CERTIFIED BY:
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SUMMARY/MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON THE ADVANCED BOILING WATER REACTORS (GE)

OCTOBER 26-27, 1993
BETHESDA, MARYLAND

PURPOSE

The purpose of this meeting was to review certain chapters of the NRC staff's final safety evaluation report (FSER) of the GE/ABWR design. The meeting began at 8:30 a.m. on October 26, 1993, and adjourned at 5:30 p.m., then it was reconvened at 8:30 a.m. on October 27, 1993, with a final adjournment at 4:00 p.m. The meeting was held entirely in open session. A copy of the meeting agenda is attached. Dr. Medhat El-Zeftawy was the Designated Federal Official (DFO) for the meeting. No written comments or requests for time to make oral statements were received from members of the public.

ATTENDEES: The principal attendees were as follows:

ACRS

C. Michelson, Chairman
P. Davis, Member
T. Kress, Member
W. Lindblad, Member
R. Seale, Member
W. Shack, Member
C. Wylie, Member
M. El-Zeftawy, Staff Engineer

GE

J. Power
J. Duncan
A. Beard
C. Buchholz
S. Visweswaran
C. Christensen

Others

N. Fletcher, DOE
K. Mali, DOE
C. Willbanks, NUS
H. Barbeito, Bechtel
R. Clark, EPA

NRC

J. Wilson, NRR
C. Poslusny, NRR
D. Thatcher, NRR
J. Knox, NRR
C. Thomas, NRR
H. Pastis, NRR
C. McCracken, NRR
J. Wigginton, NRR
R. Emich, NRR
R. Rothman, NRR
J. Lee, NRR
T. Cheng, NRR
G. Bagchi, NRR
D. Terao, NRR
H. Richings, NRR
L. Philips, NRR
T. Collins, NRR
C. Li, NRR
W. Burton, NRR
J. Raval, NRR
T. Chandrasekaran, NRR
H. Walker, NRR
K. Parczewski, NRR
R. Barrett, NRR
J. Monninger, NRR
R. Palla, NRR
G. Kelly, NRR
A. El-Bassioni, NRR

NRC (Cont'd).

M. Rubin, NRR
D. Scaletti, NRR
M. Snodderly, NRR
C. Hoxie, NRR
R. Pedersen, NRR
J. Lyons, NRR
R. Borchardt, NRR
B. Zalcman, NRR

CHAIRMAN'S OPENING REMARKS

In his opening remarks, Mr. Michelson stated that the purpose of this meeting is to review the first several chapters of the NRC staff final safety evaluation report (FSER) for the ABWR design that the ACRS has received. However, these FSER chapters are not approved and signed off by the NRC management.

Mr. Michelson stated that recently more changes to Amendment 32 were received, which indicated that there will be another amendment (No. 33). Mr. Michelson added that the current FSER chapters seem to be technically complete, however, they represent a laborious reading, at best.

NRC STAFF PRESENTATION

1. Status of FSER Items - C. Poslusny, NRR

Mr. Poslusny stated that the draft FSER identified over 600 open and confirmatory items. Currently 12 issues remain as confirmatory, are being worked by the staff and GE. Tier 1/ITAAC review effort identified over 800 concerns. Currently less than 20 ITAAC issues remain with agreed upon GE actions.

GE will provide SSAR/Amendment 33, Technical specifications, and Tier 1 submittal on November 22, 1993. The staff will verify that all issue resolutions are correctly and completely documented in GE's application by December 17, 1993. Based on evaluation, NRR will decide shortly on the release of FSER to the Commission.

