can be lost without affecting the operation of the diesel unless debris from the damaged silencer clogs the exhaust pipe. In this highly unlikely scenario, the diesel would be assumed lost and the plant shutdown could still be accomplished with either of the remaining two diesels.

The design basis for the diesel generator combustion air intake and exhaust system, regarding protection from the effects of contaminating substances related to the facility site, systems, and equipment is a follows:

(1) There are no contaminating substances available within the ABWR buildings to the combustion air intake in quantities which could degrade the diesel engine performance.

(2) Restriction contaminating substances from the plant site, which may be available to the combustion air intake is COL license information (See Subsection 9.5.13.1).

(3) The diesel engine exhaust system is capable of exhausting the products of combustion to the atmosphere.

9.5.8.2 System Description

9.5.8.2.1 (Deleted)

Although specific suppliers may differ in the final design, a typical P&ID is provided as Figure 9.5-6. See Subsection 9.5.13.5 for COL license information.

Each engine DG-A, DG-B and DG-C takes combustion air from its own inlet air cubical above the diesel generator room. The air is filtered as it enters the cubical through the outside wall above. See Section 9.4.5.5 for a description of the diesel-generator HVAC system.

Engine exhaust gases are ducted out of the building. The exhaust is ducted up through the reactor building to the roof where the silencers are mounted. Each engine has its own exhaust system.

In order to protect the crank case from accumulation of fumes and possible consequent fire and explosion, the crank case is kept at negative pressure by vacuum blowers. The gases are exhausted to an outside vent via a six-inch pipe which passes through the reactor building wall (see Figure 9.5-6). Pressure sensors will detect unacceptable high pressure conditions in the crank case, and will annunciate this condition to the operator. This signal will also shut down the diesel unless a LOCA signal is present (see Table 8.3-11).

9.5.8.3 Safety Evaluation

Both the intake and exhaust system components of all three engines are completely separate and independent. Failure in any one system has no effect on the readiness and/or operability of either of the others.

For all systems, the air intake is through the wall of the reactor building at approximately 11.5m above grade, while the exhaust gases are released to the atmosphere on the Reactor Building roof at approximately 26m above grade. Therefore, the possibility of products of combustion diluting the oxygen content of the intake air is essentially nil. Also, other gases will not be stored close enough to the diesel air intake that their release to the atmosphere would dilute the intake air and affect the performance of the diesel generators.

See the Reactor Building arrangement drawings in Section 1.2 for intake and exhaust locations, Subsection 3.8.4 for design of the Reactor building, Section 3.4 for flood protection and Section 3.6 for pipe failure protection.

The combustion air intakes are protected by grills through which the air passes vertically upward. This minimizes plugging of the filters by gross debris picked up by events such as a tornado or a hurricane. Particulate matter small enough to pass through the grill can cause plugging of the inlet filters. To monitor this condition, a differential pressure gauge is installed across each filter.

The effect of a local decrease in barometric pressure (e.g., due to a tornado or hurricane) are largely negated by the engine turbochargers.

All intake and exhaust ducting, as well as the ducting hangers, are designed and qualified to Seismic Category I requirements. Further the ducting conforms to ASME Section III, Class 3, Quality Group C requirements.

9.5.8.4 Inspection and Testing Requirements

Visual inspection of the diesel-generator combustion air intake and exhaust system may be carried out concurrently with regularly scheduled diesel-generator testing and inspection. Integrity of the ducting and joints, filter condition, intake and exhaust silencer condition inspection are included in the diesel-generator inspection procedure.

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10.4.5 Circulating Water System

The circulating water system (CWS) provides cooling water for removal of the power cycle waste heat from the main condensers and transfers this heat to the ultimate heat sink.

10.4.5.1 Design Bases

10.4.5.1.1 Safety Design Bases

The CWS does not serve or support any safety function and has no safety design basis.

10.4.5.1.2 Power Generation Design Bases

Power Generation Design Basis One - The CWS supplies cooling water at a sufficient flow rate to condense the steam in the condenser, as required for optimum heat cycle efficiency.

Power Generation Design Basis Two - The CWS is automatically isolated in the event of gross leakage into the condenser pit to prevent flooding of the turbine building.

10.4.5.2 Description

10.4.5.2.1 General Description

The circulating water system is illustrated in Figure 10.4-3. The circulating water system consists of the following components: screen house and intake screens; pumps; condenser water boxes and piping and valves; tube side of the main condenser; water box fill and drain subsystem; and related support facilities such as for system water treatment and general maintenance.

The ultimate heat sink is designed to maintain the temperature of the water entering the circulation water system within the range of $32^{O}F$ to $100^{O}F$. The circulating water system is designed to deliver water to the main condenser within a temperature range of $40^{O}F$ to $100^{O}F$. The $40^{O}F$ minimum temperature is maintained, when needed, by warm water recirculation.

The cooling water is circulated by three fixed speed motor driven pumps.

The pumps are arranged in parallel and discharge into a common header. The discharge of each pump is fitted with a butterfly valve. This arrangement permits isolation and maintenance of any one pump while the others remain in operation.

The circulating water system and condenser is designed to permit isolation of each set of the three series connected tube bundles to permit repair of leaks and cleaning of water boxes while operating at reduced power.

The circulating water system includes water box vents to help fill the condenser water boxes during startup and removes accumulated air and other gases from the water boxes during normal operation.

A chemical additive subsystem is also provided to prevent the accumulation of biological growth and chemical deposits within the wetted surfaces of the system.

10.4.5.2.2 Component Description

Codes and standards applicable to the CWS are listed in Section 3.2. The system is designed and constructed in accordance with quality group D specifications Table 10.4-3 provides design parameters for the major components of the circulating water system.

10.4.5.2.3 System Operation

The CWS operates continuously during power generation including startup and shutdown. Pumps and condenser isolation valve actuation is controlled by locally mounted hand switches or by remote manual switches located in the main control room.

The circulating water pumps are tripped and the pump and condenser valves are closed in the event of a system isolation signal from the condenser pit high-high level switches. A condenser pit high level alarm is provided in the control room. The pit water level trip is set high enough to prevent inadvertent plant trips from unrelated failures, such as a sump overflow.

Draining of any set of series connected condenser water boxes is initiated by closing the associated condenser isolation valves and opening the drain connection and water box vent valve. When the suction standpipe of the condenser drain pump is filled, the pump is manually started. A low level switch is provided in the standpipe, on the suction side of the drain pump. This switch will automatically stop the pump in the event of low water level in the standpipe to protect the pump from excessive cavitation.

10.4.5.3 Evaluation

The CWS is not a safey-related system; however, a flooding analysis of the turbine building is performed on the CWS postulating a complete rupture of a single expansion joint. The analysis assumes that the flow into the condenser pit comes from both the upstream and downstream side of the break and, for conservatism, it assumes that one system isolation valve does not fully close.

Based on the above conservative assumptions, the CWS and related facilities are designed such that the selected combination of plant physical arrangement and system protective features ensures that all credible potential circulating water spills inside the turbine building remain confined inside the condenser pit. Further, plant safety is ensured in case of multiple CWS failures or other negligible probability CWS related events by the plant safety related general flooding protection provisions that are discussed in Section 3.4.

10.4.5.4 Tests and Inspections

The CWS and related systems and facilities are tested and checked for leakage integrity prior to initial plant startup and, as may be appropriate, following major maintenance and inspection.

All active and selected passive components of the circulating water system are accessible for inspection and maintenance/testing during normal power station operation.

10.4.5.5 Instrumentation Applications

Temperature monitors are provided upstream and downstream of each condenser shell section.

Indication is provided in the control room to identify open and closed positions of motor-operated butterfly valves in the CWS piping.

All major circulating water system valves which control the flow path can be operated by local controls or by remote manual switches located on the main control board. The pump discharge isolation valves are interlocked with the circulating water pumps so that when a pump is started, its discharge valve will be opening while the pump is coming up to speed, thus assuring there is water flow through the pump. When the pump is stopped, the discharge valve closes automatically to prevent or minimize backward rotation of the pump and motor.

Level switches monitor water level in the condenser discharge water boxes and provide a permissive for starting the circulating water pumps. These level switches ensure that the supply piping and the condenser are full of water prior to circulating water pump startup thus preventing water pressure surges from damaging the supply piping or the condenser.

To satisfy the bearing lubricating water and shaft sealing water interlocks during startup, the circulating water pump bearing lubricating and shaft seal flow switches, located in the lubricating seal water supply lines, must sense a minimum flow to provide pump start permissive.

Monitoring the performance of the circulating water system is accomplished by differential pressure transducers across each half of the condenser with remote differential pressure indicators located in the main control room. Thermal element signals from the supply and discharge sides of the condenser are transmitted to the plant computer for recording, display and condenser performance calculations.

To prevent icing and freeze up when the ambient temperature of the ultimate heat sink falls below 32°F, warm water from the discharge side of the condenser is recirculated back to the screen house intake. Thermal elements, located in each condenser supply line and monitored in the main control room, are utilized in throttling the warm water recirculation valve, which maintains the minimum inlet temperature of approximately 40°F.

10.4.5.6 Flood Protection

A circulating water system pipe, waterbox, or expansion joint failure, if not detected and isolated, would cause internal turbine building flooding up to slightly over grade level, with excess flood waters potentially spilling over on site. If a failure occurred within the condensate system (condenser shell side), the resulting flood level would be less than grade level due to the relatively small hotwell water inventory relative to the condenser pit capacity. In either event the flooding of the turbine building would not affect safety related equipment since no such equipment is located inside the turbine building and all plant safety related facilities are protected against site surface water intrusion.

10.4.6 Condensate Cleanup System

The condensate cleanup system (CCS) purifies and treats the condensate as required to maintain reactor feedwater purity, using filtration to remove suspended solids including corrosion products, ion exchange to remove dissolved solids from condenser leakage and other impurities, and water treatment additions to minimize corrosion/erosion product releases in the power cycle.

operation with hydrogen water chemistry, the recommended design basis N-16 concentration in steam is 4 times the value for natural water chemistry or 200 μCi/gm.

11.1.2.2 Noncoolant Activation Products

Radionuclides are produced in the coolant by neutron activation of circulating impurities and by corrosion of irradiated system materials. Typical reactor water concentrations for the principal activation products are contained in Reference 1. The values of Reference 1 were adjusted to ABWR conditions by using the procedure described in subsection 11.1.3 and appropriate data from Tables 11.1-6 and 11.1-7. These results were arbitrarily increased by the same factor used for the design basis radioiodine concentrations (6.7) to obtain the conservative design basis reactor water concentrations shown in Table 11.1-5. The steam carryover ratio for these isotopes is estimated to be less than 0.001. A factor of 0.001 is applied to the Table 11.1-5 values to obtain the design basis concentrations in steam.

11.1.2.3 Tritium

Tritium is produced by activation of naturally occurring deuterium in the primary coolant and, to a lesser extent, as a fission product in the fuel (Reference 2). The tritium is primarily present as tritiated oxide (HTO). Since tritium has a long half-life (12 years) and will not be affected by cleanup processes in the system, the concentration will be controlled by the rate of loss of water from the system by evaporation or leakage. All plant process water and steam will have a common tritium concentration. The concentration reached will depend on the actual water loss rate; however, References 1 and 3 both specify a typical concentration of 0.01 μCi/gm which is stated in Reference 3 to be based on BWR experience adjusted to account for liquid recycle. This value is taken to be applicable for ABWR.

11.1.2.4 Argon-41

Argon-41 is produced in the reactor coolant as a consequence of neutron activation of naturally occurring Argon-40 in air which is entrained in the feedwater. The Argon-41 gas is carried out of the vessel with the steam and stripped from the system with the non-condensables in the main condenser. Observed Argon-41 levels are highly variable due to the variability in air in-leakage rates into the system. Reference 3 specifies an Argon-41 release rate from the vessel of 40 μCi/sec for a 3400Mw Reference BWR. This value bounded the available experimental data base. Based on adjusting to the ABWR thermal power, a design basis Argon-41 release rate of 46 μCi/sec is specified for the ABWR.

11.1.3 Radionuclide Concentration Adjustment

In order to determine the estimated concentrations of radionuclides in the groups classified as iodines, other non-volatile fission products, and non-coolant activation products using the ANSI/ANS-18.1 Source Term Standard (Reference 1) it is necessary to apply appropriate adjustment factors to the Reference Plant concentrations provided in the Standard.

Equilibrium concentrations in reactor water are assumed to satisfy the relationship:

$$C = \frac{s}{M(\lambda + R)} \qquad (11.1\text{-}1)$$

where:

- C = radionuclide concentration
- s = radionuclide input rate to coolant
- M = reactor water mass
- λ = radionuclide decay constant
- R = sum of removal rates of the radionuclide from the system.

Consequently, if the radionuclide input rate is taken to depend primarily on the reactor thermal power, the adjustment factors to be applied to the Reference Plant reactor water concentrations are given by:

$$\text{Adjustment Factor} = \frac{P\, M_r\, (\lambda + R_r)}{P_r\, M\, (\lambda + R)} \qquad (11.1\text{-}2)$$

where the subscript "r" refers to the Reference Plant, P is the reactor thermal power and M, λ, and R are as defined above. The removal rate from the system is the sum of the removal rates due to the reactor water cleanup system and the condensate demineralizer and is given by:

$$R = \frac{F_c E_c + F_s A B E_s}{M} \qquad (11.1\text{-}3)$$

where:

F_c = cleanup system flow rate

E_c = fraction of radionuclide removed in cleanup demineralizer

F_s = steam flow rate

A^s = ratio of radionuclide concentration in steam to concentration in water (carryover ratio)

B = fraction of radionuclide in steam which is circulated through the condensate demineralizer

E_s = fraction of radionuclide removed in condensate demineralizer.

The Reference Plant and ABWR plant parameters are shown in Table 11.1-6 and the nuclide-dependent removal rate parameters used for ABWR are shown in Table 11.1-7. The nuclide-dependent parameters are the same as those used for the Reference Plant except for the fraction circulated through the condensate demineralizer. The Reference Plant data is given for a plant without pumped-forward heater drains so that the fraction of condensate treated by the demineralizer is 1.0. In ABWR, which has pumped forward drains, the radionuclides are assumed to preferentially go with the pumped-forward flow (Reference 3). The effective treatment fractions are .18 for iodines and .01 for other fission products and non-coolant activation products (Reference 3).

11.1.4 Fuel Fission Production Inventory

Fuel fission product inventory information is used in establishing fission product source terms for accident analysis and is discussed in Chapter 15.

11.1.5 Process Leakage Sources

Process leakage results in potential release of noble gases and other volatile fission products via ventilation systems. Liquid from process leaks is collected and routed to the liquid-solid radwaste system. With the effective process offgas treatment systems now in use, the ventilation releases are relatively significant contributions to total plant releases.

Leakage of fluids from the process system results in the release of radionuclides into plant buildings. In general, the noble radiogases will remain airborne and will be released to the atmosphere with little delay via the building ventilation exhaust ducts. Other radionuclides will partition between air and water and may plate-out on metal surfaces, concrete, and paint. Radioiodines are found in ventilation air as methyl iodide and as inorganic iodine (particulate, elemental, and hypoiodous acid forms).

As a consequence of normal steam and water leakage in to the drywell, equilibrium drywell concentrations will exist during normal operation. Purging of this activity from the drywell to the environment will occur via the Standby Gas Treatment System and will make minor contributions to total plant releases.

Airborne release data from BWR building ventilation systems and the main condenser mechanical vacuum pump have been compiled and evaluated in Reference 4, which contains data obtained by utility personnel and from special in-plant studies of operating BWR plants by independent organizations and the General Electric Company. Releases due to process leakage are reflected in the airborne release estimates discussed in Section 11.3.

11.1.6 References

1. American National Standard Radioactive Term for Normal Operation of Light Water Reactors, ANSI/ANS-18.1.

2. Skarpelos, J.M. and R.S. Gilbert, *Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms,* March 1975 (NEDO-10871).

3. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors, NUREG-0016, Revision 1, January 1979.

4. Marrero, T.R., *Airborne Releases From BWRs for Environmental Impact Evaluations,* March 1976 (NEDO-21159).

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have been redesigned as a result of inservice testing.

12.1.2.2.3 Equipment Design Considerations to Limit Component Radiation Levels

(1) Equipment and piping were designed to reduce the accumulation of radioactive materials in the equipment. The piping, where possible, was constructed of seamless pipe as a means to reduce radiation accumulation on the seam. The filter demineralizers in the RWCS and FPCS are backwashed and flushed prior to maintenance.

(2) Equipment designs include provisions for limiting leaks or controlling the fluid that does leak. This includes piping the released fluid to the sumps and the use of drip pans with drains piped to the floor drains.

(3) The materials selected for use in the primary coolant system consist mainly of austenitic stainless steel, carbon steel and low alloy steel components.

471.5 (4) The system design includes a RWCS and a condensate demineralizer system on the reactor feedwater. These systems are designed to limit the radioactive isotopes in the coolant.

471.5 (5) External recirculation pumps and recirculation piping were replaced by internally mounted recirculation pumps. Such pumps can be removed easily as an integral or package unit for maintenance outside the lower drywell radiation zone.

12.1.2.3 Facility Layout General Design Considerations for Maintaining Radiation Exposures ALARA

12.1.2.3.1 Minimizing Personnel Time Spent in Radiation Areas

Facility general design considerations to minimize the amount of personnel time spent in radiation areas include the following:

(1) locating equipment, instruments, and sampling stations, which require routine maintenance, calibration, operation, or inspection, for ease of access and minimum required occupancy time in radiation areas;

(2) laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment; and

(3) providing, where practicable, for transportation of equipment or components requiring service to a lower radiation area.

