1A Response to TMI Related Matters

1A.1 Introduction

The investigations and studies associated with the TMI accident produced several documents specifying results and recommendations, which prompted the issuances by the NRC of various bulletins, letters, and NUREGs providing guidance and requiring specific actions by the nuclear power industry. In May 1980, the issuance of NUREG-0660 (Reference 1A-4) provided a comprehensive and integrated plan and listing requirements to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at TMI and the studies and investigations of the accident. NUREG-0737(Reference 1A-5), issued in November 1980, listed items from NUREG-0660 approved by the NRC for implementation, and included additional information concerning schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions. Finally, NUREG-0718 (Reference 1A-6) was issued in June 1981 to provide guidance that the NRC staff believes should be followed to account for the lessons learned from the TMI accident.

This Appendix 1A provides responses for the ABWR Standard Plant required by Section II of the NRC Standard Review Plan, those satisfying 10CFR50.34(f) are addressed in Appendix 19A. The remaining TMI issues satisfying severe accident requirements are addressed in Appendix 19B.

1A.2 NRC Positions/Responses

1A.2.1 Short-Term Accident Analysis Procedure Revision [I.C.1(3)]

NRC Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979 (References 1A-7 through 1A-11), the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also Item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on Babcock and Wilcox (B&W)-designed plants, the staff will follow up on the bulletin and order matters relating to analysis methods and results, as listed in NUREG-0660, Appendix C (Table C.1, Items 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, Items 4, 12, 17, 18, 19, 20; and Table C.3, Items 6, 35, 37, 38, 39, 41, 47, 55, 57).

Response

In the clarification of the NUREG-0737 requirement for reanalysis of transients and accidents and inadequate core cooling and preparation of guidelines for development of emergency procedures, NUREG-0737 states:

> Owners' group or vendor submittals may be referenced as appropriate to support this reanalysis. If owners' group or vendor submittals have already been forwarded to the staff for review, a brief description of the submittals and justification of their adequacy to support guideline development is all that is required.

GE participated in the BWR Owners' Group (BWROG) program to develop emergency procedure guidelines for GE BWRs. The resulting emergency procedure guidelines are generally applicable to the ABWR, as are the transient and accident analyses. Following is a brief description of the submittals to date, and a justification of their adequacy to support guideline development.

(1) Description of Submittals

(a) NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors", August 1979.

(b) NEDO-24708A, Revision 1,"Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors", December 1980. This report was issued via the letter from D. B. Waters (BWR Owners' Group) to D. G. Eisenhut (NRC) dated March 20, 1981.

(c) "BWR Emergency Procedure Guidelines (Rev. 0)"—submitted in prepublication form June 30, 1980.

(d) "BWR Emergency Procedure Guidelines (Rev. 1)"—Issued via the letter from D. B. Waters (BWR Owners' Group) to D. G. Eisenhut (NRC) dated January 31, 1981.

(e) "BWR Emergency Procedure Guidelines (Rev. 2)"—submitted in prepublication form June 1, 1982, Letter BWROG-8219 from T. J. Dente (BWR Owners' Group) to D. G. Eisenhut (NRC).

(f) "BWR Emergency Procedure Guidelines (Rev. 3)", submitted in prepublication form December 22, 1982, Letter BWROG-8262 from T. J. Dente (BWR Owners' Group) to D. G. Eisenhut (NRC).

(g) NEDO-31331, "BWR Emergency Procedure Guidelines (Rev. 4)", submitted April 23, 1987, Letter BWROG-8717, from T. A. Pickens (BWR Owners' Group) to T. Murley (NRC).

(2) Adequacy of Submittals

The submittals described in (1) above have been discussed and reviewed extensively among the BWR Owners' Group, the General Electric Company, and the NRC Staff.

The NRC has extensively reviewed the latest revision (Revision 4) of the Emergency Procedures Guidelines and issued a SER, "Safety Evaluation of BWR Owners' Group Emergency Procedure Guidelines, Revision 4, NEDO-31331, March 1987", letter from A. C. Thadani, NRC Office of Nuclear Reactor Regulation, to D. Grace, Chairman of BWR Owners' Group, dated September 12, 1988. The SER concludes that this document is acceptable for implementation. It further states that the SER closes all the open items carried from the previous revisions of the EPG.

In view of these findings, no further detailed justification of the analyses or guidelines was considered necessary. COL license information requirements pertaining to emergency procedures are discussed in Subsection 1A.3.1.

1A.2.2 Control Room Design Reviews—Guidelines and Requirements [I.D.1(1)]

NRC Position

In accordance with task Action Plan I.D.1(1), all licensees and applicants for operating licenses will be required to conduct a detailed control room design review to identify and correct design deficiencies. This detailed control room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

Response

The design of the main control room will utilize accepted human factors engineering principles, incorporating the results of a full systems analysis similar to that described in Appendix B of NUREG-0700. A DCRDR specified in NUREG-0737 is not required by SRP Section 18.1. Details are described in Chapter 18.

1A.2.3 Control Room Design—Plant Safety Parameter Display Console [I.D.2]

NRC Position

In accordance with Task Action Plan I.D.2, each applicant and licensee shall install a Safety Parameter Display System (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

Response

The functions of the SPDS will be integrated into the overall control room design, as permitted by SRP Section 18.2. Details are found in Chapter 18.

1A.2.4 Scope of Test Program—Preoperational and Low Power Testing [I.G.1]

NRC Position

Supplement operator training by completing the special low-power test program. Tests may be observed by other shifts or repeated on other shifts to provide training to the operators.

Response

The initial test program presents an excellent opportunity for licensed operators and other plant staff members to gain valuable experience and training and, in fact, these benefits are objectives of the program (Subsection 14.2.1). The degree to which the potential benefit is realized will depend on such plant-specific factors as the organizational makeup of the startup group and overall plant staff (Subsections 14.2.2 and 13.1), as well as how the test program is conducted (Subsection 14.2.4).

The BWR Owners' Group response to Item I.G.1 of NUREG-0737 is documented in a letter of February 4, 1981 from D.B. Waters to D.G. Eisenhut. For the most part, this issue concerns training requirements, although in the context of the initial test program. Thus, the BWROG response primarily deals with operator training issues. The exception is Appendix E of the BWROG response which describes additional tests to be conducted during the preoperational and/or startup phase.

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

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

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

NRC Position

Each applicant and licensee shall install Reactor Coolant System (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS, which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10CFR50, "General Design Criteria". The vent system shall be designed with sufficient redundancy to assure a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for LOCAs initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10CFR50.46.

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

Response

The capability to vent the ABWR reactor coolant system is provided by the safety/relief valves (SRVs) and reactor coolant vent line, as well as other systems. The COL applicant will develop plant-specific procedures to govern the operator's use of the relief mode for venting the reactor (Subsection 1A.3.6). The capability of these systems and their satisfaction of Item II.B.1 are discussed below.

The ABWR design includes various means of high-point venting. Among these are:

(1) Normally closed reactor vessel head vent valves, operable from the control room, which discharge to the drywell. The reactor coolant vent line is located at the very top of the reactor vessel as shown in the nuclear boiler system P&ID (Figure 5.1-3). This 50A line contains two safety-related Class 1E motor-operated valves that are operated from the control room. The location of this line permits it to vent the entire reactor core system normally connected to the reactor pressure vessel. In addition, since this vent line is part of the original design, it has already been considered in all the design basis accident (DBA) analyses contained elsewhere in this document.

