GE Nuclear Energy

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

September 14, 1988

MFN No. 64-88

Docket No. STN 50-605

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Charles L. Miller, Director
Standardization and Non-Power Reactor Project Directorate

Subject: **Submittal of Responses to Additional Information as Requested in NRC Letter from Dino C. Scaletti, Dated July 7, 1988**

Dear Mr. Miller:

Enclosed are thirty four (34) copies of the Responses to Request for Additional Information (RAI) on the Standard Safety Analysis Report (SSAR) for the Advanced Boiling Water Reactor (ABWR). These responses principally pertain to Chapters 4, 5, 6 and 15. Also included are other committed responses to RAIs from Scaletti's letter, dated February 22, 1988.

It is intended that GE will amend the SSAR with these responses in December 1988.

Sincerely,

P. W. Marriott, Manager
Licensing and Consulting Services

cc: D. R. Wilkins (GE)
F. A. Ross (DOE)
J. F. Quirk (GE)
D. C. Scaletti (NRC)

D028
1/34

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

The reactivity control systems consists of control rods and control rod drives, supplementary reactivity control in the form of a gandolinia (Section 4.3), and the standby liquid control system (described in Subsection 9.3.5).

4401 Evaluations of the reactivity control systems against the applicable General Design Criteria (GDC) are contained in the folowing subsections:

GDC	Subsection
23	3.1.2.3.4
25	3.1.2.3.6
26	3.1.2.3.7
27	3.1.2.3.8
28	3.1.2.3.9
29	3.1.2.3.10

4.6.1 Information for Control Rod Drive System

4.6.1.1 Design Bases

4.6.1.1.1 Safety Design Bases

The control rod drive CRD mechanical system shall meet the following safety design bases:

(1) The design shall provide for rapid control rod insertion (scram) so that no fuel damage results from any moderately frequent event (see Chapter 15).

(2) The design shall include positioning devices, each of which individually supports and positions a control rod.

(3) Each positioning device shall be capable of holding the control rod in position and preventing it from inadvertently withdrawing outward during any non-accident, accident, post-accident and seismic condition.

(4) Each positioning device shall be capable of detecting the separation of the control rod from the drive mechanism to prevent a rod drop accident.

(5) Each positioning device shall provide a means to prevent or limit the rate of control rod ejection from the core due to a break in the drive mechanism pressure boundary. This is to prevent fuel damage resulting from rapid insertion of reactivity.

4.6.1.1.2 Power Generation Design Basis

The control rod drive system (CRDS) design shall meet the following power generation design bases:

(1) The design shall provide for controlling changes in core reactivity by positioning neutron-absorbing control rods within the core.

(2) The design shall provide for movement and positioning of control rods in increments to enable optimized power control and core power shaping.

4.6.1.2 Description

The CRDS consists of fine motion control rod drive (FMCRD) mechanisms, and the CRD hydraulic system (including pumps, filters, hydraulic control units, interconnecting piping, instrumentation and electrical controls). The CRDS, in conjunction with the rod control and information system (RC&IS) and reactor protection system (RPS) performs the following functions:

(1) Controls changes in core reactivity by positioning neutron-absorbing control rods within the core in response to control signals from the RC&IS.

(2) Provides movement and positioning of control rods in increments to enable optimized power control and core power shape in response to control signals from the RC&IS.

(3) Provides the ability to position large groups of rods simultaneously in response to control signals from the RC&IS.

(4) Provides rapid control rod insertion (scram) in response to manual or automatic signals from the RPS so that no fuel damage results from any plant transient.

(5) Gathers rod status and rod position data for rod pattern control, performance monitoring, operator display and scram time testing.

(6) Prevents undesirable rod pattern or rod motions by imposing rod motion blocks in order to protect the fuel.

(7) Prevents and mitigates the consequences of a rod drop accident by detecting rod separation and controlling rod pattern.

(8) Provides alternate rod insertion (ARI), an alternate means of actuating motor-driven rod insertion should an anticipated transient without scram (ATWS) occur.

(9) Automatically drives in the drive mechanisms with the electric motors upon scram initiation. This provides an additional, diverse means of fully inserting a control rod.

440.3 (10) Provides selected control rod run-in (SCRRI) for reactor stability control. (See Subsection 7.7.1.2.2.(2)).

The design bases and further discussion of both the RC&IS and RPS, and their control interfaces with the CRDS, are presented in Chapter 7.

4.6.1.1 Fine Motion Control Rod Drive Mechanisms

The fine motion control rod drive (FMCRD) used for positioning the control rod in the reactor core is a mechanical/hydraulic actuated mechanism (Figures 4.6-1, 4.6-2 and 4.6-3). An electric motor driven ball nut and spindle assembly is capable of positioning the drive at a minimum of 18.3mm increments. Hydraulic pressure is used for fast scrams. The FMCRD penetrates the bottom head of the reactor pressure vessel. The FMCRD does not interfere with refueling and is operative even when the head is removed from the reactor vessel.

The fine motion capability is achieved with a ball nut and spindle arrangement driven by an electric motor. The ball nut is keyed to the guide tube (roller key) to prevent its rotation and traverses axially as the spindle rotates. A hollow piston rests on the ball nut and upward motion of the ball nut drives this piston and the control rod into the core. The weight of the control rod keeps the hollow piston and ball nut in contact during withdrawal.

A single hydraulic control unit (HCU) powers the scram action of two FMCRDs. Upon scram valve initiation, high pressure nitrogen from the HCU raises the piston within the accumulator forcing water through the scram piping. This water is directed to each FMCRD connected to the HCU. Inside each FMCRD, high pressure water lifts the hollow piston off the ball nut and drives the control rod into the core. A spring washer buffer assembly stops the hollow piston at the end of its stroke. Departure from the ball nut releases spring loaded latches in the hollow piston that engage slots in the guide tube. These latches support the control rod in the inserted position. The control rod cannot be withdrawn until the ball nut is driven up and engaged with the hollow piston. Stationary fingers on the ball nut then cam the latches out of the slots and hold them in the retracted position. A scram action is complete when every FMCRD has reached their fully inserted position.

The use of the FMCRD mechanisms in the CRD system provides several features which enhance both the system reliability and plant operations. Some of these features are listed and discussed briefly as follows:

(1) Diverse Means of Rod Insertion

The FMCRDs can be inserted either hydraulically or electrically. In response to a scram signal, the FMCRD is inserted hydraulically via the stored energy in the scram accumulators. A signal is also given simultaneously to insert the FMCRD electrically via its motor drive. This diversity provides a high degree of assurance of rod insertion on demand.

(2) Absence of FMCRD Piston Seals

The FMCRD pistons have no seals and thus, do not require maintenance.

(3) FMCRD Discharge

The water which scrams the control rod discharges into the reactor vessel and does not require a scram discharge volume, thus eliminating a potential for common-mode failure.

(4) Improved Plant Maneuverability

The fine motion capability of the FMCRD allows rod pattern optimization in response operating buffer under criteria for normal and upset events and for an abnormally operating buffer under criteria for upset events.

(5) The control rod is designed for lateral displacements due to the maximum fuel channel deflection allowed within fuel channel design criteria under upset (OBE) events and faulted (SSE) events.

4.6.2.3.2 Control Rod Drives

4.6.2.3.2.1 Evaluation of Scram Time

The rod scram function of the CRD system provides the negative reactivity insertion required by safety design basis 4.6.1.1.1(1). The scram time shown in the description is adequate as shown by the transient analyses of Chapter 15.

4.6.2.3.2.2 Analysis of Malfunction Relating to Rod Withdrawal

4403 There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod. However, if multiple malfunctions are postulated, studies show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated.

4.6.2.3.2.2.1 Drive Housing Failure

The bottom head of the reactor vessel has a penetration for each CRD location. A drive housing is raised into position inside each penetration and fastened by welding. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing.

In the unlikely event of a failure of the drive housing to vessel attachment weld (including a failure through the housing or along the fusion line of the housing to stub tube weld) or the flange bolting attaching the drive to the housing, ejection of the CRD and attached control rod is prevented by the integral internal blowout support. The details of the this internal blowout support structure are contained in Section 4.6.1.2.2.9.

4.6.2.3.2.2.2 Rupture of Hydraulic Line to Drive Housing Flange

For the case of a scram insert line break, a partial or complete circumferential opening is postulated at or near the point where the line enters the housing flange. This failure, if not mitigated by special design features, could result in rod ejection at speeds exceeding maximum allowable limits of 4 in/sec (assuming rod pattern control) or 6 inches maximum travel distance before full stop. Failure of the scram insert line would cause loss of pressure to the underside of the hollow piston. The force resulting from full reactor pressure acting on the cross-sectional area of the hollow piston, plus the weights of the control rod and hollow piston, is imposed on the ball nut. The ball nut in turn translates this resultant force into a torque acting on the spindle. When this torque exceeds the motor residual torque and seal friction, reverse rotation of the spindle will occur permitting rod withdrawal. Analyses show that the forces generated during this postulated event can result in rod ejection speeds which exceed the maximum allowable limits.

The FMCRD design provides two diverse means of protection against the results of a postulated scram insert line failure. The first means of protection is a ball check valve located in the middle flange of the drive at the scram port. Reverse flow during a line break will cause the ball to move to the closed position. This will prevent loss of pressure to the underside of the hollow piston which in turn will prevent the generation of loads on the drive which could cause rod ejection.

The second means of protection is the centrifugal brake described in Subsection 4.6.1.2.2.8. In the event of the failure of the check valve, the centrifugal brake will actuate and stop the ball spindle rotation before the maximum allowable ejection speed (approximately 506 rpm corresponding to 4 in/sec) is reached.

4.6.2.3.2.2.3 Total Failure of All Drive Flange Bolts

The FMCRD design provides an anti-rotation

material adjacent to welds in Type 304 and Type 316 stainless steel piping systems has occurred in the past. Substantial research and development programs have been undertaken to understand the IGSCC phenomenon and develop remedial measures.
252.7 For the ABWR, IGSCC resistance has been achieved through the use of IGSCC resistant materials such as Type 316 Nuclear Grade stainless steel and stabilized nickel-base Alloy 600M and 182M.

Much of the early remedy-development work focused on alternative materials or local stress reduction, but recently the effects of water chemistry parameters on the IGSCC process have received increasing attention. Many important features of the relationship between BWR water chemistry and IGSCC of sensitized stainless steels have been identified.

252.7 Laboratory studies (References 3 and 4) have shown that although IGSCC can occur in simulated BWR startup environments, most IGSCC damage probably occurs during power operation. The normal BWR environment during power operation is ~280 ^{o}C water containing dissolved oxygen, hydrogen and small concentrations of ionic and nonionic impurities
252.7 (conductivity generally below 0.3 $\mu S/cm$ at 25^{o}C). It has been well documented that some ionic impurities (notably sulfate and chloride) aggravate IGSCC, and a number of studies have been made of the effects of individual impurity species on IGSCC initiation and
252.7 growth rates (References 3 thru 7). This work clearly shows that IGSCC can occur in water at 280^{o}C with 200 ppb of dissolved oxygen, even at low conductivity (low impurity levels), but the rate of cracking decreases with decreasing impurity content. Although BWR water chemistry guidelines for reactor water cannot prevent IGSCC, maintaining the lowest practically achievable impurity levels will minimize its rate of
252.7 progression (References 5 and 9).

Stress corrosion cracking of ductile materials in aqueous environments often is restricted to specific ranges of corrosion potential*, so a number of studies of impurity effects on IGSCC have been made as a function of either corrosion potential or dissolved oxygen content (the dissolved oxygen content is the major chemical variable in BWR type water that can be used to manipulate the corrosion potential in laboratory tests) (Reference 10).

As the corrosion potential is reduced below the range typical of normal BWR power operation (+50 to -50 mV_{SHE}), a region of immunity to IGSCC appears at ~ -230 mV_{SHE}. It is apparent that a combination of corrosion potential (which can be achieved in a BWR by injecting usually < 1 ppm hydrogen into the feedwater) plus tight conductivity control (0.2 $\mu S/cm$) should permit BWRs to operate in a regime where sensitized stainless steels are immune to IGSCC. In the reactor vessel, the excess hydrogen reacts with the radioloytic oxygen and reduces the electrochemical corrosion potential (Reference 10a and 10b). The reactor water cleanup system, which processes reactor water at a rate of 2% of rated feedwater flow, removes both dissolved and undissolved impurities that enter the reactor water. The removal of dissolved impurities reduces the conductivity into the region of immunity to IGSCC. 252.7 252.7 252.7 281.3 281.10(2) 281.10(11)

Since the ABWR has no sensitized stainless steel, IGSCC control by hydrogen injection is not required. However, irradiation assisted stress corrosion cracking (IASCC) can occur in highly irradiated annealed stainless steel and nickel-base alloys. Preliminary in-reactor and laboratory studies (Reference 11) have indicated that HWC will be useful in mitigating IASCC.

In-reactor and laboratory evidence also indicates that carbon and low alloy steels also tend to show improved resistance to environmentally assisted cracking with both increasing water purity and decreasing corrosion potential (Reference 12). 252.7

5.2.3.2.2.1 Fuel Performance Considerations

Nuclear fuel is contained in Zircaloy tubes that constitute the first boundary or primary containment for the highly radioactive species generated by the fission process; therefore, the integrity of the tubes must be ensured. Zircaloy interacts with the coolant water and some coolant impurities. This results in oxidation by the water, increased hydrogen content in the Zircaloy (hydriding), and, often, buildup of a layer of crud on the outside of the tube. Ex-

252.7 **Also called electrochemical corrosion potential or ECP, see Reference 9.*

cessive oxidation, hydriding, or crud deposition may lead to a breach of the cladding wall.

Metallic impurities can result in neutron losses and associated economic penalties which increase in proportion to the amount being introduced into the reactor and deposited on the fuel. With respect to iron oxide-type crud deposits, it can be concluded that operation within the BWR water chemistry guidelines (specifically the limits on feedwater iron levels) effectively precludes the buildup of significant deposits on fuel elements.

5.2.3.2.2.2 Radiation Field Buildup

The primary long-term source of radiation fields in most BWRs is cobalt-60, which is formed by neutron activation of cobalt-59. Corrosion products are released from corroding and wearing surfaces as soluble, colloidal, and particulate species. The formation of cobalt-60 takes place after the corrosion products precipitate, adsorb, or deposit on the fuel rods. Subsequent re-entrainment in the coolant and deposition on out-of-core stainless steel surfaces leads to buildup of the activated corrosion products (such as cobalt-60) on the out-of-core surfaces. The deposition may occur either in a loosely adherent layer created by particle deposition, or in a tightly adherent corrosion layer incorporating radioisotopes during corrosion and subsequent ion exchange. Water chemistry influences all of these transport processes. The key variables are the concentration of soluble cobalt-60 in the reactor water and the characteristics of surface oxides. Thus, any reduction in the soluble cobalt-60 concentration will have positive benefits.

281.10 Item 1

As a means to reduce cobalt, GE has reduced cobalt content in alloys to be used in high fluence areas such as fuel assemblies and control rods. In addition, cobalt base alloys used for pins and rollers in control rods have been replaced with noncobalt alloys.

281.10 Item 8

The reator water cleanup system, which processes reactor water at a rate of 2% of rated feedwater flow, will remove both dissolved and undissolved impurities which can become radioactive deposits. Reduction of these radioactive deposits will reduce occupational radiation exposure during operation and maintenance of the plant components.

Water quality parameters can have an influence on radiation buildup rates. In laboratory tests, the water conductivity and pH were varied systematically from a high purity base case. In each case, impurities increased the rate of cobalt-60 uptake over that of the base case. The evidence suggests that these impurities change both the corrosion rate and the oxide film characteristics to adversely increase the cobalt-60 uptake. Thus, controlling water purity should be beneficial in reducing radiation buildup.

Prefilming of stainless steel in cobalt-60 free water, steam, or water/steam mixtures also appears to be a promising method to reduce initial radiation buildup rates. As an example, the radiation buildup rates are reduced significantly when samples are prefilmed in high temperature (288°C), oxygenated (200 ppb oxygen) water prior to exposure to cobalt-60 containing water. Mechanical polishing and electropolishing of piping internal faces should also be effective in reducing radiation buildup.

5.2.3.2.2.3 Sources of Impurities

Various pathways exist for impurity ingress to the primary system. The most common sources of impurities that result in increases in reactor water conductivity are condenser cooling water inleakage, improper operation of ion exchange units, air inleakage, and radwaste recycle. In addition to situations of relatively continuous ingress, such as from low level condenser cooling water inleakage, transient events can also be significant. The major sources of impurities during such events are resin intrusions, organic chemical intrusions, inorganic chemical intrusions, and improper rinse of resins. Chemistry transients resulting from introduction of organic substances into the radwaste system comprised a significant fraction of the transients which have occurred.

The following factors are measured for control or diagnostic purposes to maintain proper water chemistry in the ABWR.

(1) <u>Conductivity</u>

Increasing levels of many ionic impurities adversely influence both the stress corrosion cracking behavior of RCS materials, the rate of radiation field buildup and also can affect fuel performance. Therefore, conductivity levels in the reactor water should be maintained at the lowest levels practically achievable.

(2) Chloride

Chlorides are among the most potent promoters of IGSCC of sensitized stainless steels and are also capable of inducing transgranular cracking of nonsensitized stainless steels. Chlorides also promote pitting and crevice attack of most RCS materials. Chlorides normally are associated with cooling water inleakage, but inputs via radwaste processing systems have also occurred.

to have an important influence on IGSCC initiation times for smooth stainless steel specimens in laboratory tests. In addition, pH can serve as a useful diagnostic parameter for interpreting severe water chemistry transients and pH measurements are recommended for this purpose.

(10) Electrochemical Corrosion Potential

The electrochemical corrosion potential (ECP) of a metal is the potential it attains when immersed in a water environment. The ECP is controlled by various oxidizing agents including copper and radiolysis products. At low reactor water conductivities, the ECP of stainless steel should be below -0.23 V_{SHE} to suppress IGSCC.

(11) Feedwater Hydrogen Addition Rate

A direct measurement of the feedwater hydrogen addition rate can be made using the hydrogen addition system flow measurement device and is used to establish the plant-specific hydrogen flow requirements required to satisfy the limit for the ECP of stainless steel (Paragraph 10). Subsequently, the addition rate measurements can be used to help diagnose the origin of unexpected ECP changes

(12) Recirculation System Water Dissolved Hydrogen

A direct measurement of the dissolved hydrogen content in the reactor water serves as a cross check against the hydrogen gas flow meter in the injection system to confirm the actual presence and magnitude of the hydrogen addition rate.

(13) Main Steam Line Radiation Level

281.8 The major activity in the main steam line is nitrogen-16 produced by an (n, p) reaction with oxygen-16 in the reactor water. Under conditions of hydrogen water chemistry, the fraction of the nitrogen-16 that volatilizes with the steam increases with increased dissolved hydrogen. The main steam line radiation monitor readings increase with the hydrogen addition rate. During initial plant testing, the amount of hydrogen addition required to reduce the electrochemical corrosion potential to the desired range is determined at various power levels. Changes in the main steam line radiation monitor readings at the same power level indicate an over-addition (high readings) or under-addition (low readings) of hydrogen. 281.8

(14) Constant Extension Rate Test

Constant extension rate tests (CERTs) are accelerated tests that can be completed in a few days, for the determination of the susceptibility to IGSCC. It is useful for verifying IGSCC suppression during initial implementation of hydrogen water chemistry (HWC) or following plant outages that could have had an impact on system chemistry (e.g., condenser repairs during refueling).

(15) Continuous Crack Growth Monitoring Test

This test employs a reversing DC potential drop technique to detect changes in crack length in IGSCC test specimens. The crack growth test can be used for a variety of purposes, including the following:

(a) Initial verification of IGSCC suppression following HWC implementation.

(b) Quantitative assessment of water chemistry transients.

(c) Long-term quantification of the success of the HWC program.

The major impurities in various parts of a BWR under certain operating conditions are listed in Table 5.2-5. The plant systems have been designed to achieve these limits at least 90% of the time. The plant operators are encouraged to achieve better water quality by using good operating practice.

Water quality specifications require that erosion-corrosion resistant low alloy steels are to be used in susceptible steam extraction and 281.10 Item 6 & 9

281.10 Item 6 & 9 drain lines. Stainless steels are considered for baffles, shields, or other areas of severe duty. Provisions are made to add nitrogen gas to extraction steamlines, feedwater heater shells, heater drain tanks, and drain piping to minimize corrosion during layup. Alternatively, the system may be designed to drain while hot so that dry layup can be achieved.

281.10 Item 7 Condenser tubes and tubesheet are required to be made of titanium alloys.

gallon per minute, thus meeting Position C.2 requirements.

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

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

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

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

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

(1) simulation of signals into trip units;

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

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

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

These satisfy Position C.8 requirements.

Limiting unidentified leakage to the range of 1 to 5 gpm and identified to 25 gpm satisfies Position C.9.

5.2.6 Interfaces

The remainder of plant will meet the water chemistry requirements given in Table 5.2-5.

5.2.7 References

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

2. (Deleted)

3. D.A. Hale, *The Effect of BWR Startup Environments on Crack Growth in Structural Alloys*, Trans. of ASME, vol 108, January 1986.

4. F.P. Ford and M. J. Povich, *The Effect of Oxygen/Temperature Combinations on the Stress Corrosion Susceptibility of Sensitized T-304 Stainless Steel in High Purity Water*, Paper 94 presented at Corrosion 79, Atlanta, GA, March 1979.

5. *BWR Normal Water Chemistry Guidelines: 1986 Revision*, EPRI NP-4946-SR, July 1988. 281.4 281.7 281.10 Item 4 & 11

6. B.M. Gordon, *The Effect of Chloride and Oxygen on the Stress Corrosion Cracking of Stainless Steels: Review of Literature*, Material Performance, NACE, Vol. 19, No. 4, April 1980.

7. W.J. Shack, et al, *Environmentally Assisted Cracking in Light Water Reactors: Annual Report, October 1983 - September 1984*, NUREG/CR-4287, ANL-85-33, June 1985.

8. D.A. Hale, et al, *BWR Coolant Impurities Program*, EPRI, Palo Alto, CA, Final Report on RP2293-2, to be published.

9. K.S. Brown and G.M. Gordon, *Effects of BWR Coolant Chemistry on the Propensity of IGSCC Initiation and Growth in Creviced Reactor Internals Components*, paper presented at the Third International Symposium of Environmental Degradation of Materials in Nuclear Power Systems, ANS-NACE-TMS/AIME, Traverse City, Michigan, September 1987.

281.4
281.7
281.10
Item 4 & 11

10. B.M. Gordon et al, *EAC Resistance of BWR Materials in HWC*, Proceeding of Second International Symposium Environmental Degration of Materials in Nuclear Power Systems, ANS, LaGrange Park, ILL 1986.

10a. BWR Hydrogen Water Chemistry Guidelines: 1987 Revision, EPRI NP-4947-SR-LD (To be published).

281.3

10b. Guideline for Permanent BWR Hydrogen Water Chemistry Installations: 1987 Revision, EPRI NP-5203-SR-A.

11. B.M. Gordon, *Corrosion and Corrosion Control in BWRs*, NEDE-30637, December 1984.

281.4
281.7
281.10
Item 4 & 11

12. B.M. Gordon et al, *Halogen Water Chemistry for BWRs - Materials Behavior*, EPRI NP-5080, Palo Alto, CA, March 1987.

Table 5.2-5

281.10
Item 8

BWR WATER CHEMISTRY

	Concentrations* Parts Per Billion (ppb)					Conductivity		Electro-Chemical Corrosion Potential
	Iron	Copper	Chloride	Sulfate	Oxygen**	μS/cm at 25°C	pH at 25°C	V at 25°C
Condensate	<20	<2	<4	<4	<10	~0.15	~7	-----
Condensate Treatment Effluent and Feedwater	<2.2	<0.02	<0.4	<0.4	20-50	<0.059	~7	-----
Reactor Water								
(a) Normal Operation	<20	<1	<20	<20	**	<0.3	~7	<-0.23
(b) Shutdown	<20	<1	<20	<20	-	<1.2	~7	-----
(c) Hot Standby	<20	<1	<20	<20	<200	<0.3	~7	-----
(d) Depressurized	<20	<1	<20	<20	high (may be 1000 to 8000)	<1.2	5.6-8.6	-----
Control Rod Drive Cooling Water	<2.2	<0.1	<0.4	<0.4	20-50	≤0.059	~7	-----

* *These limits should be met at least 90% of the time.*

** *Some revision of oxygen values may be established after hydrogen water chemistry has been established.*

TABLES

ILLUSTRATIONS

6.2.1.1.3.1 Summary Evaluation

The key design parameter and the maximum calculated accident parameters for the pressure suppression containment are shown in Table 6.2-1.

The maximum drywell pressure would occur during a feedwater line break. The maximum drywell temperature condition would result from a main steam line break. All of the analyses assume that the primary system and containment system are initially at the maximum normal operating conditions.

6.2.1.1.3.2 Containment Design Parameters

Table 6.2-2 provides a listing of the key design parameters of the primary containment system including the design characteristics of the drywell, suppression pool and the pressure suppression vent system.