12.1.2.3.2 Minimizing Radiation Levels in Plant Access Areas and Vicinity of Equipment

Facility general design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention include the following:

(1) separating radiation sources and occupied areas where practicable (e.g., pipes or ducts containing potentially high radioactive fluids not passing through occupied areas;

(2) providing adequate shielding between radiation sources and access and service areas. Of special note, the reactor pressure vessel shield wall in the upper drywell extends to within four inches of the upper drywell ceiling thus permitting continued operation in the upper drywell during refueling and providing sheilding in the case of a refueling accident;

(3) locating equipment, instruments, and sampling sites in the lowest practicable radiation zone;

(4) providing central control panels to permit remote operation of all essential instrumentation and controls from the lowest radiation zone practicable;

(5) where practicable for package units, separating highly radioactive equipment from less radioactive equipment, instruments, and controls;

(6) providing means and adequate space for utilizing moveable shielding for sources within the service area when required;

(7) providing means to control contamination and to facilitate decontamination of potentially contaminated areas where practicable;

(8) providing means for decontamination of service areas;

Table 12.2-5

Radiation Sources (Continued)

C. Shielding Geometry in meters

Component	Room Dimensions			Wall Thickness in meters[e]					
	Length	Width	Height	East	West	North	South	Floor	Ceiling
RHR Heat Exchanger	12.6	5.6	5.6	0.8	0.6	0.6	0.6	Ground	0.8
RCIC Turbine	14.6	7.8	5.6	0.8	2	0.6	0.6	Ground	0.8
CUW Filter Demineralizer	2.8	3	7.4	0.8	1	0.8	1	0.5	Hatch
CUW Regen Heat Exchanger	7.7	3.6	6	1.4	1.4	1	1.4[a]	0.8	0.5
CUW Non-Regen Heat Exchanger	7.4	4.4	5.6	1	1	1	1[a]	Ground	0.8
LCW Collector Tank	19	1	13	1.2	0.8	0.8	1.2	Ground	0.8
LCW Filter	16.4	10.6	8	0.8	0.8	0.8	0.8	0.8	0.8
LCW Demineralizer[b]	19.6	10.6	8	0.8	0.8	0.8	0.8	0.8	0.8
LCW Sample Tank	19	10	13	1.2	0.8	1.2	0.8	Ground	0.8
HCW Collector Tank	9	11.2	5.4	0.8	0.8	0.8	1.2	Ground	0.8
HCW Demineralizer[b]	19.6	10.6	8	0.8	0.8	0.8	0.8	0.8	0.8
Offgas	9.1	11	16	1	1	1	1	2.5	1
Steam Jet Air Ejector and Recombiner Room	9.1	14.2	7	1	1	1	1	1	1
CUW Backwash Receiving Tank	6.6	7.4	5.6	1	0.8	0.8	1	Ground	0.8
CF Backwash Receiving Tank	5	5	25	1	1	1	1	2.5	Hatch
Phase Separator	16	8.4	4.6	0.8	0.8	0.8	1.2	0.8	0.8
Spent Resin Storage Tank	6.4	6.4	4.6	0.8	0.8	0.8	0.8	0.8	0.8
Concentrated Waste Tank	4.6	5	5.4	0.8	0.8	1.2	0.8	Ground	0.8
Sol Dryer Feed Tank	9.4	7.2	6.2	0.8	0.8	0.8	0.8	0.8	0.8
Sol Dryer (outlet)[c]	9.2	5.2	8	0.8	0.8	0.8	0.8	0.8	0.8
Sol Peletizer	9.2	5.2	6.8	0.8	0.8	0.8	0.8	0.8	0.8
Sol Mist Separator (steam)[c]	9.2	5.2	8	0.8	0.8	0.8	0.8	0.8	0.8
Sol Condenser	4.2	7.2	6.2	0.8	0.8	0.8	0.8	0.8	0.8
Sol Drum	3.2	3	8	0.8	0.8	0.8	0.8	0.8	0.8
FPC Filter Demineralizer	3.2	3.2	7.4	0.8	1	0.8	0.8	0.5	Hatch
Suppression Pool Cleanup Sys	3.2	3.2	7.4	0.5	0.8	0.8	0.8	0.5	Hatch
Control Rod Drive System[f]	7.6	33.4	5.8	0.6	0.6	0.6	0.6	0.8	0.6
Transverse Incore Probe	4	7.3	2.7	1	1	1	1	Mezz	0.6

Table 12.2-5

Radiation Sources (Continued)

C. Shielding Geometry in meters (Continued)

Component	Room Dimensions			Wall Thickness in meters[e]					
	Length	Width	Height	East	West	North	South	Floor	Ceiling
Reactor Internal Pumps[f]	8.2	8.5	5.8	0.6	0.6	0.6	0.6	0.8	0.6
RIP Heat Exchanger	Primary Containment								
Turbine Moisture Sep/Reheater	12.4	47.6	8.5	1	1	1	1	1	1
Turbine Condenser	14.2	36	25	3.5	2.5	1	1	2.5	Turbine
Condenser Filter	5	21.1	8	2.5[a]	1	1	1	1	Hatch
Condenser Demineralizer	9.8	17.3	9	1	1	1	1.6	1	1
SGTS Filter Train	14.4	5	8.2	0.2	0.5	0.2	0.2	2	0.6
Spent Fuel Storage	9.4	14	4.1	2	2	2	2	2	7.4[d]

Notes

[a] Moveable Wall
[b] LCW and HCW Demineralizer share same room
[c] Solid dryer and Mist Separator share same room
[d] 7.4 meter water depth above fuel elements
[e] North refers to plant 0 degree orientation, east = 90 degrees
[f] Maintenance Facility

Table 13.3-1

ABWR DESIGN CONSIDERATIONS FOR EMERGENCY PLANNING REQUIREMENTS (Continued)

Facility	Primary Document/ Section	Emergency Planning Requirements	ABWR Design Consideration
		control room and the TSC, shall be provided for operations support personnel to report in an emergency. There shall be direct communications between the OSC and control room and between the OSC and the TSC so that the personnel reporting to the OSC can be assigned to duties in emergency operations.	
Emergency Operations Center (EOC)	NUREG-0654/ II.H.6	Each licensee shall make provision to acquire data from or for emergency access to offsite monitoring equipment including geophysical phenomena and radiological monitors.	Not within the scope of the ABWR Standard Plant. However, no inpact on ABWR design.
Laboratory Facilities, Fixed or Mobile	NUREG-0654/ II.H.6.c	Provisions to acquire data from or for emergency access to offsite monitoring and analysis equipment for laboratory facilities, fixed or mobile.	Responsibility of COL applicant. ABWR design allows applicant to meet this requirement.
Post Accident Sampling System	NUREG-0737/ II.B.3	Post accident sampling capability	Post-accident sampling system of Subsection 9.3.2 meets requirements except as described in Section 1A.2.7 for the upper limit of activity in the samples at the time they are taken.

Table 13.3-1

ABWR DESIGN CONSIDERATIONS FOR EMERGENCY PLANNING REQUIREMENTS (Continued)

Facility	Primary Document/ Section	Emergency Planning Requirements	ABWR Design Consideration
Onsite Decontamination Facility	10CFR50 Appendix E/ IV.E.3	Provisions shall be made and described for facilities and supplies at the site for decontamination of onsite individuals.	Decontamination of of onsite individuals will be provided by the COL applicant in the service building adjacent to the main change rooms (See Figure 1.2-20).

TABLE 13.6-1

REACTOR BUILDING CONTROL MEASURES

SAFEGUARDS INFORMATION- Provided under separate cover.

TABLE 13.6-2

CONTROL BUILDING CONTROL MEASURES

SAFEGUARDS INFORMATION - Provided under separate cover.

power conditions

(c) proper functioning of valve positive closure devices including verification of adequate valve leak tightness; and

(d) proper functioning of vacuum breaker test features.

System operation is considered acceptable when the observed/measured performance characteristics meet the applicable design specifications.

14.2.12.1.44 Primary Containment Monitoring Instrumentation Preoperational Test

(1) Purpose

To verify the proper operation of instrumentation used for long term monitoring of the drywell and wetwell atmospheres and suppression pool temperature and level during both normal operations and accident conditions in the primary containment.

(2) Prerequisites

The construction tests have been successfully completed and the SCG has reviewed the test procedure and has approved the initiation of testing. The suppression pool shall be filled and expected to undergo measurable level and temperature changes at some point during the scheduled testing. The required interfacing systems and components are available, as needed, to support the specified testing. Additionally, any parallel testing to be performed in conjunction with the testing of this subsection is appropriately scheduled.

(3) General Test Methods and Acceptance Criteria

A description of the instrumentation required for containment monitoring is presented in Subsection 6.2.1.7. Preoperational testing of these instruments will be performed in conjunction with the testing of the applicable systems. Only that instrumentation requiring special considerations is discussed below.

Performance shall be observed and recorded during a series of individual component and integrated system tests to demonstrate the following:

(a) proper tracking of drywell pressure by all instrument channels during containment integrated leak rate testing (see Subsection 1A.2.4);

(b) proper response of all suppression pool level instrumentation during actual changes in pool level;

(c) proper tracking by all suppression pool temperature instrument channels of an actual change in pool temperature;

(d) proper functioning of associated indicators, recorders, annunciators, and alarms including those monitoring instrumentation status; and

(e) proper system trips in response to the appropriate high and/or low setpoints and inoperative conditions.

System operation is considered acceptable when the observed/measured performance characteristics, from the testing described above, meet the applicable design specifications.

14.2.12.1.45 Electrical Systems Preoperational Test

The total plant electrical distribution network is described in Chapter 8 and is comprised of the following systems:

(1) unit auxillary AC power system;
(2) unit Class 1E AC power system;
(3) safety system logic and control system power system;
(4) instrument power system;
(5) uninterruptible power system;
(6) unit auxillary DC power system; and
(7) unit class 1E DC power system.

Because of the similarities in their design and function, the testing requirements for these systems, and their respective components, can be divided into the four general categories as described below. The specific testing required for each system is described in the applicable design and testing specifications.

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SECTION 15.6

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Two specific pathways are analyzed in releasing fission products to the environment. The first pathway is leakage to the reactor building (secondary containment) via penetrations and engineered safety feature components. This leakage pathway is assumed as not greater than an equivalent release of 0.5% by weight per day of the primary containment free air weight per plant technical specification. The secondary containment is a multi-compartment self contained structure maintained at negative pressure with respect to the environment thereby providing a significant hold up volume for fission product releases. All leakage pathways from the primary containment except the main steamlines and the feedwater lines terminate in the reactor building. Leakage through the steamlines is treated separately below and leakage through the feedwater lines is assumed negligible assuming the proper isolation and filling of the feedwater lines upstream of the primary containment through the feedwater system. Flow through the reactor building/secondary containment is directed via the standby gas treatment system to the plant stack through hepa and charcoal filters. Credit is taken for hold up assuming 50% mixing in the secondary containment without plateout and other removal processes except filtration in the stand by gas treatment system (SGTS) as given in Table 15.6-8. It is assumed that for the first 20 minutes after an isolation signal, the SGTS is drawing the reactor building down to negative pressures, and therefore all leakage during this time period is assumed without effective filtration. Following this 20 minute period, full filtration is assumed for the remainder of the period.

Removal process in the primary containment and for leakages from the primary containment are described in the following sections. Section 15.6.5.5.1.1 discusses reductions in airborne iodine due to water attrition while sections 15.6.5.5.1.2 and 15.6.5.5.1.3 discuss removal processes for leakages downstream of the main steamline isolation valves.

15.6.5.5.1.1 Suppression Pool Scrubbing

The BWR suppression pool, though designed primarily as a pressure suppression mechanism for vessel blow down, serves also as an excellent medium for the intrainment and capturing of all fission products except the noble gases. The design and operational characteristics of the BWR provide for a release pathway from the vessel and drywell into the suppression pool for all cases involving vessel depressurization and therefore for removal of fission products by scrubbing in the suppression pool. The NRC has accepted the fact that the suppression pool is capable of removing fission products and provides for credit to incorporate this phenomena in design basis analysis by recourse to the requirements of Standard Review Plan 6.5.5. The requirements of SRP 6.5.5 state that any flow directed through the pool can be credited with a decontamination factor of 10 providing the requirements of subsection II are met and that the total decontamination is a combination of the decontamination applied to flow through the pool to that fraction of the release which bypasses the pool. The following paragraphs describe the determination of the bypass fraction for the calculation of overall pool decontamination.

The requirements of Regulatory Guide 1.3 stipulate an instantaneous release of fission products from the vessel to the containment atmosphere. Coincident with an instantaneous release, under LOCA conditions, the BWR pressure vessel will be depressurized resulting in the purging of the primary containment atmosphere to the suppression pool. This situation is shown in Figure 15.6-3 which show the fractions of airborne particulate as a function of time in the drywell and wetwell airspaces assuming a decontamination factor of 10 for that flow which is purged either through the horizontal vents or the safety relief valves. The figure shows that the airborne inventory is reduced by almost a factor of ten within two minutes of the initiation of the blowdown event.

However, the application of the precepts of Regulatory Guide 1.3 do not indicate the most likely train of events in a core damage event which is what is implied in the design basis release assumptions. Both Regulatory Guide 1.3 and its predecessor, TID-14844, are based upon non-mechanistic assumptions and devices and are in the process of being replaced. Therefore consideration of a range of accident progressions beyond the rigidly narrow scope of Regulatory Guide 1.3 is given below to evaluate potential suppression pool bypass under more realistic conditions.

The basic assumptions of this evaluation of suppression pool bypass conditions assumes that an event occurs which challenges the reactor core causing sufficient damage to release approximately half the fission product volatile iodines. Damage to the core is limited to this extent implying the ability to recover core cooling and limit in-vessel damage. Such an assumption complies with the intent of design basis licensing in that the exact means by which the core is challenged is not specified but given the challenge, the response and adequacy of the plant design is tested. In addition, the assumption of resumpiton of core cooling and recovery with limited release is fully justifiable since the ABWR incorporates multiple cooling modes with redundant safety grade cooling systems. Events leading to more significant core damage are not considered as design basis since they assume massive damage with multiple failures to the design safety systems. Such events are of exceedingly low probability and are described and evaluated in Chapter 19. Therefore broadly speaking events which lead to the assumed damage can be divided into two categories, break and non-break. Break events are those through which primary coolant are released directly to the primary containment atmosphere and non-break events are those in which the primary coolant boundary is not breached. Both types of events will be considered below to provide a bounding analysis for suppression pool bypass.

In considering the non-break events, core damage is primarily the result of failure to maintain proper core water level resulting in uncovering the core with subsequent release of fission products upon overheating of the fuel rods. To consider the train of events in such a case, the MAAP code (see subsection 19.E for a description of the MAAP code) was used to model vessel response. Based upon the MAAP analysis, releases would begin shortly after core water level reaches the bottom of the core and would proceed rapidly. During this period, isolation of the primary coolant system and containment would have been automatically tripped on low water level and the main steam isolation valves as well as all the other isolation valves would have tripped effectively isolating all flow from the primary containment. Therefore, the released fission products would be exposed to three primary influences: (1) plateout and removal in the dryers and separators, (2) leakage from the main steamline isolation valves into the main steamlines, and (3) flow through the safety relief valves into the suppression pool.

The release of volatile fission products would occur over a period of 10-20 minutes during which steam or hydrogem flow from the core region would be very small. Using an upper bound estimate of 2 kg/sec of steam generation during this period, the vessel flushing rate would be once every ten minutes. Therefore during this period 0.13% of the flow would bypass the pool through MSIV leakage. The remaining faction would be transported through the safety relief valves. Without recovery of cooling water after this period significant damage would occur to the core beyond that of a design basis event. With the recovery of water, the energy generated from decay heat which would be evident in overall core temperature rise and core degradation would cause a rapid pulse of steam resulting in the purging the pressure vessel of all airborne materials. Based upon the MAAP analysis it is conservatively estimated that 9×10^3 Kg of steam would be generated in a short period of time on the order of minutes resulting in a vessel purge rate of seven to eight complete exchanges. Therefore effectively all fission products remaining airborne in the vessel or lines would be purged to the suppression pool. The effective pool bypass fraction would then be 0.13% for an integrated overall DF of 9.8 without credit for plateout or 4.9 with a factor of two plateout.

The break case follows a similar logic. Initially, following a break, massive depressurization of the pressure vessel would occur causing all non-condensables in the drywell to be purged into the wetwell air space through both the horizontal vents and the safety relief valves. Isolation of the containment and associated lines would be automatically initiated on depressurization. Following this rapid depressurization there would follow a period during which the water level in the vessel would drop to the bottom of core resulting in the eventual release of fission products form the core. Since in a break case, the path of least resistance would be through the break, the fission products would be effectively purged to the drywell airspace. In this case the temperatures and surface areas involved would provide adequate plate out areas to validate the Regulatory Guide 1.3 plate out factor of 2. Like the non-break case the total release is limited implying resumption of cooling and a massive release of steam upon resumption of cooling. In the case of reflood with a break, because of the large volume of the drywell, convervatively 80% of the drywell volume is purged during the reflood period. If complete mixing is assumed which is resonable because of the dynamic flows involved, it is then found that 55.6% of the airborne fission products are purged to the suppression pool in the few minutes needed to reflood the core. Therefore in this case an integrated pool DF of 2 is calculated.

In summary, it is found that for design basis accident conditions in which credit is taken for the proper operation of redundant safety grade systems subject to the single failure proof criteria that the suppression pool is capable of reducing the elemental and particulate airborne iodine inventory by a factor of 2.

15.6.5.5.1.2 Main Steamline Modeling

The second potential release pathway is via the main steamline through leakage in the main steamline isolation valves. It is assumed that a pathway exits which permits the primary containment atmosphere or in the non-break case pressure vessel air space direct access to the main steamlines and that the main steamline isolation valves leak at the maximum technical specification. Furthermore, it is assumed that the most critical main steamline isolation valve fails in the open position. Therefore, the total leakage through the steamlines is equal to the maximum technical specification for the plant.

The main steamlines are graded (see Table 3.2-1) as Seismic Classification I Quality Group B from the pressure vessel interface to the outboard seismic restraint outboard of the downstream MSIV thereby providing a qualified safety grade mitigation system for fission product leakage which in this case is limited by the leakage criteria specific in the technical specifications for the Main Steamline Isolation Valves (MSIV). The primary purpose of this system is to stop any potential flow through the main steamlines. Down stream of the seismic restraint referred to above, the steamlines pass through the reactor building - control building interface into the steam tunnel located in the control building upper floor. This steam tunnel is a heavily shielded seismic category I structure designed primarily to shield the control building complex. From the control building the steamlines pass through the control building - turbine building interface into the turbine building steam tunnel which is a heavily shielded reinforced concrete structure designed to shield workers from main steamline radiation shine. The steamlines, their associated branch lines outboard of the last reactor building seismic restraint are Quality Group B structures. In addition, these lines and structures are required to be dynamically analyzed to SSE conditions (Table 3.2-2, note R) which determines the flexibility and structural capabilities of the lines under hypothetical SSE conditions.