(2) Normally open reactor head vent line, which discharges to a main steamline.

The conclusions from this vent evaluation are as follows:

(1) Reactor vessel head vent valves exist to relieve head pressure (at shutdown) to the drywell via remote operator action.

(2) The reactor vessel head is continuously swept to the main condenser and can be vented during operating conditions.

(3) The size of the vents is not a critical issue because BWR SRVs have substantial capacity, exceeding the full power steaming rate of the nuclear boiler.

(4) No new 10CFR50.46 conformance calculations are required, because the vent provisions are part of the plant's original design and are covered by the original design bases.

(5) Plant-specific procedures govern the operator's use of the relief mode for venting reactor pressure.

1A.2.6 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation [II.B.2]

NRC Position

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operation of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Response

A review of the radiation and shielding of the ABWR Standard Plant post- accident operations has been made. It has been found that there is adequate access to vital areas and that safety equipment is adequately protected. No need for corrective action was identified. Details of the review may be found in Attachment A to Appendix 1A.

1A.2.7 Post-Accident Sampling [II.B.3]

NRC Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 0.05 and 0.50 Sv to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to quantify (in less than 2 hours) certain radionuclides

that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

Response

Discharges From Plant and Containment— During the development of an accident, samples of liquid and gaseous discharges from both the plant and containment will be obtained. Chemical and radiochemical analyses will be performed for protection of the health and safety of the public and the plant operators. These samples will be obtained from the Process Sampling System. The Post Accident Sampling Systems will not be required to obtain these samples.

Core Damage Assessment—During this initial period, instrumentation will provide sufficient information for the operators to perform their duties. For example, the containment high range radiation meters will give instant information concerning the radiation level in containment (To obtain data from the PASS several hours may be required for sampling and analyses.). Calculations can be performed to relate containment radiation level with the probable extent of core damage. Core damage assessment instrumentation is described in Section 18.4.6. This section describes the safety parameter display system (SPDS). The principle purpose of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. The following critical safety functions are provided at the wide screen display panel in the main control room:

(1) Reactivity control

(2) Reactor core cooling and heat removal from the primary system

(3) Reactor coolant system integrity

(4) Radioactivity control

(5) Contamination conditions

This instrumentation and the PASS work together to obtain complementary information. After this initial period during the development of an accident, the ABWR PASS will be used to obtain samples of reactor water and containment atmosphere to assess the extent of core damage. The ABWR PASS has been designed to safely obtain samples with radioactivity levels up to 37,000 M Bq/g. Approximately one day after a serious core damage accident, it is expected that sample radioactivity levels will be no more than this value. Early in such an accident, the plant instrumentation in the main control room would be indicating that abnormal conditions exist. If a reactor coolant sample were obtained which had excessive radioactivity, as measured by the area radiation monitor in the PASS area, the plant operators would use this high radiation information as confirmatory evidence that severe core damage has occurred and continue following the emergency operating procedures. It would not be necessary to perform any radiochemical analyses to reach this conclusion. During less severe accidents, in which only some cladding damage has occurred, samples may be obtained from either the Process Sampling System or PASS.

NUREG-0737 Requirements— The ABWR PASS has been designed to meet the eleven requirements listed in NUREG-0737 except as noted below:

(1) The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample. Meets the requirements of NUREG-0737.

(2) There shall be onsite capability to perform the following within the 3 hour time period:

   (a) Determine the presence and amount of certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage. Meets the requirements of NUREG-0737.

   (b) Hydrogen in containment atmosphere. Hydrogen in containment atmosphere is measured by the Containment Atmospheric Monitoring System. Meets the requirements of NUREG-0737 with the exception that the design follows the guidance of RG 1.7 Rev. 3 which permits the hydrogen monitor to be classified as non-safety related.

   (c) Dissolved gases, chloride and boron in liquids. Dissolved gases are discussed in item 4 below. Meets the requirements concerning chloride and boron of NUREG-0737.

   (d) Inline monitoring capability is acceptable. No inline monitors are provided in PASS.

(3) Sampling need not depend upon an isolated auxiliary system being put into operation. Meets the requirements of NUREG-0737.

(4) Reactor coolant samples and analyses for total dissolved gases and hydrogen are required. During a severe core damage accident for the ABWR, the reactor water will become mixed with the suppression pool water. The pressure in the reactor vessel will decrease to approximately the pressure within the wetwell and the drywell. As a result of this decrease in pressure, dissolved gases will partially pass out of the water phase into the gas phase. Data on gases dissolved in the reactor water under these conditions will have little meaning in responding to the accident. During accidents in which only a small amount of cladding damage has occurred or in accidents in which the reactor vessel has not been depressurized, pressurized reactor water samples may be obtained from the Process Sampling System. Therefore, the ability to obtain pressurized or depressurized reactor water samples for dissolved gas analyses has not been provided.

(5) If both of the following are present:

(a) There is only a single barrier between primary containment and the cooling water.

(b) If the cooling water is sea water or brackish water, chloride analysis within 24 hours after sampling shall be provided. If both are not present, the time to complete the analyses is increased to 4 days. Analysis does not have to be done onsite. Meets the requirements of NUREG-0737. (Note that there are several barriers to prevent chloride intrusion from the power cycle cooling water into the reactor vessel. These barriers are: the main condenser tubing, the condensate polishing demineralizers and the pumps and valves in the condensate/feedwater systems. These pumps are stopped and these valves closed during upset conditions. Thus, because both factors are not present, the time to complete the analysis is increased to 4 days.)

(6) It must be possible to obtain and analyze a sample without radiation exposures to any individual exceeding 0.05 Sv for whole body and 0.50 Sv for extremities. Meets the requirements of 50.34(f)(2)(viii).

(7) Ability to sample and analyze for reactor coolant boron must be provided. Meets the requirements of NUREG-0737.

(8) If inline monitoring is used, backup sampling and analysis capability must be provided. Inline monitoring is not used. Meets the requirements of NUREG-0737.

(9)

(a) Capability to identify and quantify a specified number of isotopes over a range of nuclide concentrations from approximately 37,000 Bq/g to 370,000 M Bq/g. Capability is provided to identify and quantify the desired isotopes in samples over a range from approximately 37,000 Bq/g to 37,000 M Bq/g. Samples

obtained during the accident recovery phase would be within this range for most core damage accidents. If the gross radioactivity levels are higher than 37,000 M Bq/g, this would confirm that severe core damage has occurred.

(b) Restrict background levels of radiation in the laboratory and provide proper ventilation. Meets the requirements of NUREG-0737.

(10) Provide adequate accuracy, range and sensitivity to provide pertinent information. Meets the requirements of NUREG-0737.

(11)

(a) Provide sample lines with proper features for sampling during accident conditions. Meets the requirements of NUREG-0737.

(b) PASS ventilation exhaust should be filtered with charcoal adsorbers and HEPA filters. Meets the requirements of NUREG-0737.