Table 6.2-2a provides the performance parameters of the related engineered safety feature systems which supplement the design conditions of
430.16 Table 6.2-2 for containment cooling purposes during post-blowdown long-term accident operation. Performance parameters given include those applicable to full capacity operation and reduced capacities assumed for containment analyses.

6.2.1.1.3.3 Accident Response Analysis

The containment functional evaluation is based upon the consideration of several postulated accident conditions which would result in the release of reactor coolant to the containment. These accidents include:

(1) an instantaneous guillotine rupture of a feedwater line;

(2) an instantaneous guillotine rupture of a main steam line; or

(3) small break accidents.

The containment design pressure and temperature were established based on enveloping the results of this range of analyses plus providing NRC prescribed margins.

6.2.1.1.3.3.1 Feedwater Line Break

Immediately following a double-ended rupture in one of the two main feedwater lines just outside the vessel (Figure 6.2-1), the flow from both sides of the break will be limited to the maximum allowed by critical flow considerations. The effective flow area on the RPV side is given in Figure 6.2-2. During the inventory depletion period, subcooled blowdown occurs and the effective flow area at saturated condition is much less than the actual break area. The detailed calculational method is provided in Reference 1. The RPV blowdown through the break is prevented by the check valves.

The feedwater system side of the feedwater line break (FWLB) was modeled by adding a time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. The time histories of the mass flow and enthalpy were determined from the operating characteristics of a typical feedwater system.

The maximum possible feedwater flow rate was calculated to be 164% of nuclear boiler rated (NBR), based on the response of the feedwater pumps to an instantaneous loss of discharge pressure. Since the feedwater control system will respond to decreasing RPV water level by demanding increased feedwater flow, and there is no FWLB sensor in the design, this maximum feedwater flow was conservatively assumed to continue for 120 seconds, as shown in Figure 6.2-3. This is very conservative because: 1) all feedwater system flow is assumed to go directly to the drywell, 2) flashing in the broken feedwater line was ignored, 3) initial feedwater flow was assumed to be 105% NBR, and 4) the feedwater pump discharge flow will coastdown as the feedwater system pumps trip due to low suction pressure. During the inventory depletion period, the flow rate is less than 164% because of the highly subcooled blowdown. A feedwater line length of 100 M was assumed on the feedwater system side.

The enthalpy of the feedwater flow is 120% of a typical BWR/5 feedwater system inventory enthalpy. The specific enthalpy time history, assuming the break flow of Figure 6.2-3, is shown in Figure 6.2-4.

6.2.1.1.3.3.1.1 Assumptions for Short-Term Response Analysis

The response of the reactor coolant system and the containment system during the short-term blowdown period of the accident has been analyzed using the following assumptions:

(10) Actuation of SRVs is modeled.

(11) Wetwell-to-drywell vacuum breakers are not modeled.

(12) Drywell and wetwell sprays and RHR cooling mode are not modeled.

(13) The dynamic back pressure model is used.

(14) Initial drywell conditions are 15.45 psia, 135°F, and 20% relative humidity.

(15) Initial wetwell airspace conditions are 15.45 psia, 95°F and 100% relative humidity.

(16) The drywell is modeled as a single node. All break flow into the drywell is homogeneously mixed with the drywell inventory.

(17) Because of the unique containment geometry of ABWR, the inert atmosphere in the lower drywell would not transfer to the wetwell until the peak pressure in the drywell is achieved. Figure 6.2-5 shows the actual case, and the model assumption. Because the lower drywell is connected to the drywell connecting vent, no gas can escape from the lower drywell until the peak pressure occurs. This situation can be compared to a bottle whose opening is exposed to an atmosphere with an increasing pressure. The contents of the lower drywell will start transferring to the wetwell as soon as the pressure starts decreasing. A conservative credit for transfer of 50% of the lower drywell contents into the wetwell was taken.

6.2.1.1.3.3.1.2 Assumptions for Long-Term Cooling Analysis

Following the blowdown period, the ECCS discussed in Section 6.3 provides water for core flooding, containment spray, and long-term decay heat removal. The containment pressure and temperature response during this period was analyzed using the following assumptions:

(1) The ECCS pumps are available as specified in Subsection 6.1.1.3.3.1.1 (except one low pressure flooder feeding a broken feedwater line, in case of a FWLB). A single failure of one RHR heat exchanger was assumed for conservatism.

(2) The ANS decay heat is used. Fission energy, fuel relaxation heat, and pump heat are included.

(3) The suppression pool is the only heat sink available in the containment system.

(4) After 10 minutes, the RHR heat exchangers are activated to remove energy via recirculation cooling of the suppression pool with the RCWS and ultimately to the RSWS. This is a conservative assumption since, the RHR design permits initiation of containment cooling well before a 10 minute period. (See response to Question 430.26) 430.26

(5) The maximum service water temperature is assumed to be 95°F. This is a conservative assumption that maximizes the suppression pool temperature.

(6) The lower drywell flooding of 28,760 ft^3 was assumed to occur 70 seconds after scram. During blowdown phase, a portion of break flow flows into the lower drywell. This is conservative since lower drywell flooding will probably occur at approximately 110 to 120 second time period (See Figure 6.2-6).

(7) At 70 seconds, the feedwater specific enthalpy becomes 180 Btu/lb (212°F saturation fluid enthalpy).

6.2.1.1.3.3.1.3 Short-Term Accident Responses

The calculated containment pressure and temperature responses for feedwater line break are shown in Figures 6.2-6 and 6.2-7, respectively. The peak pressure (39 psig) and temperature (284°F) occur in the drywell. The containment design pressure of 45 psig is 116% of the peak pressure.

The drywell pressurization is driven by the wetwell pressurization for stable peaks. The wetwell pressurization is a function of three of channels, recording of parameters, instrument range and accuracy and post-accident monitoring equipment is discussed in Section 7.5.

6.2.2 Containment Heat Removal System

6.2.2.1 Design Bases

The containment heat removal system, consisting of the suppression pool cooling mode and the wetwell and drywell spray features are integral parts of the RHR system. The purpose of this system is to prevent excessive containment temperatures and pressures, thus maintaining containment integrity following a LOCA. To fulfill this purpose, the containment cooling system meets the following safety design bases:

(1) The system limits the long-term bulk temperature of the suppression pool to 207°F when considering the energy additions to the containment following a LOCA. These energy additions, as a function of time, are provided in the previous section.

(2) The single-failure criterion applies to the system.

(3) The system is designed to safety grade requirements including the capability to perform its function following a Safe Shutdown Earthquake.

(4) The system maintains operation during those environmental conditions imposed by the LOCA.

(5) Each active component of the system is testable during normal operation of the nuclear power plant.

6.2.2.2 Containment Cooling System Design

The containment cooling system encompasses several of the RHR operating modes, which are the low pressue flooder (LPFL) mode, the suppression pool cooling mode, and the containment spray
430.26 modes (drywell and wetwell). Containment cooling starts as soon as the LPFL injection flow begins. The suppression pool cooling mode cools the containment. The containment sprays cool the drywell and wetwell by condensing steam and the condensate running back into the suppression pool. All water that leaves the suppression pool is cooled by the RHR heat exchangers during the three operational modes indicated above. For each of the three loops, water is drawn from the suppression pool, pumped through a RHR heat exchanger and injected into the reactor vessel 430.26
for the LPFL mode. Also, for each of the three loops for the suppression pool cooling mode, water is drawn from the suppression pool, pumped through a RHR heat exchanger and delivered to the suppression pool. On two of the loops (B&C), a portion of the water returned to the suppression pool may be passed through wetwell spray headers. These two loops also have a manual feature for providing drywell spray. Water from the RCWS is pumped through the heat exchanger shell side to exchange heat with the processed water. Three cooling loops are provided, each being mechanically and electrically separate from the other to achieve redundancy. A piping and instrumentation diagram (P&ID) is provided in Section 5.4. The process diagram, including the process data, is provided for all design operating modes and conditions.

All portions of the containment cooling 430.26
system mode are designed to withstand operating loads and loads resulting from natural phenomena. All operating components can be tested during normal plant operation so that reliability can be assured. Construction codes and standards are covered in Subsection 5.4.7.

The low pressure flooder (LPFL) mode is 430.26
automatically initiated from ECCS signals. The suppression pool cooling mode is started manually or automatically. The RHR system must be realigned for suppression pool cooling by the plant operator after the reactor vessel water level has been recovered (Subsection 6.2.1). The RHR pumps are already operating. Suppression pool cooling is initiated in any of the three loops by manually closing the LPFL injection valve and opening the pool return valve. In the event that a single failure has occurred, and the action which the plant operator is taking does not result in system initiation, then the operator will place the other totally redundant system into operation by following the same initiation procedure. If the operator chooses to utilize the containment sprays, he must close the LPFL injection valves

and open and spray valves. The drywell spray mode may be initiated manually only after a high drywell pressure permissive occurs.

Preoperational tests are performed to verify individual component operation, individual logic element operation and system operation up to the containment spray spargers. A sample of the sparger nozzles is bench tested for flow rate versus pressure drop to evaluate the original hydraulic calculations. Finally, the spargers are tested by air and visually inspected to verify that all nozzles are clear. (See

Subsection 5.4.7.4 for further discussion of preoperational testing.)

6.2.2.3 Design Evaluation of the Containment Cooling System

6.2.2.3.1 System Operation and Sequence of Events

In the event of the postulated LOCA, the short-term energy release from the reactor primary system will be dumped to the suppression pool. Subsequent to the accident, fission product decay heat will result in a continuing energy input to the pool. The RHR LPFL mode and suppression pool cooling mode will remove this energy which is released into the primary containment system, thus resulting in acceptable suppression pool temperatures and containment pressures.

In order to evaluate the adequacy of the RHR system, the following is assumed:

(1) With the reactor initially operating at 102% of rated power, a LOCA occurs.

(2) A single failure of a RHR heat exchanger is the most limiting single failure.

(3) The ECCS flows assumed available are 2 HPCF, 1 RCIC, and 2 LPFL (RHR).

430.26 (4) Containment cooling is initiated after 10 minutes. (See Response to Question 430.26)

430.28 Analysis of the net positive suction head (NPSH) available to the RHR pumps in accordance with the recommendations of Regulatory Guide 1.1 is provided in Table 6.2-2b.

General compliance for Regulatory Guide 1.26 may be found in Subsection 3.2.2.

Failure modes and effects analyses for the RHR and RCWS are provided in Appendix 15B.

6.2.2.3.2 Summary of Containment Cooling Analysis

When calculating the long-term, post-LOCA pool temperature transient, it is assumed that the initial suppression pool temperature and the RHR service water temperature are at their maximum values. This assumption maximizes the heat sink temperature to which the containment heat is rejected and thus maximizes the containment temperature. In addition, the RHR heat exchanger is assumed to be in a fully fouled condition at the time the accident occurs. This conservatively minimizes the heat exchanger heat removal capacity. Even with the degraded conditions outlined above, the maximum temperature is maintained below the design limit specified in Subsection 6.2.2.1.

It should be noted that, when evaluating this long-term suppression pool transient, all heat sources in the containment are considered with no credit taken for any heat losses other than through the RHR heat exchanger. These heat sources are discussed to Subsection 6.2.1.3.

It can be concluded that the conservative evaluation procedure described above clearly demonstrates that the RHR system in the suppression pool cooling mode limits the post-LOCA containment temperature transient.

6.2.2.4 Test and Inspections

The containment cooling system is required to have scheduled maintenance. The system testing and inspection will be performed periodically during the plant normal operation and after each plant shutdown. Functional testing will be performed on all active components and controls. The system reference characteristics will be established during preoperational testing to be used as base points for checking measurements obtained from the system tests during the plant operation.

The preoperational test program of the containment cooling system is described in Subsection 14.2.12. The following functional tests will be performed. The RHR pump will be tested through the suppression pool cooling loop operation by measuring flow and pressure. Each pump will be tested individually.

Containment spray spargers will be tested during reactor shutdown by air, and by visual inspection to verify that all the nozzles are pressurized with air at a reduced test pressure P_t, which will result in a measured leakage rate, identified as L_{tm}. The second phase is then conducted at pressure P_a resulting in a measured leakage rate identified as L_{am}. The absolute method shall be employed for determining the leakage rate (see ANSI N45.4 Subsection 5.2.1 and Section 7.9). Test duration of each phase shall be sufficient for pressure and temperature stabilization. To ensure uniform temperature distribution, fans will be provided to circulate air in the containment during the test. Prior to commencement of the tests, the test prerequisites described in Subsections 6.2.6.1.2.1 and 6.2.6.1.3 will be met.

6.2.6.1.1.3 Supplement Verification Test

The accuracy of the leakage rate tests is verified by using a supplemental method of leakage measurement. Verification is obtained by superimposing a controlled and measurable leak on the normal containment leakage rate or other methods of demonstrated equivalency. The difference between the total leakage and the superimposed known-leakage results in the actual leakage rate. This leakage rate is a check against its accuracy and is acceptable provided the correlation between the supplemental test data and integrated leak test data demonstrates an agreement within $\pm$ 25%. Conduct of the verification test is normally accomplished after completion of each test phase of the ILRT. Complete descriptive details are found in Appendix C of ANSI N45.4.

6.2.6.1.1.4 Instrumentation Requirements

Instrumentation provided to monitor the containment leakage rate testing is designed, calibrated and tested to accurately ensure that the containment atmosphere parameters can be precisely measured.

6.2.6.1.1.5 Acceptance Criteria

The initial allowable leakage rate (L_{tm}) at test pressure P_t shall not exceed 75% of the maximum allowable test leakage rate (L_t), where L_t is defined as follows:

$$L_t = L_a \frac{L_{tm}}{L_{am}} \quad \text{for values of} \quad \frac{L_{tm}}{L_{am}} \leq 0.7$$

$$L_t = L_a \left(\frac{P_t}{P_a}\right)^{1/2} \quad \text{for values of} \quad \frac{L_{tm}}{L_{am}} > 0.7$$

430.38b

(P_t and P_a are psig).

The leakage L_{am} shall be less than 0.75 L_a and not greater than the design leakage rate (L_d).

6.2.6.1.2 Periodic Leakage Rate Tests

Leakage rate tests are conducted periodically in conformance to Appendix J of 10CFR50 to ensure that the integrity of the containment is maintained and to determine if any leakage increase has developed since the previous ILRT. The tests are performed at regular intervals, after major repairs or upon indication of excessive leakage, as specified in the standard technical specification for the ABWR.

6.2.6.1.2.1 Integrated Leakage Rate Test (ILRT, Type A)

Type A tests are conducted periodically, following the initial preoperational tests, at test pressure P_t only. Except for the elimination of the P_a pressure test, all ILRTs follow the same format as the initial ILRT, as outlined in Subsection 6.2.6.1.1.

In addition to the normal test prerequisites, the following requirements are mandatory prior to all periodic Type A tests:

(1) A detailed visual examination of critical areas and general inspection of the accessible interior and exterior surfaces of the containment structure and components shall be performed to uncover any evidence of structural deterioration which may affect either the structural integrity or leaktightness of the containment. If there is evidence of significant structural

deterioration, Type A tests shall not be performed until corrective action is taken in accordance with approved repair procedures. If leak repairs of testable components are performed, the reduction in leakage shall be measured (at test pressure P_t) and added to the Type A test result. Except for inspections and actions taken above, no preliminary leak detection surveys and repairs shall be performed prior to the conduct of the Type A test.

(2) Closure of containment isolation valves shall be accomplished by normal mode of actuation and without preliminary exercises or adjustments. All malfunctions and subsequent corrective actions shall be reported to the NRC.

6.2.6.1.2.2 Acceptance Criteria

The measured leakage rate L_{tm} shall not exceed 0.75 L_t as established by the initial ILRT.

(1) If during a Type A test including the supplemental test, potentially excessive leakage paths are identified which will interfere with satisfactory completion of the test, or which result in the Type A test not meeting the acceptance criteria, the Type A test shall be terminated and the leakage through such paths shall be measured using local leakage testing methods. Repairs and/or adjustments to equipment shall be made and a Type A test performed. The corrective action taken and the change in leakage rate determined from the tests and overall integrated leakage determined from the local leak and Type A tests shall be included in the report submitted to the NRC. 430.38c

(2) If any Type A test fails to meet the acceptance criteria, prior to corrective action, the test schedule applicable to subsequent Type A test shall be subject to review and approval by the NRC.

(3) If two consecutive periodic Type A tests fail to meet the applicable acceptance criteria, prior to corrective action, notwithstanding the established periodic retest schedule, a Type A test shall be performed at each plant shutdown for major refueling, or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria, after which time the previously established periodic retest schedule may be resumed. 430.38c

6.2.6.1.2.3 Test Frequency

After initial ILRT, a set of three Type A tests shall be performed at approximately equal intervals during each 10-yr service period, with the third test of each set coinciding with the end of each 10-yr major inservice inspection shutdown. In addition, any major modification or replacement of components of the primary reactor containment performed after the initial ILRT shall be followed by either a Type A or a Type B test of the area affected by the modification, with the affected area to meet the applicable acceptance criteria. The basis for the frequency of testing is established in accordance with 10CFR50, Appendix J.

6.2.6.1.3 Additional Criteria for Integrated Rate Test

(1) Those portions of fluids systems that are part of the reactor coolant pressure boundary, that are open directly to the primary reactor containment atmosphere under post-accident conditions and become an extension of the boundary of the primary reactor containment, shall be opened or vented to the containment atmosphere prior to or during the Type A test. Portions of closed systems inside containment that penetrate primary containment and are not relied upon for containment isolation purposes following a LOCA shall be vented to the containment atmosphere.

(2) All vented systems shall be drained of water to the extent necessary to ensure exposure of the system primary containment isolation valves to the containment air test pressure.

(3) Those portions of fluid systems that penetrate primary containment, that are external to containment and are not designed to provide a containment isolation barrier, shall be vented to the outside atmosphere as applicable, to assure that full post-accident differential pressure is maintained across the containment isolation barrier.

(4) Systems that are required to maintain the plant in a safe condition during the Type A test shall be operable in their normal mode and are not vented.

(5) Systems that are normally filled with water and operating under post-LOCA conditions need not be vented.

6.2.6.2 Containment Penetration Leakage Rate Test (Type B)

6.2.6.2.1 General

Containment penetrations whose designs incorporate resilient seals, bellows, gaskets, or sealant compounds, airlocks and lock door seals, equipment and access hatch seals, and electrical canisters, and other such penetrations are leak tested during preoperational testing and at periodic intervals thereafter in conformance to Type B leakage rate tests defined in Appendix J of 10CFR50. The leak tests ensure the continuing structural and leak integrity of the penetrations.

To facilitate local leak testing, a permanently installed system may be provided, consisting of a pressurized gas source (nitrogen or air) and the manifolding and valving necessary to subdivide the testable penetrations into groups of two to five. Each group is then pressurized, and if any leakage is detected (by pressure decay or flow meter), individual penetrations can be isolated and tested until the source and nature of the leak is determined. All Type B tests are performed at containment peak accident pressure, Pa. The local leak detection tests of Type B and Type C (Subsection 6.2.6.3) must be completed prior to the preoperational or periodic Type A tests.

6.2.6.2.2 Acceptance Criteria

The combined leakage rate of all components subject to Type B and Type C tests shall not exceed 60% of L_a (cfm). If repairs are required to meet this limit, the results shall be reported in a separate summary to the NRC. The summary shall include the structural conditions of the components which contributed to failure.

6.2.6.2.3 Retest Frequency

In compliance with the requirement of Section III.D.2(a) of Appendix J to 10CFR Part 50, type B tests (except for air locks) are performed during each reactor shutdown for major fuel reloading, or other convenient intervals, but in no case at intervals greater than two years.* Air locks opened when containment integrity is required will be tested in manual mode within 3 days of being opened. If the air lock is to be opened more frequently than once every 3 days, the air lock will be tested at least once every 3 days during the period of frequent openings. Air locks will be tested at initial fuel loading, and at least once every 6 months thereafter. Testing may be initiated automatically at the end of each interval by the seal test instrumentation system, with manual override of the automated sequence provided for in the associated logic. Testing involves the injection of air under pressure (15 psig) into the space between the two redundant seals in each door of the air lock. The leakdown rate is measured by sensing the pressure drop and/or flow rate necessary to maintain the pressure. Main control room readout of time to next test, test completion and test results is provided. An alarm sounds if the specified interval passes without a test being effected. No direct, safety-related function is served by the seal test instrumentation system.

430.47 a&b

6.2.6.2.4 Design Provisions for Periodic Pressurization

In order to assure the capability of the containment to withstand the application of peak accident pressure at any time during plant life for the purpose of performing ILRTs, close attention is given to certain design and maintenance provisions. Specifically, the

In compliance with the requirement of Section III.D.2(b)(iii) of Appendix J to 10CFR Part 50 effects of corrosion on the structural integrity of the containment are compensated for by the inclusion of a 60-yr service life corrosion allowance, where applicable. Other design features that have the potential to deteriorate with age, such as flexible seals, are carefully inspected and tested as outlined in Subsection 6.2.6.2.2. In this manner, the structural and leakage integrity of the containment remains essentially the same as originally accepted.

6.2.6.3 Containment Isolation Valve Leakage Rate Test (Type C)

6.2.6.3.1 General

Type C tests are required on all isolation valves. All testing is performed pneumatically, except hydraulic testing may be performed on isolation valve Type C tests using water as a sealant provided that the valves will be demonstrated to exhibit leakage rates that do not exceed those in the ABWR standard technical specifications.

Type C tests (like Type B test) are performed by local pressurization using either pressure decay or flowmeter method. The test pressure is applied in the same direction as when the valve is required to perform its safety function, unless it can be shown that results from tests with pressure applied in a different direction are equivalent or conservative. For the pressure decay method, test volume is pressurized with air or nitrogen to at least P_a. The rate of decay of pressure of the known test volume is monitored to calculate leakage rate. For the flowmeter method, required pressure is maintained in the test volume by making up air, nitrogen or water (if applicable) through a calibrated flowmeter. The flowmeter fluid flow rate is the isolation valve (or Type B test volume) leakage rate.

All isolation valve seats which are exposed to containment atmosphere subsequent to a LOCA are tested with air or nitrogen at containment peak accident pressure, P_a.

Those valves which are in lines designed to be, or remain, filled with a liquid for at least 30 days subsequent to a loss-of-coolant accident are leakage rate tested with that liquid. The liquid leakage measured is not converted to equivalent air leakage nor added to the Type B and C test total.

For Type C testing of containment penetrations, all testing, with the exception of the ECCS systems will be done in the correct direction unless it can be shown that testing in the reverse direction is equivalent, or more conservative. The correct direction for this design is defined as flow from inside the containment to outside the containment.

6.2.6.3.2 Acceptance Criteria

The combined leakage rate of all components subject to Type B and Type C (Subsection 6.2.6.3) tests shall not exceed 60% of L_a. If repairs are required to meet this limit, the results shall be reported in a separate summary to the NRC, to include the structural conditions of the components which contributed to the failure.

6.2.6.4 Scheduling and Reporting of Periodic Tests

The periodic leakage rate test schedules for Type A, B and C tests are described in Chapter 16.

Type B and C tests may be conducted at any time during normal plant operations or during shutdown periods, as long as the time interval between tests for any individual Type B or C tests does not exceed the maximum allowable interval specified in the standard technical specifications for the ABWR. Each time a Type B or C test is completed, the overall total leakage rate for all required Type B and C tests is updated to reflect the most recent test results. In addition to the periodic tests, any major modification, replacement of component which is part of the primary reactor containment boundary, or resealing a seal welded door, performed after the preopertional leakage rate test will be followed by either a Type A, Type B, or Type C test as applicable for the area effected by the modification. Type A, B and C test results shall be submitted to the NRC in the summary report approximately three months after each test. 430.51

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

6.2.6.5 Special Testing Requirements

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

Table 6.2-2

CONTAINMENT DESIGN PARAMETERS

A. Drywell and Wetwell(1)	Drywell	Wetwell
1. Internal Design Pressure (psig)	45	45
2. Negative Design Pressure (psid)	-2.0	-2.0
3. Design Temperature (^{o}F)	340	219
4. Net Free Volume (ft^3)	259, 563	210, 475
430.15 5. Maximum allowable leak rate(2) (%/day)	0.5	0.5
6. Minimum Suppression Pool Water Volume (ft^3)	--	126,427
7. Suppression pool depth (ft)		
Low Level	--	22.97
High Level	--	23.29
B. Vent System		
1. Number of Vents		30
2. Nominal Vent Diameter (ft)		2.3
3. Total Vent Area (ft^2)		125
4. Vent Centerline Submergence Low Level), (ft)		
Top Row		11.48
Middle Row		15.98
Bottom Row		20.48
5. Vent Loss Coefficient (Varies with number of vents open)		2.5 - 3.5

(1) Item A.1, A.2, A.3 and A.5 apply to related structures including lower drywell access tunnels, drywell equipment hatches, drywell personnel locks and drywell head.

430.15 (2) Corresponds to calculated peak containment pressure related to the design basis accident conditions.