The analysis of leakage from the primary containment through the main steamlines involves the determination of (1) probable and alternate flow pathways, (2) physical conditions in the pathways, and (3) physical phenomena which affect the flow and concentration of fission products in the pathways. The most probable pathway for fission product transport from the main steamlines is found to be from the outboard MSIV's into the drain lines coming off the outboard MSIV and then into the turbine building to the main condenser. A secondary path is found along the main steamlines into the turbine though flow through this pathway as described below is a minor fraction of the flow through the drain lines. Consideration of the main steamlines and drain line complex downstream of the reactor building as a mitigative factor in the analysis of LOCA leakage is based upon the following determination.

1. The main steamlines and drain lines are high quality lines inspected on a regular schedule.

2. The main steamlines and drain lines are designed to meet SSE criteria and analyzed to dynamic loading criteria.

3. The main steamlines and drain lines are enclosed in a shielded corridor which protects them from collateral damage in the event of an SSE. For those portions not enclosed in the steam tunnel complex, an as built inspection is required to verify that no damage could be expected from other components and structures in a SSE.

4. The main steamlines and drain lines are required under normal conditions to function to loads at temperature and pressure far exceeding the loads expected from an SSE. This capability inherent in the basic design of these components furnishes a level of toughness and flexibility to assure their survival under SSE conditions. A large data base of experience in the survival of these types of components under actual earthquake conditions exits which prove this

contention. (Reference 5) In the case of ABWR further margin for survival can be expected since the ABWR lines are designed through dynamic analysis to survive such events whereas in the case of the actual experience data base, the lines shown to survive were designed to lesser standards to meet only normally expected loads.

Therefore, based upon the facts above, the main steamlines and drain lines in the ABWR are used as mitigative components in the analysis of leakage from the MSIVs.

The analysis of leakage from the MSIVs follows the procedures and conditions specified in Reference 5. Two flow paths are analyzed for dose contributions. The first pathway through the drain lines is expected to dominated due to the incorporation of a safety grade isolation valve on the outboard drain line which will open the line for flow down the drain line under LOCA conditions. The second pathway through the main steamlines into the turbine is expected to carry less than 0.3% of the flow based upon a determination that the maximum leakage past the turbine stop valves with an open drain line would permit only 0.3% flow for the valves to operate within specification. Specific values used and results of the main steamline leakage analysis are given in Table 15.6-8.

The COL applicant will recalculate iodine removal credit on the basis of its design characteristics of main steamlines, drain, and main condenser. See Subsection 15.6.7.1 for COL license information requirements.

15.6.5.5.1.3 Condenser and Turbine Modeling

The condenser and turbine are modeled as detailed in Reference 5 with specific values used given in Table 15.6-8. Both volumes are modeled primarily as stagnant volumes assuming the shutdown of all active components. Both turbine and condenser are used as mitigative volumes based upon the determination that such components designed to standard engineering practice are sufficiently strong to withstand SSE conditions due wholly to their design. (Reference 5) The only requirement in the design of the condenser being that it be bolted to the building basemat to prevent walking during an earthquake. The turbine has no such restriction and may possibly move. The requirement on these components for purposes of mitigation is only that they survive as a volume and not that they provide functionality or leak tightness following an earthquake.

Release from the condenser/turbine building pathway are assumed via diffuse sources in the turbine building. The two major points of release in the turbine building are expected to be the truck doors at the far end of the turbine building and the maintenance panels located midway on the turbine building on the side opposite the service building. Releases are assumed to be ground level releases. See section 15.6.5.5.3 for applicable meteorology.

The COL applicant will recalculate iodine removal credit on the basis of its design characteristics of main steamlines, drain, and main condenser. See Subsection 15.6.7.1 for COL license information requirements.

15.6.5.5.2 Control Room

The ABWR control room is physically integrated with the reactor building and turbine buildings and is located between these structures (see Figure 15.6-4). During a LOCA, exposure to the operators will consist of contributions from airborne fission products entrained into the control room ventilation system and gamma shine from the reactor building and airborne fission products external to the control building. Of these contributions, the last two involving gamma shine are negligible since the inhabited portions of the ABWR control room are physically located underground with sufficient shielding overhead (a minimum of 1.6 meters of concrete) and in the side walls (1.2 meters) to protect the operators from the normal steamline gamma shine. Such shielding is more than sufficient to protect the operators given any amount of airborne fission products.

Therefore, exposure to the operators will consist almost entirely of fission products entrained into the control room environment from the atmosphere. The ABWR control room uses a redundant safety grade HVAC system with two inch charcoal filters for removal of iodines and two roof mounted automatically controlled intake vents. The location of the vents are given in Figure 15.6-4. Because of the location of these vents, it cannot be assumed that at least one vent will be uncontaminated given most conditions of meteorology. Therefore, no credit for dual intakes was taken. In addition, the location of these vents with respect to the potential release points show that given any wind flow condition, the vents may be contaminated only by a release from the reactor building or turbine building but not both. Nevertheless, for purposes of conservative calculations, it was arbitrarily assumed that for 30% of the time stagnant meteorological conditions were assumed such that the primary intake vent was contaminated by both sources. For the remaining 70% of the time only the more significant source was assumed to contaminate the primary intake vent.

Infiltration of airborne contamination to the control room was considered negligible owing to the pathway for access to the control complex. Entry into the control room is via the service building and a labyrinth doorway entry system through double doors into the clean portions of the service building. From the service building additional controlled access through double doors provides entry into the control room. In each of these entry/access door systems, positive pressure is maintained to vent infiltrated air to the outside and away from the control room complex. As such no contamination is anticipated beyond the initial access entry way from which infiltrating air is purged to the environment.

Control room dose is based upon fission product releases modeled as described in paragraph 15.6.5.5.1 and the values presented in Table 15.6-8. Operator exposure was based upon those conditions given in Table 15.6-8 and occupancy factors as shown below derived from SRP 6.4. Meteorology was derived as is specified in section 15.6.5.3.2.

Time	Occupancy Factor
0-1 day	1.0
1-4 days	0.6
> 4 days	0.4

15.6.7 COL License Information

15.6.7.1 Iodine Removal Credit

The COL applicant will recalculate iodine removal credit as outlined in Subsections 15.6.5.5.1.2 and 15.6.5.5.1.3.

15.6.8 References

1. F.J. Moody, *Maximum Two-Phase Vessel Blowdown from Pipes*, ASME Paper Number 65-WA/HT-1, March 15, 1965.

2. H.A. Careway, V.D. Nguyen, and P.P. Stancavage, *Radiological Accident Evaluation - The CONAC03 Code*, December 1981 (NEDO-21143-1).

3. H.A. Careway, V.D. Nguyen, and D.G. Weiss, *Control Room Accident Exposure Evaluation, CRDOS Program*, February 1981 (NEDO-23909A).

4. K.G. Murphy, and K.M. Campe, *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19*, 13th ASC Air Cleaning Conference, June 1974.

5. L.S. Lee, *BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems*, Febuary 1991, NEDC-31858P.

6. J.V. Ramsdell, *Atmospheric Diffusion for Control Room Habitability Assessments*, May 1988 (NUREG/CR-5055).

7. Ramsdell, J.V., *Alternatives to Current Procedures Used to Estimate Concentrations in Building Wakes*, 21st DOE/NRC Nuclear Air Cleaning Conference, pgs 714-729.

Table 15.6-5

STEAM LINE BREAK ACCIDENT PARAMETERS

I	Data and assumptions used to estimate source terms.		
	A.	Power level	4005 MWt
	B.	Fuel damage	none
	C.	Reactor Coolant Activity	Subsection 15.6.4.5
	D.	Steam mass released	28373 lbs
	E.	Water mass released	48397 lbs
II	Data and assumptions used to estimate activity released		
	A.	Isolation valve closure time	5.0 sec
	B.	Maximum release time	2 hr
III	Dispersion and Dose Data		
	A.	Meteorology	Table 15.6-7
	B.	Boundary and LPZ distances	Table 15.6-7
	C.	Method of Dose Calculation	Reference 2
	D.	Dose conversion Assumptions	Reference 2, RG 1.109, and ICRP 30
	E.	Activity Inventory/releases	Table 15.6-6
	F.	Dose Evaluations	Table 15.6-7

Table 15.6-6

MAIN STEAM LINE BREAK ACCIDENT ACTIVITY RELEASED TO ENVIRONMENT IN CURIES

ISOTOPE	CASE 1	CASE 2
I-131	1.97E+00	3.94E+01[a]
I-132	1.92E+01	3.83E+02
I-133	1.35E+01	2.70E+02
I-134	3.78E+01	7.55E+02
I-135	1.97E+01	3.94E+02
TOTAL HALOGENS	9.21E+01	1.84E+03
KR-83M	1.10E-02	6.60E-02
KR-85M	1.94E-02	1.16E-01
KR-85	6.11E-05	3.67E-04
KR-87	6.59E-02	3.96E-01
KR-88	6.66E-02	4.00E-01
KR-89	2.67E-01	1.60E+00
KR-90	6.90E-02	4.20E-01
XE131M	4.76E-05	2.86E-04
XE133M	9.15E-04	5.50E-03
XE133	2.56E-02	1.54E-01
XE135M	2.56E-02	1.54E-01
XE-135M	7.80E-02	4.69E-01
XE-135	7.31E-02	4.39E-01
XE-137	3.32E-01	2.00E+00
XE-138	2.55E-01	1.53E+00
XE-139	1.17E-01	7.01E-01
TOTAL NOBLE GASES	1.38E+00	8.30E+00

a 3.94E+01 = 3.94×10^{1}

Table 15.6-8

LOSS OF COOLANT ACCIDENT PARAMETERS

I Data and Assumptions used to esimate source terms.

A.	Power Level	4005 MWt
B.	Fraction of Core Inventory Released	
	Noble Gases	100%
	Iodines	50%
C.	Iodine Initial Plateout Fraction	50%
D.	Iodine Chemical Species	
	Elemental	91%
	Particulate	5%
	Organic	4%
E.	Suppression Pool Decontamination Factor - sec 15.6.5.5.1.1	
	Noble Gas	1
	Organic Iodine	1
	Elemental Iodine	2
	Particulate	2
	Pool Bypass Area	0.05 ft^2

II Data and Assumption used to estimate activity released.

A.	Primary Containment Leakage	
	(1) Penetration and ESF Equipment	0.5%/day
	(2) MSIV Leakage (Total all lines)	140 SCFH
B.	Reactor Building Leakage	
	(1) 0-20 min	150%/hr
	(2) >20 minutes	50%/day
	(3) Mixing Effeciency	50%
C.	SGTS	
	Filter Effeciency (6 inch charcoal)	97%
	Drawdown Time	20 min
D.	MSIV Leakage - see Reference 5 for standard parameters	
	Main Steam Line Length	157 ft
	Drain Line Length	235 ft
	Main Steam Line IR/OR	31.98/35.55cm
	Drain Line IR/OR	3.33/4.45cm
	Main Steam Line Insulation	12.0 cm
	Drain Line Insulation	6.5cm
	Plateout and Resuspension Factors	Ref 5.
E.	Condenser Data	
	Free Air Volume	220,000ft^3
	Fraction of Volume involved	20%
	Leakage Rate	11.6%/day
	Iodine Removal Factor	
	Elemental	0.993
	Particulate	0.993
	Organic	0

Table 15.6-8

LOSS OF COOLANT ACCIDENT PARAMETERS (Continued)

III Control Room Data	
A. Control Room Volumes	
Total Free Air Volume	5,509 m^3
Gamma Room Volume (room size)	1,400 m^3
B. Recirculation Rates	
Filtered Intake	0.1 m^3/sec
Unfiltered Intake	0.0
Filtered Recirculation	0.65 m^3/sec
Filter Efficiency (2 inch charcoal)	95%
IV Dispersion and Dose Data	
A. Meteorology	Sec 15.6.5.5.3 Tbls 15.6-13,14
B. Dose Calculation Method (semi-infinite)	Ref 2 & 3, RG 1.109
C. Dose Conversion Assumptions	Ref 2, 3
D. Activity/Releases	Tbls 15.6-9,10,11,12 Appendix F.
E. Dose Evaluation	Tbls 15.6-13,14

Table 15.6-9

IODINE ACTIVITIES

A. Primary Containment Inventory in Curies

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
I-131	1.4E+7	1.4E+7	1.4E+7	1.4E+7	1.3E+7	1.3E+7	1.3E+7	1.2E+7	9.2E+6	7.2E+5
I-132	2.0E+7	1.9E+7	1.5E+7	1.1E+7	5.9E+6	1.8E+6	5.2E+5	1.4E+4	4.3E-6	0
I-133	2.9E+7	2.8E+7	2.8E+7	2.7E+7	2.5E+7	2.2E+7	1.9E+7	1.3E+7	1.1E+6	7.5E-4
I-134	3.1E+7	2.8E+7	1.4E+7	6.5E+6	1.3E+6	5.6E+4	2.4E+3	1.8E-1	0	0
I-135	2.7E+7	2.7E+7	2.4E+7	2.2E+7	1.8E+7	1.2E+7	7.6E+6	2.1E+6	1.1E+3	0
Total	1.2E+8	1.2E+8	9.5E+7	8.0E+7	6.3E+7	4.9E+7	4.0E+7	2.7E+7	1.0E+7	7.2E+5

B. Reactor Building Inventory in Curies

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
I-131	4.7E+1	4.0E+2	2.5E+3	5.2E+3	1.0E+4	1.9E+4	2.6E+4	3.9E+4	4.6E+4	3.6E+3
I-132	6.8E+1	5.6E+2	2.8E+3	4.2E+3	4.5E+3	2.5E+3	1.0E+3	4.3E+1	2.1E-8	0
I-133	9.8E+1	8.4E+2	5.2E+3	1.0E+4	1.9E+4	3.1E+4	3.7E+4	4.0E+4	5.5E+3	3.8E-6
I-134	1.1E+2	8.2E+2	2.7E+3	2.5E+3	1.0E+3	7.9E+1	4.7E+0	5.7E-4	0	0
I-135	9.2E+1	7.8E+2	4.5E+3	8.4E+3	1.3E+4	1.6E+4	1.5E+4	6.8E+3	5.2E+0	0
Total	4.1E+2	3.4E+3	1.8E+4	3.0E+4	4.8E+4	6.8E+4	7.9E+4	8.6E+4	5.1E+4	3.6E+3

C.1 MSIV Pathway - Condenser Inventory in Curies - Elemental Iodine

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
I-131	0	0	2.1E+2	1.3E+3	5.5E+3	1.7E+4	3.0E+4	6.4E+4	1.1E+5	8.5E+2
I-132	0	0	2.3E+2	1.1E+3	2.4E+3	2.3E+3	1.2E+3	7.1E+1	5.2E-8	0
I-133	0	0	4.3E+2	2.6E+3	1.0E+4	2.8E+4	4.4E+4	6.6E+4	1.3E+4	9.0E-7
I-134	0	0	2.2E+2	6.4E+2	5.4E+2	7.3E+1	5.5E+0	9.4E-4	0	0
I-135	0	0	3.8E+2	2.2E+3	7.2E+3	1.5E+4	1.7E+4	1.1E+4	1.3E+1	0
Total	0	0	1.5E+3	7.9E+3	2.6E+4	6.3E+4	9.2E+4	1.4E+5	1.2E+5	8.5E+2

C.2 MSIV Pathway - Condenser Inventory in Curies - Organic Iodine (Primary Containment)

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
I-131	0	0	1.8E+1	1.1E+2	4.7E+2	1.5E+3	2.5E+3	5.6E+3	1.6E+4	3.6E+3
I-132	0	0	1.9E+1	9.1E+1	2.1E+2	1.9E+2	1.0E+2	6.2E+0	7.6E-9	0
I-133	0	0	3.7E+1	2.2E+2	8.6E+2	2.4E+3	3.7E+3	5.7E+3	1.9E+3	3.8E-6
I-134	0	0	1.9E+1	5.4E+1	4.6E+1	6.2E+0	4.7E-1	8.2E-5	0	0
I-135	0	0	3.2E+1	1.8E+2	6.1E+2	1.3E+3	1.5E+3	9.6E+2	1.8E+0	0
Total	0	0	1.2E+2	6.7E+2	2.2E+3	5.3E+3	7.8E+3	1.2E+4	1.8E+4	3.6E+3

Table 15.6-9

IODINE ACTIVITIES (Continued)

C.3 MSIV Pathway - Condenser Inventory in Curies - Resuspended Organic

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
I-131	0	0	7.5E-2	1.5E-1	9.2E-1	2.4E+0	7.4E+0	2.5E+1	1.3E+3	2.6E+3
I-132	0	0	6.0E-2	1.0E-1	2.3E-1	3.3E-1	1.6E-1	4.7E-2	0	0
I-133	0	0	1.5E-1	2.9E-1	1.6E+0	4.0E+0	9.6E+0	2.5E+1	1.6E+2	1.4E-5
I-134	0	0	3.9E-2	5.7E-2	3.4E-2	2.8E-2	1.5E-3	3.2E-5	0	0
I-135	0	0	1.2E-1	2.3E-1	9.7E-1	2.0E+0	3.1E+0	4.2E+0	2.6E-1	0
Total	0	0	4.4E-1	8.3E-1	3.8E+0	8.8E+0	2.0E+1	5.4E+1	1.5E+3	2.6E+3

C.4 Condenser Inventory in Curies - Combined

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
I-131	0	0	2.3E+2	1.5E+3	6.0E+3	1.9E+4	3.2E+4	7.0E+4	1.3E+5	7.1E+3
I-132	0	0	2.5E+2	1.2E+3	2.6E+3	2.5E+3	1.3E+3	7.7E+1	6.0E-8	0
I-133	0	0	4.7E+2	2.9E+3	1.1E+4	3.1E+4	4.7E+4	7.2E+4	1.5E+4	1.9E-5
I-134	0	0	2.4E+2	6.9E+2	5.9E+2	7.9E+1	5.9E+0	1.1E-3	0	0
I-135	0	0	4.1E+2	2.3E+3	7.8E+3	1.6E+4	1.9E+4	1.2E+4	1.5E+1	0
Total	0	0	1.6E+3	8.5E+3	2.8E+4	6.8E+4	1.0E+5	1.5E+5	1.4E+5	7.1E+3