The remaining FSER chapters that yet to be provided to the ACRS are:

- Chapter 1: Introduction and General Discussion
- Chapter 6.2.1.6: Suppression Pool Dynamic loads
- Chapter 9: Auxiliary Systems
- Chapter 14: Verification Programs
- Chapter 16: Technical Specifications
- Chapter 18.10: EPG Evaluations
- Chapter 19.3: Shutdown Risk
- Chapter 20: (USI/GSI/50.34(f) Evaluations)
- Appendix: (Human Factors Engineering Review Model)

2. FSER Chapter 8: Electrical Systems - Mr. J. Knox, NRR

Mr. Knox stated that the bases for evaluating the adequacy of the ABWR electrical power systems presented in the SSAR Chapter 8 were the acceptance criteria and guidelines for electric power systems contained in Standard Review Plan (SRP) Chapter 8 and Regulatory Guides (1.153 (Rev. O), "Criteria for Power, Instrumentation, and Control Portions of Safety Systems," and 1.155 (Rev. O), "Station Blackout."

Chapter 8 deals with the "offsite electric power system" and the onsite Class 1E power system."

The offsite electric power system is commonly called the "preferred" power system. The staff's evaluation of this system focused on the system's importance as the preferred supplier of electric power for the onsite power system (that is, the Class 1E ac-distribution system), which supplies power to safety systems.

For the ABWR, the preferred power system comprises the following circuits:

- normal preferred power circuit - a back-feed circuit from the transmission network to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems through the main transformer and three unit auxiliary transformers

- alternate preferred power circuit - from the transmission network through one reserve auxiliary transformer to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems

The following portions of the preferred power circuits are outside the scope of design of the ABWR standard plant:

- normal preferred power circuit from the transmission network through the main power transformer to the low-voltage terminals of the main transformer, and

- alternate preferred power circuit from the transmission network through the reserve auxiliary transformer to the low-voltage terminals of the reserve auxiliary transformer.

The following portions of the preferred power circuits are within the scope of design of the ABWR standard plant:

- preferred power circuit from the low-voltage terminals of the main transformer through the three unit auxiliary transformers to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems, and

- alternate preferred power circuit from the low-voltage terminals of the reserve auxiliary transformer to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems.

By its draft submittal to the NRC staff dated April 3 1992, GE revised the SSAR to indicate that the reserve auxiliary transformer will be separated from the unit auxiliary transformers by a minimum distance of 50 ft and that each transformer will be provided with an oil collection pit and drain to a safe disposal area. In addition, GE indicated that the reserve auxiliary transformer and its input feeders will be separated from the main power transformer and its input feeders and from the unit auxiliary transformer by a minimum of 50 ft.

The staff concluded that the 50 ft separation between the normal and alternate preferred power circuit transformers, together with each transformer's oil collection pit and automatic deluge water spray system, will minimize to the extent feasible the likelihood of simultaneous failure of both normal and alternate offsite preferred power circuits under operating and postulated accident and environmental conditions.

GE indicated that instrumentation and control cables that are affiliated with the normal and alternate preferred offsite circuits will be separated as follows:

- The instrumentation and control cables that are affiliated with the normal preferred offsite circuit will be routed in raceways corresponding to the load group of their power source.

- The instrumentation and control cables that are affiliated with the alternate preferred offsite circuit will be routed in dedicated raceways. The alternate preferred offsite instrumentation and control circuit cables will not share raceways with any other cables.

- The separation between the normal and alternate preferred offsite instrumentation and control cables will be the same as the separation between the normal and alternate preferred offsite power circuits (that is, floors, walls, or 50 ft of physical separation).

For electrical independence, GE indicated that there will be no electrical interconnections between the normal and alternate preferred power, instrumentation, and control circuits except where the power circuits connect to common Class 1E and non-Class 1E switchgear lineups. At the common switchgear, one open and one closed circuit breaker will maintain the electrical independence. These circuit breakers will be interlocked so that the closed breaker must be opened before the open breaker can be closed. Transfer from normal to alternate (or alternate to normal) preferred power circuits will be manual. Instrumentation and control circuits (including their power supply) that are affiliated with the normal preferred offsite circuit will be electrically independent from (that is, they will have no electrical interconnection with) the instrumentation and control circuits including their power supply affiliated with the alternate preferred power supply.