TABLE 6.2-2a

ENGINEERED SAFETY SYSTEMS INFORMATION FOR CONTAINMENT RESPONSE ANALYSES

	Full Capacity	Containment Analysis Value
A. Containment Spray		
1. Number of RHR Pumps	1(1)	1(1)
2. Number of Lines	1(1)	1(1)
3. Number of Heat Exchangers	1(2)	1(2)
4. Drywell Flow Rate (lb/hr)	1.81×10^6	1.81×10^6
5. Wetwell Flow Rate (lb/hr)	2.46×10^5	2.46×10^5
B. Containment Cooling System		
1. Number of RHR Pumps	3	2
2. Pump Capacity (gpm/pump)	4200	4200
3. RHR Heat Exchangers		
a. Type - U-tube,		
b. Number	3	2
c. Heat Transfer Area (ft^2/unit)	(3)	(3)
d. Overall Heat Transfer Coefficent (Btu/hr-ft^2-$^\circ$F/unit)	(3)	(3)
e. Service Water Flowrate (lb/hr)	2.63×10^6	2.63×10^6
f. Maximum Service Water Temperature ($^\circ$F)	95	95

430.16

TABLE 6.2-2a

ENGINEERED SAFETY SYSTEMS INFORMATION FOR CONTAINMENT RESPONSE ANALYSES (Continued)

NOTES

1. Two redundant loops available with one pump each.

2. One header each for drywell and wetwell.

3. The RHR heat exchanger characteristic has been defined by an overall K coeficient based on a temperature difference and the heat rate. The defining equation is: 430.16

$$Q = (K)(\Delta T)$$

$$Q, \frac{Btu}{sec} = \left(K, \frac{Btu}{sec\ ^{\circ}F}\right)\left(\Delta T, ^{\circ}F\right)$$

The K value is 195 Btu/sec$^{\circ}$F.

The applicable temperature difference occurs from the RHR heat exchanger's reactor side inlet to the service water temperature. Thus, K is a characteristic of the combined RHR and reactor service water system's heat exchangers.

TABLE 6.2-2b

NET POSITIVE SUCTION HEAD (NPSH) AVAILABLE TO RHR PUMPS

A. Suppression pool is at its minimum depth, El. -3740mm (-12.27 Ft).

B. Centerline of pump suction is at El. -7200mm (23.62 Ft).

C. Suppression pool water is at its maximum temperatue for the given operating mode; 97°C (207°F).

D. Pressure is atmospheric above the suppression pool.

E. Maximum suction strainer losses are 2.0 psi.

$NPSH = H_{ATM} + H_A - H_{VAP} - H_F$

where:

H_{ATM} = atmospheric head

H_S = static head 430.28

H_{VAP} = vapor pressure head

H_F = Frictional head including strainer

<u>Minimum Expected NPSH</u>

RHR Pump Runout is 1100 m^3/h (4843 gpm).

Maximum suppression pool temperature is 97°C (207°F)

H_{ATM} = 10.73m (35.20 Ft)

H_S = 3.46m (11.35 Ft)

H_{VAP} = 9.74m (31.95 Ft)

H_F = 1.82m (5.97 Ft)

Strainer head loss = 2.0 psi = 1.46m = 4.80 Ft

NPSH available = 10.73 + 3.46 - 9.74 - 1.82 = 2.63m (8.63 Ft)

NPSH required = 2.4m (7.87 Ft)

two channel operability. The operator has the capability to invoke bypass conditions within the following system or subsystems:

(a) Synchro A or B position bypass
(b) Rod server module channel A or B bypass
(c) Uncoupled condition bypass
(d) File control module channel A or B bypass
(e) AFBM channel A or B bypass
(f) MRBM channel A or B bypass
(g) RPC channel A or B bypass
(h) RACS channel A or B bypass
(i) DAM channel A or B bypass

(11) Scram Time Test Data Recording

The logic of the RC&IS provides the capability to automatically record individual fine motion control rod drive (FMCRD) scram timing data based upon scram timing reed switches. When a particular FMCRD scram timing switch is activated, the time of actuation is recorded by DAMs for time tagging of stored scram time test data for that particular fine motion control rod drive. The time tagged data is stored in memory until the next actuation of that particular reed switch is detected again.

The RC&IS also time tags the receipt of a reactor scram condition being activated based upon the scram following function input signals from the reactor protection system which is received via the essential multiplexing system.

The resolution of this time tagging feature is less than 10 milliseconds. Contact bounce of the reed switch inputs and the DAM inputs are properly masked to support this function. The reference real time clock for time tagging is the real time clock of the RC&IS.

When RC&IS detects a reactor scram condition, the current position of all control rods in the core are recorded, time tagged, and stored in memory. RC&IS logic stores this data in memory until a request is received from the performance monitoring and control system. The transmitted data is used by the PMCS to calculate and summarize scram time performance based on the scram timing data received from the RC&IS.

7.7.1.2.2 Other Systems Interfaces

(1) Alternate Rod Insertion (ATWS) (Anticipated Transient Without Scram)

The RC&IS logic, during an anticipated transient without scram (on receipt of signals as a result of high reactor dome pressure or low reactor water level) initiates ARI signals which controls the fine motion control rod drive motors such that all control rods are driven to their full-in position automatically. The four divisions of the nuclear boiler system provide each of the two channels of the RC&IS logic with the reactor high dome pressure and reactor low water level signals for generation of the ARI signal based on two-out-of-four logic.

The operator at the RC&IS dedicated display can take action and initiate the ARI function. Two manual actions are required to manually initiate ARI. The RC&IS logic has been designed to complete the ARI functions in the worst case non-accident environment, completely independent of reactor pressure transient conditions. This capability is accomplished with control logic for insertion of all control rods by an alternate and diverse method, based on receiving reactor high dome pressure and low water level (Level 2) signals for generating its own ATWS (anticipated transient without scram) signal. The logic of the RC&IS has been designed such that no single failure results in failure to insert more than one operable control rod when the ARI function is activated.

(2) Recirculation Flow Control System

The recirculation flow control system (RFCS) provides each of the two channels of the RC&IS with two separate isolated trip signals indicating the need for automatic selected control rod run-in (SCRRI). The signals are treated as nonsafety-related signals within the logic of the RC&IS. 440.3

The RFCS provides signals to both channels of the RC&IS that represent validated total core flow. These signals are used for part of the validity checks when performing an ARBM operating limit setpoint update. The RC&IS can obtain these signals from the RFCS via the multiplexing system of direct communication links to the RC&IS channels. These signals are also completely independent from the process computer system.

The RFCS receives reference power level signals from the neutron monitoring system and compares the reference power level signals with the nominal power level setpoint.

Selected control rod run in (SCRRI) is automatically initiated when a trip of two or more reactor internal pumps (RIPs) occur. This function is part of the stability control and protection logic.

When two or more RIPs are tripped, the trip signal is "ANDED" with the power level "AND" flow rate signals and RFCS automatically sends a request for control rod blocks to the RC&IS. When the power level signal with two or more RIPs tripped is "ANDED" with the flow rate the RFCS automatically sends four signals to the RC&IS to initiate the SCRRI function.

The SCRRI function is bypassed when power level is below the specified setpoint, or when the core flow is above the specified setpoint.

The SCRRI function if designed as a non-safety related system. The function is designed to meet the reliability requirement that no single failure shall cause a loss of the function.

The RFCS automatic initiation signal for the SCRRI function is sent as two independent sets of signals, one set to each channel of the RC&IS, each channel of the RC&IS uses the input in two-out-of-two logic to control the fine motion control rod drive (FMCRD) motors of preselected control rods. The preselected control rods are driven to their full-in position on receipt of the automatic initiation signals. Either channel of an RC&IS is capable of initiating the SCRRI function on receipt of the automatic signal from the RFCS.

The preselected control rods for an SCRRI function are selected at the RC&IS dedicated operators control panel and the CRT displays of the performance monitor control system in the main control room. The preselected SCRRI rod data is stored in memory in the rod action and position information subsystem of the RC&IS. The total control rod worth for the preselected control rods is designed to bring down the reactor power level from the 100% rod line to the 80% line.

The RC&IS dedicated operators control panel also provides control switches that requires two manual operator actions for the operator to manually initiate the SCRRI function.

For the manual or the automatic initiation of the SCRRI function the RC&IS dedicated operators panel provides status indications and alarm annunciators in the control room.

The RC&IS provides the capability for manual or automatic initiation of the SCRRI function and the total delay time to start of control rod motion for the preselected control rods is less than 350 milliseconds.

(3) Feedwater Control System

The feedwater control system provides signals to both channels of the logic of the RC&IS that represents validated total feedwater flow to the vessel, validated narrow range vessel dome pressure, and validated feedwater temperature. These signals are used as part of the validity checks when performing an ARBM operating limit setpoint update.

The RC&IS can obtain these signals from the feedwater control system via the multiplexing system of direct communication links to the RC&IS channels. These signals are also completely independent from the process computer system.

(4) Neutron Monitoring System

Each of the four divisions of the neutron monitoring system provides independent signals to both channels of the RC&IS that indicate when the following conditions are active:

440.3

440.3

(a) Startup range neutron monitor (SRNM) period alarm
(b) SRNM downscale alarm
(c) SRNM upscale alarm
(d) Average power range monitor (APRM) upscale alarm
(e) SRNM inoperative
(f) APRM downscale
(g) Flow biased APRM rod block
(h) APRM inoperative

Whether or not some of the signals result in a rod block depends on whether or not the reactor is in the RUN mode. The reactor mode status is provided to the RC&IS from the reactor protection system via the essential multiplexing system.

Each of the four divisions of NMS signals provides APRM, LPRM and core flow signals to the two channels of logic in the rod action and position information system for determining whether reactor power is above or below the low power setpoint.

The four divisions of NMS signals to the RC&IS two channel system are isolated signals between the Class 1E NMS and the nonsafety-related equipment of the RC&IS.

(5) Reactor Protection System

Each of the four divisions of the reactor protection system provides the RC&IS two channel system with separate isolated signals for indication of the reactor mode switch positions: SHUTDOWN, REFUEL, STARTUP, HOT STANDBY and RUN.

Each of the four divisions of the reactor protection system (RPS) provides to the RC&IS two separate isolated signals for each of the following conditions:

(a) Reactor scram condition. This signal remains active if initiated until the scram condition has been cleared by the RPS operators resetting the reset switch.

(b) Low charging water header pressure trip switches in bypass position.

(c) Status of hydraulic control unit (HCU) scram test switch.

The essential multiplexing system provides the above signals to the RC&IS with complete isolation between the safety-related system and the nonsafety-related system equipment.

(6) Nuclear Boiler System

The four divisions of the nuclear boiler system (NBS) provide each of the two channels of the RC&IS with the reactor high dome pressure and reactor water level signals for generation of the RC&IS alternate rod insertion (ARI) function.

NRC* Branch	Review Area	Question Number	SSAR Subsection	Response Subsection	RAI** Letter
	Materials Application	252.1	4.5.1.1(1)	20.3.1	1
		252.2	4.5.1.1(2)	20.3.1	1
		252.3	4.5.2.2	20.3.1	1
		252.4	4.5.2.3	20.3.1	1
		252.5	4.5.2.4	20.3.1	1
		252.6	4.5.2.5	20.3.1	1
		252.7	5.2.3.2.2	20.3.1	1
		252.8	5.2.3.2.3	20.3.1	1
		252.9	5.2.3.3.1	20.3.1	1
		252.8	5.2.3.2.3	20.3.1	1
		252.10	5.2.3.4.1.1	20.3.1	1
		252.11	5.2.3.4.2.3	20.3.1	1
ECEB	Chemical Technology	281.1	5.1	20.3.1	1
		281.2	5.2.3.2.2	20.3.1	1
		281.3	5.2.3.2.2	20.3.1	1
		281.4	5.2.3.2.2	20.3.1	1
		281.5	5.2.3.2.2	20.3.1	1
		281.6	5.2.3.2.2.2	20.3.1	1
		281.7	5.2.3.2.2.3(4)	20.3.1	1
		281.8	5.2.3.2.2.3(13)	20.3.1	1
		281.9	6.4.9.2	20.3.1	1
		281.10	Chap. 5	20.3.1	1
SPLB	Plant Systems	430.1	4.6	20.3.2	2
		430.2	5.2.5	20.3.2	2
		430.3	5.2.5	20.3.2	2
		430.4	5.2.5.4.1	20.3.2	2
		430.5	5.2.5	20.3.2	2
		430.6	5.2.5	20.3.2	2
		430.7	6.2	20.3.2	2
		430.8	6.2	20.3.2	2
		430.9	6.2	20.3.2	2
		430.10	6.2	20.3.2	2
		430.11	6.2	20.3.2	2
		430.12	6.2	20.3.2	2
		430.13	6.2.1.1.3	20.3.2	2
		430.14	6.2	20.3.2	2
		430.15	6.2	20.3.2	2
		430.16	6.2	20.3.2	2
		430.17	6.2.1.2.3	20.3.2	2
		430.18	6.2	20.3.2	2
		430.19	6.2	20.3.2	2
		430.20	6.2	20.3.2	2

NRC* Branch	Review Area	Question Number	SSAR Subsection	Response Subsection	RAI** Letter
		430.21	6.2	20.3.2	2
		430.22	6.2	20.3.2	2
		430.23	6.2	20.3.2	2
		430.24	6.2	20.3.2	2
		430.25	6.2	20.3.2	2
		430.26	6.2	20.3.2	2
		430.27	6.2	20.3.2	2
		430.28	6.2	20.3.2	2
		430.29	6.2.3	20.3.2	2
		430.30	6.2	20.3.2	2
		430.31	6.2	20.3.2	2
		430.32	6.2	20.3.2	2
		430.33	6.2	20.3.2	2
		430.34	6.2	20.3.2	2
		430.35	6.2	20.3.2	2
		430.36	6.2	20.3.2	2
		430.37	6.2	20.3.2	2
		430.38	6.2	20.3.2	2
		430.39	6.2.4	20.3.2	2
		430.40	6.2	20.3.2	2
		430.41	6.2	20.3.2	2
		430.42	6.2	20.3.2	2
		430.43	6.2	20.3.2	2
		430.44	6.2	20.3.2	2
		430.45	6.2	20.3.2	2
		430.46	6.2	20.3.2	2
		430.47	6.2.5.3	20.3.2	2
		430.48	6.2.6	20.3.2	2
		430.49	6.2.6	20.3.2	2
		430.50	6.2.6	20.3.2	2
		430.51	6.2.6	20.3.2	2
		430.52	6.2.6	20.3.2	2
		430.53	6.2.6	20.3.2	2
		430.54	6.4	20.3.2	2
		430.55	6.5.1	20.3.2	2
		430.56	6.5.3	20.3.2	2
		430.57	6.7	20.3.2	2
		430.58	15.7.3	20.3.2	2
SRXB	Reactor Systems	440.1	4.6	20.3.2	2
		440.2	4.6.2.3.2.2	20.3.2	2
		440.3	4.6.1.2	20.3.2	2
		440.4	4.6	20.3.2	2
		440.5	4.6	20.3.2	2
		440.6	4.6	20.3.2	2
		440.7	4.6	20.3.2	2

NRC* Branch	Review Area	Question Number	SSAR Subsection	Response Subsection	RAI** Letter
		440.8	4.6	20.3.2	2
		440.9	4.6	20.3.2	2
		440.10	4.6.2.3.1	20.3.2	2
		440.11	4.6	20.3.2	2
		440.12	4.6	20.3.2	2
PRPB	Radiological Report	470.1	15.5.2	20.3.1	1
		470.2	15.6.2	20.3.1	1
		470.3	15.6.4.5.1.1	20.3.1	1
		470.4	15.6.5.5	20.3.1	1
		470.5	15.6.5	20.3.1	1
		470.6	15.7.5	20.3.1	1
		470.7	15.7	20.3.1	1
		470.8	15.7	20.3.1	1
		470.9	15.7	20.3.1	1
		470.10	15.7	20.3.1	1

SECTION 20.2

CONTENTS

20.2.4 Chapter 4 Questions

252.1

Subsection 4.5.1.1 (1) should state: "The properties of the materials selected for the control rod drive mechanism must be equivalent to those given in Appendix I to Section III of the ASME Code, or parts A and B of Section II of the ASME Code, or are included in Regulatory Guide 1.85, except that cold-worked austenitic stainless steels should have a 0.2% offset yield strength no greater than 90,000 psi."

252.2

Subsection 4.5.1.1 (2) should state: "All materials for use in this system must be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code."

252.3

Subsection 4.5.2.2: The first sentence should read: "Core support structures are fabricated in accordance with the requirements of ASME Code, Section III, Subsection NG-4000, and the examination and acceptance criteria shown in NG-5000."

252.4

Subsection 4.5.2.3: The following statement should be added to the last sentence of the first paragraph: "The examination will satisfy the requirements of NG-5300."

252.5

Subsection 4.5.2.4 should state: "Furnace sensitized material should not be allowed."

252.6

Subsection 4.5.2.5 should state: "All materials used for reactor internals will be selected for their compatibility with the reactor coolant as shown in ASME Code Section III, NG-2160 and NG-3120. The fabrication and cleaning controls will preclude contamination of nickel-based alloys by chloride ions, fluoride ions, or lead."

430.1

Provide a failure modes and effects analysis of the control rod drive system (CRDS) in tabular form with supporting discussion to delineate the logic employed. The failure analysis should demonstrate that the CRDS can perform the intended functions with the loss of any active single component. These evaluations and assessments should establish that all essential elements of the CRDS are identified and provisions are made for isolation from nonessential CRDS elements. It should be established that all essential equipment is protected from common mode failures such as failure of moderate- and high-energy lines. The failure mode and effects analysis of the control rod drives should include water, air and electrical failures to CRDs and how the CRD system operation is affected due to air contamination or water contamination. Before finalizing the scope of the analysis, refer to ACRS subcommittee meeting proceedings on the ABWR dated June 1, 1988. It is noted that the above information is to be included in Appendix 15B of the SSAR which will be submitted at a later date. However, the evaluation of the functional design of the reactivity control systems cannot be completed until this information is provided. (4.6)

440.1

SRP 4.6 identifies the following GDCs 23, 25, 26, 27, 28 and 29 in the acceptance criteria. Confirm that the reactivity system, described in Section 4.6 of the SSAR, meet the requirements of the above GDCs.

440.2

In Section 4.6.2.3.2.2 Analysis of malfunction relating to rod withdrawal, it is stated "There are known single malfunctions that cause the unplanned withdrawal of even a single control rod." Confirm that this is a editorial mistake and correct it if so. Otherwise, explain in detail the basis for this statement and why this is acceptable.

440.3

In Section 4.6.1.2 it is stated that CRD system in conjunction with CRC&IS and RPS systems provides selected control rod run in (SCRRI) for reactor stability control. Describe in detail how SCRRI works.

440.4

In Figure 4.6-8a, CRD system P&ID, sheet 1, piping quality classes AA-D, FC-D, FD-D, FD-B, etc. are shown. Submit the document which explains these classes and relates them to ASME code classes.

440.5

In Figure 4.6-8b, the leak receiver tank is shown. What is the function of this tank? How big is this tank? Will a high level in the tank impact the operation of the control rod drive?

440.6

Identify the essential portions of the CRD system which are safety related. Confirm that the safety related portions are isolable from non-essential portions. (4.6)

440.7

In the old CRD system, the major function of the cooling water was to cool the drive mechanism and its seals to preclude damage resulting from long term exposure to reactor temperatures. What is the function of purge water flow to the drives? (4.6)

440.8

We understand that the LaSalle Unit 2 fine motion control rod drive demonstration test is still in progress. Submit the test results as soon as it is available. (4.6)

440.9

In the present CRD system design, the ball check valve ensures rod insertion in the event the accumulator is not charged or the inlet scram value fails to open if the reactor pressure is above 600 psig. Confirm that this capability still exists in the ABWR design. (4.6)

440.10

In section 4.6.2.3.1, it is stated the scram time is adequate as shown by the transient analyses of Chapter 15. Specify the scram time.

440.11

For both the low ("zero") power and operating power region describe the patterns of the control rod groups that are expected to be withdrawn simultaneously with the new rod system, and estimate the maximum for the total and differential reactivity worth of these groups. What sort of margin to period scram will exist in the low power range. (4.6)

440.12

Describe the relative core location of control rods sharing a scram accumulator. Can a failure of the scram accumulator fail to insert adjacent rods? If so, discuss the consequences of that failure. (4.6)

20.2.5 Chapter 5 Questions

210.1

In Subsection 5.2.1.2. the statement is made that Section 50.55a of 10CFR50 requires NRC staff approval of ASME code cases only for Class 1 components. Revise this statement to be consistent with the current (1987) edition of 10CFR50.55a, which requires staff approval of code cases for ASME Class 1, 2, and 3 components.

210.2

Revise Table 5.2-1 or provide additional tables in Subsection 5.2.1.2 which identify all ASME code cases that will be used in the construction and in-plant operation of all ASME Class 1, 2, and 3 components in the ABWR. All code cases in these tables should be identified by code case number, revision, and title. These tables should include those applicable code cases that are listed either as acceptable or conditionally acceptable in Regulatory Guides 1.84, 1.85, and 1.147. For those code cases listed as conditionally acceptable, verify that the construction of all applicable components will be in compliance with the additional Regulatory Guide conditions.

250.1

Subsection 5.2.4.1 should state that the system boundary includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor systems, up to and including:

(1) The outermost containment isolation valve in system piping that penetrates the primary reactor containment.

(2) The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment.

(3) The reactor coolant system and relief valves.

250.2

Subsection 5.2.4.2 should satisfy the requirements in ASME Code IWA-1500.

251.1

Subsection 5.3.1.1 should state that the materials will comply with the provisions of the ASME Code, Section III, Appendix I, and meet the specification requirements of 10CFR50, Appendix G.

251.2

Subsection 5.3.1.2 should state the specific subsection NB of ASME Code to which the manufacturing and fabrication specifications were alluded.

251.3

Subsections 5.3.1.4.4 and 5.3.1.4.5 should be rewritten; the cross-reference is unacceptable.

Subsections 5.3.1.4.7, 5.3.1.5.2, 5.3.1.6.3, and 5.3.2.1.5: Revision 2 of Regulatory Guide 1.99 should be added in these subsections.

251.4

Subsection 5.3.1.6.1: the third capsule of the vessel surveillance program is designated as a standby; however, according to ASTM 185-82, the capsule should be withdrawn at the end of life. Provide justification for this deviation.

430.2

Regarding Reactor Coolant Pressure Boundary (RCPB) leakage detection systems, provide information on the following: (5.2.5)

(a) Describe how the leakage through both the inner and outer reactor vessel head flange seals will be detected and quantified.

(b) List the sources that may contribute to the identified leakage collected in the Reactor Building Equipment Drain Sumps.

(c) Describe how potential intersystem leakages will be monitored for the (1) Low pressure Coolant Injection System, (2) High Pressure Core Spray System, (3) Reactor Core Isolation Cooling System (RCIC) - Water side and (4) Residual Heat Removal System-Inlet and discharge sides. Your response should include <u>all</u> the applicable (for the ABWR design) systems and components connected to the Reactor Coolant System that are listed in Table 1 of SRP Section 5.2.5 and other systems that are unique to ABWR (except those that you have already discussed in SSAR Subsection 5.2.5.2.2, Item 11).

430.3

Discuss compliance of reactor coolant leak detection systems with Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Positions C4, C5, C6, C8 and C9 with respect to the following items: (5.2.5)

(a) Indicators for abnormal water levels or flows in all the affected areas in the event of intersystem leakages.

(b) Sensitivity and response time of leak detection systems used for unidentified leakages outside the drywell.

(c) Qualification relating to seismic events for drywell equipment drain sump monitoring system and leak detection systems outside the drywell.

(d) Testing Procedures - Monitoring sump levels and comparing them with applicable flow rates of fluids in the sumps.

(e) Inclusion of reactor building and other areas floor and equipment drain sumps in ABWR Technical Specifications for leak detection systems.

Note that a few of the questions above arise because in Subsection 5.2.5.4.1 you state that the total leakage rate includes leakages collected in drywell, reactor building and other area floor drain and equipment drain sumps.

430.4

Clarify whether the RCIC makeup capacity is sufficient to provide also for main stem line leakage through to the main turbine stop valves. Also, clarify whether this leakage is included in the total leakage mentioned in Subsection 5.2.5.4.1.

430.5

Clarify how Position C.2 of RG 1.29, "Seismic Design Classification" is met for <u>all</u> applicable leak detection systems (also include the leak detection systems outside the drywell). (5.2.5)

430.6

Identify all the interface requirements relating to RCPB leakage detection systems. (5.2.5)

20.2.6 Chapter 6 Questions

250.3

Subsection 6.6.8 should discuss the augmented inservice inspection for those portions of high energy piping enclosed in guard pipes.

252.12

Subsection 6.1.1.1 should discuss ferritic steel welding in detail. It should also discuss the control of ferrite content in stainless steel weld metal similar to that of Regulatory Guide 1.31.

252.13

Subsections 6.1.1.1.3.1, 6.1.1.1.3.2, and 6.1.1.1.3.5 should be rewritten because the cross-reference is unacceptable.