Summary

The post-accident sampling system meets the requirements of the NRC position with the following exceptions:

(1) The upper limit of activity in the samples at the time they are taken is as follows:

(a) Liquid sample 37,000 M Bq/ml

(b) Gas sample 3700 M Bq/ml.

(2) Radiological measurements could be performed 24 hours following the accident.

(3) Boron concentration measurements could be performed 8 hours following the accident.

(4) There is no need to perform chloride measurements.

(5) There is no need to analyze dissolved gases.

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

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

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

NRC Position

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

Response

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

A generic test program was conducted through the BWR Owners' Group (Reference 1A-10) to satisfy the discharge of steam. These steam discharge test results will be used as the qualification basis for plant-specific SRV models and discharge piping that are sufficiently similar to those reported in Reference 1A-11. [*Plant-specific SRV models and discharge piping that are not similar will be tested in accordance with NUREG-0737 requirements.*]* See Subsection 1A.3.7 for COL license information.

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

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

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

NRC Position

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

Response

The ABWR Standard Plant SRVs are equipped with position sensors which are qualified as Class 1E components. These are used to monitor valve position.

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

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

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

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

* See Subsection 3.9.1.7.

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

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

1A.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

The Flammability Control System, including the recombiners, has been deleted from the ABWR design as described in Subsection 6.2.5. Accordingly, no penetrations are required for the recombiners.

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

NRC Position

(1) Containment isolation system designs shall comply with the recommendations of the 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 non-essential 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 resetting 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. The basis for classification is also presented in Section 3.2. The ESF system, with remote manual valves with leakage detection outside the containment, is delineated 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 consistent with this position.

(6) All ABWR containment purge valves meet the criteria provided in BTP CSB 6-4. The main 550A 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 presence 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]

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 3.7E+09 M BQ/cc (Xe133) 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 3.7E+09 M Bq/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 1E+06 Gy/h 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 –34.32 kPa G 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 2.27×10^6 liter capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 1.52 meters 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

The requirements of Regulatory Guide 1.97, incorporate the above requirements. Section 7.5 compares the ABWR design against this Regulatory Guide and Subsection 18.8.13 addresses the operator interface of the instrumentation.

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. The ABWR arrangement of reactor water level sensing complies with the NRC Generic Letter 84-23. The vertical drop inside the drywell of the reactor pressure vessel reference leg water level instrument lines is limited to 0.9 meters. Analog level transmitters are employed to monitor the reactor vessel water level. For the safety related functions initiated automatically upon receipt of a reactor pressure vessel water level trip signal, two-out-of-four trip initiation logic is employed, utilizing a signal from a level transmitter in each of the four instrument divisions. This provides high reliability for initiation upon demand, and high tolerance against inadvertent initiation.

To address the US NRC staff’s concerns about the potential for reactor pressure vessel water level measurement errors resulting from dissolved non-condensable gasses in the water column in the reactor pressure vessel reference leg water level instrument lines (NRC Generic Letter 92-04 and NRC Information Notice 93-27), the ABWR has implemented continuous purging of the reactor pressure vessel reference leg water level instrument lines. Water is continuously injected into the reactor water level reference leg water level instrument lines by means of the Control Rod Drive (CRD) System. This is shown in Figure 5.1-3 and discussed in Subsection 7.7.1.1.

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 additional instrumentation in the ABWR design. Subsection 18.8.14 addresses the COL license information requirements for these instruments and controls.

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

NRC Position

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

Response

The ABWR Standard Plant is designed in accordance with Regulatory Guide 1.97 Rev. 3. A detailed assessment of the Regulatory Guide, including the list of instruments, is found in Section 7.5. The hydrogen and oxygen monitors are declassified to nonsafety-related, as permitted by Regulatory Guide 1.7, Revision 3.

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

NRC Position

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

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

Response

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

In addition to the status monitoring, the COL applicant plant-specific procedures (Subsection 1A.3.2) will assure that independent verification of system lineups is applied to valve and electrical lineups for all safety-related equipment, to surveillance procedures, to restoration following maintenance and to comply with IE Bulletin 79-08. Through these procedures, approval will be required for the performance of surveillance tests and maintenance, including equipment removal from service and return to service.

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

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

NRC Position

Review and modify (as required) procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. The COL applicant must verify

the operability of safety-related systems after performing maintenance or tests as part of the test to restore a system to service.

Response

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

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

NRC Position

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

Response

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

In the nonaccident case, the RCIC System is normally the only makeup system utilized to furnish subsequent makeup water to the RPV. When the water level reaches Level 2 again due to loss of inventory through the main steam relief valves or to the main condenser, the RCIC System automatically restarts as described in Subsection 1A.2.22. This system then maintains the coolant makeup supply. RPV pressure is regulated by the automatic or manual operation of the main turbine bypass valves which discharge to the condenser.

To remove decay heat during a planned isolation event, assuming that the main condenser is not available, the SRVs are utilized to dump the residual steam to the suppression pool. Suppression pool temperature is monitored by the Suppression Pool Temperature Monitoring (SPTM) System. When the pool temperature increases to a selected setpoint, the SPTM System signals the RHR System to cause automatic initiation of the suppression pool cooling (SPC) mode of RHR. The SPC mode cools the suppression pool by routing pool water through the RHR heat exchanger to cool it, and returning it to the suppression pool. SPC may also be affected by manual alignment of the RHR System. Makeup water to the RPV is still supplied by the RCIC System.

For the accident case with the RPV at high pressure, the HPCF Systems can also be utilized to automatically provide the required makeup flow when the water level drops below Level 1.5 setpoint. No manual operations are required. If the HPCF Systems are postulated to fail at these same conditions and the RCIC capacity is insufficient, the Automatic Depressurization System

(ADS) will automatically initiate depressurization of the RPV to permit the low pressure ECCS/LPFL mode of the RHR System to provide makeup coolant.

Therefore, it can be seen that, although manual actions can be taken to mitigate the consequences of a loss of feedwater, there are no short-term manual actions which must be taken. Sufficient systems exist to automatically mitigate these consequences.

1A.2.21 Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation of Safety Systems [II.K.1(23)]

NRC Position

For boiling water reactors, describe all uses and types of reactor vessel level indication for both automatic and manual initiation of safety systems. Describe other instrumentation that might give the operator the same information on plant status (Bulletin 79-08, Item 4).

Response

The water-level measurement for BWRs, in general, is fully described in NEDO-24708A, “Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors”. An outline of this description as applicable to the ABWR Standard Plant is provided in the following paragraphs.

Figure 7.7-1 illustrates the reactor vessel elevations covered by each water-level range. Additional details may be found in Figure 5.1-3 (Nuclear Boiler System P&ID). The instruments that sense the water level are differential pressure devices calibrated to be accurate at a specific vessel pressure and liquid temperature condition. The following is a description of each water-level range.

(1) **Shutdown Water-Level Range**—This range is used to monitor the reactor water level during the shutdown condition when the reactor system is flooded for maintenance and head removal. The water-level measurement design is the condensate reference chamber leg type that is not compensated for changes in density. The vessel temperature and pressure conditions that are used for the calibration are 48.9°C and 0 kPaG water in the vessel. The two vessel instrument penetration elevations used for this water-level measurement are located at the top of the RPV head and the instrument tap just below the bottom of the dryer skirt.