The staff concluded that these provisions for electrical independence of power, instrumentation, and control circuits will minimize to the extent feasible the likelihood of simultaneous failure of both normal and alternate offsite preferred power circuits under operating and postulated accident and environmental conditions.

The ACRS in its April 13, 1993, letter to the EDO discussed a concern that SSAR Chapter 8 did not discuss any requirements or design considerations for station grounding. In response GE has committed to meet the following Electric Power Research Institute (EPRI) plant grounding guidelines:

- A station grounding grid, consisting of bare copper cables, will be provided that will limit step-and-touch potentials to safe values under all fault conditions.

- Bare copper risers will be furnished for all underground electrical ducts and equipment, and for connections to the grounding systems within buildings.

***** (new)

- The design and analysis of the grounding system will follow the procedures and recommendations specified by the latest revision of IEEE 665, "Guide for Generation Station Grounding."

- Each building will be equipped with grounding systems connected to the station grounding grid. As a minimum, every other steel column of each building perimeter will connect directly to the grounding grid.

- The plant's main generator will be grounded with a neutral grounding device. The impedance of that device will limit the maximum phase current under short-circuit conditions to a value not greater than that for a three-phase fault at its terminals.

- Provisions will be included to ensure proper groundings of the isophase buses when the generator is disconnected.

- The onsite, medium-voltage ac distribution system will be resistance grounded at the neutral point of the low-voltage windings of the unit auxiliary and reserve transformers.

- Grounding of the neutral point of the generator windings of the onsite standby power supply units (Class 1E diesel generators and non-Class 1E combustion turbine generator) will be through distribution-type transformers and loading resistors, sized for continuous operation in the event of a ground fault.

- The neutral point of the low-voltage ac distribution systems will be either solidly or impedance grounded, as necessary, to ensure proper coordination of ground fault protection.

- The dc systems will be left ungrounded.

- Each major piece of equipment, metal structure, or metallic tank will be equipped with two ground connections diagonally opposite each other.

- The ground bus of all switchgear assemblies, motor control centers, and control cabinets will be connected to the station ground grid through at least two parallel paths.

- One bare copper cable will be installed with each underground electrical duct run, and all metallic hardware in each manhole will be connected to this cable.

- Plant instrumentation will be grounded through separate radial grounding systems consisting of isolated instrumentation ground buses and insulated cables. The instrumentation grounding systems will be connected to the station grounding grid at only one point and will be insulated from all other grounding circuits.

- Separate instrumentation grounding systems shall be provided for plant analog and digital instrumentation systems.

- A lightning protection system will be provided for each major plant structure, including the containment enclosure building. The design and installation of these systems will comply with the National Fire Protection Association's Lightning Protection Code, NFPA-78, and the Nuclear Property Insurance Association's (NEPIA's) "Basic Fire Protection for Nuclear Power Plants" document.

- Lightning arresters will be provided in each phase of all tie lines connecting the plant electrical systems to the switching station(s) and offsite transmission system. These arresters will be connected to the high-voltage terminals of the main step-up and reserve transformers.

- Plant instrumentation and monitoring equipment located outdoors or connected to cabling that runs outdoors will be equipped with built-in surge suppression devices to protect the equipment from lightning-induced surges.

Based on the above ABWR design commitments, the staff concludes that plant structures, systems, and equipment will be appropriately grounded and protected from lightning.

2.1 Station Blackout (SBO) - Mr. J. Knox, NRR

Mr. Knox stated that GE planned to change their commitment for meeting the SBO rule from a design that met the SBO rule through the use of coping to a design that meets the SBO rule through the use of an alternate AC (AAC) power source. The AAC power source will:

- be a combustion turbine generator (CTG),
- be capable of automatically starting, accelerating to rated speed, of reaching nominal voltage, and to begin accepting load within two minutes of receipt of its start signal,
- be capable of being manually reconfigured such that nonsafety investment protection loads can be shed and safety related shutdown loads can be connected (via any one of the Class 1E 6.9 kV buses) from the main control room within ten minutes,
- be a self contained unit equipped with its own auxiliary control and support systems such that external ac power is not required for its operation,
- be physically and electrically independent and diverse from the Class 1E standby diesel generators such that weather-related failures, common cause failures, or single point vulnerabilities are minimized to the extent practicable,