281.9

Subsection 6.4.4.2 (page 6.4-6) discusses personnel respirator use in the event of toxic gas intrusion into the control room. However, the chlorine detection system is not discussed. Also, any control functions that are automatically triggered by a chlorine detector alarm (closing intake dampers, energizing control room HVAC system recirculation) should be identified.

430.7

In the SSAR section devoted to containment functional design, identify clearly those areas that are not part of the ABWR scope and provide relevant interface requirements. (6.2)

430.8

With respect to the design bases for the containment: (6.2)

(a) Discuss the bases for establishing the margin between the maximum calculated accident pressure or pressure difference and the corresponding design pressure or pressure difference. This includes the design external pressure, internal pressure, and pressure between subcompartment walls.

(b) Discuss the capability for energy removal from the containment under various single-failure conditions. State and justify the design basis single failure that affects containment heat removal.

430.9

The Standard Safety Analysis Report (SSAR) states that the analytical models used to evaluate the containment and drywell responses to postulated accidents and transients are included in General Electric Co. report NEDO-20533 and its supplement 1, entitled *The G.E. Mark III Pressure Suppression Containment Analytical Model*. Provide justification that these references are appropriate to use for the ABWR Containment design which is not specified as Mark III. Discuss the similarities and differences of the ABWR design to previously approved Mark II and Mark III designs as they relate to the containment and drywell responses to the postulated accidents and the analytical model used for the analyses. Include in the discussion the conservatism used in the model and assumptions, the applicable test data that support the analytical models, and the sensitivity of the analyses to key parameters. (6.2)

430.10

With regard to the design features of the containment: (6.2)

(a) Provide general arrangement drawings for the containment structure.

(b) Provide appropriate references to Section 3 of the SSAR which includes the information on the codes, standards, and guides applied in the design of the containment and containment internal structures.

(c) Discuss the possibilities of water entrapment inside containment and its effect on the accident analysis.

(d) Provide information on qualification tests that are intended to demonstrate the functional capability of the containment structures, systems and components. Discuss the status of any developmental tests that may not have been completed.

430.11

Provide a detailed discussion of the likelihood and sensitivity to steam bypass of the suppression pool for a spectrum of accidents. Include in your discussion the following information: (6.2)

(a) A comparison of the ABWR pool bypass capability with that for Mark II and Mark III designs.

(b) The measures for minimizing the potential for steam bypass and the systems provided to mitigate the consequences of pool bypass. Discuss and demonstrate the conservatism of assumptions made in the analysis of steam bypass.

(c) Identify all lines from which leakage (or rupture) could contribute to pool bypass and wetwell air space pressurization.

(d) Identify all fluid lines which traverse the wetwell air space and identify those lines which are protected by guard pipe.

(e) Discuss the rationale and basis for the wetwell spray flow capacity.

430.12

With regard to containment response to external pressure: (6.2)

(a) Describe the wetwell-to-drywell vacuum breaker system and show the extent to which the requirements of subsection NE of section III of the ASME B&PV Code are satisfied. Discuss the functional capability of the system. Provide the design and performance parameters for the vacuum relief devices.

(b) Discuss the basis for selecting a low design capability for external pressure acting across the drywell to wetwell boundary. It is not apparent that the drywell negative design pressure of 2.0 psid is desirable or sufficient.

(c) The margin between the calculated wetwell-to-reactor building negative differential pressure (-1.8 psid) and the sign differential pressure (-2.0 psid) is not considered adequate. A higher margin of 15% should be provided at this stage of the design. Further, design on containment venting to control pressure, discuss the basis for not providing wetwell-to-reactor building vacuum breakers.

(d) In the analysis of wetwell-to-reactor building negative differential pressure calculation, a 500 gpm wetwell spray flow rate was used. Provide the basis for the assumption and the design basis for the wetwell spray capacity.

430.13

Section 6.2.1.1.3 of the SSAR states that the containment functional evaluation is based upon the consideration of several postulated accident conditions including small break accidents. Provide the assumptions, analysis and results of the small break accidents considered, ad demonstrate that the identified (in the SSAR) feedwater line and steam line breaks are the limiting accidents.

430.14

Provide analyses of the suppression pool temperature for transients involving the actuation of safety/relief valves. Provide the assumptions and conservatism employed in the analyses so that an assessment could be made for conformance to the acceptance criteria set forth in NUREG-0783, *Suppression Pool Temperature Limits for BWR Containments.* (6.2)

430.15

Provide the pressure at which the maximum allowable leak rate of 0.5%/day is quoted. (6.2)

430.16

Provide engineered safety systems information for containment response analysis (full capacity operation and capability used in the containment analysis), as indicated in Table 6-7 of Regulatory Guide 1.70, Revision 3. (6.2)

430.17

In the design evaluation section for containment subcompartments (Section 6.2.1.2.3), provide the information necessary to substantiate your assessment that the peak differential pressures do not exceed the design differential pressures. Guidance for the information required is provided in Regulatory Guide 1.70, Revision 3, Section 6.2.1.2, "Containment Subcompartments", Design Evaluation.

430.18

Describe the manner in which suppression pool dynamic loads resulting from postulated loss-of-coolant accidents, transients (e.g., relief valve actuation), and seismic events have been integrated into the affected containment structures. Provide plan and section drawings illustrating all equipment and structural surfaces that could be subjected to pool dynamic loads. For each structure or group of structures, specify the dynamic loads as a function of time, and specify the relative magnitude of the pool dynamic load compared to the design basis load for each structure. Provide justification for each of the dynamic load histories by the use of appropriate experimental data and/or analyses.

Describe the manner by which potential asymmetric loads were considered in the containment design. Characterize the type and magnitude of possible asymmetric loads and the capabilities of the affected structures to withstand such a loading profile. (6.2)

430.19

Provide information to demonstrate that the ABWR design is not vulnerable to a safety relief valve discharge line break within the air space of the wetwell, coupled with a stuck open relief valve after its actuation as a result of the transient. (6.2)

430.20

Discuss suppression pool water makeup under normal and accident conditions. (6.2)

430.21

With respect to mass and energy release analyses for postulated loss-of-coolant accidents identify the sources of generated and stored energy in reactor coolant system that are considered in the analyses of loss-of-coolant accidents. Describe the methods used and assumptions made in calculations of the energy available for release from these sources. Address the conservatism in the calculation of the available energy from each source. Tabulate the stored energy sources and the amounts of stored energy. For the sources of generated energy, provide curves showing the energy release rates and integrated energy release. (6.2)

430.22

In the SSAR sections devoted to containment heat removal systems, identify clearly those areas that may not be part of the GE scope and provide relevant interface requirements. (6.2)

430.23

The SSAR states that the containment heat removal system is designed to limit the long-term temperature of the suppression pool to 207°F. The calculated peak pool temperature is 206.46°F for the feedwater line break. With respect to this analysis provide the following information: (6.2)

(a) The justification that this is the limiting accident with respect to the maximum temperature in the suppression pool.

(b) The bases for the design margin between the design and calculated temperatures.

(c) All assumptions used in the analysis and conservatism associated with each. Include the effects of potential temperature stratification in the suppression pool and its effects on heat removal capability of the system.

(d) The identification of the decay heat curve used in the analysis.

430.24

Provide the design bases for the spray features of the containment heat removal system. Provide the safety classification of the components associated with the spray feature of the system. (6.2)

430.25

Discuss the rationale for continued reliance on sprays as the sole active engineered safety feature for drywell atmosphere pressure and temperature. Discuss the merits of upgrading the design of drywell fan coolers to provide some capacity for pressure, temperature, and humidity control following an accident. (6.2)

430.26

The time period assumed for initiation of the containment heat removal system after a LOCA is 10 minutes requiring operator action. It is the staff's position that this time period is too restrictive. In fact previous BWR designs (Grand Gulf's Mark III) use 30 minutes actuation time. Provide the reasons why the ABWR does not provide more flexibility with respect to the time required for actuation. (6.2)

430.27

Describe the design features of the suppression pool suction strainers. Specify the mesh size of the screens and the maximum particle size that could be drawn into the piping. Of the systems that receive water through the suppression pool suction strainers under post accident conditions identify the system component that places the limiting requirements on the maximum size of debris that may be allowed to pass through the strainers and specify the limiting particle size that the component can circulate without impairing system performance. Discuss the potential for the strainers to become clogged with debris. Identify and discuss the kinds of debris that might be developed following a loss-of-coolant accident. Discuss the types of insulation used in the containment and describe the behavior of the insulation during and after a LOCA. Include in your discussion information regarding compliance with the acceptance criteria associated with USI A-43 as documented in NUREG-0897. (6.2)

430.28

Provide analyses of the net positive suction head (NPSH) available to the RHR pumps in accordance with the recommendations of Regulatory Guide 1.1. Compare the calculated values of available NPSH to the required NPSH of the pumps. (6.2)

430.29

In SSAR Section 6.2.3, identify clearly those areas that may not be part of the ABWR scope and provide relevant interface requirements. (6.2)

430.30

Provide a tabulation of the design and performance data for the secondary containment structure. Provide the types of information indicated in Table 6-17 of Regulatory Guide 1.70, Revision 3. (6.2)

430.31

Describe the valve isolation features used in support of the secondary containment. Specify the plant protection system signals that isolate the secondary containment and activate the standby gas treatment system. (6.2)

430.32

Identify and tabulate by size, piping which is not provided with isolation features. Provide an analysis to demonstrate the capability of the standby gas treatment system to maintain the design negative pressure following a design basis accident with all non isolated lines open and the event of the worst single failure of a secondary containment isolation valve to close. (6.2)

430.33

Discuss the design provisions that prevent primary containment leakage from bypassing the secondary containment standby gas treatment system and escaping directly to the environment. Include a tabulation of potential bypass leakage paths, including the types of information indicated in Table 6-18 of Regulatory Guide 1.70, Revision 3. Provide an evaluation of potential bypass leakage paths considering equipment design limitations and test sensitivities. Specify and justify the maximum allowable fraction of primary containment leakage that may bypass the secondary containment structure. The guidelines of BTP 6-3 should be addressed in considering potential bypass leakage paths. (6.2)

430.34

Provide a list of the secondary containment openings and the instrumentation means by which each is assured to be closed during a postulated design basis accident. (6.2)

430.35

Provide a table of design information regarding the containment isolation provisions for fluid system lines and fluid instrument lines penetrating the containment which are within the GE scope of the ABWR design. Include as a minimum the following information:

(1) General design criteria or regulatory guide recommendations that have been met or other defined bases for acceptability;

(2) System name;

(3) Fluid contained;

(4) Line size;

(5) ESF system (yes or no);

(6) Through-line leakage classification

(7) Reference to figure in SSAR showing arrangement of containment isolation barriers;

(8) Location of valve (inside/outside containment);

(9) Type C leakage test (yes or no);

(10) Valve type and operator;

(11) Primary mode of valve actuation;

(12) Secondary mode of valve actuation;

(13) Normal valve position;

(14) Shutdown valve position

(15) Postaccident valve position;

(16) Power failure valve position;

(17) Containment isolation signals;

(18) Valve closure time; and

(19) Power source. (6.2)

430.36

For isolation valve design in systems not within the ABWR scope, identify the systems and the relevant interface requirements. Include a discussion on essential and non-essential systems per Regulatory Guide 1.141 and the means or criteria provided to automatically isolate the nonessential systems by a containment isolation signal. Also, include a discussion on the requirement that the setpoint pressure which initiates containment isolation for nonessential penetrations be reduced to the minimum value compatible with normal operations. (6.2)

430.37

Specify all plant protection signals that initiate closure of the containment isolation valves. (6.2)

430.38

Describe the leakage detection means provided to identify leakage for the outside-containment remote-manual isolation valves on the following influent lines: Feedwater, RHR injection, HPCS, standby liquid control, RWCU connecting to feedwater line, RWCU reactor vessel head spray. (6.2)

430.39

The containment isolation design provisions for the recirculation pump seal water purge line do not meet the explicit requirements of GDC 55 nor does the design satisfy the GDC on some other defined basis as outlined in SRP Section 6.2.4. It is our position that the isolation design in the instance is inadequate and should be modified to satisfy GDC 55 either explicitly or on some other defined basis, with the appropriate justification. (6.2)

430.40

With respect to Figure 6.2-38a

(a) Include the isolation valve arrangement of the standby liquid control system line.

(b) Identify the line labeled in the figure as "WDCS-A" (it joins the RWCU line prior to its connection to the feedwater line), and discuss the isolation provisions for that line.

430.41

Provide a diagram or reference to figure(s) showing the isolation valve arrangement for the lines identified below. For the isolation valve design of each of these lines, provide justification for not meeting the explicit requirements of GDC 56, and demonstrate that the guidelines for acceptable alternate containment isolation provisions contained in SRP 6.2.4 are satisfied. The lines in question are:

- o HPCS and RHR test and pump miniflow bypass lines
- o RCIC pump miniflow bypass line
- o RCIC turbine exhaust and pump miniflow bypass lines
- o SPCU suction and discharge lines

430.42

Describe the isolation provisions for the containment purge supply and exhaust lines and discuss design conformance with Branch Technical Position CSB 6-4, "Containment Purge During Normal Operations." (6.2)

430.43

Discuss the closure times of isolation valves in system lines that can provide an open path from the primary containment to the environment (e.g., containment purge system). Also discuss provisions of radiation monitors in these lines having the capability of actuating containment isolation. (6.2)

430.44

Identify the system lines whose containment isolation requirements are covered by GDC 57 and discuss conformance of the design to the GDC requirements. (6.2)

430.45

For the combustible gas control systems design, identify clearly those areas that may not be part of the ABWR scope and provide relevant interface requirements. (6.2)

430.46

Accordin[illegible]ific acceptance criteria related to the concentration of hydrogen or oxygen in [illegible]heres among others are the following:

(a) The [illegible] oxygen production should be based on the parameters listed in Table 1 [illegible] 1.7 for the purpose of establishing the design basis for combustible [illegible]ol systems.

(b) The fission product decay energy used in the calculation of hydrogen and oxygen production from radiolysis should be equal to or more conservative than decay energy model given in Branch Technical Position ASB9-2 in SRP 9.2.5.

Provide justification that the assumptions used in the ABWR in establishing the design basis for the combustible gas control systems are conservative with respect to the criteria a. and b. above. (6.2)

430.47

Provide an analysis of the production and accumulation of combustible gases within the containment following a postulated loss-of-coolant accident including all applicable information specified in Section 6.2.5.3 of Regulatory Guide 1.70, Revision 3.

430.48

Regarding Containment Type A leakage testing (6.2.6)

(a) Provide the values for P_a and P_t.

(b) Include the acceptance criterion for L_t during preoperational leakage rate tests, i.e., $L_t = L_a (L_{tm}/L_{am})$, for the case when $L_a (L_{tm}/L_{am}) = 0.7$.

(c) Your acceptance criterion for L_{tm} (SSAR Subsection 6.2.6.1.2.2, Item 1) is at variance with the staff's current practice for acceptance of L_{tm}. Also, it does not comply with the 10 CFR Part 50, Appendix J, Section III, Item A.1.(a) requirement. Therefore, either provide sufficient supporting justification for the above requirement or correct the criterion as appropriate to comply with the requirement. Also, correct the stated acceptance criterion (SSAR Subsection 6.2.6.1.2.2, Item 3) as appropriate to comply with Appendix J, Section III, Item A.6.(b) requirement.

(d) Regarding ILRT, identify the systems that will not be vented or drained and provide reasons for the same.

(e) Provide P&IDs and process flow drawings for systems that will be vented or drained.

430.49

Regarding Type B tests (6.2.6)

(a) Clarify how air locks opened during periods when containment integrity is required by plant's Technical Specifications will be tested to comply with Appendix J, Section III, Item D.2.(b).(iii).

(b) Provide the frequency for periodic tests of air locks and associated inflatable seals.

(c) Provide the acceptance criteria for air lock testing and the associated inflatable seal testing.

(d) List all containment penetrations subject to Type B tests.

(e) List all those penetrations to be excluded from Type B testing and the rationale for excluding them.

430.50

Regarding Type C tests (6.2.6)

(a) Correct the statement (Subsection 6.2.6.3.1, Paragraph 1) as appropriate to ensure that the hydraulic Type C tests are performed only on those isolation valves that are qualified for such tests per Appendix J. The current statement implies that these tests are not necessarily restricted to the valves that qualify for such tests.

(b) List all the primary containment isolation valves subject to type C tests and provide the necessary P&IDs.

(c) Provide the list of valves that you propose to test in the reverse direction and justification for such testing for each of these valves.

(d) Identify the valves that you propose to test hydrostatically based on their ability to maintain a 30-day water leg seal. Also, identify other valves which you propose to test hydrostatically and provide the basis for such tests. Provide the test pressure for all the valves mentioned above.

(e) Indicate test pressures for MSIVs (with justification if it is less than P_a) and isolation valves sealed from a sealing system.

(f) Indicate how you will perform Type C leak tests for ECCS systems and RCIC system isolation valves.

(g) Confirm that the interval between two consecutive periodic Type C tests will not exceed 2 years as required by Appendix J.

(h) State what testing procedures you will follow regarding the valves that are not covered by Appendix J requirements.

430.51

Identify the reporting requirements for the tests. Note that your response should address compliance with the requirements in this regard as stated in Appendix J, Sections III.A.(a), IV.A and V. (For example, regarding follow up tests after containment modification, you have not included Type C testing for affected areas). (6.2.6)

430.52

Regarding Secondary Containment (6.2.6)

(a) Identify the special testing procedures you will follow to assure a maximum allowable in leakage of 50 percent of the secondary containment free column per day at a differential pressure of -0.25" water gauge with respect to the outdoor atmosphere (See Section 6.5.1.3.2).

(b) Identify all potential leak paths which bypass the secondary containment. (For such identification, see (BTP) CSB 6-3, "Determination of bypass Leakage Paths in Dual Containment Plants")

(c) Identify the total rate of secondary containment bypass leakage to the environment.

430.53

Identify all the interface requirements relating to containment leak testing. (6.2.6)

430.54

Regarding Control Room Habitability systems, (6.4)

(a) Provide the minimum positive pressure at which the control building envelope (which includes the mechanical equipment room) will be maintained with respect to the surrounding air spaces when makeup air is supplied to the design basis rate (295 CFM).

(b) Provide the periodicity for verification of control room pressurization with design flow rate of makeup air.

(c) Clarify whether all the potential leak paths (to be provided in Section 9.4.1) include dampers or valves upstream of recirculation fans.

(d) Identify the action to be taken when there is no flow of the equipment room return fan and consequently the equipment room is over pressurized (Table 6.4-1 contains no information on the above).

(e) Provide the actual minimum distances (lateral and vertical) of the control room ventilation inlets from major potential plant release points that have been used in your control room dose analysis. Also, provide a schematic of the location of control room intake vents.

(f) Provide Figure 6.4-5 (plan view) which you state shows the release points (SGTS vent).

(g) Section 6.4.2.4 and Figure 6.4-1 indicate only one air inlet for supplying makeup air to the emergency zone. However, Tables 6.4-2 and 15.6-8 and Section 15.6.5.5.2 indicate that there are two automatic air inlets for the emergency zone. Correct the above discrepancy as appropriate. Also describe the characteristics of these inlets with respect to their relative locations and automatic selection control features. State how both flow and isolation in each inlet assuming single active components failure will be ensured.

(h) Describe the design features for protecting against confined area releases (e.g., multiple barriers, air flow patterns in ventilation zones adjacent to the emergency zone).

(i) Describe the specific features for protecting the control room operator from airborne radioactivity outside the control room and direct shine from all radiation sources (e.g., shielding thickness for control room structure boundary, two-door vestibules).

(j) Clarify what you mean by "sustained occupancy" (See SSAR Section 6.4.1.1, Item 3) for 12 persons.

(k) Provide justification for not specifying any unfiltered infiltration of contaminated air into the control room in SSAR Table 15.6-8.

(l) Provide Subsection 6.3.1.1.6 which you state (SSAR Section 6.4.6) contains a complete description of the required instrumentation for ensuring control room habitability at all times.

(m) Give schematics for control room emergency mode of operation during a postulated LOCA (this is required for calculating control room LOCA doses).

(n) The source terms and control room atmospheric dispersion factors (X/Q values) used in the control room dose analysis (See SSAR Tables 15.6-8 and 15.6-12) to demonstrate ABWR control room compliance with GDC 19 are non-consecutive. Therefore, reevaluate control room doses during a postulated LOCA using RG 1.3 source terms and assumptions and the methodology given in Reference 4 of SSAR Section 15.6.7. Include possible dose contributions from containment shine, ESF filters and airborne radioactivity outside the control room. Also check and correct as appropriate the recirculation rate in the control room (22.4 m^3/sec) given in Table 15.6-8.

(o) Section 6.4.7.1, "External Temperature," provides design maximum external temperatures of 100°F and -10°F. How are these values used in the design and assessments related to the ABWR? What factors, such as insulation, heat generation from control room personnel and equipment and heat losses, are taken into account? Do these values represent "instantaneous" values or are they temporal and/or spatial averages?

(p) Clarify your position on potential hazardous or toxic gas sources onsite of an ABWR. If applicable, indicate the special features provided in the ABWR design in this regard, to ensure control room habitability.

(q) Identify <u>all</u> the interface requirements for control room habitability systems (e.g., instrumentation for protection against toxic gases in general and chlorine in particular; potential toxic gas release points in the environs).

430.55

Regarding ESF Atmosphere Cleanup System (6.5.1)

(a) Provide a table listing the compliance status of the standby gas treatment system (SGTS) with <u>each</u> of the regulatory positions specified under C of RG 1.52. Provide justifications for each of those items that do not fully comply with the corresponding requirements. In this context, you may note that the lack of redundancy of the SGTS filter train (the staff considers that filter trains are also active components - See SRP 6.4, Acceptance Criterion II.2.B) is not acceptable. Further, the described sizing of the charcoal adsorbers based on assumed decontamination factors for various chemical forms of iodine in the suppression pool is not acceptable (RG 1.3 assumes a decontamination factor of 1 for all forms of iodine and RG 1.52 requires compliance with the above guide for the design of the adsorber section). Therefore, revise charcoal weight and charcoal iodine loading given in SSAR Table 6.5-1 as appropriate.

(b) Specify the laboratory test criteria for methyl iodine penetration that will be identified as an interface requirement to be qualified for the adsorber efficiencies for iodine given in SSAR Table 15.6-8. Also, provide the depth of the charcoal beds for the control room emergency system.

(c) Provide a table listing the compliance status of the instrumentation provided for the SGTS for read out, recording and alarm provisions in the control room with <u>each</u> of the instrumentation items identified in Table 6.5.1-1 of SRP 6.5.1. For partial or non-compliance items, provide justifications.

(d) Clarify whether primary containment purging during normal plant operation when required to limit the discharge of contaminants to the environment will always be through the SGTS (See SSAR Section 6.5.1.2.3.3). Clarify whether such a release prior to the purge system isolation has been considered in the LOCA dose analysis.

(e) Provide the compliance status tables referred to in Items (a) and (c) above for the control room ESF filter trains. (The staff notes that you have committed to discuss control room ESF filter system under SSAR Section 9.4.1. However, since evaluation of the control room habitability system cannot be completed until the information identified above is provided, the above information is requested now.)

(f) Identify the applicable interface requirements for the SGTS and the control room ESF atmosphere cleanup system.

430.56

Regarding Fission Product Control Systems and Structures (6.5.3)

(a) Provide the drawdown time for achieving a negative pressure of 0.25 inch water gauge for the secondary containment with respect to the environs during SGTS operation. Clarify whether the unfiltered release of radioactivity to the environs during this time for a postulated LOCA has been considered in the LOCA dose analysis. (Note that the unfiltered release need not be considered provided the required negative pressure differential is achieved within 60 seconds from the time of the accident.)

(b) Provide justification (See SRP Section 6.5.3, II.4) for the decontamination factor assumed in SSAR Table 6.5-2 and 15.6-8 for iodine in the suppression pool, correct the elemental, particulate and organic iodine fractions given in the tables to be consistent with RG 1.3, and incorporate the correction in the LOCA analysis tables. Alternatively, taking no credit for iodine retention in the suppression pool, revise the LOCA analysis tables. Note that the revision of the LOCA analysis tables (this also includes the control room doses) mentioned above is strictly in relation to the iodine retention factor in the suppression pool (also, there may be need for revision of other parameter(s) given in the tables and these will be identified under the relevant SRP Sections questions).

(c) Identify the applicable interface requirements.

430.57

Regarding SSAR Section 6.7, the staff notes that the Nitrogen Supply System has been discussed under this section, instead of the Main Steam Isolation Valve Leakage Control System (MSIV-LCS) as required by the Standard Format for SARs. The staff will review the material presented in SSAR Section 6.7 along with the material that will be presented in SSAR Section 9.3.1.