(2) **Narrow Water-Level Range**—This range uses for its RPV taps the elevation above the main steam outlet nozzle and the tap at an elevation near the bottom of the dryer skirt. The instruments are calibrated to be accurate at the normal operating points. The water-level measurement design is the condensate reference chamber type, is not density compensated, and uses differential pressure devices as its primary elements. The Feedwater Control System uses this range for its water-level control and indication inputs.

(3) **Wide Water-Level Range**—This range uses for its RPV taps the elevation above the main steam outlet nozzle and the taps at an elevation near the top of the active fuel. The instruments are calibrated to be accurate at the normal power operating point. The water-level measurement design is the condensate reference type, is not density compensated, and uses differential pressure devices as its primary elements. These instruments provide inputs to various safety systems and engineered safeguards systems.

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

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

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

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

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

NRC Position

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

Response

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

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

NRC Position

Currently, the Reactor Core Isolation Cooling (RCIC) System and the High-Pressure Coolant Injection (HPCI) System both initiate on the same low-water-level signal and both isolate on the same high-water-level signal. The HPCI System will restart on low water level but the RCIC System will not. The RCIC System is a low-flow system when compared to the HPCI System. The initiation levels of the HPCI and RCIC Systems should be separated so that the RCIC System initiates at a higher water level than the HPCI System. Further, the initiation logic of

the RCIC System should be modified so that the RCIC System will restart on low-water-level. These changes have the potential to reduce the number of challenges to the HPCI System and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analyses.

Response

The ABWR Standard Plant design is consistent with this position. The High-Pressure Core Flooder (HPCF) System is initiated at Level one and one half , and the RCIC System is initiated at Level 2. At Level 8, the injection valves for the HPCF and the RCIC steam supply and injection valves will automatically close in order to prevent water from entering the main steamlines.

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

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

NRC Position

The High-Pressure Coolant Injection (HPCI)and Reactor Core Isolation Cooling (RCIC) Systems use differential pressure sensors on elbow taps in the steamlines to their turbine drives to detect and isolate pipe breaks in the systems. The pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC Systems due to the pressure spike which accompanies startup of the systems. The pipe-break-detection circuitry should be modified to that pressure spikes resulting from HPCI and RCIC System initiation will not cause inadvertent system isolation.

Response

The ABWR design utilizes the motor-driven HPCF System rather than the turbine-driven HPCI System for high pressure inventory maintenance. Therefore, this position is only applicable to the turbine-driven RCIC System.

The ABWR design for the RCIC System utilizes a flow control system that is an integral part of the pump and turbine. Pump discharge passes through a venturi. The pressure differential between the venturi inlet and throat work together with a balance piston and spring to control the steam flow to the turbine, which in turn adjusts the pump speed and flow.

This is an improvement relative to existing BWR plant designs in which flow control is performed external to the pump and turbine, where flow is measured in a flow element and evaluated in a flow controller to generate an electrical signal to an electro-hydraulic flow

control valve to signal a servo to adjust the position of the control valve. See Subsection 1A.3.8 for COL license information requirements.

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

NRC Position

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

(1) Additional anticipatory scram on loss of feedwater

(2) Revised relief-valve actuation setpoints

(3) Increased emergency core cooling (ECC) flow

(4) Lower operating pressures

(5) Earlier initiation of ECC systems

(6) Heat removal through emergency condensers

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

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

(9) Increasing the high steamline flow setpoint for main steamline isolation valve (MSIV) closure

(10) Lowering the pressure setpoint for MSIV Closure

(11) Reducing the testing frequency of the MSIVs

(12) More stringent valve leakage criteria

(13) Early removal of leaking valves

An investigation of the feasibility and constraints of reducing challenges to the relief valves by use of the aforementioned methods should be conducted. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief-valve

challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude).

Response

GE and the BWR Owners' Group responded to this requirement in Reference 1A-6. This response, which was based on a review of existing operating information on the challenge rate of relief valves, concluded that the BWR/6 product line had already achieved the "order of magnitude" level of reduction in SRV challenge rate. The ABWR relief valve system also has similar design features which also reduce the SRV challenge rate. With regard to inadvertently opened relief valves (IORV), the BWR/6 plant design evaluated for the Owners' Group report reflected a reduced level if IORV compared with the previous design because of elimination of the pilot-operated relief valve type of design. The ABWR design has also eliminated the pilot-operated relief valve type of design.

For the ABWR, which has a solid-state logic design with redundancy, the likelihood of an IORV is the same or less than the BWR/6 design evaluated in connection with the Owners' Group report. The redundant-solid state design has been selected in order that the frequency of IORV with solid state-logic becomes low enough so as to achieve the order of magnitude reduction in total SRV challenge rate required by NUREG-0737.

The redundant solid-state design for SRV operation in the pressure relief mode consists of two duplicated microprocessor channels. Each microprocessor channel activates a separate load driver and both load drivers must be activated to cause operation of the SRVs in the relief mode. Operation of the SRVs in the ADS mode also requires activation of two microprocessor channels with separate load drivers to prevent unwanted SRV operation; however, two separate dual channel systems are used to assure reliable operation in the ADS mode. Reliable operation in the pressure relief mode is assured by direct opening of the SRV against spring force.

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

NRC Position

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

Clarification

The present technical specifications contain limits on allowable outage times for ECC Systems and components. However, there are no cumulative outage time limitations on these same systems. It is possible that ECC equipment could meet present technical specification

requirements but have a high unavailability because of frequent outages within the allowable technical specifications.

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

Based on the above guidance and clarification, a detailed report should be submitted. The report should contain:

(1) Outage dates and duration of outages

(2) Causes of the outage

(3) ECC Systems or components involved in the outage

(4) Corrective action taken

Tests and maintenance outages should be included in the above listings which are to cover the last 5 years of operation. The licensee should propose changes to improve the availability of ECC equipment, if needed.

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

Response

See Subsection 1A.3.5 for COL license information requirements.

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

NRC Position

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

Response

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

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

1A.2.27 Restart of Core Spray and LPCI Systems on Low Level Design and Modification [II.K.3(21)]

NRC Position

The Core Spray and Low Pressure Coolant Injection (LPCI) Systems flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The Core Spray and LPCI system logic should be modified so that these systems will restart, if required, to assure adequate core cooling. Because this design modification affects several core-cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

Response

The ABWR Standard Plant Emergency Core Cooling System (ECCS) is made up of the High Pressure Core Flooder (HPCF) System, the Reactor Core Isolation Cooling (RCIC) System, the Automatic Depressurization System (ADS) and the low pressure flooder (LPFL) mode of the Residual Heat Removal (RHR) System.

A high water level (Level 8) signal will automatically close the HPCF injection valves and the RCIC steam supply and injection valves. These systems may also be shut down manually. Manual action is required to shut down the RHR once it is initiated.

In the unlikely event that a subsequent low level reoccurs, the RCIC steam supply and injection valves will automatically reopen to allow continued flooding of the vessel. The HPCF injection valves will also automatically reopen unless the operator previously closed them manually.

