19A Response to CP/ML Rule 10 CFR 50.34(f)

19A.1 Introduction

On January 15, 1982 (47 FR 2286) the NRC amended 10 CFR 50.34 to include paragraph (f), “Additional TMI-Related Requirements”. These additional requirements were directed to each applicant for a light-water-reactor construction permit or manufacturing license (CP/ML) whose application was pending as of February 16, 1982.

In its “Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation”, on April 13, 1983 (48 FR 16014), the NRC proposed to extend its policy such that future CP applications or reactivations of CP applications previously docketed also comply with the CP/ML rule. Finally, on August 8, 1985 the commission issued a revision to this proposed policy statement as “Policy Statement on Severe Accident Regarding Future Designs and Existing Plants”. This appendix reports responses for the ABWR Standard Plant to the CP/ML rule.

The responses demonstrate that the NRC requirements are satisfactorily fulfilled for the ABWR design. For each item, a summary of the NRC position is given and followed by a response. The response clarifies the issue as it pertains to the ABWR design and/or provides a listing of applicable Tier 2 sections, relevant correspondence, or other necessary documentation that may be referenced for complete clarification of our position. Where a particular requirement is not applicable to the ABWR Standard Plant, a statement to that effect is provided in the response.

For items that affect equipment outside the scope of the ABWR Standard Plant or utility operations and procedures, the response indicates that item will be addressed by the COL applicant. Otherwise, this appendix is complete in that all of the “Additional TMI-Related Requirements” approved for implementation by the NRC as listed in 10 CFR 50.34(f) have been favorably addressed where they apply to the ABWR design.

The bracketed item numbers at the end of each title correspond with the subsections in 10CFR50.34(f). Alphanumeric designations at the end of each “NRC Position” statement correspond to the related action plan items in NUREG-0718 and NUREG-0660 [provided in 10 CFR 50.34(f) for information only].

Table 19A-1 is provided as a convenient cross-reference which consolidates pertinent information associated with each of the 47 requirements. This includes the 10 CFR 50.34(f) subsection, the action plan numbers, the Appendix 19A subsection number, the item title, and the Tier 2 reference detailing resolution.

19A.2 NRC Positions/Responses

19A.2.1 Probabilistic Risk Assessment [Item (1) (i)]

NRC Position

Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. [II.B.8]

Response

The ABWR probabilistic risk assessment (PRA) was submitted as Appendix 19D.

19A.2.2 Auxiliary Feedwater System Evaluation [Item (1)(ii)]

NRC Position

Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWRs only) [II.E.1.1]:

(1) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.

(2) A design review of AFWS.

(3) An evaluation of AFWS flow design bases and criteria.

Response

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

19A.2.3 Impact of RCP Seal Damages Following Small-Break LOCA with Loss of Offsite Power [Item (1) (iii)]

NRC Position

Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. [II.K.2(16) and II.K.3(25)]

Response

This item is addressed in Subsection 1A.2.30.

19A.2.4 Report on Overall Safety Effect of PORV Isolation System [Item (1) (iv)]

NRC Position

Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCAs from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWRs only.) [II.K.3(2)]

Response

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

19A.2.5 Separation of HPCS and RCIC System Initiation Levels [Item (1) (v)]

NRC Position

Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI System, and of providing that both systems restart on low water level. (For plants with High Pressure Core Flood Systems in lieu of high pressure coolant injection systems, substitute the words, “high pressure core flood” for “high pressure coolant injection” and (“HPCF” for “HPCI”) (Applicable to BWRs only). [II.K.3(13)]

Response

This item is addressed in Subsection 1A.2.22.

19A.2.6 Reduction of Challenges and Failures of Safety Relief Valves—Feasibility Study and System Modification [Item (1) (vi)]

NRC Position

Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWRs only.) [II.K.3(16)]

Response

This item is addressed in Subsection 1A.2.24.