- be electrically isolated from each 6.9 kV bus by two normally open circuit breakers in series (one Class 1E and one non-Class 1E),

- will require DC control power from the Class 1E and non-Class 1E dc power systems to connect the AAC to the Class 1E 6.9 kV bus (to close the two normally open circuit breakers) from the control room within the 10 minute requirement,

- be capable of operating during and after a station blackout without any ac support systems powered from the preferred power supply or the blacked-out units Class 1E power sources affected by the event,

- have sufficient capacity and capability to power any of the non-Safety Related plant investment protection (PIP) buses or one PIP bus and any one Safety-Related bus within 10 minutes of the onset of a station blackout, such that the plant safety systems will be capable of maintaining core cooling and containment integrity,

- not normally or automatically be connected to the off-site power sources or the on-site Class 1E 6.9 kV buses thus minimizing the possibility of a common cause failure,

- be designed to assume non-Safety (PIP) loads automatically, to shed loads manually, and assume safety systems loads manually while maintaining voltage and frequency within design requirements of safety system loads,

- be capable of powering HVAC Systems, chillers, battery chargers, and other support/auxiliary equipment during the station blackout event such that the environment during and following a station blackout event will not exceed the environment for which the equipment is designed and the habitable environment for personnel,

- undergo factory testing, similar to the Class 1E diesel generator, to demonstrate its ability to reliably start, accelerate to rated speed, voltage, and supply power within two minutes,

- be subject to site acceptance testing, periodic preventive maintenance, inspection, testing and operational reliability assurance program goals,

- be designed to quality assurance requirements commensurate with its importance to safety,

- be located above the maximum flood level in the turbine building,

- will be provided with an oil storage and transfer system that will be physically and mechanically independent of the Class 1E standby diesel generator oil storage and transfer system,

- have its fuel sampled and analyzed consistent with applicable standards,

- have sufficient fuel oil stored on site to support 7 days operation, and

- be capable of being periodically inspected, tested, and maintained.

In addition, GE indicated that required plant core cooling and containment integrity during the station blackout duration (10 minutes) will not depend on any ac power sources.

The staff concludes that the use of a CTG as an AAC power source meets the guidelines of RG 1.155 and NUMARC-87-00, meets the requirements of 10 CFR Part 50.63 and NRC Policy Issue SECY-90-016 on Station Blackout, and is therefore acceptable.

3. <u>FSER Chapter 12: Radiation Protection</u> - Mr. R. Pedersen, NRR

Mr. Pedersen stated that Chapter 12 of the SSAR provides information on the radiation protection features and estimated occupation exposure associated with the ABWR design. The radiation protection measures for the ABWR are intended to ensure that internal and external occupational radiation exposures to plant personnel, contractors, and the general population, as a result of plant operations, including shutdown periods and anticipated operational occurrences (AOOs), will be within applicable limits of regulatory criteria and will be as low as is reasonably achievable (ALARA). The staff reviewed the SSAR for completeness against the guidelines of Regulatory Guide (RG) 1.79 (Rev. 3) and against the criteria of applicable sections of the SRP.

The staff's review scope consisted of:

- Ensuring that occupational radiation doses are as low as is reasonably achievable (ALARA)
- Review radiation sources
- Review radiation protection design features, and

- Dose assessment.

The staff concluded that "the radiation protection measures incorporated in the ABWR design will provide reasonable assurance that occupational doses can be maintained ALARA and below the limits of 10 CFR Part 20 during all plant operations."

Significant issues that were resolved included revised dose assessment; design acceptance criteria (DAC) regarding source term, shielding, ventilation; and design changes such as the upper drywell shielding and lower drywell access.