Regarding MSIV-LCS, the staff notes that you are committed to provide a non-safety related MSIV leakage processing pathway consistent with those evaluated in NUREG-1169, "Resolution of Generic Issue C-8," August 1986. Since the staff has not finalized its position so far on the acceptability of the NUREG findings with regard to the design of the MSIV-LCS, provide pertinent information on the system design including interface requirements to evaluate the to-be-proposed design against the acceptance criteria of SRP 6.7. (6.7)

20.2.15 Chapter 15 Questions

430.58

The accident analyzed under this section considers only the airborne radioactivity that may be released due to potential failure of a concentrated waste tank in the radwaste enclosure. The SRP acceptance criteria, however, requires demonstration that the liquid radwaste concentration at the nearest potable water supply in an unrestricted area resulting from transport of the liquid radwaste to the unrestricted area does not exceed the radionuclide concentration limits specified in 10 CFR Part 20, Appendix B Table II, Column 2. Such a demonstration will require information on possible dilution and/or decay during transit which, in turn, will depend upon site specific data such as surface and ground water hydrology and the parameters governing liquid waste movement through the soil. Additionally, special design features (e.g., steel liners or walls in the radwaste enclosure) may be provided as part of the liquid radwaste treatment systems at certain sites. The staff will, therefore, review the site specific characteristics mentioned above individually for each plant referencing the ABWR and confine its review of ABWR, only to the choice of the liquid radwaste tank. Therefore, provide information on the following: (15.7.3)

(a) Basis for determining the concentrated waste tank as the worst tank (this may very well be the case, but in the absence of information on the capacities of major tanks, particularly the waste holdup tanks, it is hard to conclude that the above tank both in terms of radionuclide concentrations and inventories will turn out to be the worst tank).

(b) Radionuclide source terms, particularly for the long-lived radionuclides such as Cs-137 and Sr-90 (these may be the critical isotopes for sites that can claim only decay credit during transit) in the major liquid radwaste tanks.

470.1

Subsection 15.6.2 of the ABWR FSAR provides your analysis for the radiological consequences of a failure of small lines carrying primary coolant outside of containment. This analysis only considers the failure of an instrument line with a 1/4-inch flow restricting orifice. Show that this failure scenario provides the most severe radioactive releases of any postulated failure of a small line. Your evaluation should include lines that meet GDC 55 as well as small lines exempt from GDC 55.

470.2

Provide a justification for your assumption that the plant continues to operate (and therefore no iodine peaking is experienced) during a small line break outside containment (Subsection 15.6.2) accident scenario. Also provide the basis for the assumption that the release duration is only two hours.

470.3

Subsection 15.6.4.5.1.1 of the FSAR gives the iodine source term (concentration and isotopic mix) used to analyze the steam line break outside of containment accident. The noble gas source term, however, is not addressed. Provide the noble gas source term used. Also, the table in Subsection 15.6.4.5.1.1 seems heavily weighted to the shorter lived activities (i.e., (I-134). Provide the bases for the isotopic mix used in your analysis (iodine and noble gas).

470.4

Subsection 15.6.5.5 states that the analysis is based on assumptions provided in Regulatory Guide 1.3 except where noted. For all assumptions (e.g., release assumed to occur one hour after accident initiation, the chemical species fractions for iodine, the temporal decrease in primary containment leakage rates, credit for condenser leakage rates, and dose conversion factors) which deviate from NRC guidance such as regulatory guides and ICRP2, provide a detailed description of the justification for the deviation or a reference to another section of the SSAR where the deviations are discussed in detail. Provide a comparison of the dose estimates using these assumptions versus those which would result from using the NRC guidance.

470.5

Provide a discussion of, or reference to, the analysis of the radiological consequences of leakage from engineered safety feature components after a design basis LOCA.

470.6

For the spent fuel cask drop accident, what is the assumed period for decay from the stated power condition? What is the justification for that assumption?

470.7

The tables in Chapter 15 should be checked and revised as appropriate. In several cases the footnotes contain typographical errors related to defining the scientific notation. Table 15.7-12 also appears to contain inappropriate references to Table 15.7-16, rather that Table 15.7-13.

470.8

It is stated that Regulatory Guides 1.3 and 1.45 were used in the calculations of X/Q values. Based on the values presented, it appears as though a Pasquill stability Class F and one meter per second wind speed were assumed, with adjustment for meander per Figure 3 of Regulatory Guide 1.145. If this is not the case, describe the assumptions and justification used in calculating the X/Q values which are used in Chapter 15 dose assessments.

470.9

The SGTS filter efficiencies of 99% for inorganic and organic iodine are higher than the 90% and 70% values, respectively, assumed in Regulatory Guide 1.25 if it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine. Provide a justification for the use of the higher values.

470.10

Dose related factors such as breathing rates, iodine conversion factors and finite versus infinite cloud assumptions for calculating the whole body dose are not stated explicitly, although reference is made to Regulatory Guide 1.25 and another document. State these assumptions explicitly and justify use of any values which deviate from Regulatory Guide 1.25.

SECTION 20.3

CONTENTS

TABLES

ILLUSTRATIONS

SECTION 20.3

CONTENTS (Continued)

ILLUSTRATIONS

20.3 QUESTIONS/RESPONSES

This subsection provides the responses for each of the NRC questions identified in Sections 20.1 and 20.2. For convenience, each question is repeated here before its corresponding response. These questions/responses are provided in groups corresponding to the NRC Requests for Additional Information (RAI) referenced in Section 20.4. Within each group, the questions/responses are presented in the numerical order of the question numbers. Tables and Figures are provided at the end of each RAI group.

20.3.1 Response to First RAI-Reference 1

QUESTION 210.1

In Subsection 5.2.1.2, the statement is made that Section 50.55a of 10CFR50 requires NRC staff approval of ASME Code Cases only for Class 1 components. Revise this statement to be consistent with the current (1987) edition of 10CFR50.55a which requires staff approval of Code Cases for ASME Class 1, 2, and 3 components.

RESPONSE 210.1

Response to this question is provided in revised Subsection 5.2.1.2.

QUESTION 210.2

Revise Table 5.2-1 or provide additional tables in Subsection 5.2.1.2 which identifies all ASME Code Cases that will be used in the construction and in-plant operations of all ASME Class 1, 2, and 3 components in the ABWR. All Code Cases in these tables should be identified by Code Case number, revision and title. These tables should include those applicable Code Cases that are listed either as acceptable or conditionally acceptable in Regulatory Guides 1.84, 1.85 and 1.147. For those Code Cases listed as conditionally acceptable, verify that the construction of all applicable components will be in compliance with the additional Regulatory Guide conditions.

RESPONSE 210.2

Response to this question is provided in revised Subsection 5.2.1.2 and Table 5.2-1.

QUESTION 250.1

Subsection 5.2.4.1 should state that the system boundary includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor coolant systems, up to and including

(A) The outermost containment isolation valve in system piping that penetrates the primary reactor containment.

(B) The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment.

(C) The reactor coolant system and relief valves.

RESPONSE 250.1

Response to this question is provided in revised Subsection 5.2.4.1.

QUESTION 252.11

Subsection 5.2.3.4.2.3 states that the ABWR design meets the intent of this Regulatory Guide (1.71) by utilizing the alternate approach given in Section 1.8. We cannot review this subsection because we have not received Section 1.8. In addition, this subsection should be rewritten because it lacks detailed discussion about welder qualification.

RESPONSE 252.11

Response to this question is provided in revised Subsection 5.2.3.4.2.3.

QUESTION 281.1

In Section 5.1 (page 5.1-2) the function of the reactor cleanup system filter demineralizer should include the removal of radioactive corrosion and fission products in addition to particulate and dissolved impurities.

RESPONSE 281.1

Response to this question is provided in revised Section 5.1.

QUESTION 281.2

In Subsection 5.2.3.2.2 (page 5.2-7) irradiation-assisted stress corrosion cracking (IASCC) of reactor internal components and its mitigation are not discussed. Present laboratory data and plant experience has shown that IASCC can be initiated even at low conductivity ($< 0.3\mu S/cm$) after long exposure to radiation.

RESPONSE 281.2

Response to this question is provided in the new Subsection 5.2.3.2.4, *IGSCC Considerations*.

QUESTION 281.3

In Subsection 5.2.3.2.2 (pages 5.2-7 and 8) the ABWR standard plant design does not clearly incorporate hydrogen water chemistry to mitigate IGSCC. Since the plant design life is 60 years, hydrogen water chemistry may be of greater importance in reducing reactor coolant electrochemical corrosion potential to prevent IGSCC as well as IASCC. If hydrogen water chemistry is the referenced ABWR standard design, the following documents should be cited:

EPRI NP-5283-SR-A, *Guidelines for Permanent BWR Hydrogen Water Chemistry Installations* - 1987 Revision.

EPRI NP-4947-SR-LD, *BWR Hydrogen Water Chemistry Guidelines* - 1987 Revision (to be published).

RESPONSE 281.3

Response to this question is provided in revised Subsection 5.2.3.2.2.

(2) Criteria for Selecting Stellite Materials:

1. Wear resistance

2. Weldability

3. Experience and service history

4. Radiation level in area of application

(3) Evaluation of Noncobalt-containing Material to Replace Stellite:

The major source of cobalt from the reactor core has been Haynes 25 and Stellite 3 (cobalt-based alloys) for pins and rollers, respectively, in BWR control rods. Replacement of the cobalt alloy pins and rollers with noncobalt alloys has been extensively investigated under a joint GE-EPRI program (Project 1331-1). The results of this investigation are documented in the report, EPRI NP-2329, *Project 1331-1, Final Report*, March 1982. The current design noncobalt materials are alloy X-750 for control rod rollers and 13-8 PH for the pins.

QUESTION 281.7

Subsection 5.2.3.2.2.3(4) (page 5.2-10) states that control of reactor water oxygen during startup/hot standby may be accomplished by utilizing the de-aeration capabilities of the condenser. In addition, this section states that independent control of control rod drive (CRD) cooling water oxygen concentrations of < 50 ppb during power operation is desirable to protect against IGSCC of CRD materials. Are either one or both of the above dissolved oxygen controls incorporated in the ABWR standard plant design?

RESPONSE 281.7

In Subsection 5.2.3.2.2.3, control of reactor water oxygen by using the condenser and control of control rod drive water were mentioned as dissolved oxygen control methods. These two plant features are not in the Nuclear Island scope. However, an interface requirement has been added (see new Subsection 5.2.5) that requires the remainder of the plant to meet the water quality requirements of Table 5.2-5.

QUESTION 281.8

In Subsection 5.2.3.2.2.3(13) (page 5.2-11) it states that the main steam line radiation monitor indicates an excessive amount of hydrogen being injected. An explanation of this occurrence should be discussed.

RESPONSE 281.8

Reponse to this question is provided in revised Subsection 5.2.3.2.2.3(13).

QUESTION 281.9

Subsection 6.4.4.2 (page 6.4-6) discusses personnel respirator use in the event of toxic gas intrusion into the control room. However, the chlorine detection system is not discussed. Also, any control functions that are automatically triggered by a chlorine detector alarm (closing intake dampers, energizing control room HVAC system recirculation) should be identified.

RESPONSE 281.10

Item 1

Response to Item 1 of this question is provided in revised Subsection 5.2.3.2.2.2.

Item 2

Response to Item 2 of this question is provided in revised Subsection 5.2.3.2.2.

Item 3

Information is being obtained and evaluated from operating plants with GEZIP. However, this feature is not in the Nuclear Island scope.

Item 4

This feature is not in the Nuclear Island scope. However, an interface requirement has been added (see new Subsection 5.2.6) that requires the remainder of the plant to meet the water quality requirements in Table 5.2-5.

Item 5

New and improved water quality monitoring instrumentation is being constantly developed and introduced for use in BWR plants. Several useful instruments have been developed and introduced within the past few years. GE will evaluate the state of the art when a BWR is undergoing detailed design and will incorporate such instruments that are necessary to assure proper water quality.

Item 6

Response to Item 6 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 7

Response to Item 7 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 8

Response to Item 8 of this question is provided in revised Subsection 5.2.3.2.2.2 and Table 5.2-5.

Item 9

Response to Item 9 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 10

Response to Item 10 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 11

Response to Item 11 of this question is provided in revised Subsection 5.2.3.2.2.3.

Item 12

This design feature is not in the Nuclear Island scope. However, an interface requirement has been added (see new Subsection 5.2.6) that requires the remainder of the plant to meet the water quality requirements in Table 5.2-5.

QUESTION 470.1

Subsection 15.6.2 of the ABWR FSAR provides your analysis for the radiological consequences of a failure of small lines carrying primary coolant outside of containment. This analysis only considers the failure of an instrument line with a 1/4-inch flow restricting orifice. Show that this failure scenario provides the most severe radioactive releases of any postulated failure of a small line. Your evaluation should include lines that meet GDC 55 as well as small lines exempt from GDC 55.

RESPONSE 470.1

The analysis for failure of a small line carrying primary coolant was conservatively analyzed as a failure of an instrument line with full flow for a period of two hours. This analysis is deemed conservative for the reason given below.

Of all the lines carrying coolant penetrating the primary containment wall, only the instrument lines are exempt from GDC 55. All other lines use some form of check valve/motor-operated valve combination to stop the flow of primary coolant in the event of a line break. Typically, the motor-operated valves close at the rate of two inches per ten seconds. Considering a two-inch line and assuming that a flow of 175 pounds per second would result in operator action within 60 seconds, the total mass released over the 70 second period would be approximately 12,000 pounds or about one half of the assumed release over two hours from the instrument line. Using this logic and these simplified calculations, it is found that a two-hour instrument line break bounds releases for small lines.

QUESTION 470.2

Provide a justification for your assumption that the plant continues to operate (and therefore no iodine peaking is experienced) during a small line break outside containment (Subsection 15.6.2) accident scenario. Also provide the basis for the assumption that the release duration is only two hours.

TABLE 20.3-1

SENSITIVITY STUDY OF PARAMETERS FOR LOCA ANALYSIS

(RESPONSE TO QUESTION 470.4)

		Site Boundary 24 Hr. Dose at 300 m (REM)		LPZ Dose for 30 Days at 800 m (REM)	
		Thyroid	Whole Body	Thyroid	Whole Body
1.	LOCA Results	1.5	0.62	22.	12.
2.	No Initial 1 Hr. Hold-up	1.5	0.90	22	13
3.	No Pressure Reduction @ 24 Hrs	NC	NC	22	13
4.	Iodine Species Consistent with Regulatory Guide 1.3	10.0	0.64	1700	13
5.	No Suppression Pool Scrubbing	140.	0.92	930	13
6.	No Steamline Plateout	1.5	0.62	23	12
7.	No Steamline Plateout or Hold-up	1.5	0.64	23	12
8.	No Condenser Plateout	2.3	0.62	340	12
9.	No Condenser Plateout or Hold-up	280	41	1300	70

NOTE:

All evaluations are made independently of each other.

20.3.2 Response to Second RAI-Reference 2

QUESTION 430.1

Provide a failure modes and effects analysis of the control rod drive system (CRDS) in tabular form with supporting discussion to delineate the logic employed. The failure analysis should demonstrate that the CRDS can perform the intended functions with the loss of any active single component. These evaluations and assessments should establish that all essential elements of the CRDS are identified and provisions are made for isolation from nonessential CRDS elements. It should be established that all essential equipment is protected from common mode failures such as failure of moderate-and high-energy lines. The failure mode and effects analysis of the control rod drives should include water, air and electrical failures to CRDs and how the CRD system operation is affected due to air contamination or water contamination. Before finalizing the scope of the analysis, refer to ACRS subcommittee meeting proceedings on the ABWR dated June 1, 1988. It is noted that the above information is to be included in Appendix 15B of the SSAR which will be submitted at a later date. However, the evaluation of the functional design of the reactivity control systems cannot be completed until this information is provided. (4.6)

RESPONSE 430.1

FMEAs for the CRDS and other selected systems will be submitted by December 31, 1988. The scope of CRDS FMEA will include appropriate consideration of the June 1, 1988 ACRS subcommittee meeting proceedings.

QUESTION 430.2

Regarding Reactor Coolant Pressure Boundary (RCPB) leakage detection systems provide information on the following: (5.2.5)

(a) Describe how the leakage through both the inner and outer reactor vessel head flange seals will be detected and quantified.

(b) List the sources that may contribute to the identified leakage collected in the Reactor Building Equipment Drain Sumps.

(c) Describe how potential intersystem leakages will be monitored for the (1) Low Pressure Coolant Injection System, (2) High Pressure Core Spray System, (3) Reactor Core Isolation Cooling System (RCIC) - Water side and (4) Residual Heat Removal System - Inlet and discharge sides. Your response should include all the applicable (for the ABWR design) systems and components connected to the Reactor Coolant System that are listed in Table 1 of SRP Section 5.2.5 and other systems that are unique to ABWR (except those that you have already discussed in SSAR Subsection 5.2.5.2.2, Item 11).

RESPONSE 430.2

Response to this question will be provided by November 11, 1988.

QUESTION 430.3

Discuss compliance of reactor coolant leak detection systems with Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", Positions C4, C5, C6, C8, and C9 with respect to the following items: (5.2.5)

(a) Indicators for abnormal water levels or flows in all the affected areas in the event of intersystem leakages.

(b) Sensitivity and response time of leak detection systems used for unidentified leakages outside the drywell.

(c) Qualification relating to seismic events for drywell equipment drain sump monitoring system and leak detection systems outside the drywell.

(d) Testing Procedures - Monitoring sump levels and comparing them with applicable flow rates of fluids in the sumps.

(e) Inclusion of reactor building and other areas floor and equipment drain sumps in ABWR Technical Specifications for leak detection systems.

Note that a few of the questions above arise because in Subsection 5.2.5.4.1 you state that the total leakage rate includes leakages collected in drywell, reactor building and other area floor drain and equipment drain sumps.

RESPONSE 430.3

Response to this question will be provided by November 11, 1988.

QUESTION 430.4

Clarify whether the RCIC makeup capacity is sufficient to provide also for main turbine stop valves. Also, clarify whether this leakage is included in the total leakage mentioned in Subsection 5.2.5.4.1.

RESPONSE 430.4

The RCIC system has sufficient capacity to account for this leakage. The total leakage mentioned in Subsection 5.2.5.4.1 does not account for this leakage.

QUESTION 430.5

Clarify how Position C.2 of RG 1.29, "Seismic Design Classification" is met for all applicable leak detection systems (also include the leak detection systems outside the drywell). (5.2.5)

RESPONSE 430.5

All elements of the leak detection and isolation system (LDS) and supporting systems that must accomplish a safety function or whose failure could prevent accomplishment of a safety function will be designed to accommodate a SSE and remain functional. All such equipment will be designated as Seismic Category I equipment.

All LDS equipment related to isolating functions and all equipment of interfacing systems, either providing input signals to the LDS, or which receive LDS isolation signals and accomplish the safety functions related to isolating the reactor coolant pressure boundary (RCPB) or the primary containment vessel (PCV) will thus conform to Position C.2 of RG 1.29. Such conformance shall be applied to the LDS itself and to all systems which support the LDS in monitoring for leaks from the RCPB, internal to the drywell or external to the drywell, e.g., the nuclear boiler system and the process radiation monitoring system provide such support.

The LDS and associated safety systems will also conform to the RG 1.100 position related to satisfying requirements of IEEE 344. Note that RG 1.100 effects interfacing mechanical systems (e.g., the isolation valves and motor control centers, etc., of these systems) to a greater degree than it effects the LDS.

The airborne particulate radioactivity monitoring system of the LDS will also meet the guidelines of RG 1.45, Position C-6 and will be designed to remain functional when subjected to a SSE.

QUESTION 430.6

Identify all the interface requirements relating to RCPB leakage detection systems. (5.2.5)

RESPONSE 430.6

There are no RCPB leakage detection system safety-related interfaces for the ABWR Standard Plant. This will be reflected in Section 1.9.

QUESTION 430.7

In the SSAR section devoted to containment functional design, identify clearly those areas that are not part of the ABWR scope and provide relevant interface requirements. (6.2)

RESPONSE 430.7

There are no containment safety-related interfaces for the ABWR Standard Plant. This will be reflected in Section 1.9.

QUESTION 430.8

With respect to the design bases for the containment: (6.2)

QUESTION 430.8a

Discuss the bases for establishing the margin between the maximum calculated accident pressure or pressure difference and the corresponding design pressure or pressure difference. This includes the design external pressure, internal pressure, and pressure between subcompartment walls.

RESPONSE 430.8a

The containment pressure response to a postulated accident is divided into three different time periods: vent-clearing; short-term; and long-term. The most dynamic processes occur during vent clearing and result in a very rapid rise in containment pressure and the maximum differential pressure across the diaphragm floor. Because these processes are so dynamic, a margin of 30% between the maximum calculated pressure and the design pressure is specified for design purposes. The peak containment pressures are reached during the short-term period. For this time period a margin of 15%

to the maximum calculated pressure is specified. This 15% margin is judged to be adequate, since the blowdown and containment response are relatively stable and predictable. The short-term maximum calculated pressure will bound the long-term pressure response.

The 30 and 15% margins described above are the same as those recommended by the Standard Review Plan.

QUESTION 430.8b

Discuss the capability for energy removal from the containment under various single-failure conditions. State and justify the design basis single failure that affects containment heat removal.

RESPONSE 430.8b

The containment heat removal system, which comprises of three independent loops, has energy removal capability to keep the suppression pool temperature within the acceptable limits and other guidelines. The design basis of the heat removal system assumes a single failure of a RHR heat exchanger which is the most limiting single failure.

QUESTION 430.9

The Standard Safety Analysis Report (SSAR) states that the analytical models used to evaluate the containment and drywell response to postulated accidents and transients are included in the General Electric Co. report NEDO-20533 and its supplement 1, entitled "The G.E. Mark III Pressure Suppression Containment Analytical Model". Provide justification that these references are appropriate to use for the ABWR Containment design which is not specified as Mark III. Discuss the similarities and differences of the ABWR design to previously approved Mark II and Mark III designs as they relate to the containment and drywell responses to the postulated accidents and the analytical model used for the analyses. Include in the discussion the conservatism used in the model and assumptions, the applicable test data that support the analytical models, and the sensitivity of the analyses to key parameters. (6.2)

RESPONSE 430.9

The analytical models described in the NEDO-20533 are appropriate to calculate the ABWR (containment and drywell) short-term responses to postulated accidents. Though originally written for prediction of Mark III transients, these models, which simulate from first principles the transient conditions in the containment, can be adapted for the ABWR containment configuration. These models have the capacity to model the reactor pressure vessel, drywell, vent systems, and wetwell (suppression pool and airspace). They are, therefore, adaptable to other containment configuration having the same basic components. Comparison of these analytical models with test data is described and contained in NEDO-20533. In calculating the ABWR containment responses to postulated accidents, these models are used with conservative modeling assumptions. These assumptions are described in Subsection 6.2.1.1.3.3.

The ABWR design, basically, utilizes combined features of Mark II and Mark III design, with the exception of a unique feature of two drywell volumes (upper and lower). The vent system is a combination of vertical (Mark II design) and horizontal (Mark III design) vent system, and the wetwell (suppression pool and airspace) is similar to Mark II. The above models have capabilities to predict the containment and drywell responses to the postulated accidents. The vent system (combined vertical and horizontal vents) can be modeled by employing appropriate vent loss coefficient values. The unique lower drywell feature of ABWR can be modeled by taking credit for transfer of a conservative fraction of the lower drywell contents into the wetwell airspace. Because the lower drywell is connected to the drywell connecting vents, the inert atmosphere in the lower drywell would not transfer to the wetwell until the peak pressure in the drywell is achieved.

QUESTION 430.10

With regard to the design features of the containment: (6.2)

QUESTION 430.10a

Provide general arrangement drawings for the containment structure.

RESPONSE 430.10a

The general configuration and the major dimensions of the containment are shown in Figure 3.8-18. The nomenclature for various part of the containment and the internal structures are shown in Figure 3.8-17. The horizontal cross-sections of the reactor building and the containment are shown in Figures 3.8-1 through 3.8-7; the vertical cross-sections are shown in Figures 3.8-10 and 3.8-11. The code jurisdictional boundary for various codes is shown in Figure 3.8-12.

QUESTION 430.10b

Provide appropriate references to Section 3 of the SSAR which includes the information on the codes, standards, and guides applied in the design of the containment and containment internal structures.

RESPONSE 430.10b

The applicable codes, standards, and specifications applied in the design of the containment and internal structures are provided in the following subsections of Chapter 3:

Item	Subsection
Concrete Containment	3.8.1.2
Steel Components of the Reinforced Concrete Containment	3.8.2.2
Concrete and Steel Internal Structures of the Concrete Containment	3.8.3.2

QUESTION 430.10c

Discuss the possibilities of water entrapment inside containment and its effect on the accident analysis.

RESPONSE 430.10c

The ABWR containment unique design feature - lower and upper drywell volumes - has potential for some water entrapment inside containment. Water could be trapped in the lower drywell cavity and the wetwell equipment and personnel tunnel from two possible sources: (1) from the suppression pool drawdown through the suppression pool return path (see Figure 3.8-18) or (2) directly from the reactor pressure vessel (RPV). Effect of this possible water entrapment was considered as described below.