19A.2.7 Modification of ADS Logic-Feasibility Study and Modification for Increased Diversity of Some Event Sequences [Item (1) (vii)]

NRC Position

Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modification that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWRs only.) [II.K.3(18)]

Response

This item is addressed in Subsection 1A.2.26.

19A.2.8 Restart of Core Spray and LPCI Systems on Low Level—Design and Modification [Item (1) (viii)]

NRC Position

Perform a study of the effect on all core-cooling modes under accident conditions of designing the core-spray and low pressure coolant in section systems to ensure that the systems will automatically re-start on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWRs only.) [II.K.3(21)]

Response

This item is addressed in Subsection 1A.2.27.

19A.2.9 Confirm Adequacy of Space Cooling Study for HPCS and RCIC [Item (1) (ix)]

NRC Position

Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and hi-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, “high pressure core spray” for “high pressure coolant injection” and “HPCS” for “HPCI”.) (Applicable to BWRs only.) [II.K.3(24)]

Response

This item is addressed in Subsection 1A.2.29.

19A.2.10 Verify Qualification of Accumulators on ADS Valves [Item (1) (x)]

NRC Position

Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety-related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWRs only.) [II.K.3(28)]

Response

This item is addressed in Subsection 1A.2.31.

19A.2.11 Evaluate Depressurization with Other Than Full ADS [Item (1) (xi)]

NRC Position

Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cool down. (Applicable to BWRs only.) [II.K.3(45)]

Response

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

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

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

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

NRC Position

Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of 10 CFR 50.34(f). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

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

(2) For the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of 10 CFR 50.34.

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

Response

The ABWR primary containment is inerted and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation. In fact, increasing amounts of hydrogen moves the primary containment oxygen concentration further from the flammable regime. Radiolysis is the only potential source of oxygen in the ABWR primary containment.

The deletion of the Flammability Control System, including the recombiners, from the ABWR design, and the design's capability to accommodate oxygen from radiolysis, is described in Subsection 6.2.5.

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

NRC Position

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

Response

COL license information, see Subsection 19A.3.1. This will be addressed as part of simulator design which falls under operator training (Section 18.8.8).

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

NRC Position

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

Response

COL license information, see Subsection 19A.3.2.

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

NRC Position

Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. [I.D.1]

Response

This item is addressed in Subsection 1A.2.2.

19A.2.16 Plant Safety Parameter Display Console (SPDS) [Item (2) (iv)]

NRC Position

Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. [I.D.2]

Response

This item is addressed in Subsection 1A.2.3.

19A.2.17 Safety System Status Monitoring [Item (2) (v)]

NRC Position

Provide for automatic indication of the bypassed and inoperable status of safety systems. [I.D.3]

Response

The ABWR Standard Plant design fully complies with Regulatory Guide 1.47 (Subsection 7.1.2.10.2). The automatic indication of bypassed and inoperable status of safety systems is, therefore, inherent in the design. Details on human factors are not addressed specifically, however, will be addressed by the COL applicant during the conduct of the HSI design implementation process described in Section 18.E.1.

19A.2.18 Reactor Coolant System Vents [Item (2) (vi)]

NRC Position

Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. [II.B.1]

Response

This issue is addressed in Subsection 1A.2.5.

19A.2.19 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation [Item (2) (vii)]

NRC Position

Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. [II.B.2]

Response

This item is addressed in Subsection 1A.2.6.

19A.2.20 Post-Accident Sampling [Item (2) (viii)]

NRC Position

Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 0.05 Sv to the whole-body or 0.50 Sv to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. [II.B.3]

Response

This item is addressed in Subsection 1A.2.7.

19A.2.21 Hydrogen Control System Preliminary Design [Item (2) (ix)]

NRC Position

Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal-water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1) (xii) of 10 CFR 50.34(f) is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: [II.B.8]

(1) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(2) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(3) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.

(4) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

Response

Per the response to Item (1)(xii), refer to Subsection 6.2.5 for a detailed description of the inerting system.

19A.2.22 Testing Requirements [Item (2) (x)]

NRC Position

Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. [II.D.11]

Response

This item is addressed in Subsection 1A.2.9.

