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TOM KRESS 11/3/93

SUMMARY/MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON SEVERE ACCIDENTS
SEPTEMBER 22-24, 1993
PORTLAND, OREGON

PURPOSE

The ACRS Subcommittee on Severe Accidents held a meeting in the Columbian-D Room, Sheraton Portland Airport Hotel, Portland, Oregon, on September 22-24, 1993. The purpose of this meeting was to continue the discussion of severe accident and PRA issues associated with the GE Advanced Boiling Water Reactor (ABWR) design certification effort. Copies of the meeting agenda and selected slides from the presentations are attached. The meeting began at 8:30 am each day and adjourned at 5:30 pm, 4:30 pm and 4:30 pm on September 22, 23 and 24, respectively. The meeting was held entirely in open session. No written comments or requests for time to make oral statements were received from members of the public. The principal attendees were as follows:

ATTENDEES

ACRS

T. Kress, Chairman
J. Carroll, Member
I. Catton, Member
P. Davis, Member
W. Lindblad, Member
C. Michelson, Member
R. Seale, Member
M. Corradini, Consultant (9/22,23)
V. Dhir, Consultant
W. Kerr, Consultant
N. Zuber, Consultant
D. Houston, Cognizant Staff Engineer

GENERAL ELECTRIC

C. Buchholz
J. Duncan (9/24)
S. Visweswaran (9/24)
J. Power (9/24)
J. Gabor, Consultant
M. Kenton, Consultant

NRC STAFF

C. Poslusny, NRR

DOE

N. Fletcher

No other persons attended this meeting.

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DISCUSSION

Chairman's Opening Remarks

In his opening remarks, Dr. Kress stated that the ACRS had met a number of times with General Electric to discuss the features of the Advanced Boiling Water Reactor (ABWR) design that are associated with prevention or mitigation of severe accidents. He indicated that the focus of this meeting was not on those features but on the analytical tools and calculational results that show how effective these features might be. He further indicated that this was the Subcommittee's first chance to become familiar with the Modular Accident Analysis Program (MAAP) code and how it models various severe accident issues. Lastly, he indicated that this meeting would focus on Chapters 15 and 19 of the GE ABWR Standard Safety Analysis Report (SSAR).

Presentations by General Electric

Introduction - Severe Accident Review

Ms. Buchholz (GE) presented an introduction to the topics to be covered in this meeting and she discussed the following issues in detail:

- o ABWR features which improve core damage mitigation
- o Primary modeling tools - MAAP, TRACG and SAFER
- o PRA Level 2 - severe accident analysis
- o PRA Level 3 analysis
- o Probabilistic methodology

She indicated that for each of the severe accident issues, the discussion would follow a general pattern of first showing how the issue is modelled in MAAP and then how the issue is treated in the ABWR analysis. She noted that fuel failure was defined as a clad temperature of 1200°K but indicated that core uncovery is not expected if any single pump is working. Fuel relocation is initiated when the fuel temperature reaches 2500°K.

Primary and Containment Thermal Hydraulics

Dr. Kenton (GE Consultant) discussed the thermal hydraulic models used in the ABWR analysis. For the pre-core uncovery phase or success criteria, the TRAC code was used. For the failure mode, the MAAP code was applied. In regard to MAAP, he discussed the following areas:

- o Calculation of pressure, gas temperature and water temperature
- o Flashing and rainout rates
- o Heat transfer to walls
- o Engineered safety features - containment sprays, heat exchanger model
- o Inter-node gas flow rates

In closing, he noted the benchmarking efforts for comparing MAAP to FIST, SAFE and TRACG. He presented some figures by S. Levy, Inc. that showed fairly good agreement between MAAP calculations and the experimental data from FIST for downcomer water levels and fuel rod temperatures.

Core Melt Progression/Hydrogen Generation

Mr. Gabor (GE Consultant) described the core heatup model after core uncovery. He discussed the following phenomena:

- o Axial and radial power distribution
- o Two phase level in each radial region
- o Steam generation in covered regions
- o Heatup due to decay heat, gas transfer and radiation
- o Clad rupture/failure criteria
- o Hydrogen generation
- o Fuel/clad/can motion due to eutectic melting

He discussed some comparative MAAP/MELCOR results for key event times and corium effects in containment. He also discussed the modelled relationship between core relocation and hydrogen generation. He concluded the following:

- o For low pressure scenarios, there is a factor of 2 decrease in hydrogen generation with complete blockage.
- o For high pressure scenarios, there is a factor of 7 reduction due to complete core blockage.