For the short-term response analysis which determines sizing of the suppression pool, water entrapment was not considered in the analysis. It was found that the short-term blowdown is practically over before the spill-over from the suppression pool through the return path starts. Any drawdown directly from the RPV to the lower drywell cavity will result in reduced pool heatup which, in turn, will require a smaller pool volume. Therefore, for conservatism, no water entrapment was considered in determining the minimum suppression pool volume required. For the long-term response analysis which determines maximum pool temperature rise, water entrapment was considered in the pool temperature response analysis. This is conservative since water entrapment reduces the suppression pool heat sink capacity and therefore maximizes the pool temperature rise.

QUESTION 430.10d

Provide information on qualification tests that are intended to demonstrate the functional capability of the containment structures, systems and components. Discuss the status of any developmental tests that may not have been completed.

RESPONSE 430.10d

The structural integrity pressure test is discussed in Subsection 3.8.1.7.1. The preoperational and inservice integrated leak rate test is discussed in Subsection 6.2.1.6. The shop tests related to reinforced concrete containment vessel which were performed in Japan between 1981 and 1987 are listed below:

(I) Fundamental Test

1. Transverse Shear
2. Openings in RCCV
3. Rebar Joints

(II) Partial Test

1. Top slab
2. Liner and liner anchors
3. Diaphragm floor slab joint
4. Penetrations

(III) Total Test

1. Large scale (1/6) model

All of the developmental tests are complete.

QUESTION 430.11

Provide a detailed discussion of the likelihood and sensitivity to steam bypass of the suppression pool for a spectrum of accidents. Include in you discussion the following information: (6.2)

RESPONSE 430.11

The ABWR design uses a pressure suppression-type containment which is similar to that used in the Mark I, II, and III containment designs. In a pressure suppression-type containment, any steam released from the primary system following a postulated LOCA will be condensed by the suppression pool. However, the potential exists for steam to bypass the suppression pool through leakage paths between the drywell and the wetwell airspace. The steam from the drywell leaking directly into the wetwell airspace would produce pressurization of the ABWR containment as is the case for the other (Mark II and III) containment designs.

Large primary system ruptures generate high pressure differentials across the assumed leakage path which, in turn, give proportionally higher leakage flow rates. However, large breaks also rapidly depressurize the reactor and terminate the blowdown. As the size of the assumed primary system rupture decreases, the magnitude of the differential pressure across any leakage path also decreases. Small breaks, however, result in an increasingly longer reactor blowdown period, which in turn, results in longer durations of the leakage flow. The limiting case is a very small reactor system break which will not automatically result in reactor depressurization. For larger breaks the maximum allowable area of the leakage path is larger, since leakage into the wetwell airspace is of limited duration.

QUESTION 430.11a

A comparison of the ABWR pool bypass capability with that for Mark II and Mark III designs.

RESPONSE 430.11a

The ABWR containment design has a steam bypass capability for small breaks of the order of 0.05 ft^2 $(A/\sqrt{K})$, same as for the Mark II plants.

QUESTION 430.11b

The measures for minimizing the potential for steam bypass and the systems provided to mitigate the consequences of pool bypass. Discuss and demonstrate the conservatism of assumptions made in the analysis of steam bypass.

RESPONSE 430.11b

The potential leakage paths for steam bypass incorporate design features which help in minimizing the potential for steam bypass. The ABWR design includes a wetwell spray system to mitigate the consequences (wetwell airspace pressurization) of suppression pool bypass. Detailed analysis results, discussing the analysis assumptions will be provided by December 31, 1988.

QUESTION 430.11c

Identify all lines from which leakage (or rupture) could contribute to pool bypass and wetwell air space pressurization.

RESPONSE 430.11c

Response to the question will be provided by December 31, 1988.

QUESTION 430.11d

Identify all fluid lines which traverse the wetwell air space and identify those lines which are protected by guard pipe.

RESPONSE 430.11d

Response to the question will be provided by December 31, 1988.

QUESTION 430.11e

Discuss the rationale and basis for the wetwell spray flow capacity.

RESPONSE 430.11e

The primary purpose of the wetwell spray system (manually operated) is to provide mitigation for the adverse consequences of the steam bypass. The basis for the wetwell spray flow capacity (500 gpm) is to assure that the maximum containment pressure due to pool bypass does not exceed the containment design pressure

QUESTION 430.12

With regard to containment response to external pressure: (6.2)

QUESTION 430.12a

Describe the wetwell-to-drywell vacuum breaker system and show the extent to which the requirements of subsection NE of section III of the ASME B&PV Code are satisfied. Discuss the functional capability of the system. Provide the design and performance parameters for the vacuum relief devices.

RESPONSE 430.12a

The wetwell-to-drywell vacuum breaker system (WDVBS) is safety-related consisting of eight (8) 20-inch vacuum breaker valves. Seven valves are required to open to provide an effective flow area adequate to keep the differential pressure between the drywell and wetwell within the negative design value of 2 psi during all operating and accident transients. Therefore, the system design accounts for the single failure case in which one valve fails to open. Each vacuum breaker valve shall open fully within 1.0 second (start to open at a pressure differential of 0.2 psi and fully open at 0.5 psi).

The vacuum breaker valves shall be installed on the RPV pedestal in separate penetrations from the lower drywell to the suppression chamber airspace, with one valve per penetration. The vacuum breaker valves shall be swinging disk valves which will be actuated by the differential pressure across the valve ports. No external power shall be utilized to open the valves. Valves shall be capable of being manually operated and remotely operated with air operated piston to verify the movement of valve disk. The valve shall be supplied with a position indicator switch in the control room that will permit remote indication of valve position in control room.

QUESTION 430.12b

Discuss the basis for selecting a low design capability for external pressure acting across the drywell to wetwell boundary. It is not apparent that the drywell negative design pressure of 2.0 psid is desirable or sufficient.

RESPONSE 430.12b

The drywell negative design pressure of 2.0 psid is specified mainly for designing the steel liner. The ABWR primary containment vessel (PCV) is a steel-lined reinforced concrete containment vessel (RCCV). The main purpose of steel liner is to provide the leaktightness required. This design value of 2.0 psid, which has also been specified for the Mark II design, is judged to be adequate based on the experience for the Mark II plants.

Engineering analyses were performed (with no vacuum breakers) to calculate the negative differential pressure between the wetwell and the reactor building. All possible wetwell depressurization events which may result in the negative differential pressure were considered, and analyses were conducted for the limiting transient event. The negative differential pressure was determined to be 1.8 psid, which is below (by 10%) the negative design pressure of 2.0 psid.

QUESTION 430.12c

The margin between the calculated wetwell-to-reactor building negative differential pressure (-1.8 psid) and the design differential pressure (-2.0 psid) is not considered adequate. A higher margin of 15% should be provided at this stage of the design. Further, given the reliance of the BWR pressure suppression design on containment venting to control pressure, discuss the basis for not providing wetwell to reactor building vacuum breakers.

RESPONSE 430.12c

Experience indicates that a margin of 10% between the calculated and the design differential pressure should be adequate. As noted in response to Part b of this question this design pressure is for the steel liner and this is not a load carrying component to provide structural integrity of the primary containment boundary. The reinforced concrete walls (about 6 ft. thick) are the main load carrying components whose design is controlled by the internal design pressure 45 psig which is carried by rebar. The concrete walls (are not vulnerable) are subjected to compression under the 2.0 psid negative design pressure. Therefore, it is not necessary to provide wetwell to reactor building vacuum breakers.

QUESTION 430.12d

In the analysis of wetwell-to-reactor building negative differential pressure calculation, a 500 gpm wetwell spray flow rate was used. Provide the basis for the assumption and the design basis for the wetwell spray capacity.

RESPONSE 430.12d

See response to Question 430.11e.

QUESTION 430.13

Section 6.2.1.1.3 of the SSAR states that the containment functional evaluation is based upon the consideration of several postulated accident conditions including small break accidents. Provide the assumptions, analysis and results of the small break accidents considered, and demonstrate that the identified (in the SSAR) feedwater line and steam line breaks are the limiting accidents.

RESPONSE 430.13

Response to this question will be provided by December 31, 1988.

QUESTION 430.14

Provide analyses of the suppression pool temperature for transients involving the actuation of safety/relief valves. Provide the assumptions and conservatism employed in the analyses so that an assessment could be made for conformance to the acceptance criteria set forth in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments." (6.2)

RESPONSE 430.14

Suppression pool temperature analyses, for transients involving the actuation of safety relief valves (SRVs) to show conformance to NUREG-0783 are not required. Recent studies conclude that the pool temperature limit for SRV discharge is not necessary and may be eliminated. Results of these studies are documented in the GE Report NEDO-30832, Class I, December 1984. This report has been provided to the NRC Staff via BWR Owners Group letter, BWROG-8513, of March 21, 1985.

A temperature limit for BWR suppression pools during SRV discharge was specified in NUREG-0783. This limit was established because of concerns about unstable condensation and associated high loads on the containment structure at high suppression pool temperatures. The concern was raised because of experience in BWRs with prolonged SRV discharge without quencher devices. The NRC established the temperature limits in NUREG-0783 based on data available at the time it was issued in 1981. At that time sufficient data was not available to confirm that quenchers were effective in eliminating the unstable condensation loads.

Since NUREG-0783 was issued, scaling laws have been developed and confirmed for the discharge and condensation of steam in a suppression pool. Also, the subscale data base has been expanded over a range of pool temperatures up to saturation temperature with both straight pipe geometries and with quencher devices. The confirmation of the scaling laws and the expanded data base now provide strong support for the elimination of the pool temperature limit for SRV discharge with quenchers (T- and X-quenchers). For details, please refer to the GE Report NEDO-30832, Class I, December 1984, *Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers.*

The ABWR design utilizes X- quencher discharge devices which are the same as that used for the Mark II and Mark III designs and evaluated in the recent study noted above. This study determined that the dynamic pressures (loads) due to SRV discharge decrease as the pool temperature approaches saturation temperature, and concluded that the pool temperature limit, specified in NUREG-0783, for SRV discharge is not necessary and may be eliminated.

Therefore, the acceptance criteria set forth in NUREG-0783 are not necessary and, hence, suppression pool transient analyses involving the actuation of SRVs are not needed.

QUESTION 430.15

Provide the pressure at which the maximum allowable leak rate of 0.5%/day is quoted. (6.2)

RESPONSE 430.15

Response to this question is provided in revised Table 6.2-2.

QUESTION 430.16

Provide engineered safety systems information for containment response analysis (full capacity operation and capability used in the containment analysis), as indicated in Table 6-7 of Regulatory Guide 1.70, Revision 3. (6.2)

RESPONSE 430.16

Response to this question is provided in revised Subsection 6.2.1.1.3.2 and new Table 6.2-2a.

QUESTION 430.17

In the design evaluation section for containment subcompartments (Section 6.2.1.2.3), provide the information necessary to substantiate your assessment that the peak differential pressures do not exceed the design differential pressure. Guidance for the information required is provided in Regulatory Guide 1.70, Revision 3, Section 6.2.1.2., "Containment Subcompartments", Design Evaluation.

RESPONSE 430.17

Response to this question will be provided by December 31, 1988.

QUESTION 430.18

Describe the manner in which suppression pool dynamic loads resulting from postulated loss-of-coolant accidents, transients (e.g., relief valve actuation), and seismic events have been integrated into the affected containment structures. Provide plan and section drawings of the containment illustrating all equipment and structural surfaces that could be subjected to pool dynamic loads. For each structure or group of structures, specify the dynamic loads as a function of time, and specify the relative magnitude of the pool dynamic load compared to the design basis load for each structure. Provide justification for each of the dynamic load histories by the use of appropriate experimental data and/or analyses.

Describe the manner by which potential asymmetric loads were considered in the containment design. Characterize the type and magnitude of possible asymmetric loads and the capabilities of the affected structures to withstand such a loading profile. (6.2)

RESPONSE 430.18

Response to this question will be provided by November 11, 1988.

QUESTION 430.19

Provide information to demonstrate that the ABWR design is not vulnerable to a safety relief valve discharge line break within the air space of the wetwell, coupled with a stuck open relief valve after its actuation as a result of the transient. (6.2)

RESPONSE 430.19

Response to this question will be provided by November 11, 1988.

QUESTION 430.20

Discuss suppression pool water makeup under normal and accident condition. (6.2)

RESPONSE 430.20

Under normal conditions, make-up water to the suppression pool can be added by the suppression pool clean-up (SPCU) system. Suction is taken from the condensate storage pool (CSP) through a line that primarily supplies the high pressure core flooder (HPCF) system and the reactor core isolation cooling (RCIC) system. The SPCU pump outlet is piped to the suppression pool.

Under loss-of-coolant accident conditions the ECCS systems (HPCF and RCIC) take primary suction from the CSP and secondary suction from the suppression pool. Suction from the CSP is the preferred source of water. The containment accident response (pressure and temperature) analyses neglect this source of make up water for conservatism.

For post-accident suppression pool makeup or containment flooding, the HPCF system can take suction from the CSP and pump water through the HPCF suppression pool return line. This will provide makeup to the suppression pool or fill the containment to a water level consistent with containment design pressure. For the extreme situation where containment flooding is desired, additional water can be added to the CSP using fire hoses or another alternate source of water. For containment flooding the suppression pool is completely filled and the drywell flooded to the desired level.

QUESTION 430.21

With respect to mass and energy release analyses for postulated loss-of-coolant accidents identify the sources of generated and stored energy in the reactor coolant system that are considered in the analyses of loss-of-coolant accidents. Describe the methods used and assumptions made in calculations of the energy available for release from these sources. Address the conservatism in the calculation of the available energy from each source. Tabulate the stored energy sources and the amounts of stored energy. For the sources of generated energy, provide curves showing the energy release rates and integrated energy release. (6.2)

RESPONSE 430.21

The energy released for postulated loss-of-coolant accidents is comprised of (1) the energy generated by fission product decay, and (2) stored energy in the reactor system. For short-term response analyses, ANS-5 decay heat curve plus 20% margin is used for added conservatism. The rate of release of core decay heat is provided in Table 20.3-2 as a function of time after accident initiation, and Table 20.3-3 provides integrated decay heat release rate. For long-term analyses ANS-5 decay heat curve with no added margin is used.

The sensible stored energy in the reactor coolant system is made available to the reactor coolant by modeling the heat sources as heat capacity modes in the analyses. Following each postulated accident event, the total stored energy is made available for transfer to the reactor coolant. An estimated total amount of available stored energy is about 200 x 10^6 Btu.

QUESTION 430.22

In the SSAR sections devoted to containment heat removal systems, identify clearly those areas that may not be part of the GE scope and provide relevant interface requirements. (6.2)

RESPONSE 430.22

There are no containment heat removal system safety-related interfaces for the ABWR Standard Plant. This will be reflected in Section 1.9.

QUESTION 430.23

The SSAR states that the containment heat removal system is designed to limit the long-term temperature of the suppression pool to 207°F. The calculated peak pool temperature is 206.46°F for the feedwater line break. With respect to this analysis provide the following information: (6.2)

QUESTION 430.23a

The justification that this is the limiting accident with respect to the maximum temperature in the suppression pool.

RESPONSE 430.23a

In determining energy removal capability of the containment heat removal system, various potential bounding transient and accident event were analyzed. The events analyzed are:

(1) Potential Bounding Transients on Suppression Pool Temperature

- Inadvertent Open Relief Valve (IORV)
- Loss-of-Coolant Accident (LOCA) (whole spectrum of LOCAs)

(2) Normal Shutdown Cooling

(3) Emergency Shutdown Cooling

(4) Anticipated Transients Without Scram (ATWS)

A feedwater line break (FWLB), which is the largest liquid break, was determined to be the most bounding event for pool temperature response. Liquid breaks are expected to be more bounding than steam breaks, since liquid breaks are expected to result in pool drawdown. Pool drawdown will substantially reduce the heat sink capacity of the suppression pool.

QUESTION 430.23b

The bases for the design margin between the design and calculated temperatures.

RESPONSE 430.23b

The wetwell design temperature is $219^{\circ}F$ (see Table 6.2-1). The long-term pool temperature of $207^{\circ}F$ is to assure sufficient net positive suction head (NPSH) for the pumps. The calculated peak pool temperature of $206.46^{\circ}F$ demonstrates that the containment heat removal system has adequate energy removal capability.

QUESTION 430.23c

All assumptions used in the analysis and conservatism associated with each. Include the effects of potential temperature stratification in the suppression pool and its effects on heat removal capability of the system.

RESPONSE 430.23c

Analysis assumptions are listed in subsection 6.2.1.1.3.3.1.2.

During the LOCA blowdown, there exists a potential for temperature stratification in the suppression pool. During this period most of the mass and energy is release to the pool through the top horizontal vents. As a result, the top portion of the pool will be heated more than the lower portion. The temperature in the lower part of the pool where the RHR suction is located can be expected to be lower than the bulk pool temperature thus, the heat removal through the RHR heat exchanger may be less than that expected if a uniformly mixed pool temperature at the RHR suction is assumed.

The long-term pool temperature analyses assume a well mixed uniform suppression pool temperature. It is believed that the location of the RHR suction and return lines in the suppression pool, and other conservatisms in the analyses will more than offset the effect of potential pool stratification. The RHR suction and return line configuration will be designed (similar to Mark III design) to provide adequate pool mixing and reduce the pool thermal stratification. The long-term analyses conservatively model and use a lower than expected suppression pool volume; no credit for heat sinks in the drywell and wetwell; and no credit for the ECCS suction from the condensate storage pool. Furthermore, based on design practices, the RHR heat exchanger thermal performance is considerably better than the design minimum.

QUESTION 430.23d

The identification of the decay heat curve used in the analysis.

RESPONSE 430.23d

ANS-5 decay heat curve.

QUESTION 430.24

Provide the design bases for the spray features of the containment heat removal system. Provide the safety classification of the components associated with the spray feature of the system. (6.2)

RESPONSE 430.24

The drywell spray performs iodine removal which is not a NRC requirement. The drywell spray design is based on Japan Atomic Energy Research Institute (JAERI) testing. JAERI has tested the iodine capability of PCV spray with $0.1 < F/V < 0.4$ and have determined that, as a minimum, this range is acceptable,

where F = spray flow rate, m^3/hr
V = free air volume (drywell), m^3

For ABWR $\frac{F}{V} = \frac{840}{7350} = 0.11$, which is within the acceptable range.

(Note: $840\ m^3/hr = 1.81 \times 10^6$ lb/hr is from Table 6.2-2a).

The design bases for the wetwell spray is provided in the Response to Question 430.11e.

Both wetwell and drywell spray headers are located inside the primary containment vessel and are classified as Safety Class 3.

QUESTION 430.25

Discuss the rationale for continued reliance on sprays as the sole active engineered safety feature for drywell atmosphere pressure and temperature. Discuss the merits of upgrading the design of drywell fan coolers to provide some capacity for pressure, temperature, and humidity control following an accident. (6.2)

RESPONSE 430.25

The ABWR containment design does not require nor does it rely upon sprays for controlling drywell pressure and temperature below their design values following design basis loss-of-coolant accident (LOCA) conditions. The primary design objective of the drywell sprays (initiated by operator action) is to provide removal of the fission products released in the drywell during LOCA. As an option, drywell sprays can be utilized in controlling equipment environmental conditions in the drywell.

The ABWR drywell cooling system design is non-safety grade. Upgrading the design to safety to provide some capacity for controlling drywell thermodynamic conditions following an accident is not regarded as cost effective. Control of drywell conditions through the suppression pool cooling (RHR heat exchangers) is an order of magnitude more effective in overall containment heat removal than the drywell cooling system. It is not necessary to have the drywell cooling system available for controlling conditions in the drywell following an accident. The RHR heat exchangers have adequate heat removal capability.

In order to upgrade the drywell cooling system to safety grade, extensive design modification will be required. The entire cooling system (cooling units, pipings, ducts, source of cooling water, etc) design will be required to withstand seismic loads and other loads due to a high energy pipe break. In addition, this upgrading will require an increase in the emergency diesel generator capacity. Considering that it is not necessary to have the drywell cooling system available following an accident and the upgrading requires extensive design modifications, it is concluded that there is no technical merit in upgrading the drywell fan coolers.

QUESTION 430.26

The time period assumed for initiation of the containment heat removal system after a LOCA is 10 minutes requiring operator action. It is the staff's position that this time period is too restrictive. In fact previous BWR designs(Grand Gulf's Mark III) use 30 minutes actuation time. Provide the reasons why the ABWR does not provide more flexibility with respect to the time required for actuation. (6.2)

RESPONSE 430.26

Response to this question is provided in revised Subsections 6.2.1.1.3.3.1.2, 6.2.2.2 and 6.2.2.3.1. In addition, the following clarification is provided.

For the RHR response to a LOCA, 10 minutes was assumed as the time following the LOCA initiation when containment cooling is initiated. The ABWR RHR is designed with its heat exchanger always in series with the pump. As soon as RHR injection flow initiates after depressurization the RHR heat exchanger is in the flow path and cooling the water. For a large break depressurization can occur in 3 to 5 minutes, at which time containment cooling begins as RHR injection starts. For the large break analysis, 10 minutes was conservatively assumed as the start of containment cooling.

The question mentioned the previous Grand Gulf design. Unlike the ABWR, the Grand Gulf design required operator action to perform valve alignment to bring the RHR heat exchanger into the flow path to initiate containment cooling.

The ABWR design requirement for core cooling is that the ECCS shall be completely automatic in operation (i.e., no operator action required) for at least 30 minutes following a LOCA.

QUESTION 430.27

Describe the design features of the suppression pool suction strainers. Specify the mesh size of the screens and the maximum particle size that could be drawn into the piping. Of the systems that receive water through the suppression pool suction strainers under post accident conditions, identify the system component that places the limiting requirements on the maximum size of debris that may be allowed to pass through the stainers and specify the limiting particle size that the component can circulate without impairing system performance. Discuss the potential for the strainers to become clogged with debris. Identify and discuss the kinds of debris that might be developed following a loss-of-coolant accident. Discuss the types of insulation used in the containment and describe the behavior of the insulation during and after a LOCA. Include in your discussion information regarding compliance with the acceptance criteria associated with USI A-43 as documented in NUREG-0897. (6.2)

RESPONSE 430.27

Response to this question will be provided by December 31, 1988.

QUESTION 430.28

Provide analyses of the net positive suction head (NPSH) available to the RHR pumps in accordance with the recommendations of Regulatory Guide 1.1. Compare the calculated values of available NPSH to the required NPSH of the pumps. (6.2)

RESPONSE 430.28

Response to this question is provided in revised Subsection 6.2.2.3.1 and new Table 6.2-2b.

QUESTION 430.29

In SSAR Section 6.2.3, identify clearly those areas that may not be part of the ABWR scope and provide relevant interface requirements.

RESPONSE 430.29

There are no secondary containment safety-related interfaces for the ABWR Standard Plant. This will be reflected in Section 1.9.

QUESTION 430.30

Provide a tabulation of the design and performance data for the secondary containment structure. Provide the types of information indicated in Table 6-17 of Regulatory Guide 1.70, Revision 3. (6.2)

RESPONSE 430.30

Response to this question will be provided by December 31, 1988.

QUESTION 430.31

Describe the valve isolation features used in support of the secondary containment. Specify the plant protection system signals that isolate the secondary containment and activate the standby gas treatment system. (6.2)

RESPONSE 430.31

Response to this question will be provided by November 11, 1988.

QUESTION 430.32

Identify and tabulate by size, piping which is not provided with isolation features. Provide an analysis to demonstrate the capability of the Standby Gas Treatment System to maintain the design negative pressure following a design basis accident with all non isolated lines open and the event of the worst single failure of a secondary containment isolation valve to close. (6.2)

RESPONSE 430.32

Response to this question will be provided by December 31, 1988.

QUESTION 430.33

Discuss the design provisions that prevent primary containment leakage from bypassing the secondary containment standby gas treatment system and escaping directly to the environment. Include a tabulation of potential bypass leakage paths, including the types of information indicated in Table 6-18 of Regulatory Guide 1.70, Revision 3. Provide an evaluation of potential bypass leakage paths considering equipment design limitations and test sensitivities. Specify and justify the maximum allowable fraction of primary containment leakage that may bypass the secondary containment structure. The guidelines of BTP 6-3 should be addressed in considering potential bypass leakage paths. (6.2)

RESPONSE 430.33

Response to this question will be provided by November 11, 1988.

QUESTION 430.34

Provide a list of the secondary containment openings and the instrumentation means by which each is assured to be closed during a postulated design basis accident. (6.2)

RESPONSE 430.34

Response to this question will be provided by November 11, 1988.

QUESTION 430.35

Provide a table of design information regarding the containment isolation provisions for fluid system lines and fluid instrument lines penetrating the containment which are within the GE scope of the ABWR design. Include as a minimum the following information:

(1) General design criteria or regulatory guide recommendations that have been met or other defined bases for acceptability;

(2) System name;

(3) Fluid contained;

(4) Line size;

(5) ESF system (yes or no);

(6) Through-line leakage classification;

(7) Reference to figure in SSAR showing arrangement of containment isolation barriers;

(8) Location of valve (inside/outside containment);

(9) Type C leakage test (yes or no);

(10) Valve type and operator;

(11) Primary mode of valve actuation;

(12) Secondary mode of valve actuation;

(13) Normal valve position;

(14) Shutdown valve position;

(15) Post accident valve position;

(16) Power failure valve position;

(17) Containment isolation signals;

(18) Valve closure time; and

(19) Power source. (6.2)

RESPONSE 430.35

Response to this question will be provided by November 11,1988.