D.1 Control Room Inventory in Curies

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
I-131	1.2E-5	1.1E-3	2.8E-3	1.8E-3	9.0E-4	7.1E-4	6.2E-4	9.9E-4	8.9E-4	5.2E-5
I-132	1.8E-5	1.5E-3	3.0E-3	1.4E-3	4.0E-4	9.5E-5	2.5E-5	1.1E-6	0	0
I-133	2.5E-5	2.2E-3	5.6E-3	3.5E-3	1.7E-3	1.2E-3	9.0E-4	1.0E-3	1.1E-4	0
I-134	2.8E-5	2.2E-3	2.9E-3	8.5E-4	8.9E-5	3.0E-6	1.1E-7	1.5E-11	0	0
I-135	2.4E-5	2.1E-3	4.9E-3	2.9E-3	1.2E-3	6.2E-4	3.6E-4	1.7E-4	1.0E-7	0
Total	1.1E-4	9.0E-3	1.9E-2	1.0E-2	4.3E-3	2.6E-3	1.9E-3	2.2E-3	9.9E-4	5.2E-5

D.2 Control Room Integrated Activity in Curies-seconds

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
I-131	2.4E-4	2.2E-1	9.2E+0	8.0E+0	9.1E+0	1.0E+1	9.0E+0	3.5E+1	2.2E+2	4.2E+2
I-132	3.5E-4	3.1E-1	1.1E+1	7.6E+0	5.7E+0	2.7E+0	7.0E-1	3.3E-1	1.1E-2	3.2E-12
I-133	5.1E-4	4.6E-1	1.9E+1	1.6E+1	1.7E+1	1.8E+1	1.4E+1	4.2E+1	9.1E+1	5.8E+0
I-134	5.5E-4	4.6E-1	1.3E+1	6.0E+0	2.4E+0	3.3E-1	1.2E-2	5.4E-4	6.0E-8	0
I-135	4.8E-4	4.3E-1	1.7E+1	1.4E+1	1.3E+1	1.1E+1	6.5E+0	1.1E+1	4.9E+0	1.9E-3
Total	2.1E-3	1.9E+0	7.0E+1	5.1E+1	4.8E+1	4.3E+1	3.0E+1	8.9E+1	3.2E+2	4.3E+2

Table 15.6-12

NOBLE GAS ACTIVITY RELEASE TO ENVIRONMENT

A. Reactor Building Release to Environment in Curies

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
Kr-83m	7.2E-1	6.3E+1	2.5E+2	3.3E+2	5.2E+2	7.7E+2	8.6E+2	9.0E+2	9.0E+2	9.0E+2
Kr-85	6.9E-2	6.3E+0	2.7E+1	4.1E+1	9.8E+1	3.2E+2	6.6E+2	2.2E+3	1.8E+4	1.5E+5
Kr-85m	1.5E+0	1.4E+2	5.7E+2	8.2E+2	1.6E+3	3.5E+3	5.0E+3	7.2E+3	7.8E+3	7.8E+3
Kr-87	2.9E+0	2.5E+2	9.8E+2	1.2E+3	1.7E+3	2.1E+3	2.2E+3	2.2E+3	2.2E+3	2.2E+3
Kr-88	4.2E+0	3.7E+2	1.5E+3	2.1E+3	3.7E+3	6.7E+3	8.4E+3	9.8E+3	9.9E+3	9.9E+3
Kr-89	4.5E+0	1.3E+2	1.8E+2	1.8E+2	1.8E+2	1.8E+2	1.8E+2	1.8E+2	1.8E+2	1.8E+2
Xe-131m	3.6E-2	3.3E+0	1.4E+1	2.1E+1	5.1E+1	1.6E+2	3.4E+2	1.1E+3	8.2E+3	3.7E+4
Xe-133	1.3E+1	1.1E+3	4.9E+3	7.5E+3	1.8E+4	5.7E+4	1.2E+5	3.7E+5	2.4E+6	6.7E+6
Xe-133m	5.3E-1	4.8E+1	2.0E+2	3.1E+2	7.3E+2	2.3E+3	4.5E+3	1.4E+4	7.0E+4	1.1E+5
Xe-135	1.6E+0	1.5E+2	6.2E+2	9.2E+2	2.0E+3	5.2E+3	9.0E+3	1.8E+4	2.8E+4	2.8E+4
Xe-135m	2.3E+0	1.6E+2	4.7E+2	4.8E+2	4.9E+2	4.9E+2	4.9E+2	4.9E+2	4.9E+2	4.9E+2
Xe-137	9.9E+0	3.4E+2	5.2E+2	5.2E+2	5.2E+2	5.2E+2	5.2E+2	5.2E+2	5.2E+2	5.2E+2
Xe-138	1.0E+1	7.0E+2	2.0E+3	2.0E+3	2.0E+3	2.0E+3	2.0E+3	2.0E+3	2.0E+3	2.0E+3
Totals	5.1E+1	3.5E+3	1.2E+4	1.6E+4	3.1E+4	8.1E+4	1.5E+5	4.3E+5	2.6E+6	7.1E+6

B. Condenser Release to Environment in Curies

Isotope	1 Min	10 Min	1 Hr	2 Hrs	4 Hrs	8 Hrs	12 Hrs	1 Day	4 Days	30 Days
Kr-83m	0	0	1.6E-1	2.1E+0	1.2E+1	3.5E+1	4.6E+1	5.2E+1	5.2E+1	5.2E+1
Kr-85	0	0	2.0E-2	3.5E-1	3.4E+0	2.5E+1	7.1E+1	3.6E+2	6.3E+3	1.6E+5
Kr-85m	0	0	4.0E-1	6.2E+0	4.8E+1	2.3E+2	4.3E+2	8.2E+2	9.6E+2	9.6E+2
Kr-87	0	0	5.5E-1	6.5E+0	3.0E+1	6.5E+1	7.4E+1	7.6E+1	7.6E+1	7.6E+1
Kr-88	0	0	1.0E+0	1.5E+1	1.0E+2	3.8E+2	6.1E+2	8.5E+2	8.7E+2	8.7E+2
Kr-89	0	0	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4
Xe-131m	0	0	1.1E-2	1.8E-1	1.8E+0	1.3E+1	3.6E+1	1.8E+2	2.8E+3	3.4E+4
Xe-133	0	0	3.7E+0	6.4E+1	6.2E+2	4.4E+3	1.2E+4	6.0E+4	8.2E+5	4.9E+6
Xe-133m	0	0	1.5E-1	2.7E+0	2.5E+1	1.8E+2	4.8E+2	2.2E+3	2.2E+4	5.4E+4
Xe-135	0	0	4.6E-1	7.4E+0	6.5E+1	3.8E+2	8.8E+2	2.6E+3	5.3E+3	5.3E+3
Xe-135m	0	0	7.9E-2	2.8E-1	3.4E-1	3.4E-1	3.4E-1	3.4E-1	3.4E-1	3.4E-1
Xe-137	0	0	9.1E-4	9.5E-4	9.5E-4	9.5E-4	9.5E-4	9.5E-4	9.5E-4	9.5E-4
Xe-138	0	0	2.8E-1	8.6E-1	1.0E+0	1.0E+0	1.0E+0	1.0E+0	1.0E+0	1.0E+0
Totals	0	0	6.8E+0	1.1E+2	9.1E+2	5.7E+3	1.5E+4	6.8E+4	8.6E+5	5.2E+6

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17.0 INTRODUCTION

Section 17.1 of this Standard Safety Analysis Report describes the Quality Assurance (QA) Program which is implemented by GE for the ABWR project. It is based upon the standard GE QA Program documented in the GE Nuclear Energy topical report NEDO-11209-04A (Reference 1) and the additional information in this chapter describing and clarifying GE's interfaces and responsibilities with its technical associates on the ABWR. These technical associates are major international corporations who are licensees of GE's technology and have extensive independent experience in the design and construction of nuclear power stations.

The standard program is used throughout GE Nuclear Energy on all other nuclear power plant work and has been accepted by the Nuclear Regulatory Commission. It is in compliance with Title 10, Code of Federal Regulations, Part 50, Appendix B; ANSI/ASME N45.2; ANSI/ASME N45.2-series standards; and NRC Regulatory Guides with some NRC-accepted GE Nuclear Energy alternate positions.

The QA Program described in this chapter meets Regulatory Guide 1.28, Revision 3 and is organized to show its relationship to Reference 1, ANSI/ ASME NQA-1-1983 and NQA-1a-1983, and GE's interfaces with its technical associates. The terms and definitions of supplement S-1 of NQA-1a-1983 apply. Table 17.0-1 summarizes ABWR compliance with the quality related Regulatory Guides.

The COL applicant/holder is responsible to prepare and implement a quality assurance program for the construction phase of Section 17.1 and the operations phase of Section 17.2 that also meets the requirements of ANSI/ASME NQA-1-1983 and NQA-1a-1983 and the quality related Regulatory Guides listed in Table 17.0-1. See Subsection 17.0.1 for COL license information.

17.0.1 COL License Information

17.0.1.1 QA Programs for Construction and Operation

The COL applicant/holder shall prepare and implement a quality assurance program for the construction phase of Section 17.1 and the operations phase of Section 17.2. They will meet the requirements of ANSI/ASME NQA-1-1983 and NQA-1a-1983 and the quality related Regulatory Guides listed in Table 17.0-1. (See Section 17.0)

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ILLUSTRATIONS

SECTION 17.3

ILLUSTRATIONS (Continued)

17.3 RELIABILITY ASSURANCE PROGRAM DURING DESIGN PHASE

This section presents the ABWR Design Reliability Assurance Program (D-RAP).

17.3.1 Introduction

The ABWR Design Reliability Assurance Program (D-RAP) is a program that will be performed by during detailed design and specific equipment selection phases to assure that the important ABWR reliability assumptions of the probabilistic risk assessment (PRA) will be considered throughout the plant life. The plant owner/operator will complete the D-RAP and will also have an operational RAP (O-RAP) that tracks equipment reliability to demonstrate that the plant is being operated and maintained consistent with PRA assumptions so that overall risk is not unknowingly degraded. The PRA evaluates the plant response to initiating events to assure that plant damage has a very low probability and risk to the public is very low. Input to the PRA includes details of the plant design and assumptions about the reliability of the plant risk-significant structures, systems and components (SSCs) throughout plant life. Appendix 19K, PRA Based Reliability and Maintenance, identifies certain risk-significant SSCs. The results of Appendix 19K can be used as a starting point for the D-RAP.

The D-RAP will include the design evaluation of the ABWR. It will identify relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for owner/operator consideration in assuring safety of the equipment and limited risk to the public. The COL applicant will specifiy the policy and implement procedures for using the D-RAP information. See Subsection 17.3.13.1 for COL license information.

Also included in this explanation of the D-RAP is a descriptive example of how the D-RAP will apply to one potentially important plant system, the standby liquid control system (SLCS). The SLCS example shows how the principles of D-RAP will be applied to other systems identified by the PRA as being significant with respect to risk.

17.3.2 Scope

The ABWR D-RAP will include the future design evaluation of the ABWR, and it will identify relevant aspects of plant operation, maintenance, and performance monitoring of plant risk-significant SSCs. The PRA for the ABWR and other industry sources will be used to identify and prioritize those SSCs that are important to prevent or mitigate plant transients or other events that could present a risk to the public.

17.3.3 Purpose

The purpose of the D-RAP is to assure that the plant safety as estimated by the probabilistic risk analysis (PRA) is maintained as the detailed design evolves through the implementation and procurement phases and that pertinent information is provided in the design documentation to the future owner/operator so that equipment reliability, as it affects plant safety, can be maintained through operation and maintenance during the entire plant life.

17.3.4 Objective

The objective of the D-RAP is to identify those plant SSCs that are significant contributors to risk, as shown by the PRA or other sources, and to assure that, during the implementation phase, the plant design continues to utilize risk- significant SSCs whose reliability is commensurate with the PRA assumptions. The D-RAP will also identify key assumptions regarding any operation, maintenance and monitoring activities that the owner/operator should consider in developing its O-RAP to assure that such SSCs can be expected to operate throughout plant life with reliability consistent with that assumed in the PRA.

A major factor in plant reliability assurance is risk-focused maintenance, by which maintenance resources are focused on those SSCs that enable the ABWR systems to fulfill their essential safety functions and on SSCs whose failure may directly initiate challenges to safety systems. All plant modes are considered, including equipment directly relied upon in Emergency Operating Procedures (EOPs). Such a focus of maintenance will help to maintain an acceptably low level of risk, consistent with the PRA.

17.3.5 GE-NE Organization for D-RAP

The D-RAP definition, reliability analyses, and the PRA, including Appendix 19K, were performed by GE Nuclear Energy (GE-NE).

Responsibility for the design of key equipment, components and subsystems was shared by GE-NE together with external organizations, including the Architect Engineer. The manager assigned the responsibility of managing and integrating the D-RAP Program had direct access to the ABWR Project Manager and kept him abreast of D-RAP critical items, program needs and status. He had organizational freedom to:

(1) Identify D-RAP problems.
(2) Initiate, recommend or provide solution to problems through designated organizations.
(3) Verify implementation of solution.
(4) Function as an integral part of the final design process.

The COL applicant completing its detailed design and equipment selection during the design phase, must submit its specific D-RAP organization for NRC review. See Subsection 17.3.13.2 for COL license information.

17.3.6 SSC Identification /Prioritization

The PRA prepared for the ABWR will be the primary source for identifying risk- significant SSCs that should be given special consideration during the detailed design and procurement phases and/or considered for inclusion in the O-RAP. The method by which the PRA is used to identify risk-significant SSCs is described in Chapter 19. It is also possible that some risk-significant SSCs will be identified from sources other than the PRA, such as nuclear plant operating experience, other industrial experience, and relevant component failure data bases.

17.3.7 Design Considerations

The reliability of risk-significant SSCs, which are identified by the PRA, will be evaluated at the detailed design stage by appropriate design reviews and reliability analyses. Current data bases will be used to identify appropriate values for failure rates of equipment as designed, and these failure rates will be compared with those used in the PRA. Normally the failure rates will be similar, but in some cases they may differ because of recent design or data base changes. Whenever failure rates of designed equipment are significantly greater than those assumed in the PRA, an evaluation will be performed to determine if the equipment is acceptable or if it must be redesigned to achieve a lower failure rate.

For those risk-significant SSCs, as indicated by PRA or other sources, component redesign (including selection of a different component) will be considered as a way to reduce the CDF contribution. (If the system unavailability or the CDF is acceptably low, less effort will be expended toward redesign.) If there are practical ways to redesign a risk-significant SSC, it will be redesigned and the change in system fault tree results will be calculated. Following the redesign phase, dominant SSC failure modes will be identified so that protection against such failure modes can be accomplished by appropriate activities during plant life. The design considerations that will go into determining an acceptable, reliable design and the SSCs that must be considered for O-RAP activities are shown in Figure 17.3-1.

GE-NE will identify in the PRA or other design documents to the plant owner/operator the risk-significant SSCs and the associated reliability assumptions, including any pertinent bases and uncertainties considered in the PRA. GE-NE will also provide information for the plant owner/operator to incorporate into the O-RAP to help assure that PRA results will be achieved over the life of the plant. This information can be used by the owner/operator for establishing appropriate reliability targets and the associated maintenance practices for achieving them.

17.3.8 Defining Failure Modes

The determination of dominant failure modes of risk-significant SSCs will include historical information, analytical models and existing requirements. Many BWR systems and components have compiled a significant historical record, so an evaluation of that record comprises Assessment Path A in Figure 17.3-2. Details of Path A are shown in Figure 17.3-3.

For those SSCs for which there is not an adequate historical basis to identify critical failure modes, an analytical approach is necessary, shown as Assessment

Path B in Figure 17.3-2. The details of Path B are given in Figure 17.3-4. The failure modes identified in Paths A and B are then reviewed with respect to the existing maintenance activities in the industry and the maintenance requirements, Assessment Path C in Figure 17.3-2. Detailed steps in Path C are outlined in Figure 17.3-5.

17.3.9 Operational Reliability Assurance Activities

Once the dominant failure modes are determined for risk-significant SSCs, an assessment is required to determine suggested O-RAP activities that will assure acceptable performance during plant life. Such activities may consist of periodic surveillance inspections or tests, monitoring of SSC performance, and/or periodic preventive maintenance (Ref. 1). An example of a decision tree that would be applicable to these activities is shown in Figure 17.3-6. As indicated, some SSCs may require a combination of activities to assure that their performance is consistent with that assumed in the PRA.

Periodic testing of SSCs may include startup of standby systems, surveillance testing of instrument circuits to assure that they will respond to appropriate signals, and inspection of passive SSCs (such as tanks and pipes) to show that they are available to perform as designed. Performance monitoring, including condition monitoring, can consist of measurement of output (such as pump flow rate or heat exchanger temperatures), measurement of magnitude of an important variable (such as vibration or temperature), and testing for abnormal conditions (such as oil degradation or local hot spots).

Periodic preventive maintenance is an activity performed at regular intervals to preclude problems that could occur before the next PM interval. This could be regular oil changes, replacement of seals and gaskets, or refurbishment of equipment subject to wear or age related degradation.

Planned maintenance activities will be integrated with the regular operating plans so that they do not disrupt normal operation. Maintenance that will be performed more frequently than refueling outages must be planned so as to not disrupt operation or be likely to cause reactor scram, ESF actuation, or abnormal transients. Maintenance planned for performance during refueling outages must be conducted in such a way that it will have little or no impact on plant safety, on outage length or on other maintenance work.

The COL applicant will provide a complete O-RAP to be reviewed by the NRC. See Subsection 17.3.13.3 for COL license information.