19A.2.23 Relief and Safety Valve Position Indication [Item (2) (xi)]

NRC Position

Provide direct indication of relief and safety valve position (open or closed) in the control room. [II.D.3]

Response

This item is addressed in Subsection 1A.2.10.

19A.2.24 Auxiliary Feedwater System Automatic Initiation and Flow Indication [Item (2) (xii)]

NRC Position

Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWRs only.) [II.E.1.2]

Response

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

19A.2.25 Reliability of Power Supplies for Natural Circulation [Item (2) (xiii)]

NRC Position

Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWRs only.) [II.E.3.1]

Response

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

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

NRC Position

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

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

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

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

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

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

Response
This item is addressed in Subsection 1A.2.14.

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

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

Response
The ABWR primary containment vessel (PCV) operates with an inert atmosphere. During normal operation, all large valves in containment ventilation lines are closed with the exception of two large valves in the overpressure/protection where flow is prevented by a rupture disk in the piping.

Only small 50A (2-inch) pipe size nitrogen-makeup valves are opened during power operation. These are air-operated valves with rapid closure times, presenting little opportunity for substantial releases from the PCV in the event of a transient requiring containment isolation. Note that under the technical specifications, containment inerting and purging with the larger ventilation lines is permitted during power operation above 15% for limited periods at either end of the operating cycle. The process of purging the containment with air also serves to remove any potential activity for ALARA considerations prior to actual personnel entry into the PCV.

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

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

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

Response

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

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

NRC Position

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

Response

This item is addressed in Subsection 1A.2.15.

19A.2.30 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling [Item (2) (xviii)]

NRC Position

Provide instruments that provide in the control room an unabiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR’s, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs. [II.F.2]

Response

This item is addressed in Subsection 1A.2.16.

19A.2.31 Instrumentation for Monitoring Accident Conditions (Regulatory Guide 1.97) [Item (2) (xix)]

NRC Position

Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. [II.F.3]

Response

This item is addressed in Subsection 1A.2.17 and Subsection 7.5.

19A.2.32 Power Supplies for Pressurizer Relief Valves, Block Valves and Level Indication [Item (2) (xx)]

NRC Position

Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWRs only.) [II.G.1]

Response

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

19A.2.33 Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable [Item (2) (xxi)]

NRC Position

Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWRs only.) [II.K.1(22)]

Response

This item is addressed in Subsection 1A.2.20.

19A.2.34 Analysis of Upgrading of Integrated Control System [Item (2) (xxii)]

NRC Position

Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only.) [II.K.2(9)]

Response

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

19A.2.35 Hand-Wired Safety-Grade Anticipatory Reactor Trips [Item (2) (xxiii)]

NRC Position

Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only.) [II.K.2(10)]

Response

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

19A.2.36 Central Water Level Recording [Item (2) (xxiv)]

NRC Position

Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWRs only.) [II.K.3(23)]

Response

In the ABWR design, the RPV water level wide range instruments and fuel zone instruments are utilized to provide this PAM indication. The four divisions of wide range instruments cover the range from above the core to the main steam lines. The two channels of fuel zone instruments cover the range from below the core to the top of the steam separator shroud. Two channels of enhanced water level indication are provided which use a wide range level transmitter, a fuel zone level transmitter, an RPV pressure transmitter and six temperature elements. The signals are input to a microprocessor which computes compensated level and provides a level signal to an indicator and a recorder. This design avoids the ambiguity of varying process and/or ambient temperatures, of a instrument line break, or boiling in an instrument line. If one of the enhanced water level indication channels fails, reliable indication of vessel level may be regained utilizing the 4 individual wide range indicators and the 2 individual fuel zone range level indicators as described below.