QUESTION 430.36

For isolation valve design in systems not within the ABWR scope, identify the systems and the relevant interface requirements. Include a discussion on essential and non-essential systems per Regulatory Guide 1.131 and the means or criteria provided to automatically isolate the nonessential systems by a containment isolation signal. Also, include a discussion on the requirement that the setpoint pressure which initiates containment isolation for nonessential penetrations be reduced to the minimum value compatible with normal operations. (6.2)

RESPONSE 430.36

Response to this question will be provided by November 11, 1988.

QUESTION 430.37

Specify all plant protection signals that initiate closure of the containment isolation valves. (6.2)

RESPONSE 430.37

Response to this question will be provided by November 11, 1988.

QUESTION 430.38

Describe the leakage detection means provided to identify leakage for the outside-containment remote-manual isolation valves on the following influent lines: Feedwater, RHR injection, HPCS, standby liquid control, RWCU connecting to feedwater line, RWCU reactor vessel head spray. (6.2)

RESPONSE 430.38

Response to this question will be provided by November 11, 1988.

QUESTION 430.39

The containment isolation design provisions for the recirculation pump seal water purge line do not meet the explicit requirements of GDC 55 nor does the design satisfy the GDC on some other defined basis as outlined in SRP Section 6.2.4. It is our position that the isolation design in the instance is inadequate and should be modified to satisfy GDC 55 either explicitly or on some other defined basis, with the appropriate justification. (6.2)

RESPONSE 430.39

The ABWR RIP purge lines penetrating the primary containment are currently equipped with one check valve each inside and outside containment and are currently 15 mm (1/2 inch) size. This size is less than the ABWR instrument line size of 20 mm (3/4 inch). Therfore, the same design criteria (GDC 55, Reg. Guide 1.11, and SRP 6.2.4) apply to the RIP purge line design.

Paragraph C.1b(2) of Regulatory Guide 1.11 coolant loss must be within the reactor coolant makeup system capability. The ABWR RCIC system provides <u>normal</u> reactor coolant makeup and is capable of "making up" coolant to the reactor with a nominal 1" diameter broken pipe discharging reactor coolant. Therefore, due to the small 1/2" size of the RIP purge lines, the current containment isolation valve configuration is in accordance with current NRC requirements.

QUESTION 430.40

With respect to Figure 6.2-38a

(a) Include the isolation valve arrangement of the standby liquid control system line.

(b) Identify the line labeled in the figure as "WDCS-A" (it joins the RWCU line prior to its connection to the feedwater line), and discuss the isolation provisions for that line.

RESPONSE 430.40

Response to this question will be provided by November 11,1988.

QUESTION 430.41

Provide a diagram or reference to figure(s) showing the isolation valve arrangement for the lines identified below. For the isolation valve design of each of these lines, provide justification for not meeting the explicit requirements of GDC 56, and demonstrate that the guidelines for acceptable alternate containment isolation provisions contained in SRP 6.2.4 are satisfied. The lines in question are:

- o HPCS and RHR test and pump miniflow bypass
- o RCIC pump miniflow bypass line
- o RCIC turbine exhaust and pump miniflow bypass lines
- o SPCU suction and discharge lines (6.2)

RESPONSE 430.41

Response to this question will be provided by November 11, 1988.

QUESTION 430.42

Describe the isolation provisions for the containment purge supply and exhaust lines and discuss design conformance with Branch Technical Position CSB 6-4, "Containment Purge During Normal Operations."

RESPONSE 430.42

The containment purge supply and exhaust lines connect to both the drywell and the wetwell. There is one purge supply penetration for the drywell and one purge supply penetration for the wetwell. Similarly, there is one exhaust penetration through each the drywell and wetwell. The purge supply line connection to each or both of the drywell and wetwell has two inboard isolation valves in parallel, located outside of but as close as possible to the primary containment. One of these valves is intended for use for (high volume) inerting and purge. The other, a two-inch valve, is used for any necessary nitrogen makeup during power operation. The outboard isolation valves are located in each of the lines for purge supply, nitrogen inerting, and nitrogen makeup. The exhaust line has a similar parallel arrangement for the two valves located nearest to the wetwell penetration and the two valves located nearest the drywell penetration. Outboard isolation valves are located in each of the lines to the plant vent and the standby gas treatment system. All isolation valves are air operated and fail in the closed position, the signal sent from the leak detection and isolation system.

As described above, these isolation valves are in conformance to the supplemental guidance of Branch Technical Position CSB6-4 on containment purge during normal operation.

QUESTION 430.43

Discuss the closure times of isolation valves in system lines that can provide an open path from the primary containment to the environment (e.g., containment purge system). Also discuss provisions of radiation monitors in these lines having the capability of actuating containment isolation. (6.2)

RESPONSE 430.43

Response to this question will be provided by Decemb

QUESTION 430.44

Identify the system lines whose containment iso' ation requirements are covered by GDC 57 and discuss conformance of the design to the GDC requirements. (6.2)

RESPONSE 430.44

Response to this question will be provided by November 11, 1988.

QUESTION 430.45

For the combustible gas control systems design, identify clearly those areas that may not be part of the ABWR scope and provide relevant interface requirements. (6.2)

RESPONSE 430.45

There are no combustible gas control system safety-related interfaces for the ABWR Standard Plant. This will be reflected in Section 1.9.

QUESTION 430.46

According to SRP 6.2.5 specific acceptance criteria related to the concentration of hydrogen or oxygen in the containment atmosphere among others are the following:

(a) The analysis of hydrogen and oxygen production should be based on the parameters listed in Table 1 of Regulatory Guide 1.7 for the purpose of establishing the design basis for combustible control systems.

(b) The fission product decay energy used in the calculation of hydrogen and oxygen production from radiolysis should be equal to or more conservative than the decay energy model given in Branch Technical Position ASB9-2 in SRP 9.2.5.

Provide justification that the assumptions used in the ABWR in establishing the design basis for the combustible gas control systems are conservative with respect to the criteria a. and b. above. (6.2)

RESPONSE 430.46

Response to this question will be provided by December 31, 1988.

QUESTION 430.47

Provide an analysis of the production and accumulation of combustible gases within the containment following a postulated loss-of-coolant accident including all applicable information specified in Section 6.2.5.3 of Regulatory Guide 1.70, Revision 3.

RESPONSE 430.47

Response to this question will be provided by December 31, 1988.

QUESTION 430.48

Regarding Containment Type A leakage testing, (6.2.6)

QUESTION 430.48a

Provide the values for P_a and P_t.

RESPONSE 430.48a

P_a approximately 40 psig and $0.5\ P_a < P_t < P_a$.

QUESTION 430.48b

Include the acceptance criterion for L_t during preoperational leakage rate tests, i.e., L_t - $L_a\ (L_{tm}/L_{am})$, for the case when $L_a\ (L_{tm}/L_{am}) = 0.7$.

RESPONSE 430.48b

Response to this question is provided in revised Subsection 6.2.6.1.1.5.

QUESTION 430.48c

Your acceptance criterion for L_{tm} (SSAR Subsection 6.2.5.1.2.2, Item 1) is at variance with the staff's current practice for acceptance of L_{tm}. Also, it does not comply with the 10 CFR Part 50, Appendix J, Section III, Item A.1.(a) requirement. Therefore, either provide sufficient supporting justification for the exemption from compliance with the above requirement or correct the criterion as appropriate to comply with the requirement. Also, correct the stated acceptance criterion (SSAR Subsection 6.2.6.1.2.2, Item 3) as appropriate to comply with Appendix J, Section III, Item A.5.(b) requirement.

RESPONSE 430.48c

Response to this question is provided in revised Subsection 6.2.6.1.2.2.

QUESTION 430.48d

Regarding ILRT, identify the systems that will not be vented or drained and provide reasons for the same.

RESPONSE 430.48d

Response to this question will be provided by December 31, 1988.

QUESTION 430.48e

Provide F&IDs and process flow drawings for systems that will be vented or drained.

RESPONSE 430.48e

Response to this question will be provided by December 31, 1988.

QUESTION 430.49

Regarding Type B tests, (6.2.6)

QUESTION 430.49a

Clarify how air locks opened during periods when containment integrity is required by plant's Technical Specifications will be tested to comply with Appendix J, Section III, Item D.2.(b).(iii).

RESPONSE 430.49a

Response to this question is provided in revised Subsection 6.2.6.2.3.

QUESTION 430.49b

Provide the frequency for periodic tests of air locks and associated inflatable seals.

RESPONSE 430.49b

Response to this question is provided by revised Subsection 6.2.6.2.3.

QUESTION 430.49c

Provide the acceptance criteria for air lock testing and the associated inflatable seal testing.

RESPONSE 430.49c

Response to this question will be provided by December 31, 1988.

QUESTION 430.49d

List all containment penetrations subject to Type B tests.

RESPONSE 430.49d

Response to this question will be provided by December 31, 1988.

QUESTION 430.49e

List all those penetrations to be excluded from Type B testing and the rationale for excluding them.

RESPONSE 430.49e

Response to this question will be provided by December 31, 1988.

QUESTION 430.50

Regarding Type C tests (6.2.6)

(a) Correct the statement (Subsection 6.2.6.3.1, Paragraph 1) as appropriate to ensure that the hydraulic Type C tests are performed only on those isolation valves that are qualified for such tests per Appendix J. The current statement implies that these tests are not necessarily restricted to the valves that qualify for such tests.

(b) List all the primary containment isolation valves subject to Type C tests and provide the necessary P&IDs.

(c) Provide the list of valves that you propose to test in the reverse direction and justification for such testing for each of these valves.

(d) Identify the valves that you propose to test hydrostatically based on their ability to maintain a 30-day water leg seal. Also, identify other valves which you propose to test hydrostatically and provide the basis for such tests. Provide the test pressure for all the valves mentioned above.

(e) Indicate test pressures for MSIVs (with justification if it is less than P_a) and isolation valves sealed from a sealing system.

(f) Indicate how you will perform Type C leak tests for ECCS systems and RCIC system isolation valves.

(g) Confirm that the interval between two consecutive periodic Type C tests will not exceed 2 years as required by Appendix J.

(h) State what testing procedures you will follow regarding the valves that are not covered by Appendix J requirements.

RESPONSE 430.50

Response to this question will be provided by December 31, 1988.

QUESTION 430.51

Identify the reporting requirements for the tests. Note that your response should address compliance with requirements in this regard as stated in Appendix J, Sections III.A(a), IV.A and V. (For example, regarding follow up tests after containment modification, you have not included Type C testing for affected areas). (6.2.6)

RESPONSE 430.51

Response to this question is provided in revised Subsection 6.2.6.4.

QUESTION 430.52

Regarding Secondary Containment, (6.2.6)

(a) Identify the special testing procedures you will follow to assure a maximum allowable in leakage of 50 percent of the secondary containment free volume per day at a differential pressure of -0.25" water gauge with respect to the outdoor atmosphere (see Section 6.5.1.3.2).

(b) Identify all potential leak paths which bypass the secondary containment. (For such identification, see (BTP) CSB 6-3, *Determination of Bypass Leakage Paths in Dual Containment Plants*).

(c) Identify the total rate of secondary containment bypass leakage to the environment.

RESPONSE 430.52

Response to this question will be provided by December 31, 1988.

QUESTION 430.53

Identify all the interface requirements relating to containment leak testing. (6.2.6)

RESPONSE 430.53

There are no containment leak testing safety-related interfaces for the ABWR Standard Plant. This will be reflected in Section 1.9.

QUESTION 430.54

Regarding Control Room Habitability systems, (6.4)

(a) Provide the minimum positive pressure at which the control building envelope (which includes the mechanical equipment room) will be maintained with respect to the surrounding air spaces when makeup air is supplied to the envelope at the design basis rate (295 CFM).

(b) Provide the periodicity for verification of control room pressurization with design flow rate of makeup air.

(c) Clarify whether all the potential leak paths (to be provided in Section 9.4.1) include dampers or valves upstream of recirculation fans.

(d) Identify the action to be taken when there is no flow of the equipment room return fan and consequently the equipment room is over pressurized (Table 6.4-1 contains no information on the above).

(e) Provide the actual minimum distances (lateral and vertical) of the control room ventilation inlets from major potential plant release points that have been used in your control room dose analysis. Also, provide a schematic of the location of control room intake vents.

(f) Provide Figure 6.4-5 (plan view) which you state shows the release points (SGTS vent).

(g) Section 6.4.2.4 and Figure 6.4-1 indicate only one air inlet for supplying makeup air to the emergency zone. However, Tables 6.4-2 and 15.6-8 and Section 15.6.5.5.2 indicate that there are two automatic air inlets for the emergency zone. Correct the above discrepancy as appropriate. Also describe the characteristics of these inlets with respect to their relative locations and automatic selection control features. State how both flow and isolation in each inlet assuming single active component failure will be ensured.

(h) Describe the design features for protecting against confined area releases (e.g., multiple barriers, air flow patterns in ventilation zones adjacent to the emergency zone).

(i) Describe the specific features for protecting the control room operator from airborne radioactivity outside the control room and direct shine from all radiation sources (e.g., shielding thickness for control room structure boundary, two-door vestibules).

(j) Clarify what you mean by "sustained occupancy" (see SSAR Section 6.4.1.1, Item 3) for 12 persons.

(k) Provide justification for not specifying any unfiltered infiltration of contaminated air into the control room in SSAR Table 15.6-8.

QUESTION 430.53

Identify all the interface requirements relating to containment leak testing. (6.2.6)

RESPONSE 430.53

There are no containment leak testing safety-related interfaces for the ABWR Standard Plant.

QUESTION 430.54

Regarding Control Room Habitability systems, (6.4)

(a) Provide the minimum positive pressure at which the control building envelope (which includes the mechanical equipment room) will be maintained with respect to the surrounding air spaces when makeup air is supplied to the envelope at the design basis rate (295 CFM).

(b) Provide the periodicity for verification of control room pressurization with design flow rate of makeup air.

(c) Clarify whether all the potential leak paths (to be provided in Section 9.4.1) include dampers or valves upstream of recirculation fans.

(d) Identify the action to be taken when there is no flow of the equipment room return fan and consequently the equipment room is over pressurized (Table 6.4-1 contains no information on the above).

(e) Provide the actual minimum distances (lateral and vertical) of the control room ventilation inlets from major potential plant release points that have been used in your control room dose analysis. Also, provide a schematic of the location of control room intake vents.

(f) Provide Figure 6.4-5 (plan view) which you state shows the release points (SGTS vent).

(g) Section 6.4.2.4 and Figure 6.4-1 indicate only one air inlet for supplying makeup air to the emergency zone. However, Tables 6.4-2 and 15.6-8 and Section 15.6.5.5.2 indicate that there are two automatic air inlets for the emergency zone. Correct the above discrepancy as appropriate. Also describe the characteristics of these inlets with respect to their relative locations and automatic selection control features. State how both flow and isolation in each inlet assuming single active component failure will be ensured.

(h) Describe the design features for protecting against confined area releases (e.g., multiple barriers, air flow patterns in ventilation zones adjacent to the emergency zone).

(i) Describe the specific features for protecting the control room operator from airborne radioactivity outside the control room and direct shine from all radiation sources (e.g., shielding thickness for control room structure boundary, two-door vestibules).

(j) Clarify what you man by "sustained occupancy" (see SSAR Section 6.4.1.1, Item 3) for 12 persons.

(k) Provide justification for not specifying any unfiltered infiltration of contaminated air into the control room in SSAR Table 15.6-8.

(l) Provide Subsection 6.3.1.1.6 which you state (SSAR Section 6.4.6) contains a complete description of the required instrumentation for ensuring control room habitability at all times.

(m) Give schematics for control room emergency mode of operation during a postulated LOCA (this is required for calculating control room LOCA doses).

(n) The source terms and control room atmospheric dispersion factors (X/Q values) used in the control room dose analysis (see SSAR Tables 15.6-8 and 15.6-12) to demonstrate ABWR control room compliance with GDC 19 are non-conservative. Therefore, reevaluate control room doses during a postulated LOCA using RG 1.3 source terms and assumptions and the methodology given in Reference 4 of SSAR Section 15.6.7. Include possible dose contributions from containment shine, ESF filters and airborne radioactivity outside the control room. Also check and correct as appropriate the recirculation rate in the control room (22.4 M^3/sec) given in Table 15.6-8.

(o) Section 6.4.7.1, "External Temperature," provides design maximum external temperatures of 100 F and -10 F. How are these values used in the design and assessments related to the ABWR? What factors, such as insulation, heat generation from control room personnel and equipment and heat losses, are taken into account? Do these values represent "instantaneous" values or are they temperal and/or spatial averages?

(p) Clarify your position on potential hazardous or toxic gas sources onsite of an ABWR. If applicable, indicate the special features provided in the ABWR design in this regard, to ensure control room habitability.

(q) Identify <u>all</u> the interface requirements for control room habitability systems (e.g., instrumentation for protection against toxic gases in general and chlorine in particular; potential toxic gas release points in the environs).

RESPONSE 430.54

Response to this question will be provided by December 31, 1988.

QUESTION 430.55

Regarding ESF Atmosphere Cleanup Systems, (6.5.1)

(a) Provide a table listing the compliance status of the Standby Gas Treatment System (SGTS) with <u>each</u> of the regulatory positions specified under C of RG 1.52. Provide justifications for each of those items that do not fully comply with the corresponding requirements. In this context, you may note that the lack of redundancy of the SGTS filter train (the staff considers that filter trains are also active components - See SRP 6.4, Acceptance Criterion II.2.b) is not acceptable. Further, the described sizing of the charcoal adsorbers based on assumed decontamination factors for various chemical forms of iodine in the suppression pool is not acceptable (RG 1.3 assumes a decontamination factor of 1 for all forms of iodine and RG 1.52 requires compliance with the above guide for the design of the adsorber section). Therefore, revise charcoal weight and charcoal iodine loading given in SSAR Table 6.5-1 as appropriate.

(b) Specify the laboratory test criteria for methyl iodine penetration that will be identified as an interface requirement to be qualified for the adsorber efficiencies for iodine given in SSAR Table 15.6-8. Also, provide the depth of the charcoal beds for the control room emergency system.

(c) Provide a table listing the compliance status of the instrumentation provided for the SGTS for read out, recording and alarm provisions in the control room with each of the instrumentation items identified in Table 6.5.1-1 of SRP 6.5.1. For partial or non-compliance items, provide justifications.

(d) Clarify whether primary containment purging during normal plant operation when required to limit the discharge of contaminants to the environment will always be through the SGTS (See SSAR Section 6.5.1.2.3.3). Clarify whether such a release prior to the purge system isolation has been considered in the LOCA dose analysis.

(e) Provide the compliance status tables referred to in Items (a) and (c) above for the control room ESF filter trains. (The staff notes that you have committed to discuss control room habitability system cannot be complete until the information identified above is provided. the above information is requested now.)

(f) Identify the applicable interface requirements for the SGTS and the control room ESF atmosphere cleanup system.

RESPONSE 430.55

Response to this question will be provided by November 11, 1988.

QUESTION 430.56

Regarding Fission Product Control Systems and Structures, (6.5.3)

(a) Provide the drawdown time for achieving a negative pressure of 0.25 inch water gauge for the secondary containment with respect to the environs during SGTS operation. Clarify whether the unfiltered release of radioactivity to the environs during this time for postulated LOCA has been considered in the LOCA dose analysis. (Note that the unfiltered release need not be considered provided the required negative pressure differential is achieved within 60 seconds from the time of the accident).

(b) Provide justification (See SRP Section 6.5.3, II.4) for the decontamination factor assumed in SSAR Tables 6.5-2 and 15.6-8 for iodine in the suppression pool, correct the elemental, particulate and organic iodine fractions given in the tables to be consistent with RG 1.3, and incorporate the correction in the LOCA analysis tables. Alternatively, taking no credit for iodine retention in the suppression pool, revise the LOCA analysis tables. Note that the revision of the LOCA analysis tables (this also includes the control room doses) mentioned above is strictly in relation to the iodine retention factor in the suppression pool (also, there may be need for revision of other parameter(s) given in the tables and these will be identified under the relevant SRP Sections questions).

(c) Identify the applicable interface requirements.

RESPONSE 430.56

Response to this question will be provided by November 11, 1988.

QUESTION 430.57

Regarding SSAR Section 6.7, the staff notes that the Nitrogen Supply System has been discussed under this section, instead of the Main Steam Isolation Valve Leakage Control System (MSIV-LCS) as required by the Standard Format for SARS. The staff will review the material presented in SSAR Section 6.7 along with the material that will be presented in SSAR Section 9.3.1.

Regarding MSIV-LCS, the staff notes that you are committed to provide a non-safety related MSIV leakage processing pathway consistent with those evaluated in NUREG-1169, "Resolution of Generic Issue C-8," August 1986. Since the staff has not finalized its position so far on the acceptability of the NUREG findings with regard to the design of the MSIV-LCS,, provide pertinent information on the system design including interface requirements to evaluate the to-be-proposed design against the acceptance criteria of SRP 6.7. (6.7)

RESPONSE 430.57

In accordance with Section 8.9 of the GE ABWR Licensing Review Bases (Murley to Artigas dated August 7, 1987) GE committed to a design that provides a non-safety related main steam isolation valve (MSIV) leakage process pathway consistent with those evaluated in NUREG-1169. Accordingly, the drains and vents are routed to the main condenser for leakage control to take advantage of fission product plateout and holdup in the main steam line, drain line, and the main condenser. Fission products are removed by plateout on the relatively cool condenser tubes. The earlier BWR designs, where the fission products are routed through the reactor building to the standby gas treatment system, had the disadvantage of increasing the dose rate to plant personnel. In addition there was no holdup or removal of noble gases, so that dose rate to the public may be higher.

The earlier BWR designs also had the disadvantage of being ineffective if the MSIVs greatly exceeded the design leak rate (typically 11.5 standard cubic feet an hour). Because of no uncovery in the ABWR design, the ABWR would have less fission product generation during a postulated loss-of-coolant accident than earlier BWR designs. As a consequence the ABWR design is better able to handle leakages beyond the technical specification limits.

The ABWR design is also passive requiring no operator actions. The valves on the drain lines open automatically when the reactor is at less than 40 percent steam flow to vent to the main condenser. In addition, the valves fail open on loss of air or electrical power to ensure that this pathway exists during an accident. These valves and drain lines are illustrated in the Nuclear Boiler system P&ID (Figures 5.1-3).

In conclusion, the ABWR design provides a passive non-safety related means for controlling and mitigating the release of fission product leakage through the MSIVs and meets the GE ABWR Licensing Review Bases.

QUESTION 430.58

The accident analyzed under this section considers only the airborne radioactivity that may be released due to potential failure of a concentrated waste tank in the radwaste enclosure. The SRP acceptance criteria, however, requires demonstration that the liquid radwaste concentration at the nearest potable water supply in an unrestricted area resulting from transport of the liquid radwaste to the unrestricted area does not exceed the radionuclide concentration limits specified in 10 CFR Part 20, Appendix B Table II, Column 2. Such a demonstration will require information on possible dilution and/or decay during transit which, in turn, will depend upon site specific data such as surface and ground water hydrology and the parameters governing liquid waste movement through the soil. Additionally, special design features (e.g., steel liners or walls in the radwaste enclosure) may be provided as part of the liquid radwaste treatment systems at certain sites. The staff will, therefore, review the site specific characteristics mentioned above individually for each plant referencing the ABWR and confine its review of ABWR, only to the choice of the liquid radwaste tank. Therefore, provide information on the following: (15.7.3)

(a) Basis for determining the concentrated waste tank as the worst tank (this may very well be the case, but in the absence of information on the capacities of major tanks, particularly the waste holdup tanks, it is hard to conclude that the above tank both in terms of radionuclide concentrations and inventories will turn out to be the worst tank).

(b) Radionuclide source terms, particularly for the long-lived radionuclides such as Cs-137 and Sr-90 (these may be the critical isotopes for sites that can claim only decay during transit) in the major liquid radwaste tanks.

RESPONSE 430.58

Response to this question will be provided by December 31, 1988.

QUESTION 440.1

SRP 4.6 identifies the following GDCs 23, 25, 26, 27, 28, and 29 in the acceptance criteria. Confirm that the reactivity system, described in Section 4.6 of the SSAR, meet the requirements of the above GDCs.

RESPONSE 440.1

Section 4.6 has been revised to reference the evaluation of the reactivity system against the requirements of the above GDCs contained in Subsection 3.1.2.

QUESTION 440.2

In Section 4.6.2.3.2.2 analysis of malfunction relating to rod withdrawal, it is stated, "There are known single malfunctions that cause the unplanned withdrawal of even a single control rod." Confirm that this is a editorial mistake and correct it if so. Otherwise, explain in detail the basis for this statement and why this is acceptable.

RESPONSE 440.2

This editorial error has been corrected in Subsection 4.6.2.3.2.2.