17.3.10 Owner/Operator's Reliability Assurance Program

The O-RAP that will be prepared and implemented by the ABWR owner/operator will make use of the information provided by GE-NE. This information will help the owner/operator determine activities that should be included in the O-RAP. Examples of elements that might be included in an O-RAP are:

1. Reliability Performance Monitoring: Measurement of the performance of equipment to determine that it is accomplishing its goals and/or that it will continue to operate with low probability of failure.

2. Reliability Methodology: Methods by which the plant owner/operator can compare plant data to the SSC data in the PRA.

3. Problem Prioritization: Identification, for each of the risk- significant SSCs, of the importance of that item as a contributor to its system unavailability and assignment of priorities to problems that are detected with such equipment.

4. Root Cause Analysis: Determination, for problems that occur regarding reliability of risk-significant SSCs, of the root causes, those causes which, after correction, will not recur to again degrade the reliability of equipment.

5. Corrective Action Determination: Identification of corrective actions needed to restore equipment to its required functional capability and reliability, based on the results of problem identification and root cause analysis.

6. Corrective Action Implementation: Carrying out identified corrective action on risk-significant equipment to restore equipment to its intended function in such a way that plant safety is not

compromised during work.

7. Corrective Action Verification: Post-corrective action tasks to be followed after maintenance on risk significant equipment to assure that such equipment will perform its safety functions.

8. Plant Aging: Some of the risk-significant equipment is expected to undergo age related degradation that will require equipment replacement or refurbishment.

9. Feedback to Designer: The plant owner/operator will periodically compare performance of risk-significant equipment to that specified in the PRA and D-RAP, as mentioned in item 1. above, and, at its discretion, may feedback SSC performance data to plant or equipment designers in those cases that consistently show performance below that specified.

10. Programmatic Interfaces: Reliability assurance interfaces related to the work of the several organizations and personnel groups working on risk-significant SSCs.

The plant owner/operator's O-RAP will address the interfaces with construction, startup testing, operations, maintenance, engineering, safety, licensing, quality assurance and procurement of replacement equipment.

17.3.11 D-RAP Implementation

An example of implementation of the D-RAP is given for the standby liquid control system (SLCS). The purpose of the SLCS is to inject neutron absorbing poison into the reactor, upon demand, providing a backup reactor shutdown capability independent of the control rods. The system is capable of operating over a wide range of reactor pressure conditions. The SLCS may or may not be identified by the final PRA as a significant contributor to CDF or to offsite risk. For the purpose of this example it is assumed that the SLCS is identified as a significant contributor to CDF or to offsite risk.

17.3.11.1 SLCS Description

During normal operation the SLCS is on standby, only to function in event the operators are unable to control reactivity with the normal control rods. The SLCS consists of a boron solution storage tank, two positive displacement pumps, two motor operated injection valves (provided in parallel for redundancy), and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV).

The borated solution is discharged through the 'B' high pressure core flooder (HPCF) subsystem sparger. A schematic diagram of the SLCS, showing major system components, is presented in Figure 17.3-7. Some locked open maintenance valves and some check valves are not shown. Key equipment performance requirements are:

a.	Pump flow	50 gpm per pump
b.	Maximum reactor pressure (for injection)	1250 psig
c.	Pumpable volume in storage tank (minimum)	6100 U.S. gal

Design provisions to permit system testing include a test tank and associated piping and valves. The tank can be supplied with demineralized water which can be pumped in a closed loop through either pump or injected into the reactor.

The SLCS uses a dissolved solution of sodium pentaborate as the neutron-absorbing poison. This solution is held in a heated storage tank to maintain the solution above its saturation temperature. The SLCS solution tank, a test water tank, the two positive displacement pumps, and associated valving are located in the secondary containment on the floor elevation below the operating floor. This is a Seismic Category I structure, and the SLCS equipment is protected from phenomena such as earthquakes, tornados, hurricanes and floods as well as from internal postulated accident phenomena. In this area, the SLCS is not subject to conditions such as missiles, pipe whip, and discharging fluids.

The pumps are capable of producing discharge pressure to inject the solution into the reactor when the reactor is at high pressure conditions corresponding to the system relief valve actuation. Signals indicating storage tank liquid level, tank outlet valve position, pump discharge pressure and injection valve position are available in the control room.

The pumps, heater, valves and controls are powered from the standby power supply or normal offsite power.

The pumps and valves are powered and controlled from separate buses and circuits so that single active failures will not prevent system operation. The power supplied to one motor operated injection valve, storage tank discharge valve, and injection pump is from Division I, 480 VAC. The power supply to the other motor-operated injection valve, storage tank outlet valve, and injection pump is from Division II, 480 VAC. The power supply to the tank heaters and heater controls is connectable to a standby power source. The standby power source is Class 1E from an on-site source and is independent of the off-site power.

All components of the system which are required for injection of the neutron absorber into the reactor are classified Seismic Category I. All major mechanical components are designed to meet ASME Code requirements as shown below.

Component	ASME Code Class	Design Conditions Pressure	Temperature
Storage Tank	2	Static Head	150°F
Pump	2	1560 psig	150°F
Injection Valves	1	1560 psig	150 °F
Piping Inboard of Injection Valves	1	1250 psig	575 °F

17.3.11.2 SLCS Operation

The SLCS is initiated by one of three means: (a) manually initiated from the main control room, (b) automatically initiated if conditions of high reactor pressure and power level not below the ATWS permissive power level exist for 3 minutes, or (c) automatically initiated if conditions of RPV water level below the level 2 setpoint and power level not below the ATWS permissive power level exist for 3 minutes. The SLCS provides borated water to the reactor core to introduce negative reactivity effects during the required conditions.

To meet its negative reactivity objective, it is necessary for the SLCS to inject a quantity of boron which produces a minimum concentration of 850 ppm of natural boron in the reactor core at 68 F. To allow for potential leakage and imperfect mixing in the reactor system, an additional 25% (220 ppm) margin is added to the above requirement. The required concentration is achieved accounting for dilution in the RPV with normal water level and including the volume in the residual heat removal shutdown cooling piping. This quantity of boron solution is the amount which is above the pump suction shutoff level in the storage tank thus allowing for the portion of the tank volume which cannot be injected.

17.3.11.3 Major Differences From Operating BWRs

The SLCS design is very similar to that of operating BWRs. Automatic actuation of the ABWR SLCS is similar to that incorporated in some operating BWRs. Because of the larger ABWR RPV volume, the pumping capacity has been increased from 43 to 50 gpm per pump. Injection of SLCS solution through the HPCF sparger has been shown by boron mixing tests to give better mixing than the operating plant injection through a standpipe.

Injection valves of operating plants are leak proof explosive valves to keep boron out of the reactor during SLCS testing. In the ABWR the injection valves are motor operated and a suction pipe fill system keeps the lines filled with distilled water at slightly higher pressure than that of the boron storage tank to preclude entry of boron into the reactor. The motor operated injection valves provide the following advantages over explosive valves:

a. Radiation exposure to personnel is potentially reduced during testing and maintenance because less work will be required at the valves.

b. Post-injection containment isolation capability is enhanced because the motor operated valves can be closed following boron injection. Explosive valves cannot be reclosed to provide containment isolation.

17.3.11.4 SLCS Fault Tree

The top level fault tree for the SLCS is shown in Figure 17.3-8, with the top gate defined as failure to deliver 50 gpm of borated water from the storage tank to the RPV. Details providing input to most of the events in Figure 17.3-8 are contained in the several additional branches to the fault tree.

It is assumed that the SLCS has been identified by the PRA as a system making significant contribution to CDF. A listing of the SLCS components or events by Fussell-Vesely Importance was made, and those SSCs with greatest importance are given in Table 17.3-1. No SSCs appear to be risk-significant because of aging or common cause considerations. The seven most significant components are listed in Table 17.3-2, so these SSCs should be considered as risk-significant candidates for O-RAP activities.

17.3.11.5 System Design Response

The seven SLCS risk-significant components identified in Table 17.3-2 as having high importance in the SLCS fault tree are now considered for redesign or for O-RAP activities, as noted above. The flow chart of Figure 17.3-1 guides the designer.

Two of the events in Table 17.3-2 result from flow of SLCS fluid being diverted through relief valves back to pump suction rather than into the RPV. Since gate and check valve failures (which could result in relief valve operation) are accounted for by separate events, the relief valve failures of concern can be considered to be valve body failures or inadvertent opening of the relief valves. Plugging of the suction lines from the storage tank could result from some contamination of the tank fluid or collection of foreign matter in the tank. The pump failures to start upon demand could result from electrical or mechanical problems at the pumps or their control circuits.

Two AC electrical system failures that contribute to SLCS system failure are identified in Table 17.3-2. No further details of electrical system failures or maintenance are included here. That leaves the five components noted above for special attention with regard to reducing the risk of system failure.

a. Redesign

The design evaluation of Figure 17.3-1 is used by the designer. The design assessment shows that the component failure rates are the same as those used in the PRA, so there is no need to recalculate the PRA. Also, no one SSC has a major impact on SLCS system unavailability, so redesign or reselection of components is not required and the seven components are identified for consideration by the O-RAP.

Redesign considerations, if they had been required, would have included trying to identify more reliable relief valves and pumps and suction lines less likely to plug. The latter might be achieved by using larger diameter pipes or multiple suction lines. Pump and valve reliability might be enhanced by specific design changes or by selection of a different component. Any such redesign would have to be evaluated by balancing the increase in reliability against the added complication to plant equipment and layout.

b. Failure Mode Identification

If redesign is not necessary, or after redesign has been completed, the appropriate O-RAP activities would be identified for the three SLCS component types identified by the fault tree and discussed above. This begins with determining the likely failure modes that will lead to loss of function, following the steps in Figure 17.3-2. The components of SLCS have adequate failure history to identify critical failure modes, so Assessment Paths A and C (Figures 17.3-3 and 17.3-5, respectively) would be followed to define the failure modes for consideration.

For the SLCS relief valves past experience with similar valves shows that the major failure modes are fluid leakage from the valve body and a spurious opening as result of failure of the spring, the spring fastener, the valve stem or the disk. Past pump failures fall into two general categories, electrical problems resulting in failure to start on demand and mechanical problems that cause a running pump to stop or fail to provide rated flow. The plugging of fluid lines generally results from presence of sediment or precipitation of compounds from saturated fluid.

Following the flow chart of Figure 17.3-3, the designer would determine more details about each failure mode, including pieceparts most likely to fail and the frequency of each failure mode category or piecepart failure. This would result in a list of the dominant failure modes to be considered for the O-RAP. ASME Section XI requirement for inservice inspection and other mandated inspections and test would be identified, as indicated in Figure 17.3-5.

Examples of the types of failure modes that could impact reliability of these identified components are shown in Table 17.3-3. The table is not a complete listing of important failure modes, but is intended to indicate the types of failures that would be considered.

c. Identification of Maintenance

Requirements

For each identified failure mode the appropriate maintenance tasks will be identified to assure that the failure mode will be (a) avoided, (b) rendered insignificant, or (c) kept to an acceptably low probability. The type of maintenance and the maintenance frequencies are both important aspects of assuring that the equipment failure rate will be consistent with that assumed for the PRA. As indicated in Figure 17.3-6, the designer would consider periodic testing, performance testing or periodic preventive maintenance as possible O-RAP activities to keep failure rates acceptable.

For the SLCS relief valves, which normally have no cycles during operation, A visual inspection for leakage and periodic inspections of internals are judged to be appropriate. The pumps can be functionally tested periodically for ability to start and run and vibration can be measured during functional tests to detect potential mechanical problems. Detailed disassembly, inspection and refurbishment would be done less frequently. To prevent line plugging the storage tank can be sampled for sediment and/or liquid saturation, with appropriate cleaning or temperature increase as necessary. Examples of maintenance activities and frequencies are shown in Table 17.3.3 for each identified failure mode. The D-RAP will include documentation of the basis for each suggested O-RAP activity.

17.3.12. Glossary of Terms

ATWS — Anticipated Transient Without Scram.

CDF — The core damage frequency as calculated by the PRA.

D-RAP — Design Reliability Assurance Program performed by the plant designer to assure that the plant is designed so that it can be operated and maintained in such a way that the reliability assumptions of the PRA apply throughout plant life.

Fussell-Vesely Importance — A measure of the component contribution to system unavailability. Numerically, the percentage contribution of component to system unavailability.

GE-NE — GE Nuclear Energy, ABWR plant designer.

Owner/Operator — The utility or other organization that owns and operates the ABWR following construction.

O-RAP — Operational Reliability Assurance Program performed by the plant owner/operator to assure that the plant is operated and maintained safely and in such a way that the reliability assumptions of the PRA apply throughout plant life.

Piecepart — A portion of a (risk-significant) component whose failure would cause the failure of the component as a whole. The precise definition of a "piecepart" will vary between component types, depending upon their complexity.

PRA — Probabilistic risk assessment performed to identify and quantify the risk associated with the ABWR.

Risk-Significant — Those SSCs which are identified as contributing significantly to the system unavailability.

SSCs — Structures, systems and components identified as being important to the plant operation and safety.

17.3.13 COL License Information

17.3.13.1 Policy and Implementation Procedures for D-RAP

The COL applicant will specifiy the policy and implementation procedures for using D-RAP information. (See Subsection 17.3.1)

17.3.13.2 D-RAP Organization

The COL apllicant completing its detailed design and equipment selection during the design phase, must submit its specific D-RAP organization for NRC review. (See Subsection 17.3.5)

17.3.13.3 Provision for O-RAP

The COL applicant will provide a complete O- RAP to be reviewed by the NRC. (See Subsection 17.3.9)

17.3.14 Reference

(1) E.V. Lofgren, et. al., *A Process for Risk- Focused Maintenance*, SAIC, NUREG/CR-5695, March 1991.