Evaluation has concluded that two channels of fuel zone level instrumentation provide adequate post accident monitoring capability. Post accident operator actions will be in accordance with detailed procedures developed based upon the BWR Owners' Group emergency operating procedure (EOP) guidelines. In the event the vessel water level is below the range of the wide range level (WRL) sensors (i.e., the water level is in the full zone range) and the two channels of fuel zone level instrumentation disagree, the EOPs instruct the operator to return the water level back up into the range of the instrumentation. Using the four divisions of WRL instruments, an unambiguous indication of vessel water level can be determined, despite a postulated failure of a single instrument channel or division, and the operator could safely continue the execution of appropriate accident instigation activities as defined by the EOPs.

19A.2.37 Upgrade License Emergency Support Facility [Item (2) (xxv)]

NRC Position

Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a near site Emergency Operations Facility. [IIIA.1.2]

Response

The design features for the onsite Technical Support Center and the onsite Operational Support Center are provided in Subsection 13.3. The near site Emergency Operations Facility is provided by the COL applicant, Subsection 19A.3.4.

19A.2.38 Primary Coolant Sources Outside the Containment Structure [Item (2) (xxvi)]

NRC Position

Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. [III.D.1.1]

Response

This issue is addressed in Subsection 1A.2.34.

19A.2.39 Inplant Radiation Monitoring [Item (2) (xxvii)]

NRC Position

Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. [III.D.3.31]

Response

COL license information, personal monitoring radiation and portable instrumentation, training and procedures (Subsections 12.5.2, 12.5.3.1, 12.5.3.2, and 19A.3.5). Airborne radiation monitoring equipment (nonportable), Subsection 12.3.4.

19A.2.40 Control Room Habitability [Item (2) (xxviii)]

NRC Position

Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source term release, and make necessary design provisions to preclude such problems. [III.D.3.4]

Response

This item addressed in Subsection 1A.2.36.

19A.2.41 Procedures for Feedback of Operating, Design and Construction Experience [Item (3) (i)]

NRC Position

Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. [I.C.5]

Response

COL license information, see Subsection 19A.3.6.

19A.2.42 Expand QA List [Item (3) (ii)]

NRC Position

Ensure that the quality assurance (QA) list required by Criterion 11, App. B. 10 CFR Part 50 includes all structures, systems, and components important to safety. [I.F.1]

Response

Quality system requirements are identified in Table 3.2-1 Classification Summary. In addition, COL license information requirements, Section 1.9, ensure that quality system requirements will be provided during construction and operation.

19A.2.43 Develop More Detailed QA Criteria [Item (3) (iii)]

NRC Position

Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) Performing quality assurance/quality control functioning at construction sites to the maximum feasible extent; (C) Including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) Establishing criteria for determining QA programmatic requirements; (E) Establishing qualification requirements for QA and QC personnel; (F) Sizing the QA staff commensurate with its duties and responsibilities; (G) Establishing procedures for maintenance of "as-built" documentation; and (H) Providing a QA role in design and analysis activities. [I.F.2]

Response

For the initial ABWR design certification, GE provided the following response:

(1) NEDO-11209-04A, "GE Nuclear Energy Quality Assurance Program Description", conforms to this requirement. See Paragraph 1.1 on page 1-1.

(2) GE-NE services performed at the construction site are under the Owners' QA program. GE-NE provides QA program support to the Owner as described in NEDO-11209-04A, pages 1-3,1-7, 11-1, and 11-2.

(3) The GE-NE Nuclear Quality Assurance(NQA) is responsible for preparing the top level GE-NE quality policy and instructions for issue by the Vice President and General Manager, GE-NE. NQA is also responsible for preparing and issuing several GE-NE quality procedures. These documents are identified on pages 2-2 and 2-3 of NEDO-11209-04A.

    In addition, NQA is responsible for developing, issuing, and controlling NEDO-11209.

    The GE-NE line QA organizations are responsible for developing and documenting a quality system in compliance with GE-NE policies, instructions and procedures, and applicable codes, standards, and regulatory requirements. See NEDO-11209-04A, Section 1.3, "QA Functional Responsibilities" and Section 2.2 for typical line—QA procedure manuals.

