CERTIFIED

acrs-2795

PDR 9-25-92

BY: C. MICHELSON - 4/22/92

DATE ISSUED: 4/13/92

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
ADVANCED BOILING WATER REACTORS SUBCOMMITTEE
MEETING SUMMARY/MINUTES
JANUARY 23-24, 1992
BETHESDA, MARYLAND

Purpose: The purpose of this subcommittee meeting was to hear presentation by GE representatives regarding Chapter 8, "Electric Power System," Chapter 15, "Accident Analysis" and Chapter 19, "Response to Severe Accident Policy Statement" of the GE/Standard Safety Analysis Report (SSAR) for the ABWR design. In addition, the Subcommittee held discussion with the NRC staff regarding SECY-91-294 and SECY-309 that address Chapter 7, "Instrumentation and Control," and Chapter 19, "Response to Severe Accident Policy Statement," of the ABWR/SSAR, respectively.

Attendees: Principal meeting attendees included:

ACRS

C. Michelson, Chairman
I. Catton, Member
W. Kerr, Member
D. Ward, Member
C. Wylie, Member
R. Costner, Consultant
P. Davis, Consultant
M. El-Zeftawy, Staff

Others

R. Strong, GE
J. Maxwell, GE
M. Ross, GE
C. Sawyer, GE
S. Visweswaran, GE
J. Duncan, GE
C. Buchholz, GE
H. Stevens, B&W
V. San Angelo, Bechtel
T. Meisenheimer, Bechtel
M. Cluttin, Newman & Holtinger
R. Sherry, GICA
J. Cabor, GICA
R. Youngblood, BNL
J. Jo, BNL
C. Hsu, BNL

NRC

C. Abbate, NRR
C. Poslusny, NRR
J. Wilson, NRR
D. Thatcher, NRR
S. Bajwa, NRR
A. El-Bassioni, NRR
S. Newberry, NRR
J. Stewart, NRR
R. Van Houten, SECY
D. O'Neal, NRR
R. Palla, NRR
W. Beckner, NRR
R. Nease, NRR
V. McCree, NRR
M. Chairamal, NRR

R501 1/0

Meeting Highlights, Agreements and Requests

1. Mr. Michelson, Subcommittee Chairman, stated that the Commission via the staff requirements memorandum (SRM) of February 15, 1991, indicated its position on what level of design detail the application for design certification should include: (a) reflect a design which, for all structures, systems or components that can affect safe operation of the plant, be complete, except to the extent that some further adjustment to the design within established design envelopes may be necessary -- during what the staff has referred to as the design reconciliation process -- to accommodate actual, as-procured hardware characteristics; (b) encompass a depth of detail no less than that in an FASR at the operating stage for a recently licensed plant, except for site-specific, as procured, and as-built information; (c) be sufficient to allow the staff to evaluate the resolution of severe accident issues in the design, as well as to incorporate the experience from operating events in current designs which the Commission wants to prevent in the future; and (d) provide a sufficient level of detail to ascertain how the risk insights from the design-specific PRA are addressed in the design.

Today GE representatives will brief the subcommittee members regarding Chapter 8, "Electric Power System," Chapter 15, "Accident Analysis," and Chapter 19, "Response to severe accident policy statement" of the SSAR for the ABWR design. In addition, the Subcommittee will hold discussion with the NRC staff regarding SECY-91-294 and SECY-91-309, that address chapter 7, "Instrumentation and Control," and chapter 19, "Response to severe accident policy statement."

Mr. Michelson stated that the Subcommittee has received no written comments or requests to make oral statements from members of the public.

2. Mr. E. Maxwell, GE, briefed the subcommittee members regarding Chapter 8. This chapter describes the on-site and off-site electrical power systems. The scope of the on-site electrical power system includes the entire system on the plant side of the low voltage terminals of the main power transformer and the connection at the higher voltage bushings of the reserve transformer. The main power transformer is not in scope as well as the utility and grid description. The combustion turbine generator (CTG) is within scope.

   There are four unit auxiliary transformers, two to feed the non-Class 1E buses and two to feed the Class 1E buses. The "normal preferred" power feed is from the unit auxiliary transformers so that there normally are no bus transfers required when the unit is tripped off the line. The "preferred power system" is also called the "off-site" power system.