4. <u>FSER Chapter 2 - Site Characteristics</u>

(Mr. R. Rothman, NRR)

Mr. Rothman stated that the NRC staff reviewed the site-related parameters contained in the ABWR/SSAR Section 2.0. The staff found that GE's list of site characteristics is consistent with the appropriate sections of the standard review plan, and 10CFR Parts 50, 52, and 100. The staff will perform a detailed review of a specific site during the COL phase or the early site permit stage. SECY-93-087 was used for elemination of the operating basis earthquake (OBE). The safe shutdown earthquake (SSE) is 0.3g.

The COL applicant shall provide site-specific information related to site location and description, exclusion area authority and control and population distribution, and flood design consideration.

GE imposed no limit for surface faulting. However, it is the staff position that "site that include capable faults are not suitable."

5. FSER Chapter 4 - Reactor

(Mr. H. Richings, NRR)

Mr. Richings described the reactor assembly for the ABWR design. He stated that the reactor assembly consists of the reactor pressure vessel, pressure containing appurtenances that include control rod drive (CRD) housings, in-core instrumentation housing, and the head vent and spray assembly. The reactor pressure vessel includes the reactor internal pump (RIP) casing and flow restrictors in each of the steam outlet nozzles and the shroud support and pump deck that form the partition between the RIP suction and discharge.

The major reactor internal components are the core (fuel, channels, control blades, and instrumentation), the core support structure (including the shroud, top guide, and core plate), the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, and core flooding spargers. Except for the Ziracloy in the reactor core, these reactor internals are stainless steel or other corrosion-resistant alloys. The fuel assemblies (including fuel rods and channel), control blades, shroud head and steam separator assembly, and steam dryers and in-core instrumentation dry tubes are removable when the reactor vessel is opened for refueling or maintenance.

A fuel and control rod design and core loading pattern typical of many currently operating BWRs was used as the basis for the core design for the first cycle and for the system response analysis in the SSAR. These elements of the core design meet criteria approved by the NRC as presented in SSAR Appendices 4B, 4C and 4D. Mr. Richings stated that the fuel and control rod first cycle core designs are similar to current reactors (e.g. BWR6). Some features were added to the design during the course of the review such as loose parts monitor and the stability long term solution. The TRACG calculations for the ABWR show significantly smaller oscillations for BWROG large oscillation events than BWR6. The ABWR proposes EPGs similar to current BWROG recommendations. Fuel and control rod design descriptions are Tier 1 material. The remaining open item includes fuel burnup limit in fuel criteria.

6. FSER Chapter 5 - Reactor Coolant System and Connected Systems (Mr. G. Thomas, NRR)

Mr. Thomas provided an overview of the reactor coolant systems and connected subsystems. He stated that the staff reviewed the measures used to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) and other pressure-retaining components and their supports that are important to safety for the plant design lifetime.

The RCPB will be provided with a pressure relief system to:

° prevent the pressure in the RCPB from rising beyond 110 percent of the design value and,

° provide automatic depressurization if small breaks in the nuclear system should occur together with failure of the high-pressure core flooder (HCPF) and Reactor Core Isolation Cooling System (RCIC). (This depressurization will allow the operation of the low-pressure flooder systems to protect the fuel barrier.)

To be acceptable, the pressure relief system must permit verification of its operability and withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, and faulted conditions.

Overpressure protection in the ABWR will be provided using 18 safety relief valves (SRVs). Eight of the Eighteen SRVs are part of the ADS. The 18 SRVs will be divided into six nominal pressure set points groups and mounted on the four main steamlines between the reactor vessel and the first isolation valve inside the drywell. The SRVs will discharge through piping to the suppression pool. The ABWR pressure relief system design is similar to that for boiling water reactor (BWR) Class 4, 5, and 6 plants.

Mr. Thomas stated that the ABWR uses 10 rector internal pumps (RIPs) that replaced the external recirculation pumps. The rupture of large bore external pipes was eliminated as a design basis accident (DBA), and no core uncovery during DBA.