(4) NEDO-11209-04A responds to each of the QA programmatic requirements of 10CFR50, Appendix B, and the requirements of the regulatory guides and industry standards identified in Table 2-1. In addition, the GE-NE QA program conforms to the requirements of the ASME Code.

(5) NEDO-11209-04A, Section 2-1, fourth paragraph, describes the qualification of training of GE-NE personnel who perform activities affecting quality. See also Subsection 1.4, “QA Personnel Responsibilities and Qualifications”.

(6) The NRC has evaluated the GE-NE QA Program implementation for several years and has found that the program, including sizing of the QA staff, is being implemented satisfactorily. See NRC letters in Docket No. 99900403.

(7) NEDO-11209-04A, Section 17, describes the GE-NE commitments related to “as-built” documentation. The GE-NE commitments are further detailed on pages 2-10, 2-11, and 2-13 thru 2-15.

(8) NQA has the following responsibilities that are documented in NEDO-11209-04A, Subsection 1.3:

   (a) Develop GE-NE policies and procedures related to project and services management, engineering, manufacturing, procurement, field service and construction QA.

   (b) Conduct or participate in independent design reviews.

   (c) Conduct independent audits of the GE-NE design control program.

Based on the foregoing evaluation, it is demonstrated that the GE-NE QA program as described in NEDO-11209-04A, and as currently accepted by the NRC, includes full consideration of the matters identified in this item.

For Toshiba’s renewal of the ABWR design certification, the following response applies:

(1) Toshiba Document No. 4401-4, “Nuclear Energy QA Program Description” (QAPD), conforms to the requirement that the organization performing checking functions is independent of the organization performing the functions. See Section 2.1 and Figure 2.1.

(2) Toshiba Corporation, Power Systems Company, Nuclear Energy (PSNE) provides on-site QA and quality control (QC) as described in several sections of Toshiba’s QAPD, including Sections 2.2, Figure 2.1, 2.2.3, 4.6 and Chapters 9, 10, 11, 12, 13, 14, 15, 16, 17 and 18.

(3) PSNE Quality Assurance staff are responsible for establishing and issuing the QA Program Description and PSNE top level policies and procedures, and Standards

which are applied to all activities affecting quality within PSNE. A quality system document structure is described in Toshiba's QAPD, Section 3.2.1 and Figure 3.1. All quality related procedures, including those pertaining to design, construction and installation, are required to conform to Toshiba's QAPD.

PSNE has established an in-house quality-committee to provide QA communication within the organizations PSNE. Functions of the committee include deliberation of PSNE level quality system documents and NED quality manuals & procedures, and providing medium for evaluating revision/addendum of applicable regulatory, code & standard requirements to assure PSNE quality program coverage, uniformity consistency and continuity.

PSNE QA organizations are responsible for developing and documenting a quality system in compliance with PSNE policies, instructions and procedures, and applicable codes, standards and regulatory requirements. See Sections 1.2 (page 4) and 2.2 of Toshiba's QAPD for QA organization structure and responsibilities.

(4) Toshiba's QAPD conforms to the QA programmatic requirements of 10 CFR 50 Appendix B, and the requirements of QA related regulatory guides, ASME NQA-1, IAEA GS-R-3 and the ASME Boiler and Pressure Vessel code. See Sections 1.1, 3.3 and Appendix 1 of Toshiba's QAPD.

(5) Toshiba's QAPD, Section 3.4, describes the qualification requirements and training of PSNE personnel who perform activities affecting quality including QA and QC personnel. Qualification requirements for "inspection personnel" are described in Section 11.2 of Toshiba's QAPD, and qualification requirements for "QA Audit Personnel" are described in Section 19.11 of Toshiba's QAPD.

(6) Toshiba's QAPD discusses the various QA and QC positions that ensure that the requirements of the QAPD and the responsibilities of the QA Department are fulfilled. See Chapters 1 and 2, and Figure 2.1. NRC has evaluated Toshiba QA program implementation for STP-3/4 and found that the program, including sizing of QA staff is being implemented satisfactorily.