3. Mr. C. Sawyer, GE, briefed the subcommittee members regarding Chapter 15. This chapter describes GE's approach to accident analysis. In this chapter, the effects of anticipated process disturbances and postulated component failures are examined to determine their consequence and to evaluate the capability built into the plant to control or accommodate such failures and events. The system response analysis is based upon the core loading and is used to identify the limiting events for the ABWR.

GE has developed a unique systematic approach to plant safety consistent with the GE boiling water reactor technology base. The key to the GE approach to plant safety is the Nuclear Safety Operational Analysis (NSOA). A generic nuclear safety operational analysis has been developed for each of the recent GE boiling water reactor product lines. It has then been modified to be compatible with the specific plant configuration being evaluated. Key inputs into the nuclear safety operational analysis are derived from the applicable regulations and through industry codes and standards.

GE has evaluated a wide spectrum of events in the nuclear safety operational analysis to establish the most limiting or design basis events in a meaningful manner. The considered events are:

- Decrease in reactor coolant temperature
- Increase in reactor pressure
- Decrease in reactor coolant system flow rate
- Reactivity and power distribution anomalies
- Increase in reactor coolant inventory
- Decrease in reactor coolant inventory
- Radioactive release from subsystems and components, and
- Anticipated transients without scram (ATWS).

Mr. Sawyer presented a comparison of recirculation system between the ABWR and BWR-6 designs. He stated that the ABWR has 10 reactor internal pumps (RIP), while the BWR-6 has 2 external pumps with jet pumps. Number of electric buses are 4 vs. 2 for the BWR. As a consequence of loss of AC power, the ABWR design will have 4 RIPs tripped followed by 6 RIPs tripped at 3 seconds. The BWR-6, however, will have tripped all pumps. In addition, the ABWR has additional power sources such as the M/G sets for 6 RIPs. Other advantages of the ABWR design are:

- The M/G sets shall be capable of holding the RIPs at their original speeds for at least one second, then the RIP shall coast down at a speed of less than or equal to 10%/sec for two seconds.

- This capability will be verified during startup tests.

- No single failure could lead to a trip of all 10 RIPs at the same time.

- Probability of multiple failures which could lead to a trip of all 10 RIPs simultaneously is very low (less than 1×10^{-6}).

Mr. Sawyer also stated that the ABWR design has substantial core design margins.

Currently there are no major open issues identified for Chapter 15, and all open issues are being resolved.

4. Mr. S. Visweswaran, GE, described the objective and scope of the PRA for the ABWR designs. He stated that the objective of the PRA is to assess the probability of core damage and risk associated with the ABWR as defined in the SSAR. This is accomplished by evaluating the frequency and consequence of postulated accident sequences.

The PRA analyzes the ABWR at an average site. The analysis assumes that the plant is at full power prior to the initiation of an accident. The risk associated with fuel handling, storage and waste disposal accidents are judged to be insignificant and are not evaluated.

All of the plant system design detail which is usually required to complete a PRA was not available at the time of study.

The expected frequency of transient events is based upon operating BWR experience and incorporates the design requirement prescribed in the Advanced Light Water Requirements Document of a maximum of one anticipated transient per year which results in reactor scram. The expected manual shutdown frequency of one per year is based upon a 1985 analysis of operating plant data. LOCA initiation frequencies are the same as those used in the GESSAR II PRA.

Five factors are considered and explicitly incorporated in the analysis of system interactions and common cause failures: (a) Component commonality at the system level, such as a common initiating signal; (b) Common divisional services such as common electric power buses or common service water loops; (c) System dependency, such as ADS dependency on the operability of at least one of the five (two high pressure and three low pressure) emergency core cooling system pumps; (d) Past experience of losing on-site or off-site power; and (e) Human errors.

The probability of human error is incorporated throughout the analysis by explicit inclusion in the fault trees and event trees. Two types of errors have been considered: (a) Errors resulting from operator failure to act as directed by normal or emergency procedures; and (b) errors that contribute to component failure to perform as intended because the component has not been properly calibrated or restored to its operational state as required by plant procedures.

Evaluation of external consequences is performed using the CRAC-2 computer code. This evaluation involves:

- Amount and type of fission product release.
- behavior of the fission products after release from the plant.
- Effects on the population exposed to the fission products.
- Input data for the CRAC analysis include containment release data, weather data, demographic data, health physics data, and evacuation assumptions.
- The calculation of accident consequences starts with the postulated release of fission products to the environment. Following the postulated release, the computer code calculates hourly dispersion, cloud depletion, and ground contamination concurrently with population evacuation. Using the resulting air and ground contamination along with population location with respect to the moving plume and dosimetric models based on the health physics data, individual radiological doses are calculated in terms of early and latent exposure for populations within a 25 mile radius of the site.