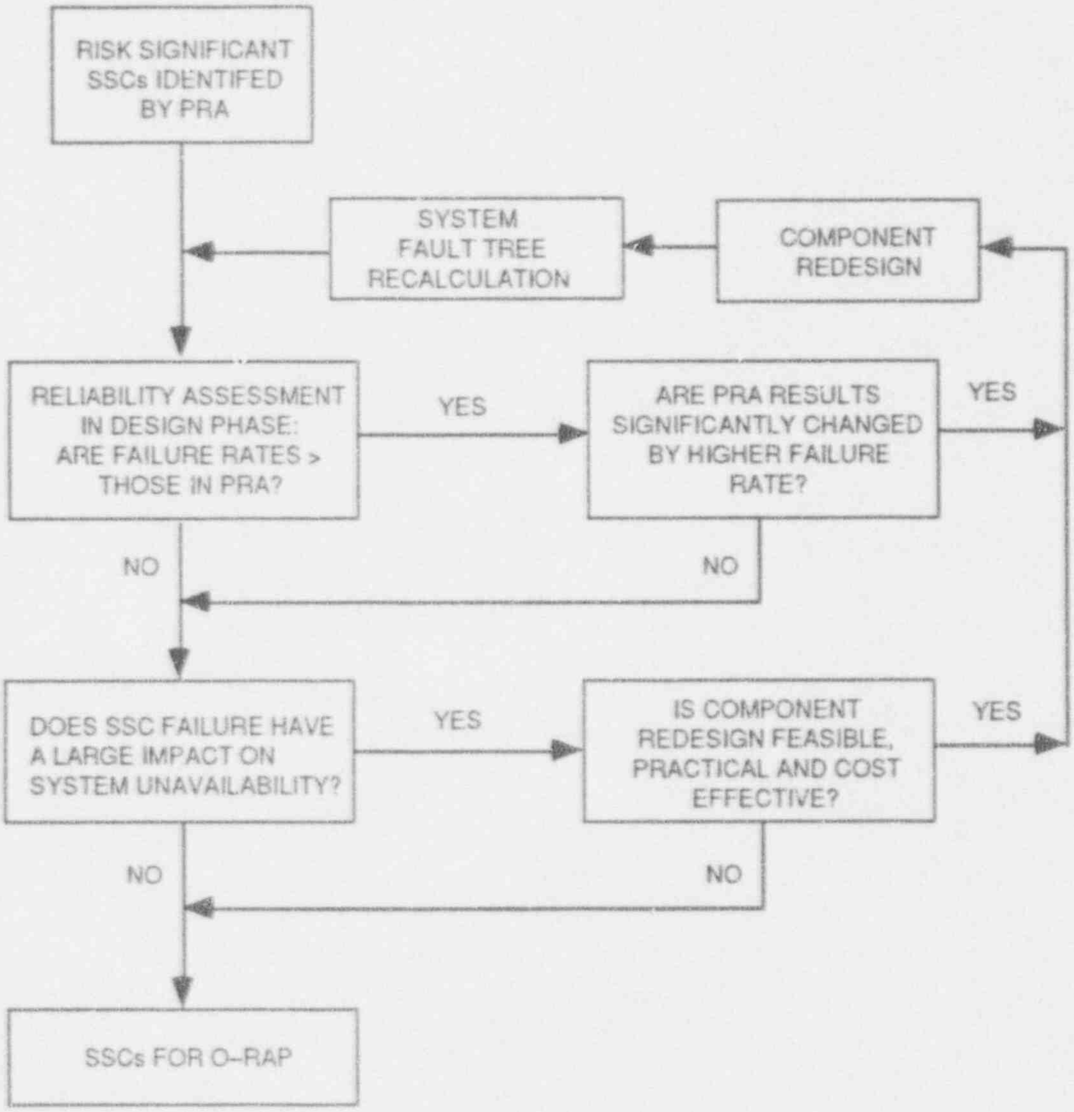

Figure 17.3–1. Design Evaluations for SSCs

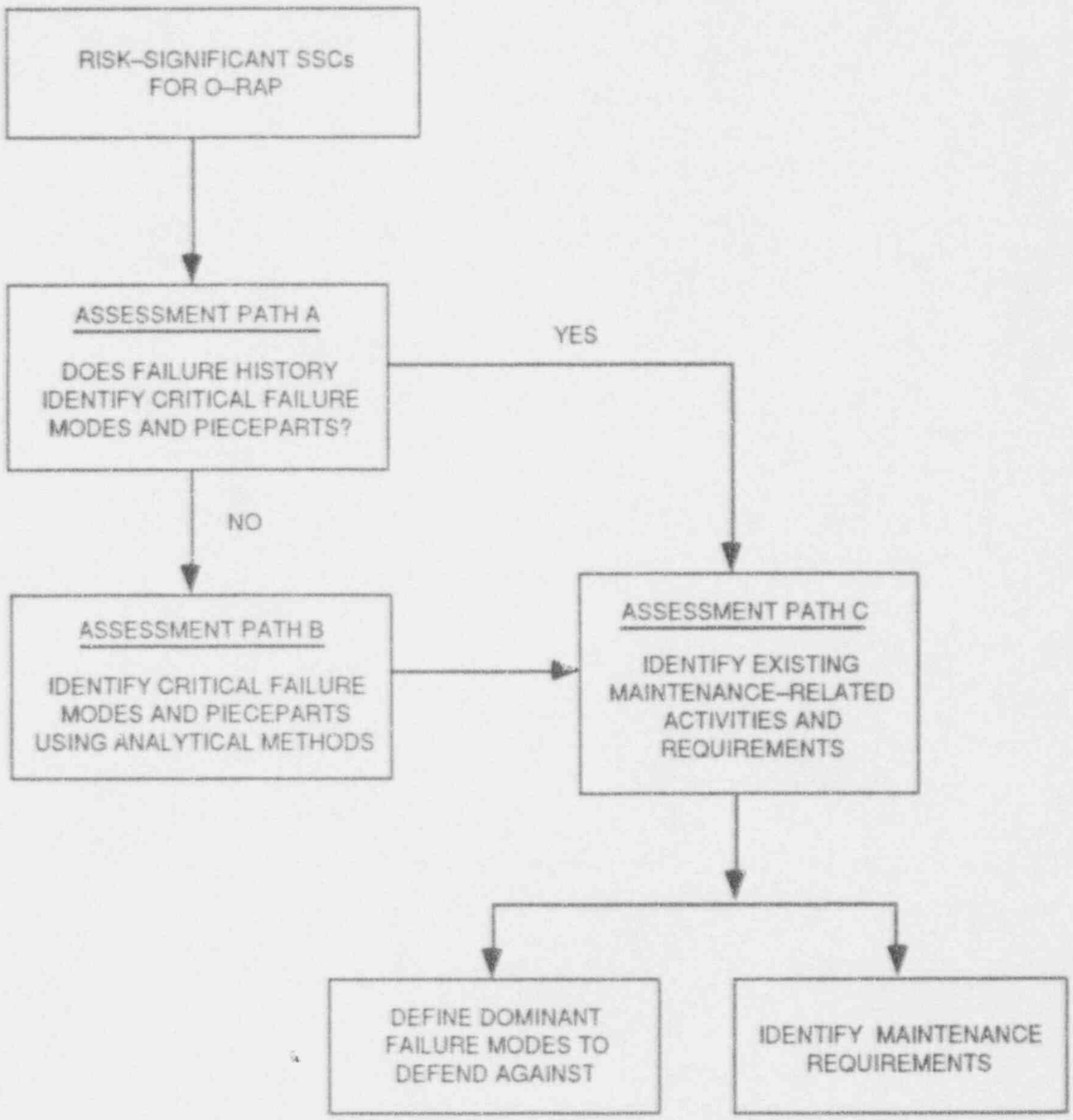

Figure 17.3–2. Process for Determining Dominant Failure Modes of Risk–Significant SSCs

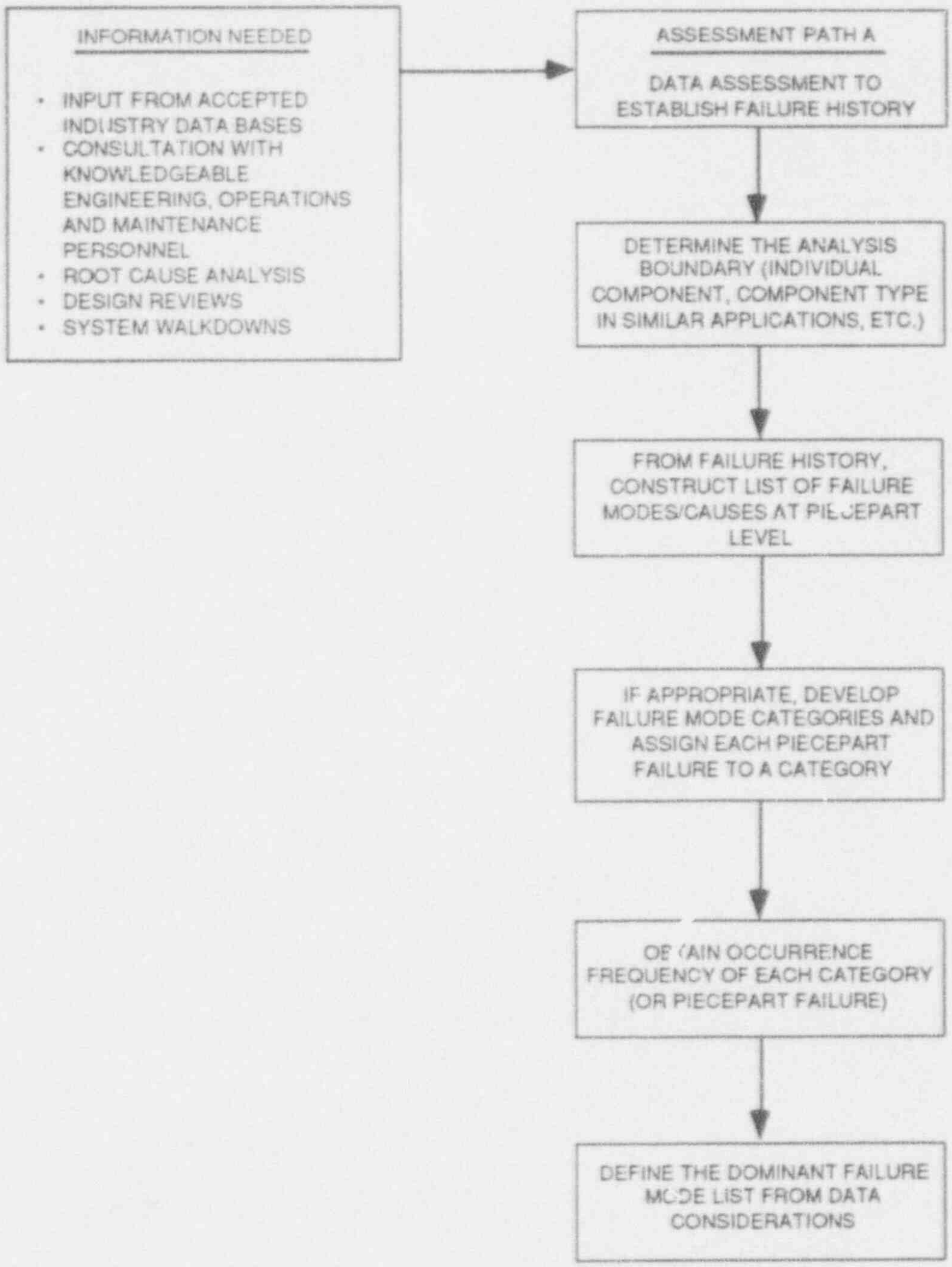

Figure 17.3-3. Use of Failure History to Define Failure Modes

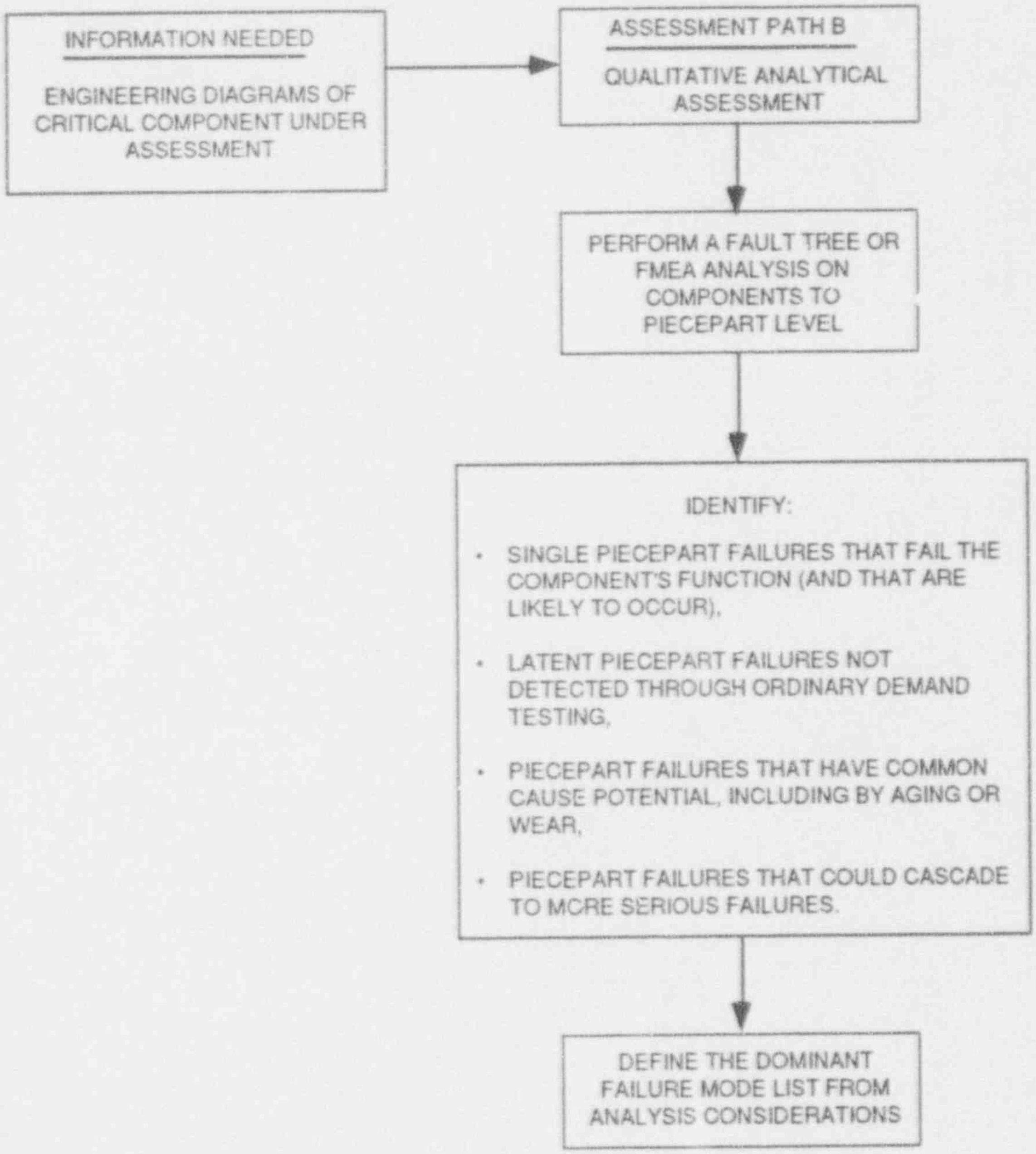

Figure 17.3–4. Analytical Assessment to Define Failure Modes

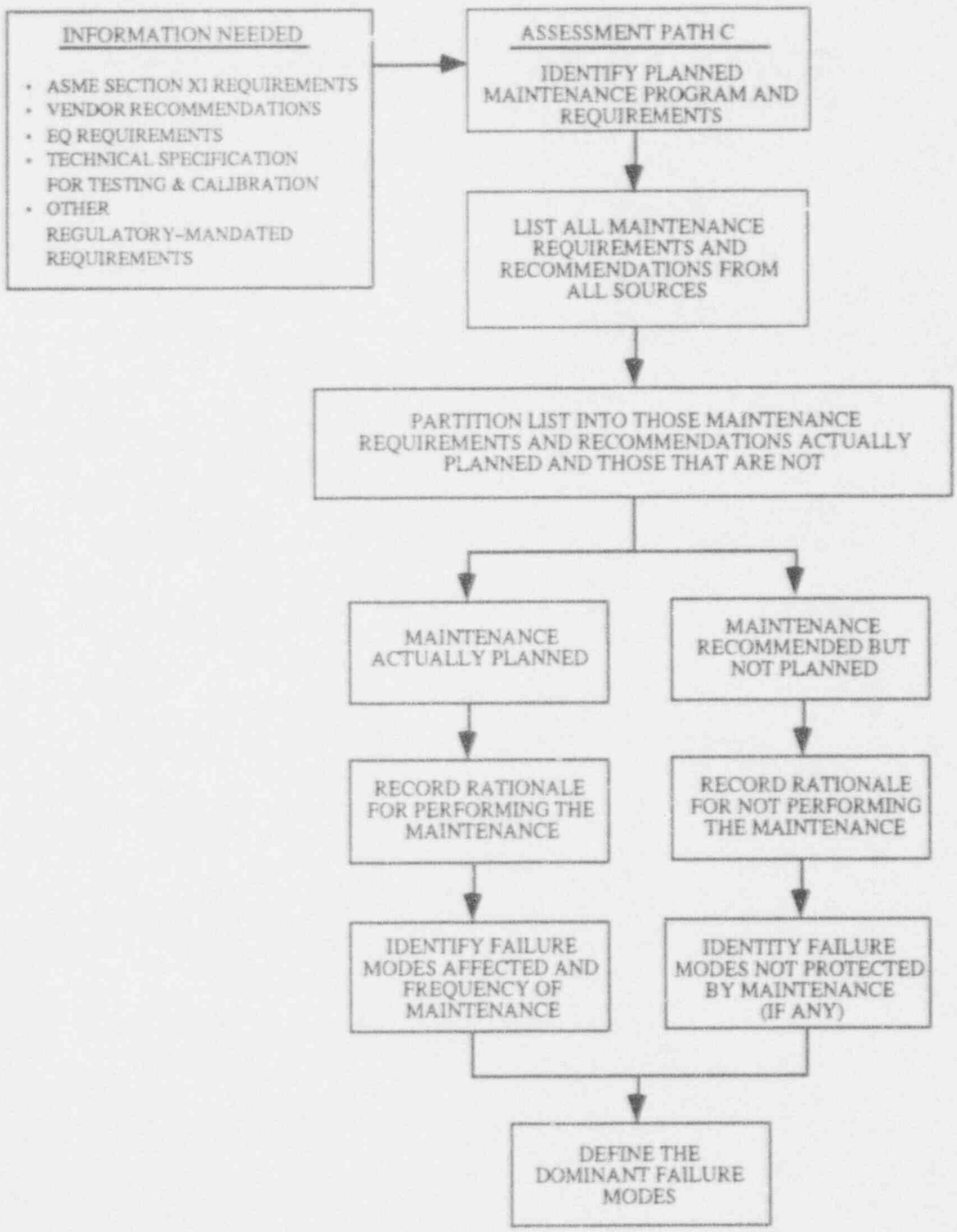

Figure 17.3–5. Inclusion of Maintenance Requirements in the Definition of Failure Modes