QUESTION 440.3

In Section 4.6.1.2 it is stated that CRD system in conjunction with CRC&IS and RPS systems provides selected control rod run in (SCRRI) for reactor stability control. Describe in detail how SCRRI works.

RESPONSE 440.3

Response to this question is provided in revised Subsections 4.6.1.2(10) and 7.7.1.2.2(2).

QUESTION 430.

In Figure 4.6-8a, CRD system P&ID, sheet 1, piping quality classes AA-D, FC-D, FD-D, FD-B, etc. are shown. Submit the document which explains these classes and relates them to ASME code classes.

RESPONSE 440.4

This information is scheduled to be included in Section 1.7. Essentially, the first two letters of the codes specify the pipe primary pressure rating (150 lb., 900 lb., etc.) the type of service (condensate or reactor water, steam, etc.), and material (carbon or stainless steel). The symbols "A", "B" and "C" represent ASME Section III, code Classes 1, 2, and 3, respectively. The symbol "D" represents ASME Section 8, or ANSI B31.1 or other equivalent codes.

QUESTION 440.5

in Figure 4.6-8b, the leak receiver tank is shown. What is the function of this tank? How big is this tank? Will a high level in the tank impact the operation of the control rod drive?

RESPONSE 440.5

This leakage collection tank is no longer part of the design. The intent of the leakage collection system was to assist the operator in identifying which drives were potential candidates for seal replacement during plant outages, which would facilitate plant maintenance planning. However, the design could not provide the level of differentiation of leakage between individual drives needed for this purpose and was therefore deleted. An updated P&ID (Figure 4.6-8b) will be provided by December 31, 1988 to document this change.

QUESTION 440.6

Identify the essential portions of the CRD system which are safety related. Confirm that the safety related portions are isolable from non-essential portions. (4.6)

RESPONSE 440.6

The essential portions of the CRD system which are safety-related are:

(a) The hydraulic control units (HCUs),

(b) The scram insert piping from the HCUs to the fine motion control rod drives (FMCRDs), and

(c) The FMCRDs (except the motors)

The non-essential portions of the CRD system interface with the essential portions at the following connections to the HCUs:

(a) The accumulator charging water line

(b) The FMCRD purge water line, and

(c) The scram valve air supply from the scram air header.

The safety-related portions of the HCU and the scram function are protected against failure in the non-essential portions of the charging water and purge water lines by check valves. Also, instrumentation in the charging water line provides signals to the reactor protection system to cause reactor scram in the event of loss of charging water pressure. Loss of pressure in the scram air header causes the scram valves to actuate, resulting in reactor scram. This fail-safe feature is the same as provided on current BWR designs using locking piston-type control rod drive.

QUESTION 440.7

In the old CRD system, the major function of the cooling water was to cool the drive mechanism and its seals to preclude damage resulting from long term exposure to reactor temperatures. What is the function of purge water flow to the drives? (4.6)

RESPONSE 440.7

The function of the purge water flow to the fine motion control rod drives is to prevent reactor water from entering the drive housing during operation. This will minimize crud buildup in the drive housing and reduce operator exposure during drive maintenance.

QUESTION 440.8

We understand that the LaSalle Unit 2 fine motion control rod drive demonstration test is still in progress. Submit the test results as soon as it is available.

RESPONSE 440.8

At the current time, the LaSalle Unit 2 fine motion control rod drive demonstration test is expected to be terminated in October 1988. The final report for the FMCRD In-Plant Test Program, which will include the LaSalle Test results, will be formally issued in September 1989.

QUESTION 440.9

In the present CRD system design, the ball check valve ensures rod insertion in the event the accumulator is not charged or the inlet scram value fails to open if the reactor pressure is above 600 psig. Confirm that this capability still exists in the ABWR design. (4.6)

RESPONSE 440.9

The ABWR control rod design does not have the capability of the locking piston control rod design to insert hydraulically using reactor pressure in the event of a failure in the hydraulic control units (i.e., scram valve fails or accumulator is not charged). However, the fine motion control rod drive (FMCRD) has a diverse means of inserting the control rod using electric motor run-in if hydraulic scram fails. This feature provides the FMCRD with the capability to insert the control rod over the entire range of reactor operating pressures.

QUESTION 440.10

In section 4.6.2.3.1, it is stated the scram time is adequate as shown by the transient analyses of Chapter 15. Specify the scram time. (4.6.2.3.2.1)

RESPONSE 440.10

The average maximum scram time of all control rods in the core under the reactor conditions with accumulator available and reactor steady state pressure as measured at the vessel bottom below 76.3 Kg/cm^2g (1085) psig) shall meet the following requirements: (all times are after deenergizing of scram solenoids)

Insertion %	Time (seconds)
10	≤ 0.42
40	≤ 1.00
60	≤ 1.44
100	≤ 2.80

QUESTION 440.11

For both the low ("zero") and operating power region describe the patterns of the control rod groups that are expected to be withdrawn simultaneously with the new rod system, and estimate the maximum for the total and differential reactivity worth of these groups. What sort of margin to period scram will exist in the low power range. (4.6)

RESPONSE 440.11

(1) Summary of rod withdrawal strategy

The ABWR rod groups are assigned as shown in Figures 20.3-1 and 20.3-2. The FMCRD step size is 18.3 mm (0.5% of full CRD stroke), with a nominal speed of 30mm/sec. The number of rods per gang for rod groups #1, 2, 3, 4 is 26, i.e., the whole group of 26 rods will be moved simultaneously as one gang. Group 1 and 2 will be moved continuously from full in to full out. Group 3 and 4 which cover the rod pattern condition from cold critical to hot critical, will be moved in jog cycle in one step at a time. The peripheral rods of group 5 and 6 will be moved as one gang. For the remaining 7, 8, 9, 10 groups, rods are divided into 4-rod gangs and 8-rod gangs.

A BWR/6 type banked position withdrawal sequence (BPWS) constraint, called grouped wit' quence (GWS), is applied in ABWR as the rod withdrawal sequence guideline. It is in et. to the low power setpoint (LPSP), or 25% power. Above LPSP, the rod withdrawal sequence is based on core-management pre-developed rod withdrawal sequence in 4 and 8 rod gangs.

(2) Typical rod patterns at various power level

(a) Hot criticality after hot recovery (EOIC, rated condition Xe)

Rod pattern: Figure 20.3-3 (quarter core only, same for all Figure 20.3-4 thru 20.3-9)

Rod position of each group: Table 20.3-4

(b) 5% power*, Cold Startup, equilibrium Xe, BOEC

Rod pattern: Figure 20.3-4

(c) 10% power*, Cold Startup, equilibrium Xe, BOEC

Rod pattern: Figure 20.3-5

(d) 25% power*, Cold Startup, equilibrium Xe, BOEC

Rod pattern: Figure 20.3-6

(e) 40% power*, Cold Startup, equilibrium Xe, BOEC

Rod pattern: Figure 20.3-7

(f) 53% power*, Cold Startup, equilibrium Xe, BOEC

Rod pattern: Figure 20.3-8

(g) 100% power 100% flow, Cold Startup, equilibrium Xe, BOEC

Rod pattern: Figure 20.3-9

*minimum core flow

(3) Estimates of maximum reactivity worth

Reactivity Worth Estimates

Group		Whole Group	Max. Worth 1st Rod	Max. Worth 1st Gang
1		---		
2		---		
3		2.1%		
4		1.5%		
5		---		
6		---		
7, 8, 9	max	3.2%	≤ 1.2%	≤ 1.5%
10				

(4) Margin to period Scram estimates

For 3% total rod worth (full in to full out), the shortest period per step is ~60 seconds. For 2% total rod worth, the shortest period per step is ~100 seconds.

So, for step-wise withdrawal, there is plenty of margin to period scram (10 second scram setpoint)

QUESTION 440.12

Describe the relative core location of control rods sharing a scram accumulator. Can a failure of the scram accumulator fail to insert adjacent rods? If so, discuss the consequences of that failure. (4.6)

RESPONSE 440.12

The grouped HCU to control rod drive assignment and their relative core locations are shown in Figure 20.3-10. As can be seen, the two control rods sharing a scram accumulator are separated by several core cell locations. A failure of an HCU cannot result in the failure to insert adjacent rods.

TABLE 20.3-2

CORE DECAY HEAT[1] FOLLOWING LOCA SHORT-TERM ANALYSES

(Response to Question 430.21)

Time (sec)	Normalized Core Heat[2]
0	1.084
2	0.5566
6	0.5501
10	0.3859
20	0.1239
30	0.0772
31	0.0771
60	0.0472
100	0.0427
120	0.04
121	0.039
200	0.0358
600	0.0279
1000	0.0245

NOTES

(1) Based on 1973 ANS Standard with 20% margin.

(2) Normalized to 102% of rated thermal power.

TABLE 20.3-3

INTEGRATED CORE DECAY HEAT VALUES (1) SHORT-TERM ANALYSES

(Response to Question 430.21)

Time t_s (sec)	Integrated Decay Heat in Full Power Seconds (2)
0.0	0.0
0.1	0.1093
1	1.0172
2	1.7083
4	2.8408
6	4.0872
8	5.3096
10^1	6.4256
2	10.092
4	12.554
6	13.736
8	14.700
10^2	15.569
2	19.445
4	26.045
6	31.864
8	37.234
10^3	42.293

NOTES

(1) Based on 1973 ANS Standard with 20% margin.

(2) Full Power = 3.797×10^6 Btu/sec.

TABLE 20.3-4

HOT STARTUP CRITICALITY ROD SEQUENCE

(Response to Question 440.11)

Group #	Gang #	Rod Withdrawn To Notch Position
1	(1)	0 - 48 (12 ft)
2	(2)	0 - 48
3	(3)	0 - 48
4	(4)	0 - 48
5	(5)	0 - 18 (4.5 ft)
6	(6)	0 - 18
7	A (7)	0 - 12 (3 ft)
	B (8)	0 - 12
	C (9)	0 - 10 (2.5 ft)
	D (10)	0 - 12
8	A (11)	0 - 12
	B (12)	0 - 10
	C (13)	0 - 10
9	A (14)	0
	B (15)	0
	C (16)	0
10	A (17)	0
	B (18)	0
	C (19)	0
	D (20)	0
	E (21)	0

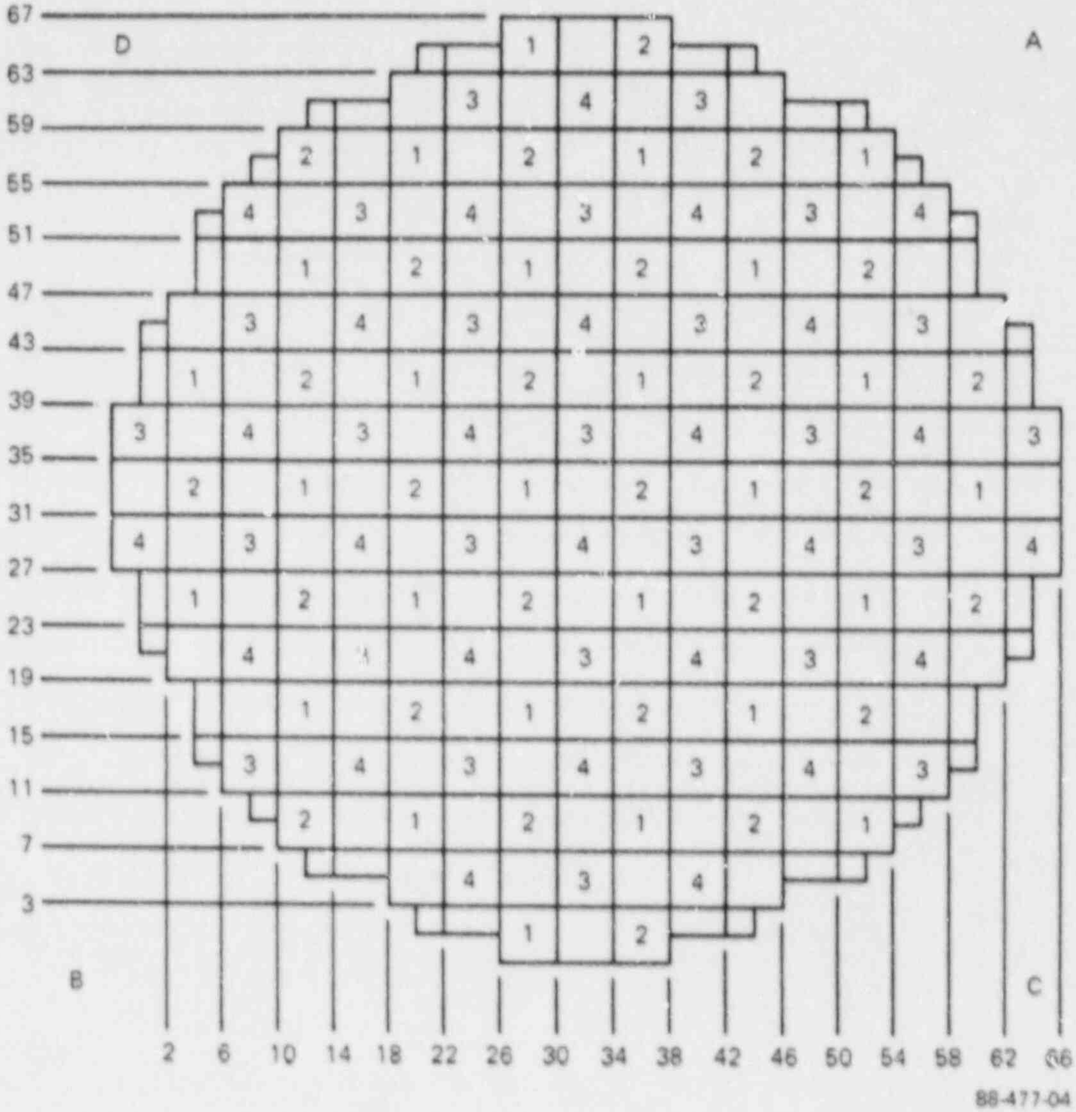

Figure 20.3-1 ROD GROUPS 1–4, SEQUENCE A
(Response to Question 440.11)

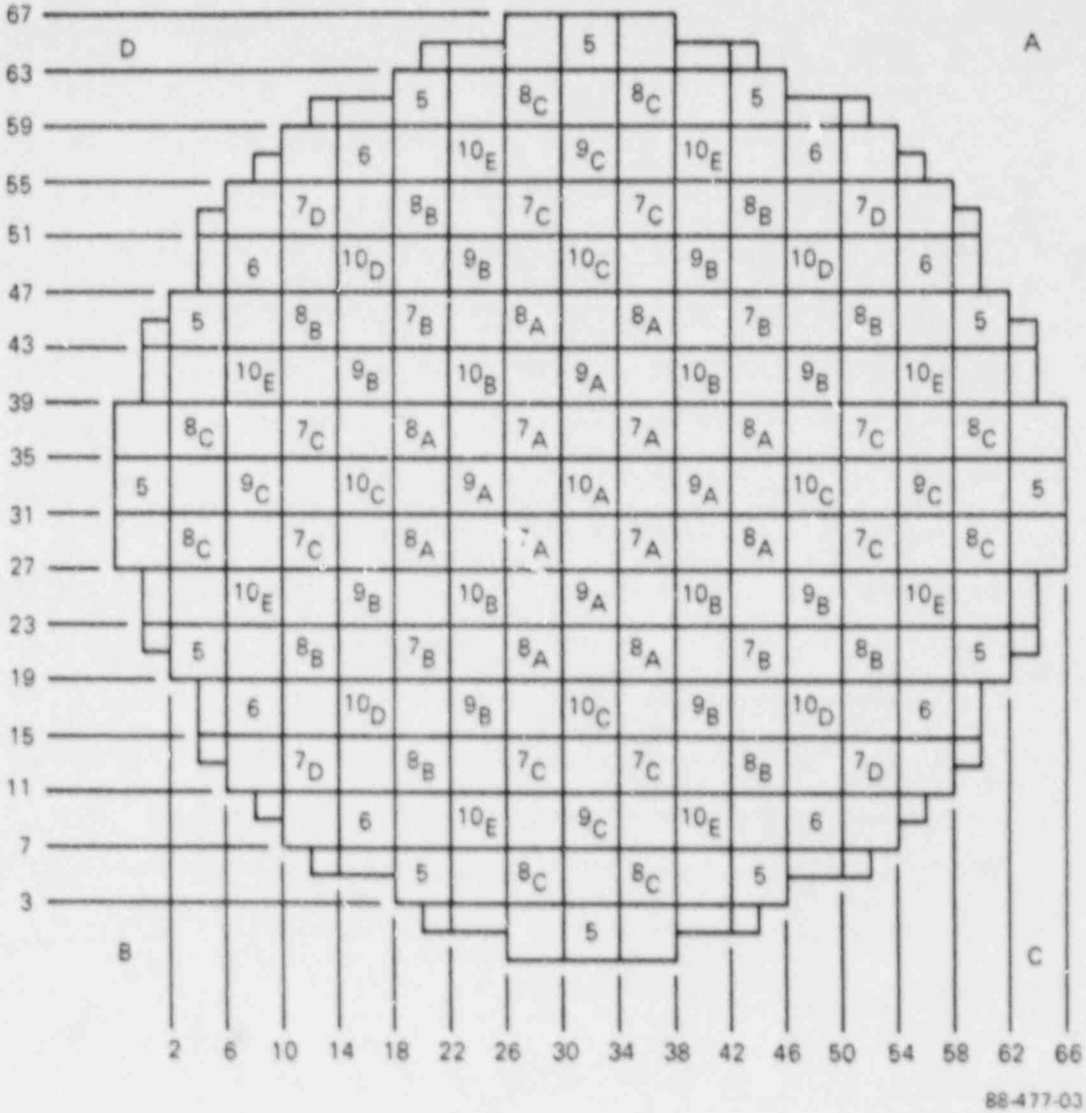

Figure 20.3-2 ROD GROUPS 5–10, SEQUENCE A
(Response to Question 440.11)

CONTROL ROD CONFIGURATION
IN NOTCHES WITHDRAWN

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	
67									18									1
63						18		10										3
59					18		0		0									5
55				12		10		10										7
51			18		0		0		0									9
47		18		10		12		12										11
43			0		0		0		0									13
39		10		10		12		12										15
35	18		0		0		0		0									17
31																		19

Figure 20.3-3 HOT RECOVERY CRITICALITY CONTROL
(Response to Question 440.11)

CONTROL ROD CONFIGURATION
IN NOTCHES WITHDRAWN

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	
67									0									1
63						0		8										3
59					0		0		0									5
55				6		8		6										7
51			0		0		0		0									9
47		0		8		6		8										11
43			0		0		0		0									13
39		8		6		8		6										15
35	0		0		0		0		0									17
31																		19

88-477-05

Figure 20.3-4 5% POWER CONTROL ROD PATTERN
(Response to Question 440.11)

CONTROL ROD CONFIGURATION
IN NOTCHES WITHDRAWN

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	
67									12									1
63						12		8										3
59					12		0		0									5
55				8		8		8										7
51			12		0		0		0									9
47		12		8		8		8										11
43			0		0		0		0									13
39		8		8		8		8										15
35	12		0		0		0		0									17
31																		19

Figure 20.3-5 10% POWER CONTROL ROD PATTERN
(Response to Question 440.11)

CONTROL ROD CONFIGURATION
IN NOTCHES WITHDRAWN

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	
67																		1
63								20										3
59							0		0									5
55				20		20		20										7
51					0		0		0									9
47				20		20		20										11
43			0		0		0		0									13
39		20		20		20		20										15
35			0		0		0		0									17
31																		19

88-477-01

Figure 20.3-6 25% POWER CONTROL ROD PATTERN
(Response to Question 440.11)

CONTROL ROD CONFIGURATION
IN NOTCHES WITHDRAWN

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	
67																		1
63																		3
59							8		10									5
55																		7
51					8		10		6									9
47																		11
43			8		10		6		10									13
39																		15
35			10		6		10		6									17
31																		19

Figure 20.3-7 40% POWER CONTROL ROD PATTERN
(Response to Question 440.11)

CONTROL ROD CONFIGURATION
IN NOTCHES WITHDRAWN

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	
67																		1
63																		3
59							10		22									5
55																		7
51					10		22		8									9
47																		11
43			10		22		10		22									13
39																		15
35			22		8		22		8									17
31																		19

88-477-06

Figure 20.3-8 53% POWER CONTROL ROD PATTERN
(Response to Question 440.11)

CONTROL ROD CONFIGURATION
IN NOTCHES WITHDRAWN

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	
67																		1
63																		3
59																		5
55																		7
51					8				10									9
47																		11
43							8											13
39																		15
35					10				8									17
31																		19

88-477-07

Figure 20.3-9 100% POWER CONTROL ROD PATTERN
(Response to Question 440.11)

DIVISION	HCU NO.	CR
A	1-25	50
B	26-51	51
C	52-77	52
D	78-103	52

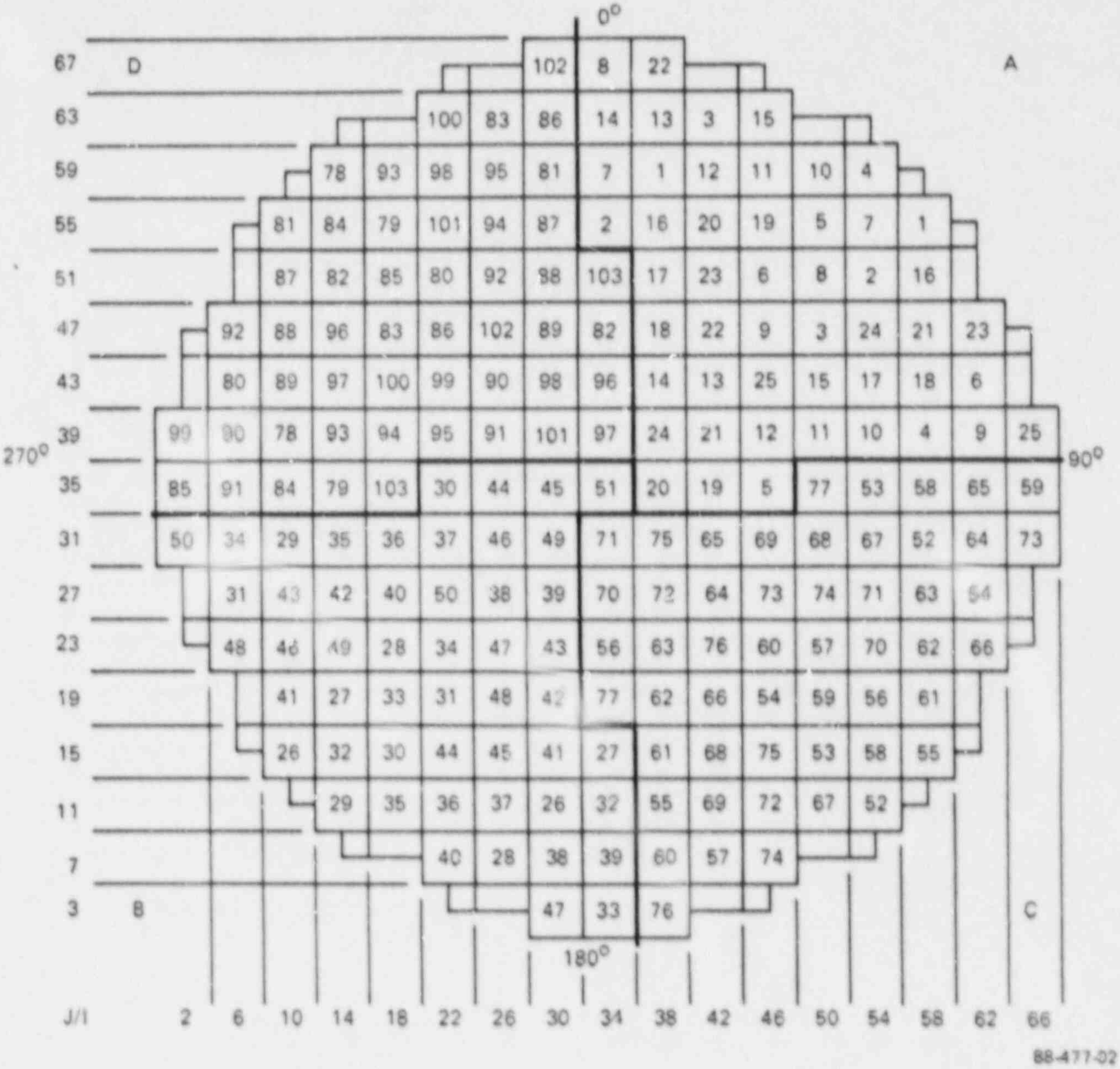

Figure 20.3-10 GROUPED HCU TO CONTROL ROD DRIVE ASSIGNMENTS
(Response to Question 440.12)

20.4 REFERENCES

1. Dino C. Scaletti to Ricardo Artigas, *Request for Additional Information Regarding the General Electric Company Application for Certification of the ABWR Design*, February 22, 1988.

2. Dino C. Scaletti to J.S. Gay, *Request for Additional Information Regarding the General Electric Company Application for Certification of the ABWR Design*, July 7, 1988.