(7) Toshiba's QAPD, Chapter 18, describes Toshiba's commitments related to "as-built" documentation.

(8) PSNE Quality Assurance staff are responsible for several quality related activities pertaining to design and analysis, as documented throughout Toshiba's QAPD and summarized in Section 2.2.1. The role of QA/QC personnel in design and analysis

activities is further discussed in Chapter 4 of Toshiba's QAPD. In summary, the responsibilities of QA/QC personnel include:

(a) Establishing and issuance of the QA Program Description and PSNE top level policies and procedures which are applied to all activities affecting quality throughout plant life.

(b) Conducting or participating in independent design reviews (see Toshiba's QAPD, Section 4.5.1.(4)).

(c) Conducting independent audits of the Toshiba design control program.

Based on the foregoing evaluation, it is demonstrated that the Toshiba QA program, as described in Toshiba Document No. 4401-4, includes full consideration of the matters identified in this item.

COL licensing information, see Subsection 19A.3.8.

19A.2.44 Dedicated Containment Penetrations Equivalent to a Single 3-Foot Diameter Opening [Item (3) (iv)]

NRC Position

Provide one or more dedicated containment penetrations, equivalent in size to a single 91 cm (3-foot) diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. [II.B.8]

Response

The Containment Overpressure Protection System is described in Subsection 6.2.5.2.6 and is analyzed in the PRA. The sizing of the system is developed in Subsection 19E.2.8.1.3 and precludes the need for a dedicated penetration equivalent in size to a single 91-cm (3-foot) diameter opening.

19A.2.45 Containment Integrity [Item (3) (v)]

NRC Position

Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: [II.B.9]

(1) (a) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added

pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 0.412 MPa (45 psig). Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(b) Subarticle NE-3220, Division 1, and Subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f)(3)(v)(A)(1) and (Q (3)(v)(B)(1) of 10 CFR 50.34, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission’s Public Document Room, 1717 H St., NW., Washington, D.C.

(2) (a) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, Service Load Category.

(c) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.

Response

(1) The containment design basis accident pressure is 0.412 MPa. The peak pressure resulting from 100% fuel-clad metal water reaction is about 0.618 MPa (Subsection 19E.2.3.2). The containment is capable of withstanding 0.618 MPa internal pressure together with dead load by meeting the code requirements (Subsection 19E.2.3.2).

(2) ABWR does not employ post accident inerting; thus, item (2) does not apply.

19A.2.46 Dedicated Penetration [Item (3) (vi)]

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. [II.E.4.1]

Response

This item is not applicable to the ABWR design.

19A.2.47 Organization and Staffing to Oversee Design and Construction [Item (3) (vii)]

NRC Position

Provide a description of the management plant for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) Technical resources directed by the applicant; (C) Details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) Proposed procedures for handling the transition to operation; (E) The degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. [II.J.3.1]

Response

COL license information, see Subsection 19A.3.7.

19A.3 COL License Information

19A.3.1 Long-Term Training Upgrade

Simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs shall be provided. (Subsection 19A.2.13.) COL License Information regarding operator training is in Section 18.8.8.

19A.3.2 Long-Term Program of Upgrading of Procedures

A long-term program of upgrading procedures shall be established to begin during construction and following term program of upgrading procedures shall be established to begin during construction and follow into operation for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analysis, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Subsection 19A.2.14.) COL License Information is in Section 13.5.3.1.b.

19A.3.3 Purge System Reliability

A testing program shall be provided to ensure that the large ventilation valves close within the limits assured in the radiologic design bases. (Subsection 19A.2.27.)

19A.3.4 Licensing Emergency Support Facility

The COL applicant has a requirement to provide a near site Emergency Operational Facility (EOF) (See Subsection 19A.2.37).

19A.3.5 In-Plant Radiation Monitoring

Personal monitoring and portable instrumentation of in-plant radiation and airborne radioactivity as well as training and procedures appropriate for a broad range of routine and accident conditions shall be provided (Subsections 12.5.2, 12.5.3.1, 12.5.3.2, and 19A.2.39).