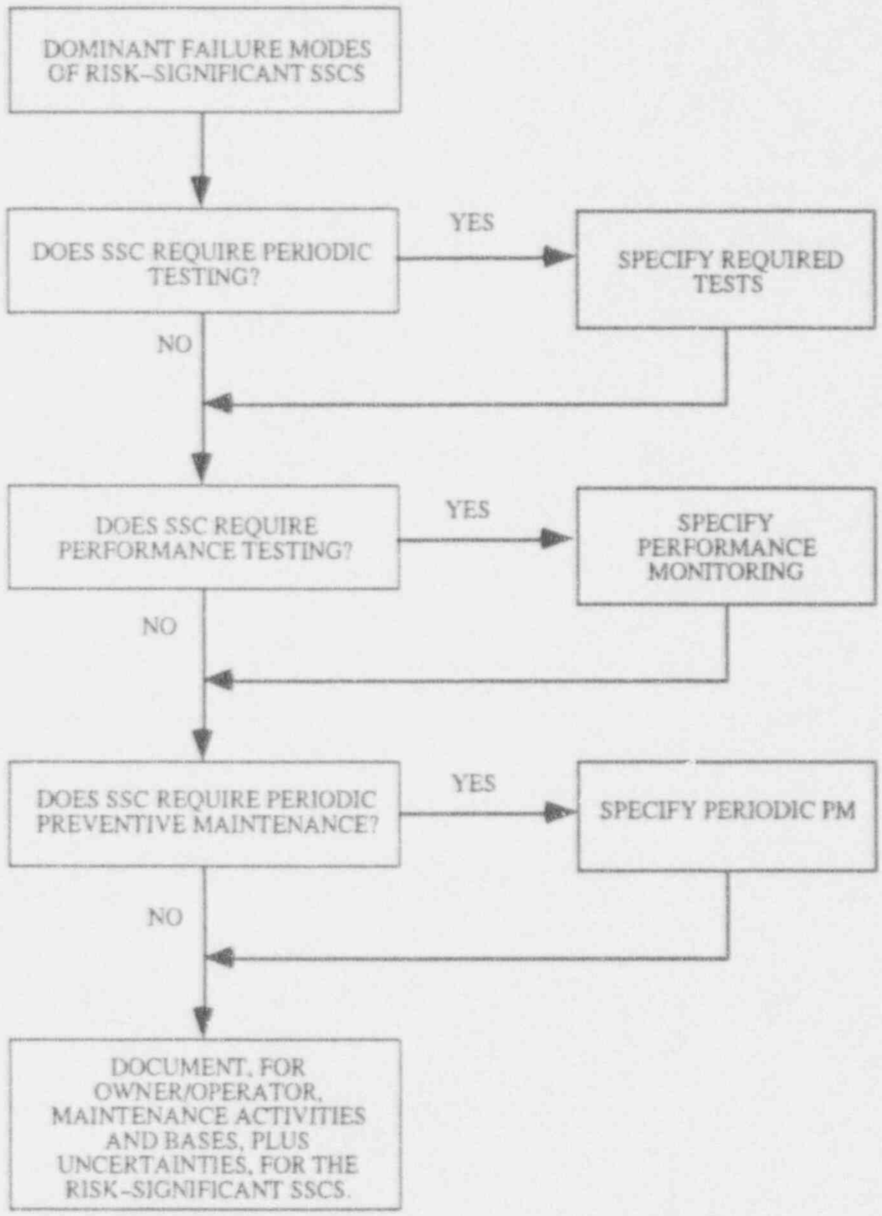

Figure 17.3–6. Identification of Risk–Significant SSC O–RAP Activities

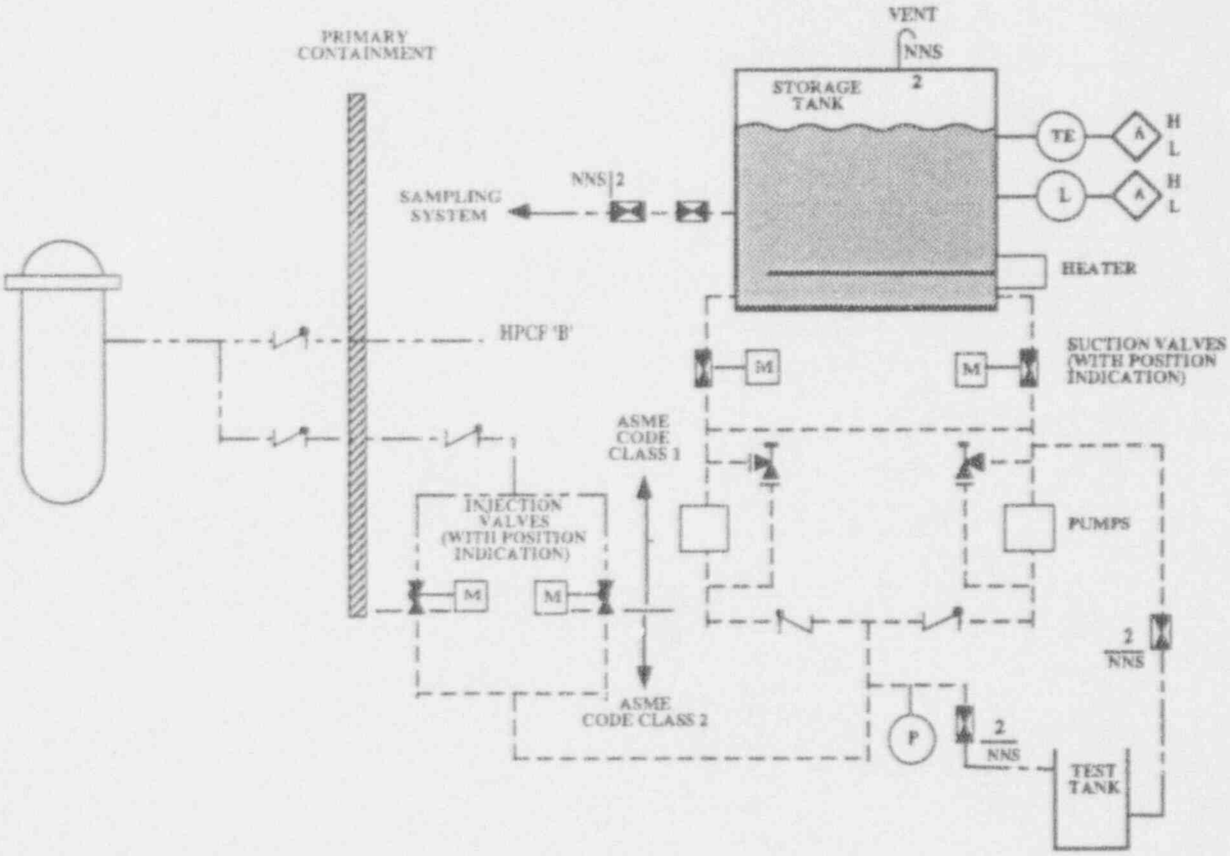

Figure 17.3–7. Standby Liquid Control System (Standby Mode)

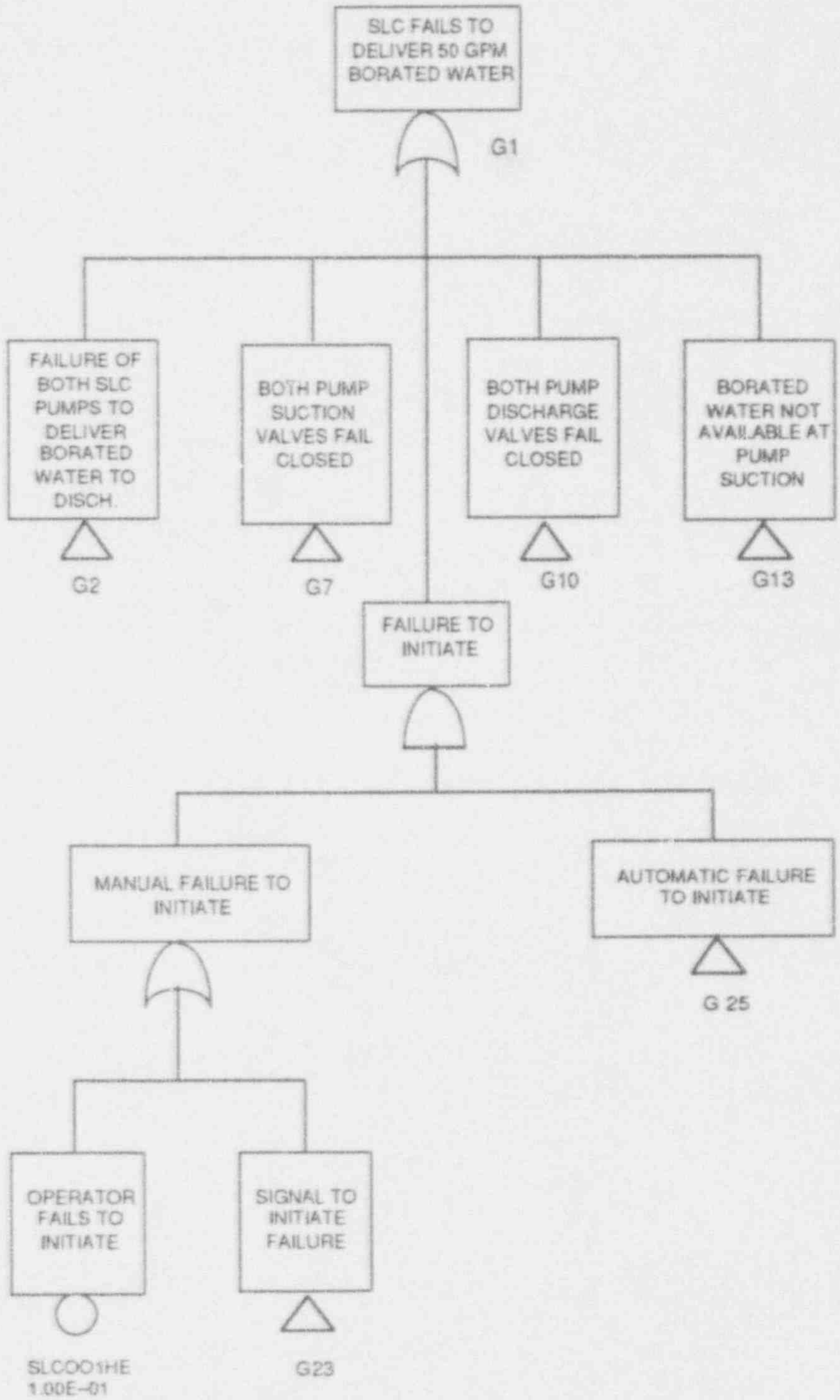

Figure 17.3-8. Standby Liquid Control System Top Level Fault Tree

18.4 CONTROL ROOM STANDARD DESIGN FEATURES

18.4.1 Introduction

This section presents, in Subsection 18.4.2, the standard design features of the HSI in the control room. These basic design features are based upon proven technologies and have been demonstrated, through broad scope control room dynamic simulation tests and evaluation, to satisfy the ABWR operator interface design goals and design bases as given in Section 18.2. The specific technologies utilized in the main control room HSI are listed in Subsection 18.4.3. Appendix 18C presents an example of a control room HSI design implementation which incorporates these design features. Validation of the implemented main control room design will include evaluation of the standard design features and will be performed as part of the design implementation process as defined by the acceptance criteria presented in Tables 18E.2-1 through 18E.2-4.

18.4.2 Standard Design Feature Descriptions

18.4.2.1 Listing of Features

The ABWR control room HSI design incorporates the following standard features:

a. A single, integrated control console staffed by two operators; the console has a low profile such that the operators can see over the console from a seated position.

b. The use of plant process computer system driven on-screen control video display units (VDUs) for safety system monitoring and non-safety system control and monitoring.

c. The use of a separate set of on-screen control VDUs for safety system control and monitoring and separate on-screen control VDUs for non-safety system control and monitoring; the operation of these two sets of VDUs is entirely independent of the process computer system. Further, the first set of VDUs and all equipment associated with their functions of safety system control and monitoring are divisionally separate and qualified to Class 1-E standards.

d. The use of dedicated function switches on the control console.

e. Operator selectable automation of pre-defined plant operation sequences.

f. The incorporation of an operator selectable semi-automated mode of plant operations, which provide procedural guidance on the control console VDUs.

g. The capability to conduct these all plant operations in an operator manual mode.

h. The incorporation of a large display panel which presents information for use by the entire control room operating staff.

i. The inclusion on the large display panel of fixed-position displays of key plant parameters and major equipment status.

j. The inclusion in the fixed-position displays of both 1E-qualified and non-1E display elements.

k. The independence of the fixed-position displays from the plant process computer.

l. The inclusion within the large display panel of a large video display unit which is driven by the plant process computer system.

m. The incorporation of a "monitoring only" supervisor's console which includes VDUs on which display formats available to the operators on the main control console are also available to the supervisors.

n The incorporation of the safety parameter display system (SPDS) function as part of the plant status summary information which is continuously displayed on the fixed-position displays on the large display panel.

o. The use of fixed-position alarm tiles on the large display panel.

p. The application of alarm processing logic to prioritize alarm indications and to filter unnecessary alarms.

q. A spatial arrangement between the large display panel, the main control console and the shift supervisors' console which allows the entire control room operating crew to conveniently view the information presented on the Class 1E large display panel.

r. The use of VDUs to provide alarm information in addition to the alarm information provided via the fixed-position alarm tiles on the large display panel.

The remainder of this subsection provides further descriptions of these standard design features.

18.4.2.2 Main Control Console

The main control console comprises the work stations for the two control room plant operators. It is configured such that each operator is provided with controls and monitoring information necessary to perform their assigned tasks and allows the operators to view all of the displays on the large display panel (see Subsection 18.4.2.7) from a seated position.

The main control console, in concert with the large display panel, provides the controls and displays required to operate the plant during normal plant operations, abnormal events and emergencies. These main control console controls and displays include the following:

1. On-screen control VDUs for safety system monitoring and non-safety system control and monitoring which are driven by the plant process computer system (see Subsection 18.4.2.3).

2. A separate set of on-screen control VDUs for safety system control and monitoring and separate on-screen control VDUs for non-safety system control and monitoring; the operation of these two sets of VDUs is entirely independent of the process computer system. Further, the first set of VDUs and all equipment associated with their functions of safety system control and monitoring are divisionally separate and qualified to Class 1E standards (see Subsection 18.4.2.4).

3. Dedicated function switches (see Subsection 18.4.2.5).

The main control console is also equipped with a limited set of dedicated displays for selected functions (e.g., the standby liquid control system and the synchronization of the main generator to the electrical grid).

In addition to the above equipment, the main control console is equipped with both intra-plant and external communications equipment and a laydown space is provided for hard copies of procedures and other documents required by the operators during the performance of their duties.

18.4.2.3 Process Computer Driven VDUs

A set of on-screen control VDUs is incorporated into the main control console design to support the following activities:

1. monitoring of plant systems, both safety and non-safety related,

2. control of non-safety system components,

3. presentation of system and equipment alarm information,

This set of VDUs is driven by the plant process computer system. Thus, data collected by the process computer is available for monitoring on these VDUs. All available display formats can be displayed on any of these VDUs.

18.4.2.4 Process Computer Independent VDUs

A set of VDUs which are independent of the process computer are also installed on the main control console. These VDUs are each driven by independent processors. They are divided into two subsets:

The first subset consists of those VDUs which are dedicated, divisionally separated devices. The VDUs in this group can only be used for monitoring and control of equipment within a given safety division. The VDUs are qualified, along with their supporting display processing equipment, to Class 1E standards.

The second subset of process computer independent VDUs are used for monitoring and control of non-safety plant systems. The VDUs in this subset are not qualified to Class 1E equipment standards.

18.4.2.5 Dedicated Function Switches

Dedicated function switches are installed on the main control console. These devices provide faster access and feedback compared to that obtainable with soft controls. These dedicated switches are implemented in hardware, so that they are located in a fixed-position and are dedicated in the sense that each individual switch is used only for a single function, or two very closely related functions (e.g., valve open/close).

The dedicated function switches on the main control console are used to support such functions as initiation of automated sequences of safety and non-safety system

TABLE 18B-1 (Cont'd)

DIFFERENCES BETWEEN BWROG EPG REVISION 4 AND ABWR EPG

ABWR EPG STEP	BWROG EPG REV. 4 STEP	DIFFERENCES FROM BWROG REV. 4 EPG	BASIS FOR DIFFERENCES
RC/Q-7.2	RC/Q-7.2	• Deleted the following phrases: " drain the scram discharge volume, ", "Increase CRD cooling water differential pressure", and "Vent control rod drive overpiston volumes".	• These steps are applicable to the conventional hydraulic locking-piston drives and are not applicable to the ABWR FMCRDs.
Primary Containment Control Entry Conditions	Primary Containment Control Entry Conditions	• Deleted the phrase: "Containment temperature above [90 °F(containment temperature LCO)]".	• This entry condition is applicable only to BWRs with Mark III containments. Refer to basis for deleting the CN/T subsection given below.
–	CN/T	• Deleted entire section.	• The control functions specified in this section, operation of containment cooling, initiation of suppression pool sprays, and performing an RPV depressurization when containment temperature cannot be maintained below a prescribed limit, are control functions that are already specified in subsection SP/T of the ABWR EPGs. Subsection CN/T of the BWROG EPGs is developed specifically for the BWR/6 Mark III containment where temperature can be controlled by the previously stated control functions. The ABWR containment, although it incorporates the concept of a Mark III suppression pool, is analogous to a Mark II BWR containment for the purpose of controlling the wetwell space temperature.
DW/T, DW/T-1	DW/T, DW/T-1	• Added phrase: "shutdown the reactor" at the end of Step DW/T, and added a second part to DW/T-1, "When drywell temperature cannot be maintained below [103 °C (Saturation temperature corresponding to high drywell pressure scram setpoint)], enter [procedure developed from the RPV Control Guideline] at [Step RC-1] and execute it concurrently with this procedure".	• The BWROG EPGs assumed that other plant procedure steps will shutdown the plant at the Technical Specification LCO limit. Adding the instruction to shutdown the reactor allows shutdown by running back the recirculation pumps and inserting control rods, and then proceed to scram the reactor as specified in the second paragraph of Step DW/T-1. Adding these steps does not change the intent of the EPGs and makes DW/T consistent with the other primary containment sections.

TABLE 18B-1 (Cont'd)

DIFFERENCES BETWEEN BWROG EPG REVISION 4 AND ABWR EPG

ABWR EPG STEP	BWROG EPG REV. 4 STEP	DIFFERENCES FROM BWROG REV. 4 EPG	BASIS FOR DIFFERENCES
DW/T-2	DW/T-2	• Replaced phrase, "elevation of bottom of internal suppression chamber to drywell vacuum breakers less vacuum breaker opening pressure in feet of water", with the phrase, "elevation of the bottom of suppression pool-to-lower-drywell vent".	• In the ABWR containment, vents are provided connecting the upper drywell to the lower drywell. When the wetwell-to-drywell vacuum breakers open, flow is from the wetwell to the lower drywell and then from the lower drywell to the upper drywell through these vents. The vaccum breakers are located above the vents. Water can also spill to the lower drywell from the suppression pool if pool level reaches the vents. Water can also flow from the lower drywell to the suppression pool if lower drywell is flooded to the elevation of thse vents. For these reasons, it is appropriate to spray the drywell only when suppression pool water level is below the bottom of the upper drywell to lower drywell vents to preclude drywell differential pressure capability to be exceeded.
		• Deleted phrase "recirculation pumps" from instruction to shutoff recirculation pumps and drywell cooling fans prior to drywell spray initiation.	• The ABWR has internal recirculation pumps, driven by motors located below the RPV in the lower portion of the drywell. Drywell spray only sprays the upper portion of the drywell. An explicit instruction to shut down the recirculation pumps is not required.
		• Specify RHR pumps used for drywell spray as "RHR subsystems B and C".	• RHR subsystems B and C provide drywell and wetwell spray capability. Initiation of sprays is by manual control action. It is possible to initiate spray when RHR B or C is operating in other modes by opening spray valves.
		• Specified the use of the firewater addition system if RHR(B) and RHR(C) are not available for drywell sprays.	• The firewater addition system is described in Subsection 5.4.7.1.1.10. The specific purpose of the fire addition system is to provide makeup to the RPV to extend the station blackout capability of the ABWR, but it can be used for drywell and wetwell sprays if no other systems are available for sprays.

of the M-MIS. Then, top level requirements for reliability and maintainability procedures, verification and validation procedures, and configuration control procedures are addressed.

(7) Operator Team; Chapter 10, Section 3.1.2, Rev. A

The M-MIS shall be designed to eliminate task requirements that are difficult for members of the operator team to consistently satisfy; task responsibilities shall be distributed evenly so that some members of the operator team are not overburdened while others are idle; and task responsibilities shall be assigned to different positions within the operator team to provide career opportunities while remaining compatible with personnel qualification and training requirements.

Engineering Rationale

Task requirements, or what members of the operator team are required to do, are determined by system and organization functions. System functions dictate what and how well something has to be done to safely operate the plant. Organizational functions dictate what has to be done to plan, coordinate, and supervise M-MIS operations. Both sets of functions will be considered when making decisions about how functions will be allocated between men an machines and when defining task requirements, along with related staffing levels and training requirements. A survey and analysis of work structures in nuclear power plants is provided in EPRI NP-3141.