The reactor core isolation cooling (RCIC) includes a steam inlet bypass start feature and a full flow test capability from suppression pool. The steamline isolation valves functional capability will be verified by ITAAC. As far as the RICI capability during station blackout, the staff accepted the resolution that the COL applicant will show 8 hours at built capability based upon best estimate assumptions. ITAAC will verify formal design requirement of 2 hours.

The ABWR residual heat removal (RHR) system consists of three independent mechanical divisions with a three separate suction lines from reactor for shutdown cooling. The pump suction piping design pressure was upgraded from 200 psig to 410 psig due to ISLOCA concerns.

7. <u>FSER Chapter 10 - Steam and Power Conversion Systems</u>

(Mr. J. Lyons, NRR)

Mr. Lyons provided a brief summary regarding the steam and power conversion system. This system is designed to remove heat energy from the reactor and to generate electric power in the turbine generator. After the steam passes through the high- and low-pressure turbines, the main condensers (MCs) will condense and deaerate the low-pressure turbine exhaust and transfer the rejected heat to the circulating water system, which, in turn, will reject the heat to the power cycle heat sink (cooling tower basin, where applicable). The condensate will be reheated and returned as feedwater to the reactor. The entire system is designed for the maximum expected energy from tne nuclear steam supply system. A turbine steam bypass system will be provided to discharge up to 33 percent of the reactor's design steam flow directly to the condenser during certain transient conditions.

The draft FSER regarding this chapter identified 8 open items, 3 confirmatory items, and 3 COL action items. Most of the open items were resolved through ITAAC.

8. FSER Chapter 11 - Radioactive Waste Management
(Mr. J. Lyons, NRR)

Mr. Lyons provided a brief summary of the radioactive waste management systems. The ABWR design has three radioactive waste management systems: the liquid waste management system, the gaseous waste management system, and the solid waste management system. The ABWR radioactive waste management systems are designed to provide for the controlled handling and treatment of liquid, gaseous, and solid wastes. The liquid radioactive waste system will collect and process liquid wastes from equipment and floor drains; sampling, decontamination and laboratory wastes; reactor water cleanup decant wastes; chemical wastes; and detergent wastes. The gaseous waste system will provide (a) catalytic recombiners to reduce the volume of offgases from the main condenser air ejector, (b) holdup capacity in the form of charcoal delay beds to allow decay of short-lived noble gases from the main condenser air ejector and to absorb radioiodines, and (c) high-efficiency particulate air (HEPA) filters to retain particulates present in the offgas stream. Thus, the system controls the release of gaseous radioactive effluents to the site environs so as to maintain the exposure of persons in unrestricted areas to as low as reasonably achievable (ALARA) in accordance with 10 CFR Part 20, 10 CFR 50.34a, and Appendix I to 10 CFR Part 50.

The draft FSER regarding this chapter identified 8 open items, 3 confirmatory items, and 3 COL action items. Most of the open items were resolved through the addition of appropriate ITAACs.

9. <u>FSER Chapter 19.1 - Probabilistic Safety Assessment</u>

(Dr. R. Palla, NRR, and Mr. G. Kelley, NRR)

Dr. Palla stated that GE submitted a level-3 PRA (i.e., the PRA calculated core damage fequencies, conditional containment failure probabilities, and conditional offsite consequences) that addresses internal initiating events. The PRA also evaluates seismic, internal flood, and fire initiating events.

The NRC reviewed the ABWR PRA to both investigate design insights and determine the quality of the PRA. The NRC concludes that the quality of the ABWR PRA is adequate for its intended functions such as supporting and improving the ABWR design process.

The NRC finds that there is an acceptable balance of preventative and mitigative features in the ABWR design. The core damage frequency estimates for internal events reported in the ABWR PRA are on the order of 10^{-7}/RY. Areas not modeled or incompletely modeled include errors of commission, sabotage, rare initiating events, construction errors, design errors, control systems, ageing, systems interactions, human interaction with smart control rooms, and human errors.