The NRC has suggested certain modifications to the LPCI (LPFL for the ABWR) and core flooder systems provided as part of the ECCS network. These NRC suggestions center on control system logic modifications that would provide automatic restart capability following manual termination of system operation.

General Electric and the BWR Owners' Group reviewed this issue on a generic basis in 1980, and concluded that the NRC suggestions were not required for plant safety considerations. Justification is provided in the December 29, 1980, BWR Owners' Group submittal to the NRC (Reference 1A-8). Plant variations between the BWR and the ABWR designs are not significant to the overall technical conclusions of the study.

This conclusion is based on the adequacy of the current ECCS logic design coupled with the potentially negative impact on overall safety of the proposed changes. For the low pressure ECCS, these negative impacts include a significant escalation of control system complexity and restricted operator flexibility when dealing with anticipated events.

A full understanding of the significance of these logic changes must be based on a recognition that these systems must consider the possible interactive effects among the other systems making up the overall ECCS network. This must also include the potential impact on supporting systems such as the standby power supplies and the shutdown service water systems. Furthermore, the LPFL is one mode of the RHR System which has other safety-related functions such as suppression pool cooling and wetwell/drywell spray cooling. Clearly, these other safety functions must not be compromised by any changes in the LPFL mode of operation.

The referenced systems analysis took into consideration these potential interactive effects, impacts on supporting systems, plant instrumentation and emergency procedure guidelines. The study concluded that auto-restart of these systems is not necessary or appropriate. Therefore, no modifications are necessary with respect to automatic restart of the low pressure ECCS.

1A.2.28 Automatic Switchover of Reactor Core Isolation Cooling System Suction—Verify Procedures and Modify Design [II.K.3(22)]

NRC Position

The Reactor Core Isolation Cooling (RCIC) System takes suction from the condensate storage tank (CST) with manual switchover to the suppression pool when the CST level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC System suction from the condensate storage tank to the suppression pool.

Response

The RCIC System provided in the ABWR Standard Plant includes an automatic switchover feature which will change the pump suction source from the condensate storage pool to the suppression pool. The safety-grade switchover will automatically occur upon receipt of a low-level signal from the condensate storage tank or a high-level signal from the suppression pool.

See Subsection 7.3.1.1.1.3 for additional information.

1A.2.29 Confirm Adequacy of Space Cooling for High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems [II.K.3(24)]

NRC Position

Long-term operation of the Reactor Core Isolation Cooling (RCIC) and High-Pressure Coolant Injection (HPCI) Systems may require space cooling to maintain the pump-room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating-current power. The RCIC and HPCI Systems should be designed

to withstand a complete loss of offsite alternating-current power to their support systems, including coolers, for at least 2 hours.

Response

The ABWR HPCF and the RCIC systems are provided space cooling via individual room safety grade air-conditioning systems (Subsection 9.4.5). If all offsite power is lost, space cooling for the HPCF and RCIC System equipment would not be lost because the motor power supply for each system is from separate essential power supplies.

1A.2.30 Effect of Loss of Alternating Current Power on Pump Seals [II.K.3(25)]

NRC Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (AC) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

Response

The ABWR design features internal recirculation pumps called Reactor Internal Pumps (RIP) which do not require shaft seals. During a Loss of Preferred Power (LOPP), the RIPs shutdown automatically but there are no shaft seals which require cooling water restoration.

A plant AC power failure would temporarily disrupt the operation of the reactor recirculation subsystems but their failure would not generate a LOCA, as the following describes.

(1) **RMC Failure**—Subsection 5.4.1.3.1 describes the RMC operation during normal running or stopped condition. This operation assumes that the Reactor Building Cooling Water (RCW) System is in operation continuously during these operating or stopped conditions. Normal LOCA and LOPP operation of the RCW are described in Subsection 9.2.11.2.

A loss of AC power or Loss of preferred power (LOPP) will stop all RIPs. The LOPP will also temporarily stop the RCW and RSW pumps. The onsite emergency power sources will automatically restart the RCW/RSW pumps, which will restore cooling for the stopped RIP motors. The RCW primary containment isolation valves will not close on LOPP (only on LOCA).

The RMC Subsystem for each RIP includes motor cooling inlet and outlet temperature detectors. The outlet temperature detectors will automatically run back individual RIPs to minimum speed and subsequently trip on high-high coolant temperature and prevent motor damage.

The RIP motors are designed and will be plant tested to not be damaged in the stopped hot standby condition indefinitely with RCW cooling available.

(2) **RMP Failure**—Subsection 5.4.1.3.2 describes the RMP operation. Since the RIP and motor have no seals, the water in the RIP motor communicates directly with the reactor water at the same pressure but at much lower temperature. There is no possibility of this water escaping from the coolant pressure boundary such as in conventional pumps, which include seals.

The RMP water is supplied from the CRD System. The CRD pumps will stop temporarily during a LOPP, which will cause the normal RMP flow to each RIP to temporarily stop. The CRD pumps are subsequently restarted automatically, after several minutes time delay, powered by onsite power sources and the RMP water will restart.

This temporary interruption of RMP flow will not initiate a LOCA. The only effect of loosing the RMP flow temporarily to the RIPs, from a LOPP, is that it will allow reactor contamination, by diffusion, to enter the RIP motor during the RMP flow interruption.

(3) **RMISS Failure—**Subsection 5.4.1.3.3 describes the operation of the RMISS, which is used only during plant shutdown and RIP maintenance. The power source for the inflatable seal is a portable air-operated water pump which is moved from RIP to RIP. A LOPP would therefore not cause a direct loss of RMISS pressure, since (1) the plant air system has a finite passive storage capacity in the air receivers, and (2) the RMISS air-operated pump only operates when the RMISS pressure drops below a preset value.

The RMISS is a secondary seal. Even with a long time RMISS failure RIP maintenance, the passive backseat seal of the RIP shaft on the stretch tube will preclude draining the reactor.

1A.2.31 Study and Verify Qualification of Accumulators on Automatic Depressurization System Valves [II.K.3(28)]

NRC Position

Safety analysis reports claim that air or nitrogen accumulators for the Automatic Depressurization System (ADS) valves are provided with sufficient capacity to cycle the valves open five times at normal drywell pressures. GE has also stated that the Emergency Core Cooling (ECC) Systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. The licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable.

Response

The accumulators for the ADS valves are sized to provide one actuation at drywell design pressure and five actuations at normal drywell pressure. This cyclic capability is validated during preoperational testing at the station. The accumulators are safety-grade components.

The 100-day, post-accident functional operability requirement is met through conservative design and redundancy; eight ADS valves are provided with code-qualified accumulators and Seismic Category I piping within primary containment. Two redundant seven-day supplies of bottled air are available to compensate for leakage during long-term usage, with replacement capability being provided for the remainder of the postulated accident to assure system functional operability. A maximum of three of eight ADS valves need to function to meet short-term demands (Subsection 19.3.1.3.1) and the functional operability of only one ADS valve will fulfill longer term needs. Loss of pneumatic supply pressure to the ADS SRV accumulator is alarmed to provide the reactor operator with indication of the failure of any of the redundant systems under hostle environmental conditions.

The BWR Owners' Group sponsored an evaluation of the adequacy of the ADS configurations. Evaluation results are summarized in the following paragraph.