(8) Data Processing; Chapter 10, Sections 3.1.3.1 and 3.1.3.2, Rev. A

Data processing shall be performed on a plant wide, common data base which shall contain all of the data needed to produce integrated displays, alarms, and controls for members of the operator team as well as all of the information needed for automatic controls, including the engineered safety feature (ESF) system.

Engineering Rationale

The M-MIS performance objective can only be achieved when a comprehensive data base is provided. A comprehensive data base will permit individual signals to be validated before subsequent processing and minimize the potential for providing conflicting information among different elements of the M-MIS.

Section 3.1.3.2

Signal validation shall be a central and important part of data processing. Signal validation shall be performed for all incoming signals affecting monitoring, controlling, and protection functions. The results of signal validation shall be attached to the process variable in the form of a data quality code. The code shall be used to determine the quality of subsequent derived variables. Operators shall be able to determine the quality of data when needed and shall be alerted to any data for which the quality is suspect.

Engineering Rationale

Signal validation is very important if faulty sensors, cabling, or processing are to be rapidly detected and corrected. This requirement minimizes the possible selection of a control action that is based upon invalid or questionable process variables. Also, salient indication of poor quality process data values will facilitate early elimination of the source of the faulty data and shall improve the operator's ability to make effective decisions.

(9) Codes, Standards, and Requirements; Chapter 10, Section 3.2.1, Rev. A

The M-MIS design shall meet all current NRC requirements. Consideration shall also be given to good design and implementation practices, as outlined in Table 10.3-1, by IEEE, ANS, IEC, and ANSI standards.

Engineering Rationale

Adherence to current NRC requirements will increase the probability of achieving a licensable design. Use of the various industry standards as a design basis will address the technical considerations of various professional societies and lead to a degree of standardization.

ABWR Resolution

An on-going program for the design of instrumentation and control systems, and man-machine interface incorporates all the stated ALWR human factors engineering requirements. The design bases, approach, and acceptance criteria are given in Chapter 18 of the SSAR. In addition, a COL license information requirement is included in Section 19B.3.8 to ensure the establishment of an interdisciplinary design review group and reviews for site specific design and construction work. Therefore, this issue is resolved for the ABWR.

19B.2.26 Maintenance and Surveillance Program

Maintenance and Surveillance Program [HF.02.1][HF8]

Issue Summary

The NRCs current regulatory approach to nuclear power plant maintenance is concentrated on: (1) quality assurance ring design, construction, and operation for safety related structures, systems and components (10CFR50, Appendix B), and (2) surveillance requirements to assure that the necessary availability and quality of such systems and components is maintained (10CFR50.36). NRC additionally requires reporting of significant events (10CFR50.72), including personnel errors and procedural inadequacies which could prevent fulfillment of safety functions and allow exceeding of technical specification limits. However, the NRC is concerned that their rules and regulations provide no clear programmatic treatment of maintenance.

The maintenance and Surveillance Plan is intended to integrate the NRCs effort to assure effective nuclear power plant maintenance. The program is to address the problems and issues which exist and to propose development of alternative NRC approaches to regulating nuclear utility maintenance activities. The scope of the program includes all aspects of maintenance required to carry out a systematic maintenance and surveillance program. It includes conventional maintenance and repair plus such things as surveillance and test activities, equipment isolation, post-maintenance testing, independent verification, maintenance management, administrative control, personnel selection and training, procedures, and technical documentation.

NRC Resolution Summary

The maintenance and surveillance program is resolved by the Commission Published Final Policy Statement on Maintenance of Nuclear Power Plants on March 23, 1988. No additional requirements were warranted.

ABWR Resolution

The NRC resolution being a regulatory impact is not an LWR design issue. Therefore, this issue should be dropped for the ABWR.

19B.2.27 Seismic Design Criteria

Seismic Design Criteria [A-40]

Issue Summary

The current seismic design sequence includes many conservative factors. Certain aspects of the sequence may not be conservative for all plant sites. At present, it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner, this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving current seismic design requirements and NRRs capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

NRC Resolution

This issue was resolved by the publication of Revision 2 to SRP Section 2.5.2, 3.7.1, 3.7.2 and 3.7.3 in August 1989.

SECTION 19F.2

CONTENTS

SECTION 19F.2

TABLES

ILLUSTRATIONS

19J.4 OPERATOR ACTIONS

Three types of operator actions are critical to the SCETs. The first is the use of the firewater system to mitigate the effects of the accident. The second is the manual initiation of the RHR system after some loss of power cases. The third case is the closure of the RHR suction valves after the earthquake causes a pipe break in this system.

19J.4.1 Firewater Injection Initiation

The firewater system is used in two ways in this analysis. The first mode is to inject water into the vessel following loss of all core cooling. Except in cases where the RCIC is initially available there is a limited period of time available to begin injection into the vessel, about 20 minutes to diagnose and begin firewater injection. Based on this time a value of 0.9 was selected for the combined reliability of the operator and the system to perform this action.

Secondly, for the cases where the RCIC is initially available, about 8 hours is available for the operator to prepare to initiate the firewater system to prevent core damage. Under these conditions the operator is aware that the RCIC may not operate for more than 8 hours, so he would prepare to initiate the firewater system when required. This would include ensuring that pumping power was available by acquiring a fire truck if necessary. Therefore, a value of 0.999 was used for the operator reliability under these circumstances (See Subsection 19K.4.2.5). The availability of the firewater system to inject if called upon is based on the results from the seismic accident events analysis.

The third type of firewater use is that which follows a loss of all core cooling where the operator failed to begin the firewater addition system before vessel failure occurred. In this case the firewater system is to be used in drywell spray mode. Several hours are allowed for this action. Using the logic about for vessel injection where the RCIC was initially available, a value of 0.999 was used for the operator reliability. The firewater system availability is based on the seismic accident events analysis.

19J.4.2 Power Transformer Bypass

A mechanism for loss of core cooling following an earthquake is the loss of the divisional 480V AC power or the divisional 120V AC power. The weakest elements in these systems are identified in the seismic fault tree, Figure 19I.2-7, as the transformer, ETR6C1, with median capacity 1.5g and the inverter, EIV0F1H, with median capacity 1.3g. These elements are required to power initiation of the ECC and RHR systems. However, it is possible to manually initiate either of these systems as described below.

The first step that the operator should take is to manually open the required valves for the system to operate. In the case of the RHR system, the operator should open the injection valves of the RHR system and the appropriate service water valves. These actions would normally be provided automatically using 120V AC.

After the valves are properly aligned, the operator should manually close the breakers to provide power to the pumps. This action, normally initiated automatically using 480V AC may be performed from the remote shutdown panel if necessary. The pumps then run directly from the 6.9kV power.

The actions required of the operators to perform these actions are very similar to those to initiate the firewater addition system described above. Therefore, the reliability used for the firewater system is also used for this transformer bypass. For the same reason, no credit is taken for the firewater system if the operator fails to perform the transformer bypass.

19J.4.3 RHR Isolation to Prevent Suppression Pool Drain

If the RHR heat exchanger anchorage fails, it is possible for a pipe to break, allowing flow from the suppression pool to the pump room. If power is available the operator has sufficient means and indication to isolate the RHR system, preventing the suppression pool from being drained. The reliability of the operator to isolate the RHR is discussed in the seismic event tree analysis, Subsection 19I.4.2.6.

(d) Clarify whether primary containment purging during normal plant operation when required to limit the discharge of contaminants to the environment will always be through the SGTS (See SSAR Section 6.5.1.2.3.3). Clarify whether such a release prior to the purge system isolation has been considered in the LOCA dose analysis.

(e) Provide the compliance status tables referred to in Items (a) and (c) above for the control room ESF filter trains. (The staff notes that you have committed to discuss control room ESF filter system under SSAR Section 9.4.1. However, since evaluation of the control room habitability system cannot be completed until the information identified above is provided, the above information is requested now.)

(f) Identify the applicable interface requirements for the SGTS and the control room ESF atmosphere cleanup system.

430.56

Regarding Fission Product Control Systems and Structures (6.5.3)

(a) Provide the drawdown time for achieving a negative pressure of 0.25 inch water gauge for the secondary containment with respect to the environs during SGTS operation. Clarify whether the unfiltered release of radioactivity to the environs during this time for a postulated LOCA has been considered in the LOCA dose analysis. (Note that the unfiltered release need not be considered provided the required negative pressure differential is achieved within 60 seconds from the time of the accident.)

(b) Provide justification (See SRP Section 6.5.3, II.4) for the decontamination factor assumed in SSAR Table 6.5-2 and 15.6-8 for iodine in the suppression pool, correct the elemental, particulate and organic iodine fractions given in the tables to be consistent with RG 1.3, and incorporate the correction in the LOCA analysis tables. Alternatively, taking no credit for iodine retention in the suppression pool, revise the LOCA analysis tables. Note that the revision of the LOCA analysis tables (this also includes the control room doses) mentioned above is strictly in relation to the iodine retention factor in the suppression pool (also, there may be need for revision of other parameter(s) given in the tables and these will be identified under the relevant SRP Sections questions).

(c) Identify the applicable interface requirements.

430.57

Regarding SSAR Section 6.7, the staff notes that the Nitrogen Supply System has been discussed under this section, instead of the Main Steam Isolation Valve Leakage Control System (MSIV-LCS) as required by the Standard Format for SARs. The staff will review the material presented in SSAR Section 6.7 along with the material that will be presented in SSAR Section 9.3.1.

Regarding MSIV-LCS, the staff notes that you are committed to provide a non-safety related MSIV leakage processing pathway consistent with those evaluated in NUREG-1169, "Resolution of Generic Issue C-8," August 1986. Since the staff has not finalized its position so far on the acceptability of the NUREG findings with regard to the design of the MSIV-LCS, provide pertinent information on the system design including interface requirements to evaluate the to-be-proposed design against the acceptance criteria of SRP 6.7. (6.7)

440.75

In the ABWR design, the HPCF is tested by taking suction from and returning water to the suppression pool. Normally the suppression pool water is a lower quality than that of the CST; therefore, draining, flushing and refilling the system is required prior to returning the system to standby after testing. Please discuss the pros and cons of using the CST for testing the HPCF system. (6.3)

440.76

Address the following TMI-2 action items related to ECCS. (6.3)

(a) II.K.1.5

(b) II.K.1.10

(c) II.K.3.17

(d) II.K.3.18

(e) II.K.3.21

(f) II.K.3.25

(g) II.K.3.30

(h) II.K.3.31

440.77

Confirm that the HPCF system meets the guidelines of Regulatory Guide 1.1 regarding pump Net Positive Suction Head (NPSH). (6.3)

440.78

SRP 6.3 identifies GDCs 35, 36, and 37 in the acceptance criteria. Confirm that the HPCF system, described in Chapter 6.3 of the SSAR, meets the requirements of the above GDCs. (6.3)

440.79

Normally, the HFCF pump takes suction from the Condensate Storage Tank (CST). But, the CST is not seismically qualified or safety related. Confirm that the system piping and level transmitters, which interface with CST, will be designed and installed such that the automatic switchover to the suppression pool takes place without failure. (6.3)

440.80

What is the minimum quantity of water required in the condensate storage tank (CST) for HPCF operation? Give the basis for the required quantity of water in the CST. (6.3)

20.2.15 Chapter 15 Questions

420.28

Section 15.A.2.2 defines "Safety" and "Power Generation." The staff did not locate definitions for "important to safety" and "safety related" which are used in Chapter 7. (15A)

420.96

The safety system auxiliaries (Figure 15A.6-1) should be modified to include any HVAC required to assure continued operation of the electronics. (15A.6)

420.118

Describe when appropriate operator action in seconds is required to prevent significant radiological impact. (15.2.4.5.1)

420.122

Is the instrumentation required for the operator to verify bypass valve performance and relief valve operation 1E or N-1E? (15.2.2.2.1.4)

420.123

SSAR 15B.4 describes the essential multiplexing system (EMS) in some detail. SSAR Figure 7A.2-1 states that the design is not limited to this configuration. It is our understanding that the EMS design is still in a preliminary design stage. Is SSAR 15B.4 still accurate and is the design limited to that configuration? (15B4)

420.124

The FMEA submitted in SSAR 15B.4 is inadequate for a safety evaluation supporting the design certification. The FMEA appears to the staff to be oversimplified with one line item each for component failures and does not address potential software complications. The staff requests clarification of how this FMEA was developed given that the system design has not been finalized. The staff also believes that software failures need to be evaluated. The failure modes investigated should include, as a minimum, stall, runaway, lockup, interruption/restoration, clock and timing faults, counter overflow, missing/corrupt data, and effects of hardware faults on software. (15B4)

430.58

The accident analyzed under this section considers only the airborne radioactivity that may be released due to potential failure of a concentrated waste tank in the radwaste enclosure. The SRP acceptance criteria, however, requires demonstration that the liquid radwaste concentration at the nearest potable water supply in an unrestricted area resulting from transport of the liquid radwaste to the unrestricted area does not exceed the radionuclide concentration limits specified in 10 CFR Part 20, Appendix B Table II, Column 2. Such a demonstration will require information on possible dilution and/or decay during transit which, in turn, will depend upon site specific data such as surface and ground water hydrology and the parameters governing liquid waste movement through the soil. Additionally, special design features (e.g., steel liners or walls in the radwaste enclosure) may be provided as part of the liquid radwaste treatment systems at certain sites. The staff will, therefore, review the site specific characteristics mentioned above individually for each plant referencing the ABWR and confine its review of ABWR, only to the choice of the liquid radwaste tank.

Therefore, provide information on the following: (15.7.3)

(a) Basis for determining the concentrated waste tank as the worst tank (this may very well be the case, but in the absence of information on the capacities of major tanks, particularly the waste holdup tanks, it is hard to conclude that the above tank both in terms of radionuclide concentrations and inventories will turn out to be the worst tank).

(b) Radionuclide source terms, particularly for the long-lived radionuclides such as Cs-137 and Sr-90 (these may be the critical isotopes for sites that can claim only decay credit during transit) in the major liquid radwaste tanks.

440.108

Provide further justification for the fact that the input parameters and initial conditions for analyzed events are conservative. Provide a list of what paramenters will be checked at startup and which will be in the Technical Specifications. You should define the range of operating conditions and fuel types for which your input parameters will remain valid. For example, would these

RESPONSE 430.31

The plant protection signals that automatically isolate the secondary containment and activate the SGTS are:

(1) Secondary containment high radiation signal.

(2) Refueling floor high radiation signal.

(3) Drywell pressure high signal.

(4) Reactor water level low signal.

(5) Secondary containment HVAC supply/exhaust fans stop.

Isolation of the secondary containment is accomplished by closure of the secondary containment HVAC supply/exhaust line ducts which pass through the secondary containment boundary. The HVAC isolation valves consist of two valves in series in each of the supply/exhaust lines. These valves are air-operated, normally-open, fail closed butterfly valves.

Further details are provided in Subsection 6.2.3, 9.4.5.1 and Section 6.5

QUESTION 430.32

Identify and tabulate by size, piping which is not provided with isolation features. Provide an analysis to demonstrate the capability of the Standby Gas Treatment System to maintain the design negative pressure following a design basis accident with all non isolated lines open and the event of the worst single failure of a secondary containment isolation valve to close. (6.2)

RESPONSE 430.32

Response to this question will be provided in revised Subsection 6.5.1.3.1 and new Subsection 6.5.5.1.

QUESTION 430.33

Discuss the design provisions that prevent primary containment leakage from bypassing the secondary containment standby gas treatment system and escaping directly to the environment. Include a tabulation of potential bypass leakage paths, including the types of information indicated in Table 6-18 of Regulatory Guide 1.70, Revision 3. Provide an evaluation of potential bypass leakage paths considering equipment design limitations and test sensitivities. Specify and justify the maximum allowable fraction of primary containment leakage that may bypass the secondary containment structure. The guidelines of BTP 6-3 should be addressed in considering potential bypass leakage paths. (6.2)

RESPONSE 430.33

The secondary containment completely surrounds the primary containment except at the basemat. In addition the lower third of the secondary containment is surrounded by soil, thereby reducing leakage paths. No measurable leakage is expected through its walls except at penetrations. The secondary containment will be maintained at subatmospheric conditions to prevent leakage from bypassing the secondary containment. Only valve leakage through process piping can bypass the secondary containment. This leakage will be monitored via the containment leakage test type C on the outboard containment isolation valves. The secondary containment leak rate calculation is provided in the response to Question 430.52c.

tube bundle, initiate power reduction and faulty tube bundle drain down if required, and arrange for water box entry and leak repair at the earliest appropriate time.

QUESTION 430.72

Provide the permissible cooling water inleakage rate and the allowed time of operation with inleakage. (10.4.1)

RESPONSE 430.72

The polishing system is sized to meet the chemistry requirements for continuous operation while operating continuously with a condenser leak of 0.001 gpm and to maintain water quality during an orderly unit shutdown (not longer than 8 hours) with a leak of 0.1 gpm until repairs can be made. The design is adequate to clean up the feed and condensate system during plant heatup and low power operation without limiting plant startup time. The number and sizing of the ion exchangers are such that the functional requirements are met while permitting the replacement of resin in one ion exchanger at a time. The ABWR Standard Plant design features facilitate replacement of ion exchange resin.

QUESTION 430.73

Provide information on the following items:(10.4.1)

(a) Provisions incorporated into the main condenser design to preclude component or tube failure due to steam blowdown from the turbine bypass system.

(b) Worst possible flood level in the applicable buildings due to complete failure of main condenser and provisions for protecting safety related equipment located in the buildings against such flooding (note that ABWR SSAR Section 3.4 does not discuss the turbine building).

RESPONSE 430.73

(a) Specific provisions inside the condenser to preclude condenser tube damage due to turbine bypass steam impingement are to be defined by the condenser vendor for each project. Typically the provision inside the condenser consists of a horizontal perforated steam distribution pipe enclosed in a perforated guard pipe designed to protect the condenser internals from steam impingement. The perforated pipe and its guard pipe run the full length of the condenser and are supported above the condenser tube bundle.

(b) See revised Subsection 10.4.5.6 for the response to this question.

QUESTION 430.74

Discuss how the components of the main condenser evacuation system (MCES) conform to the guidelines of Regulatory Guide 1.26, 1.33, and 1.123 with respect to quality group classification and quality assurance programs.(10.4.2)