The GE's estimate of ABWR internal event core damage frequency (CDF) is approximately 1 x 10^{-7}/RY. The staff has reviewed the ABWR/PRA submittal. The staff's estimate of CDF is 5.9×10^{-7}/RY.

5. Ms. C. Buchholz, GE, outlined the process for uncertainty analysis as follows:

   - Break down the event into detailed precursors and phenomena

   - Assign uncertainty values to each sub-event

   - Perform deterministic analysis of each branch to determine the value of the critical parameter (e.g., peak pressure)

   - Determine probability of containment failure for each sequence

   - Draw histogram of critical parameter vs. conditional probability to indicate uncertainty and determine confidence limits.

   - Show the impact of the confidence limits on off-site dose.

   Examples of uncertainty analysis would be:

   - High pressure melt ejection and direct containment heating

   - Critical parameter is drywell pressure

   - Important precursor phenomena

- Vessel pressure at time of vessel failure
- Containment pressure at time of vessel failure

- Key uncertainties include
  - Amount of molten corium
  - Fragmentation of corium

6. Mr. Duncan, GE, described the GE/ABWR seismic PRA approach. For seismic hazard analysis, GE uses the "GESSAR" curve, which is lower than the EPRI and LLNL values.

   The assessment of seismic-initiated core damage frequency and offsite risk consists of four primary tasks; the establishment of a seismic hazard curve, the determination of the seismic capability of critical components and structures, an assessment of the core damage frequency, and an estimate of the offsite risk.

   The first step in the analysis is to identify systems and components that are important to safety during severe accidents and that may be vulnerable (to some extent) to seismic shock. In performing this step, use is made of the internal event analysis and a general knowledge of component fragilities. The objective is to limit the size of the analysis by screening-out many components that can obviously withstand a severe earthquake without damage.

   The remaining components are then subjected to component fragility analysis. The location of components is the plant configuration in relation to structures that may fail is also established. A structural fragility analysis is then conducted for all structures that contain important safety components. The component and structural fragilities are determined in terms of the median value of ground acceleration that would result in failure of the component or structure.

A seismic hazard curve (GESSAR) is used to represent the frequency distribution of expected earthquakes as a function of intensity for the location of the plant site.

The seismic hazard curve is then integrated with the component and structure fragilities to provide an expected frequency of failure of the components and structures. The seismic core damage frequency is determined by constructing and evaluating seismic fault trees and event trees.

7. Mr. W. Beckner, NRC/NRR, presented the NRC evaluation of the ABWR/PRA overview. He stated that the ABWR significantly reduces the CDF of sequences normally found to be dominant for boiling water reactors. The staff finds the ABWR/PRA does not reflect the current state-of-the-art in PRA and have several major deficiencies. Review indicates that GE should devote further attention to the following:

   - The potential for direct containment heating and ex-vessel fuel-coolant interactions

   - The potential for attack of the pedestal by molten debris after flooder operation.

   - The impact of drywell-wetwell bypass on containment performance and the necessity of early venting in cases where RHR is lost.

   A systemic assessment of uncertainties in these areas is viewed as necessary to supplement the ABWR risk estimates.

8. Mr. Stewart, NRR, stated that GE has developed design and performance information relative to the instrumentation and control aspects of the safety-related systems for the ABWR design. Instrumentation and Control (I&C) systems are designated as either non-safety related systems or safety related systems depending on their function. Chapter 7 of the SSAR, presents the I&C systems in accordance to the NRC Reg. Guide 1.70, Rev. 3 (RPS), ESF systems, systems required for safe shutdown, safety-related display instrumentation, and all other instrumentation systems required for safety.

   Generally, for the GE/ABWR design, each individual safety-related system utilizes redundant channels of safety-related instruments for initiating safety action. The automatic decision making and trip logic functions associated with the safety action of several safety-related systems are accomplished by a four-division correlated and separated protection logic complex called the safety system logic and control (SSLC). The SSLC multidivisional complex includes divisionally separate control room and other panels which house the SSLC equipment for controlling the various safety function actuation devices. The SSLC receives input signals from the redundant channels of instrumentation in the safety-related system, and uses the input information to perform logic functions in making decisions for safety actions.