The accumulators are designed to provide one actuation at drywell design pressure and five actuations at normal drywell pressure. See Subsection 6.7.1 for a description of the ADS N_2 pneumatic supply.

1A.2.32 Revised Small-Break Loss-of-Coolant-Accident Methods to Show Compliance with 10CFR50, Appendix K [II.K.3(30)]

NRC Position

The analysis methods used by Nuclear Steam Supply System (NSSS) vendors and/or fuel suppliers for small break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10CFR50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Response

GE has evaluated the NRC request requiring that the BWR small-break LOCA analysis methods are to be demonstrated to be in compliance with Appendix K to 10CFR50 or that they be brought into compliance by analysis methods changes. The specific NRC concerns are contained in NUREG-0626, Appendix F. The specific NRC concerns identified in Subsection 4.2.10 of NUREG-0626 (Appendix F) relate to the following: counter current flow limiting (CCFL) effects, core bypass modeling, pressure variation in the reactor pressure vessel, integral experimental verification, quantification of uncertainties in predictions, the recirculation line inventory modeling, and the homogeneous/equilibrium model.

The response to the NRC small-break model concerns was provided at a meeting between the NRC and GE on June 18, 1981. Information provided at this meeting showed that, based on the TLTA small-break test results and sensitivity studies, the existing GE small-break LOCA model already satisfies the concerns of NUREG-0626 and is in compliance with 10CFR50, Appendix K. Therefore, the GE model is acceptable relative to the concerns of Item II.K.3(30), and no model changes need be made to satisfy this item.

Documentation of the information provided at the June 18, 1981 meeting was provided via the letter from R. H. Buchholz (GE) to D. G. Eisenhut (NRC), dated June 26, 1981.

1A.2.33 Plant-Specific Calculations to Show Compliance with 10CFR50.46 [II.K.3(31)]

NRC Position

Plant-specific calculations using NRC-approved models for small break loss-of-coolant accidents (LOCAs) as described in Item II.K.3.30 to show compliance with 10CFR50.46 should be submitted for NRC approval by all licensees.

Response

The ABWR standard safety small-break LOCA calculations are discussed in Subsection 6.3.3.7.

The references listed in Subsection 6.3.6 describe the currently approved Appendix K methodology used to perform these calculations. Compliance with 10CFR50.46 has previously been established for that methodology.

Since, as noted in the previous Item (1A.2.32), no model changes are needed to satisfy NUREG-0737, Item II.K.3(30), there is no need to revise the calculations presented in Subsection 6.3.3.7.

1A.2.33.1 Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure [II.K.3 (44)]

NRC Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which occur in a stuck-open relief valve should be included in this category. The results of the evaluation are due January 1, 1981.

Response

GE and the BWROG have concluded, based on a representative BWR/6 plant study, that all anticipated transients in Regulatory Guide 1.70, Revision 3, combined with the worst single failure, the reactor core remains covered with water until stable conditions are achieved. Furthermore, even with more degraded conditions involving a stuck-open relief valve in addition to the worst transient (loss of feedwater) and worst single failure (of high pressure core spray), studies show that the core remains covered and adequate core cooling is available during the whole course of the transient (NEDO-24708, March 31, 1980). The conclusion is applicable to the ABWR. Since the ABWR has more high pressure makeup systems (2HPCFs and 1 RCIC), the core covering is further assured.

Other discussions of transients with single failure is presented in the response to NRC Question 440.111.

1A.2.33.2 Evaluate Depressurization Other Than Full ADS [II.K.3 (45)]

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 cooldown (Applicable to BWRs only).

Response

This response is provided in Subsection 19A.2.11.

1A.2.33.3 Responding to Michelson Concerns [II.K.3 (46)]

NRC Position

General Electric should provide a response to the Michelson concerns as they relate to boiling water reactors.

Clarification: General Electric provided a response to the Michelson concerns as they relate to boiling water reactors by letter dated February 21, 1980. Licensees and applicants should assess applicability and adequacy of this response to their plants.

Response

All of the generic February 21, 1980 GE responses were reviewed and updated for the ABWR Standard Plant. The specific responses are provided in Table 1A-1.

1A.2.34 Primary Coolant Sources Outside Containment Structure [III.D.1.1(1)]

NRC Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate Leak Reduction

   (a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment

   (b) Measure actual leakage rates with systems in operation and report them to the NRC

(2) Continuing Leak Reduction—establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response

Leak reduction measures of the ABWR Standard Plant include a number of barriers to containment leakage in the closed systems outside the containment. These closed systems include:

(1) Residual Heat Removal

(2) High Pressure Core Flooder

(3) Low Pressure Core Flooder

(4) Reactor Core Isolation Cooling

(5) Suppression Pool Cleanup

(6) Reactor Water Cleanup

(7) Fuel Pool Cooling and Cleanup

(8) Post-Accident Sampling

(9) Process Sampling

(10) Containment Atmospheric Monitoring

(11) Fission Product Monitor (Part of LDS)

(12) NOT USED

(13) Standby Gas Treatment

Plant procedures will prescribe the method of leak testing these systems. The testing will be performed on a schedule appropriate to 10CFR50 Appendix J type B and C penetrations (i.e., at each refueling outage). When leakage paths are discovered, including during these tests, they will be investigated and necessary maintenance will be performed to reduce leakage to its lowest practical level.

In addition, lines which penetrate the primary containment are equipped with inboard and outboard isolation valves that are designed in accordance with GDC 55, 56 and 57 to provide reliable isolation in the event of a line break or leakage. The containment isolation provisions are discussed in detail in Subsection 6.2.4, which also identifies all the system lines that penetrate the containment, together with their respective isolation valves.

Leakage within and outside the primary containment are continuously monitored by the Leak Detection and Isolation System (LDS) for breach in the integrity of the containment. Upon detection of a leakage parameter, the LDS will automatically initiate the necessary control

functions to isolate the source of the break and alerts the operator for corrective action. The MSL tunnel area is monitored for high ambient temperatures that are indicative of steam leakage. The Turbine Building is also monitored for high area ambient temperatures for MSL leakage. The resulting action causes isolation of the MSIVs and subsequent shutdown of the reactor.

The radiation levels in the HVAC air exhaust ducts of the Reactor Building and the fuel handling area are monitored for use in isolating the secondary containment. The results in closure of the HVAC air ducts in the Reactor Building, closure of the containment purge and vent lines, and activation of the Standby Gas Treatment System (SGTS).

The leak detection methods and associated instrumentation that monitor leakage from the reactor coolant pressure boundary are described in Subsection 5.2.5.

For small line breaks in the secondary containment that could cause significant release of radioactive material will be detected by process radiation monitors in the Reactor Building HVAC air ducts. As indicated above, a high radiation level will activate the SGTS (Subsection 6.5.1) prior to the release of radiation to the environment. Also, any fluid leakage will drain into the sumps in the Reactor Building and will be monitored by sumps instrumentation for level and flow rate. The operator will be alerted to any abnormal condition for action. All lines which pass outside of the secondary containment contain leakage detection systems or loop seals. These systems allow the SGTS to maintain a negative pressure relative to the environment and thus limit the amount of leakage through the secondary containment. These systems are discussed in Subsection 6.2.3. Finally, expected liquid leakoff from equipment outside the containment is directed to equipment drain sumps and processed by the Radwaste System. These multiple design features of the ABWR Standard Plant provide substantial capability to limit any potential release to the environment from systems likely to contain radioactive material.