19A.3.6 Feedback of Operating, Design and Construction Experience

Administrative procedures for evaluating design and construction experience and for ensuring that applicable important industry experiences shall be provided in a timely manner to those designing and constructing the ABWR Standard Plant. (Subsection 19A.2.41) COL license information regarding incorporation of operator experience into training and procedures is found in Sections 13.2.3 and 13.5.3, respectively.

19A.3.7 Organization and Staffing to Oversee Design and Construction

A description of the management plan for design and construction activities shall be provided. It will include:

(1) Organizational and management structure singularly responsible for direction of design and construction for the plant

(2) Technical resources directed by the applicant referencing the ABWR design

(3) Details of the interaction of design and construction within the organization of the applicant referencing the ABWR design and the associated organization by which integration of the total project is ensured

(4) Procedures for handling the transition to operation

(5) The degree of top level management oversight and technical control will be exercised during design and construction including the preparation and implementation of procedures necessary to guide the effort (Subsection 19A.2.47)

19A.3.8 Develop More Detailed QA Criteria

Establish a quality assurance (QA) program in accordance with the requirements in Subsection 19A.2.43.

Table 19A-1 ABWR—CP/ML Rule Cross Reference

CP/ML Rule Section	Item Action Plan	Appendix Section	Title	Tier 2 Reference
(1) (i)	II.B.8	19A.2.1	Probabilistic Risk Assessment	Appendix 19D
(ii)	II.E.1.1	19A.2.2	Auxiliary Feedwater System Evaluation	Not Applicable (PWR Only)
(iii)	II.K.2(16) & II.K.3(25)	19A.2.3	Impact of RCP Seal Damages Following Small-Break LOCA with Loss of Offsite Power	Subsection 1A.2.30
(iv)	II.K.3(2)	19A.2.4	Report on Overall Safety Effect on PORV Isolation System	Not Applicable (PWR Only)
(v)	II.K.3(13)	19A.2.5	Separation of HPCF and RCIC System Initiation Levels	Subsection 1A.2.22
(vi)	II.K.3(16)	19A.2.6	Reduction of Challenges and Failures of Safety Relief Valves Feasibility Study and System Modification	Subsection 1A.2.24
(vii)	II.K.3(18)	19A.2.7	Modification of ADS Logic-Feasibility Study and Modification for Increased Diversity of Some Event Sequences	Subsection 1A.2.26
(viii)	II.K.3(21)	19A.2.8	Restart of Core Flood and LPCI Systems on Low Level-Design and Modification	Subsection 1A.2.27
(ix)	II.K.3(24)	19A.2.9	Confirm Adequacy of Space Cooling Study for HPCF and RCIC	Subsection 1A.2.29
(x)	II.K.3(28)	19A.2.10	Verify Qualification of Accumulators on ADS Valves	Subsection 1A.2.31
(xi)	II.K.3(45)	19A.2.11	Evaluate Depressurization with Other than Full ADS	Subsection 19A.2.11
(xii)	—	19A.2.12	Evaluation of Alternative Hydrogen Control Systems	Subsection 19A.2.12
(2) (i)	IA.4.2	19A.2.13	Long-Term Training Upgrade	Subsection 19A.3.1
(ii)	I.C.9	19A.2.14	Long-Term Program of Upgrading of Procedures	Subsection 19A.3.2/13.5.3.1
(iii)	I.D.1	19A.2.15	Control Room Design Reviews	Subsection 1A.2.2/18.8.1
(iv)	I.D.2	19A.2.16	Plant Safety Parameter Display Console	Subsection 1A.2.3/18.8.4
(v)	I.D.3	19A.2.17	Safety System Status Monitoring	Subsection 19A.2.17/18.8.9
(vi)	II.B.1	19A.2.18	Reactor Coolant System Vents	Subsection 1A.2.5

Table 19A-1 ABWR—CP/ML Rule Cross Reference (Continued)