For seismic initiating events, GE submitted a PRA-based seismic margins analysis. This method eliminates the uncertainties associated with picking an appropriate seismic hazard curve, while still providing the insights needed to judge the ability of the design to withstand beyond design bases earthquakes. With a PRA-based seismic margins analysis, rather than developing an estimated core damage frequency, the method estimates the margin the design has beyond the design basis safe shutdown earthquake (SSE) (which is 0.3g for the ABWR) and identifies any weak links in the design. GE reported that all sequences leading to core damage from a purely seismic event were found to have a high confidence with low probability of failure (HCLPF) value of 0.6g or higher.

For internal floods that occur at power, GE performed a PRA internal flood analysis that assumed that once flood water reached a level high enough to fail any piece of equipment in an area, then all the equipment in that area instantly fails and is unrecoverable. This analysis estimated that the core damage frequency from internal floods was on the order of 10^{-8}/RY.

For fires, GE submitted a fire analysis that was a combination of the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by EPRI and the internal events PRA. This analysis assumed that, if a fire occurred in any portion of a fire area, all equipment in the area failed instantly.

The GE fire analysis estimated the core damage fequency from fires to be on order of 10^{-6}/RY.

GE made a number of design modifications to the ABWR both early in the design and later during the NRC's review of the ABWR PRA that were motivated by the results of the PRA. GE and the NRC have drawn a substantial number of significant safety insights from the ABWR PRA that have or will affect the design, construction, operation, maintenance, and regulation of the ABWR.

It is the NRC's subjective view that the mean core damage frequency for the ABWR from internal, external, and shutdown events is probably on the order of 10^{-6} or less. However, it should be emphasized that there are large uncertainties in internal event core damage frequency estimates; the external event analyses for the ABWR were quasi-probabilistic and were designed to uncover vulnerabilities in the design.

10. FSER Chapter 19.2 - Severe Accident Performance
(Mr. R. Barrett, NRR, Mr. J. Monninger, NRR)

Mr. Barrett stated that the purpose of section 19.2 is (a) to consolidate the NRC's approach to resolution of severe accident issues for advanced light water reactors as specified in SECY-90-016, SECY-93-087, and the corresponding SRMs and (b) to evaluate the approach proposed by GE for resolution of severe accident issues for the ABWR.

To provide adequate protection of the public health and safety, current NRC regulations require conservatism in design, construction, testing, operation, and maintenance of nuclear power plants. A defense-in-depth approach has been mandated in order to prevent accidents from happening and to mitigate their consequences. Siting in less populated areas is emphasized. Furthermore, emergency response capabilities are mandated to provide additional defense-in-depth protection to the surrounding population.

The reactor and containment design provide a vital link in the defense-in-depth philosophy. This level of pedigree encompasses, but is not limited to, requirements for meeting single failure criterion, redundancy, diversity, quality assurance, and utilization of conservative models. The staff concludes that existing requirements ensure a safe containment design.

The NRC also has requirements to address conditions beyond the design basis spectrum such as Anticipated Transients Without Scram (ATWS) (10 CFR 50.62), Station Blackout (SBO) (10 CFR 50.63), and Combustible Gas Control (10 CFR 50.44); however, a definitive set of regulatory requirements for addressing specific severe accident phenomenon does not exist. Existing regulations which require conservative analyses and inclusion of features for design basis events provide margin for severe accident challenges. This design basis margin coupled with regulatory guidance to address severe accidents in the form of policy positions ensure a robust design.

In addition, the staff, in keeping with the Commission's policy expectation that future designs for nuclear power plants achieve a higher standard of severe accident safety performance, concluded that severe accidents should be considered to provide an additional level of assurance that the containment function will be met.

Mr. Barrett described the ABWR containment response as passive in nature and it provides an increased level of protection relative to severe accidents.