Additionally, pressure boundary components of radioactive waste systems are purchased as augmented Class D systems to assure their capability to provide integrity.

1A.2.35 In-Plant Radiation Monitoring [II.D.3.3(3)]

NRC Position

(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Response

(1) See Subsection 1A.3.3 for COL license information requirements.

(2) Not applicable.

1A.2.36 Control Room Habitability [III.D.3.4(1)]

NRC Position

In accordance with Task Action Plan Item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, “Control Room” of Appendix A, General Design Criteria for Nuclear Power Plants, to 10CFR50).

Response

This requirement is satisfied for the ABWR by the provisions of the Control Building Atmospheric Control System(ACS).

Section 7.1 describes the Control Building ACS instrumentation and controls for assuring control room habitability. Subsection 6.4 provides HVAC design details. As demonstrated by the analyses provided in these subsections, these systems satisfy Criterion 19, Appendix A of 10CFR50.

The ABWR design satisfies this item.

1A.3 COL License Information

1A.3.1 Emergency Procedures and Emergency Procedures Training Program

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

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

Procedures shall be reviewed and modified (as required) for removing safety-related systems from service (and restoring to service) to assure operability status is known (Subsections 1A.2.18 and 1A.2.19).

1A.3.3 In-Plant Radiation Monitoring

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

1A.3.4 Reporting Failures of Reactor System Relief Valves

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

1A.3.5 Report on ECCS Outages

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

(1) Outage date

(2) Duration of outage

(3) Cause of outage

(4) Emergency core cooling system or component involved

(5) Corrective action taken

The above information shall be assembled into a report, which will also include a discussion of any changes, proposed or implemented, deemed appropriate, to improve the availability of the emergency core cooling equipment (Subsection 1A.2.25).

1A.3.6 Procedure for Reactor Venting

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

1A.3.7 Testing of SRV and Discharge Piping

The COL applicant will confirm that any SRVs or discharge piping installed that is not similar to those that have been tested will be tested in accordance with Subsection 1A2.9.

1A.3.8 RCIC Bypass Start System Test

The COL applicant shall perform a RCIC start system test to ensure the pressure spikes described in Subsection 1A.2.23 do not occur. This test will be conducted during plant startup.

1A.4 References

1A-1 Memo, C. Michelson to D. Okrent, “Possible Incorrect Operator Action Such as Pipe Break Isolation”, June 4,1979.

1A-2 Letter, D. G. Eisenhut (NRC)to R. L. Gridley(GE), “Potential for Break Isolation and Resulting GE-Recommended BWR/3 ECCS Modifications”, June 14, 1978.

1A-3 “Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors”, NEDO-24708, August 1979.

1A-4 U. S. Nuclear Regulatory Commission, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, USNRC report NUREG-0660, Vols. 1 and 2, May 1980.

1A-5 U. S. Nuclear Regulatory Commission, “Clarification of TMI Action Plan Requirements”, USNRC Report NUREG-0737, November 1980.

1A-6 U. S. Nuclear Regulatory Commission, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License”, NUREG-0718, Revision 1, June 1981.

1A-7 Letter, R. H. Buchholz (GE) to D. F. Ross (NRC), Subject: Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, November 30, 1979, MFN-290-79.

1A-8 Letter, D. B. Waters (Chairman, BWR Owners’ Group) to NRC staff, dated December 29, 1980, “BWR Owners’ Group Evaluation of NUREG-0737 Requirements.”

1A-9 Letter, D. B. Waters (Chairman, BWR Owners’ Group) to D. G. Eisenhut (NRC), dated March 31, 1981, “BWR Owners’ Group Evaluation of NUREG-0737 Requirements II.K.3.16 and II.K.3.18.”

1A-10 Letter, D.B.Waters (Chairman, BWR Owners’ Group) to R.H. Vollmer (NRC), dated September 17, 1980, NUREG-0578 “Requirements 2.12-Performance Testing of BWR and PWR Relief and Safety Valves.”

1A-11 NEDE-24988-P, “Analysis of Generic BWR Safety/Relief Valve Operability Test Results”, Proprietary Document, October 1981.

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)]

Question 1

Pressurizer level is an incorrect measure of primary coolant inventory.

Response 1

BWRs do not have pressurizers. BWRs measure primary coolant inventory directly using differential pressure sensors attached to the reactor vessel. Thus, this concern does not apply to the ABWR.

Question 2

The isolation of small breaks (e.g., letdown line; PORV) not addressed or analyzed.

Response 2

Automatic isolation only occurs for breaks outside the containment. Such breaks are addressed in Section 3.1.1.1.2 of NEDO-24708. It was shown that if high pressure systems are available, no operator actions are required. If it is assumed that all high pressure systems fail, the operator must manually depressurize to allow the low pressure systems to inject and maintain vessel water level. Analyses in Section 3.5.2.1 of NEDO-24708 show that the operator has sufficient information and time to perform these manual actions. The necessary manual actions have been included in the operator guidelines for small-break accidents.

Question 3

Pressure boundary damage due to loadings from (1) bubble collapse in subcooled liquid and 2) injection of ECC water in steam-filled pipes.

Response

The BWR has no geometry equivalent to that identified in Michelson’s report on B&W reactors relative to bubble collapse (steam bubbling upward through the pressurizer surge line and pressurizer). Thus, the first concern is not applicable to the ABWR.

ECC injection in the ABWR at high pressure is either into the reactor vessel throuh water-filled lines (RHR-B+C;HPCF-B+C) or into the feedwater lines (RHR-A:RCIC). The feedwater lines are normally filled with relatively cold liquid (251.5^{o}C or less). ECCS injection in the ABWR at low pressure is either directly into the reactor vessel or into the feedwater lines. Thus, the second concern is not applicable to the ABWR.

Question 4

In determining need for steam generators to remove decay heat, consider that break flow enthalpy is not core exit enthalpy.

Response 4

BWRs do not use steam generators to remove decay heat, so this concern does not apply to the ABWR.

Question 5

Are sources of auxiliary feedwater adequate in the event of a delay in cooldown subsequent to a small LOCA?

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)] (Continued)

Response 5

BWRs do not need feedwater to remove heat from the reactor following a LOCA, whether the subsequent cooldown is delayed or not. Therefore, this concern is not applicable to the ABWR. BWRs have a closed cooling system in which vessel water flows out the postulated break to the suppression pool. The suppression pool is cooled and water is pumped back to the vessel with ECCS pumps.

Question 6

Is the recirculation mode of operation of the HPCI pumps at high pressure an established design requirement?

Response 6

The high-pressure injection systems utilized in the ABWR are the Reactor Core Isolation Cooling (RCIC) and High Pressure Core Flooder (HPCF) Systems.

The RCIC and HPCF Systems normally take suction from the condensate storage tank and have an alternate suction source from the suppression pool. A recirculation mode of operation of these systems is established when the system suction is from the suppression pool. Following a LOCA when system suction is from the suppression pool, water injected into the reactor is discharged through the break and flows back to the suppression pool, forming a closed recirculation loop.