   Divisional separation is also applied to the essential multiplexing system (EMS), which provides data highways for the sensor input to the logic units and for the logic output to the system actuators. Systems which utilize the SSLC are the reactor protection (trip) system, the high pressure core flooder system, the residual heat removal system, the automatic depressurization system, the leak detection and isolation system and the reactor core isolation cooling system.

The NRC staff reviewed Chapter 7 of the ABWR/SSAR and indicated that the SSAR does not contain sufficient design detail, as required by 10 CFR 52 and clarified by the February 15, 1991 staff requirements memorandum. The level of detail available for review is not adequate for the staff to resolve all safety questions. The staff requested the applicants to provide additional information and will continue to work with the applicants to resolve any open issues. The prototype testing required for the design certification of the microprocessor based monitoring, control and protection system, in accordance with 10 CFR Part 52, paragraph 52.47 (b) (2) (i) (B) and (2) (ii) is currently under review. This item is a major factor in establishing the level of detail required for design certification. Based on the information currently available, the staff believes that prototypes will be needed to demonstrate acceptable performance of new technology.

9. Subcommittee Comments and Observations

During the Subcommittee meeting, various concerns were raised by the members and consultants. These are given as follows in random order.

- Dr. Kerr expressed concern regarding the adequacy of the ABWR/PRA. He stated that in the severe accident policy statement, the Commission indicated that a PRA would be required for each new design and the result of this PRA would be part of the submittal that guides the staff in its safety determination. The NRC staff has yet to produce the required guidance.

- Mr. Michelson expressed concern regarding the adequacy and completeness of the SSAR. He cited the reactor water cleanup (RWCU) system as an example, and the apparent lack of consideration of RWCU rupture outside primary containment.

- In regard to PRA studies, a concern was expressed that some accident sequences are being missed because the analyst is relying on failure data for plant equipment that is determined under normal operating conditions while little or no data are known for the equipment under accident loads or environments. A specific example was failure data for Motor-Operated Valves (MOV's).

- A concern was expressed regarding the interpretation of "silent consent" approach from the staff regarding certain issues. The staff responded by stating that the silent approach only implies no conflict with existing regulations governing those issues.

- Mr. Michelson expressed concern regarding the heavy load handling during reactor pressure vessel opening and closing operations. The opening and closing of the Reactor Pressure Vessel (RPV) for refueling requires the handling of massive shield plugs, and the reactor vessel head, steam dryer, and steam separator. The primary containment drywell head must be removed before the RPV head can be removed. The hazards associated with possible accidents during RPV operations are likely to be greater with only secondary containment available to confine the consequences.

- Dr. Kerr commented that GE should explain the discrepancies between its PRA results and BNL. GE representatives agreed.

- Dr. Davis stated that it appears from GE analysis, that no probability of containment failure is allocated for any severe accident pressure below 112 psig. Given that the design pressure is only 45 psig, this appears to be an excessively optimistic assumption.

- Dr. Davis stated that he agrees with the Staff that the ABWR contains potentially important omissions and deficiencies. These include: no fire or flood risks considered, no uncertainties estimated, no shutdown risks evaluated, inadequate human error model, omission of potentially important containment integrity threats, excessively low scram failure rate, inadequate seismic hazard curve, drywell/wetwell leakage and potential seal degradation, important differences in the severe accident code MAAP as compared to MELCOR, and low transient frequency (the transient frequency assumed is 1/yr. for a "mature" plant), which suggests that the results are not valid early in the plant life.

- Dr. Davis expressed concern regarding the use of the CRAC2 consequence code (which was replaced several years ago with the more sophisticated MAACS code).

<u>Future Action</u>

The ABWR Subcommittee will continue to review the GE/ABWR design as the documentation becomes available.

Reviewing Documents Provided for the Subcommittee Meeting

1. GE/ABWR - SSAR / Chapter 19, "PRA."

2. BNL Report - A review of the Advanced Boiling Water Reactor probabilistic Risk Assessment / Vol. 1 and 2 - February 1991.

3. SECY-91-309 / DSER covering Chapter 19 of the SSAR.

4. SECY-91-294 / DSER covering Chapter 7 of the SSAR.

5. GE Viewgraphs.

6. NRC Staff Viewgraphs.

* * * * * * * * * * * * * * * * *

NOTE: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, NW, Washington, DC 20006, (202) 634-3273, or can be purchased from Ann Riley and Associates, Ltd., 1612 K Street, NW, Suite 300, Washington, DC 20006, (202) 293-3950.