CP/ML Rule Section	Item Action Plan	Appendix Section	Title	Tier 2 Reference
(vii)	II.B.2	19A.2.19	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	Subsection 1A.2.6
(viii)	II.B.3	19A.2.20	Post-Accident Sampling	Subsection 1A.2.7
(ix)	II.B.8	19A.2.21	Hydrogen Control System Preliminary Design	Subsection 19A.2.21
(x)	II.D.1	19A.2.22	Testing Requirements	Subsection 1A.2.9
(xi)	II.D.3	19A.2.23	Relief and Safety Valve Position Indication	Subsection 1A.2.10
(xii)	II.E.1.2	19A.2.24	Auxiliary Feedwater System Automatic Initiation and Flow (Indication	Not Applicable (PWR Only)
(xiii)	I.E.3.1	19A.2.25	Reliability of Power Supplies for Natural Circulation	Not Applicable (PW R Only)
(xiv)	II.E.4.2	19A.2.26	Isolation Dependability	Subsection 1A.2.14
(xv)	II.E.4.4	19A.2.27	Purging	Subsections 19A.2.27 and 19A.3.3
(xvi)	II.E.5.1	19A.2.28	Design Evaluator	Not Applicable (B&W Only)
(xvii)	II.F.1	19A.2.29	Additional Accident-Monitoring Instrumentation	Subsection 1A.2.15/18.8.13
(xviii)	II.F.2	19A.2.30	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	Subsection 1A.2.16/18.8.14
(xix)	II.F.3	19A.2.31	Instrumentation for Monitoring Accident Conditions (Regulatory Guide 1.97)	Subsection 1A.2.17 and Section 7.5
(xx)	II.G.1	19A.2.32	Power Supplies for Pressurizer Relief Valves, Block Valves and Level Indication	Not Applicable (PWR Only)
(xxi)	II.K.1(22)	19A.2.33	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW Systems not Available	Subsection 1A.2.20
(xxii)	II.K.2(9)	19A.2.34	Analysis of Upgrading of Integrated Control System	Not Applicable (P&W Only)

Table 19A-1 ABWR—CP/ML Rule Cross Reference (Continued)

CP/ML Rule Section		Item Action Plan	Appendix Section	Title	Tier 2 Reference
	(xxiii)	II.K.2.(10)	19A.2.35	Hand-Wired Safety-Grade Anticipatory Reactor Trips	Not Applicable (P&W Only)
	(xxiv)	II.K.3(23)	19A.2.36	Central Water Level Recording	Subsection 19A.2.26
	(xxv)	III.A.1.2	19A.2.37	Upgrade License Emergency Support Facility	Subsection 19A.3.4
	(xxvi)	III.D.1.1	19A.2.38	Primary Coolant Sources Outside the Containment Structure	Subsection 1A.2. 34
	(xxvii)	III.D.3.3	19A.2.39	In-Plant Radiation Monitoring	Subsection 19A.3.5
	(xxviii)	II.D.3.4	19A.2.40	Control Room Habitibility	Subsection 1A.2. 36
(3)	(i)	I.C.5	19A.2.41	Procedures for Feedback of Operating, Design and Construction Experience	Subsection 19A.3.6/13.2.3.1 /13.5.3.3.f
	(ii)	I.F.1	19A.2.42	Expand QA List	Subsection 19A.2.42
	(iii)	I.F.2	19A.2.43	Develop More Detailed QA Criteria	Subsection 19A.2.43
	(iv)	II.B.8	19A.2.44	Dedicated Containment Penetrations, Equivalent to a Single 3-foot Diameter Opening	Subsection 19A.2.44
	(v)	II.B.8	19A.2.45	Containment Integrity	Subsection 19A.2.45
	(vi)	II.E.4.1	19A.2.46	Dedicated Penetration	Not Applicable
	(vii)	II.J.3.1	19A.2.47	Organization and Staffing to Oversee Design and Construction	Subsection 19A.3.7