Other recirculation modes include test modes (e.g., suction from and discharge to the suppression pool) and system operation on low flow bypass with discharge to the suppression pool.

All of these modes are established design requirements.

Question 7

Are the HPCI pumps and RHR pumps run simultaneously? Do they share common piping?/suction? If so, is the system properly designed to accommodate this mode of operation (i.e., are any NPSH requirements violated, etc...)?

Response 7

For the ABWR, the high-pressure injection systems (RCIC/HPCF) do not share any common suction piping with the low pressure RHR and they can operate simultaneously with this low pressure system.

The RCIC and HPCF Systems share common suction line from the condensate storage tank. The RHR shutdown cooling operating mode does not share any common suction piping with the RCIC or HPCF Systems. It is an established design requirement to size the suction piping, including shared piping, such that adequate NPSH is available to RCIC and HPCF pumps for all operating modes of these systems.

Pre-operational and/or startup tests are conducted that demonstrate that the NPSH requirements are met.

Question 8

Mechanical effects of slug flow on steam generator tubes need to be addressed (transitioning from solid natural circulation to reflux boiling and back to solid natural circulation may cause slug flow in the hot leg pipes).

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)] (Continued)

Response 8

BWRs do not have steam generators, so this concern does not apply to ABWR.

Question 9

Is there minimum flow protection for the HPCI pumps during the recirculation mode of operation?

Response 9

For the ABWR, the RCIC, RHR, and HPCF pumps all contain valves, piping, and automatic logic that bypasses flow to the suppression pool as required to provide minimum flow protection for all design basis operating modes of the systems.

Question 10

The effect of the accumulators dumping during small-break LOCAs is not taken into account.

Response 10

BWRs do not use accumulators to mitigate LOCAs. Therefore, this concern does not apply to the ABWR.

Question 11

What is the impact of continued running of the RC pumps during a small LOCA?

Response 11

The impact of continued running of the recirculation pumps has been addressed in Subsections 3.3.2.2, 3.3.2.3, and Subsection 3.5.2.1.5.1 of NEDO-24708. The conclusions were that continued running of the recirculation pumps results in little change in the time available for operator actions and does not significantly change the overall system response.

Question 12

During a small break LOCA in which offsite power is lost, the possibility and impact of pump seal damage and leakage has not been evaluated.

Response 12

The RCIC, HPCF, and RHR pumps are provided with mechanical seals. No external cooling from auxiliary support systems, such as site service water or room air coolers, is required for RCIC pump seals. The HPCF and RHR seals are cooled by connections to the three separate divisions of the Reactor Building Cooling Water (RCW) System to protect against single failures. RHR Divisions A, B and C, and HPCF-B and C are connected to their corresponding RCW divisions. If offsite power is lost, onsite diesel generators maintain the RCW three-divisional function. These types of seals have demonstrated (in nuclear and other applications) their capability to operate for an extended period of time at temperatures in excess of those expected following a LOCA.

Should seal failure occur, it can be detected by room sump high level alarms. The RCIC, HPCF, and RHR individual pumps are arranged, and motor-operated valves provided, so that a pump with a failed seal can be shut down and isolated without affecting the proper operation of the other redundant pumps/systems.

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)] (Continued)

Considering the low probability of seal failure during a LOCA, the fact that a pump with a failed seal can be isolated without affecting other redundant equipment, and the substantial redundancy provided in the BWR emergency cooling systems, pump seal failure is not considered a significant concern.

Question 13

During transitioning from solid natural circulation to reflux boiling and back again, the vessel level will be unknown to the operators, and emergency procedures and operator training may be inadequate. This needs to be addressed and evaluated.

Response 13

There is no similar transition in the BWR case. In addition, the BWR has water level measurement within the vessel and the indication of the water level is incorporated into the operator guidelines. Consequently, this concern does not apply to ABWR.

Question 14

The effect of non-condensable gas accumulation in the steam generators and its possible disruption of decay heat removal by natural circulation needs to be addressed.

Response 14

The effect of non-condensable gas accumulation is addressed in Subsection 3.3.1.8.2 of NEDO-24708. For a BWR, vapor is present in the core during both normal operation and natural circulation conditions. Non-condensables may change the composition of the vapor but would have an insignificant effect on the natural or forced circulation itself, since the non-condensables would rise with the steam to the top of the vessel after leaving the steam separators.

Concern 15

Delayed cooldown following a small-break LOCA could raise the containment pressure and activate the containment spray system.

Response 15

The ABWR containment spray system is manually initiated. All essential equipment inside the containment is required to be qualified for the environmental conditions resulting from the initiation of the containment spray system.

Question 16*

This concern relates to the possibility that an operator may be included and perhaps even trained to isolate, where possible, a pipe break LOCA without realizing that it might be an unsafe action leading to high pressure, and short-term core bakeout. For example, if a BWR should experience a LOCA from a pressure boundary failure somewhere between the pump suction and discharge valve for either reactor recirculation pump, it would be possible for the operator to close these valves following the reactor blowdown to repressurize the reactor coolant system. Before such isolation should be permitted, it is first necessary to show by an appropriate analysis that the high pressure ECCS is adequate to reflood the uncovered core without assistance from the low pressure ECCS, which can no longer deliver flow because of the repressurization. Otherwise, such isolation action should be explicitly forbidden in the emergency operating instructions.

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)] (Continued)

Response 16
The ABWR does not have recirculation lines. However, there are other systems where it is possible for the operator to close these valves following the reactor blowdown to low pressure and thereby isolate the break. An example of this would be a break in the reactor water cleanup piping between the shutdown suction line valve and the containment boundary. In Reference 1A-2, the NRC concluded that, based on information provided by GE, break isolation is not a problem. In order for the reactor vessel to repressurize following isolation of a line break, the isolation would have to occur before initiation of ADS due to a high drywell pressure in concurrence with low water Level 1 condition. Isolation of a line break prior to obtaining a high drywell pressure signal might occur for very small breaks (area << 9.3 cm^2), which may require several hundred seconds following the break to reach the high drywell pressure setpoint. In this case it has been shown in Reference 1A-3 that the high pressure systems (RCIC, HPCF and feedwater) are sufficient to maintain the water level above the top of the core. If isolation of the break were to occur prior to reaching Level 1 but after the high drywell pressure signal, the vessel would pressurize to the SRV setpoint following isolation of the main steamlines and then oscillate as the SRVs cycle opened and closed. If no high pressure systems were available, the loss of mass out the SRVs would cause the level to continue dropping and result in automatic ADS actuation shortly after reaching Level 1. This would depressurize the vessel and allow the low pressure systems to begin injecting. This capability was demonstrated in NEDO-24708, explicitly to provide for manual depressurization in the event of low reactor water level with high pressure systems unable to maintain level for any reason. In summary, in order to repressurize the vessel following break isolation, the isolation would have to occur prior to ADS blowdown. For these cases, high pressure systems would maintain inventory. If no high pressure system was available, the low pressure systems would control the vessel water level following automatic or manual vessel depressurization.

* Excerpt from Reference 1A-1