ACTIONS, AGREEMENTS, AND COMMITMENTS

The following items are requests for action items that were raised by several subcommittee members, as indicated parenthetically:

GE ACTIONS:

1. Provide a better description of electrical spatial separation provided in and in the vicinity of the control room area (for control and power circuits). (C. Michelson)

2. Clarify how the piping from diesel storage tanks to the diesels is routed (e.g., underground or in tunnels). (C. Michelson)

3. Clarify how non-1E power is routed from the turbine building to the 6900KV area in the reactor building. (C. Michelson)

4. Describe the periodic testing of the CTG. (C. Wylie)

5. Provide an explanation of the parallel operation of class 1E circuits. (C. Wylie)

6. Provide a discussion of how the use of cobalt might be restricted in valve design. (R. Seale)

7. Discuss the physical impact aspects of dropping a fuel bundle or shield plug on the refueling cavity seal. (C. Michelson)

8. Concern over the melt through potential of TIP penetrations during severe accident conditions. (C. Michelson/T. Kress)

9. Concern about water-tightness of building joints and

penetration seals. (C. Michelson)

10. Clarify the sensitivity of the reactor pressure boundary leak detection system outside of primary containment. (C. Michelson)

11. Effect of RWCU and feedwater line breaks on associated ion beds and the activity that could be released given the break. (C. Michelson)

12. Clarify the design basis for number of divisions required for safe shutdown following each design basis event (i.e., do you need 1 or 2 divisions as a minimum and is the answer different for different systems or divisions). (C. Michelson)

13. Discuss how secondary containment and divisional separation modeling includes a RWCU break. (C. Michelson)

14. Discuss the design basis for primary containment electrical penetrations relative to Service Level C loading. (C. Michelson)

15. Discuss the potential for water release in the lower drywell area based on the amount of water containing piping and equipment located there. (C. Michelson)

16. What is your best estimate of the completeness of design (i.e., what percentage of total design effort required for a COL is completed of the SSAR effort)? [**For February 1994 full Committee.**] (C. Michelson)

STAFF ACTIONS:

1. Discuss dropping of a fuel bundle or shield plug on the refueling cavity seal. (C. Michelson)

2. Present to the Committee a discussion of the impacts of the ABWR review on current or future research program requirements. (R. Seale)

3. Clarify the design basis for number of divisions required for safe shutdown following each design basis event (i.e., do you need 1 or 2 divisions as a minimum and is the answer different for different systems or divisions). [**This should be discussed at the November 1993 Subcommittee meeting as well as the February 1994 full Committee meeting.**] (C. Michelson)

4. Discuss the staff's basis for approving the test frequency of the vacuum breakers. (C. Michelson)

5. What is your best estimate of the completeness of design (i.e., what percentage of total design effort required for a COL is completed of the SSAR effort)? [**For February 1994 full Committee.**] (C. Michelson/J. Carroll)

6. Comment on GE's responses to all of the above and previous requests. (C. Michelson)

7. Describe significant changes made to SSAR in amendment No. 33; no later than the March 1994 full Committee meating. (C. Michelson)

FUTURE ACTIONS

Future subcommittee meetings will be scheduled as follows:

1. November 2, 1993. (Computer and Ad Hoc DAC subcommittees.)

Re: Review FSER Chapter 7 and DACs.)

2. November 3, 1993: (Safeguards and security subcomittee). Re: Safeguards requirements for the ABWR.

3. November 17, 1993. (ABWR s/c). Re: Review FSER Chapters 6, 13, 17, 18, 18A, and 19.3.

4. December 15, 1993 (ABWR subcommittee). Re: Review FSER Chapters 3, 15, 13.6, and 5.4.8.

5. January 25-26, 1993 (ABWR subcommittee). Re: Review FSER Chapters 1, 9, 14, 16, 20, and 22.

6. February 1994. ACRS Full Committee.

ACTION

Pending the results of the subcommittee review, Mr. Michelson is planning to brief the full Committee during the December 1993 ACRS meeting regarding the review process and the plans for future actions to complete the ABWR review and preparation of the ACRS report to the Commission regarding this matter.

DOCUMENTS

The review document for this subcommittee meeting was the Advanced Boiling Water Reactor Standard Safety Analysis Report (SSAR) to (Amendment 31/August 1993), and the NRC staff's Final Safety Evaluation Report (FSER).