Fuel Coolant Interactions

Ms. Buchholz (GE) discussed the issue of fuel coolant interactions (FCI) and steam explosions in regard to the ABWR design. First, she indicated that the MAAP code does not contain a detailed steam explosion model. Secondly, she indicated that the probability of water being in the lower drywell at the time of vessel failure was very low. She based this conclusion on the fact that only 0.3% of the core damage frequency (CDF) comes from sequences where water would be in the lower drywell, mostly from SBLOCA (Bottom Drain Line Break). She reviewed the experiments performed in this area, such as, MACE, WETCOR, HIPS, BETA and tests at SNL. She indicated that with a simplified bounding analysis, conditions were not satisfied for a steam explosion.

She discussed the potential implications of FCI and presented the results of an independent assessment using the Texas-II FCI model. The calculated results of this assessment showed that the impulse loads at the cavity wall would be much less the 25 kPa-sec, the smallest load calculated for pedestal failure.

Direct Containment Heating

Mr. Gabor indicated that the MAAP code models high pressure melt ejection for BWRs but does not model direct containment heating (DCH). Therefore, a separate analysis was performed to: address major parameters that impact DCH pressure rise, calculate peak pressure associated with variation in these parameters, and determine the probability of failure based on containment fragility curves. Based on this analysis, the containment failure probability resulting from DCH was calculated to be well below the Conditional Containment Failure Probability (CCFP) goal of 0.1.

Debris-Concrete & Non-Explosive FCI in Containment

Dr. Kenton and Ms. Buchholz discussed non-explosive debris-water and debris-concrete (CCI) interactions inside containment. In the discussion, they presented the following topics:

- o Models and assumptions in the MAAP analysis
- o Upward heat transfer from debris for cases of no water
- o Chemistry model for fission product release
- o Comparisons of MAAP CCI model to experiments
- o Impact of CCI model assumptions on ABWR PRA results
- o Evaluation of debris coolability and CCI in ABWR PRA
- o Analysis of pedestal structural failure

The pedestal analysis gave a calculated stress of 0.4 Fy as compared to a code allowable of 0.6 Fy. They concluded that pedestal failure is highly unlikely.

MAAP Modeling of Fission Product Transport and Settling

Dr. Kenton discussed the necessary components for fission product calculations. These included the following:

- o Release rate from fuel or debris
- o Revaporization
- o Aerosol settling
- o Inter-node transport

He indicated that the MAAP code does not model the following: aerosol resuspension, chemical reactions of settled fission products with surfaces, or chemical reactions within pools. He stated that a special version of MAAP had been developed that appears to give correlations that are well suited for PRA use.

ABWR Suppression Pool Carryover

Ms. Buchholz discussed the containment response to the rupture disc opening. The concern was based on fission product release from the suppression pool, the time of rupture and the analysis considered the rate of containment depressurization, pool swell, and carryover of fission products through the vent due to entrainment. She indicated that the conclusions from the analysis were as follows:

- o Pool swell is not a threat to the COPS piping
- o Carryover of fission products will not contribute to offsite consequences.

Summary of Containment Performance

Ms. Buchholz reviewed the containment performance goals in terms of core damage frequency, probability of release >25 rem at the site boundary, individual fatality risk, cancer risk and frequency of a "large release." She indicated that the ABWR satisfies all goals with a large margin, typically by two to four orders of magnitude. She also presented a comparison of individual risks and CDF between the ABWR and the NUREG-1150 plants. In each category, the ABWR gave a lower value by an order of magnitude for CDF and three orders of magnitude for individual risks.

BNL Review of MAAP 3.0B

Mr. Gabor discussed the three phase review of the MAAP 3.0B code evaluation and how the Brookhaven National Laboratory (BNL) study (Phase 2) findings and recommendations were applicable to the ABWR. He briefly described how each severe accident issue was addressed in the ABWR PRA. Included in this discussion were the following topics:

- o Success criteria based on TRACG or SAFER analysis
- o Parametric model for DCH
- o Core blockage sensitivity studies
- o Assumption of complete core drop at vessel failure
- o CCI sensitivity studies
- o ATWS power analyzed by TRACG

In summary, he indicated that the ABWR evaluation addressed the key findings and recommendations from the BNL review. Additionally sensitivity analysis were performed to show the robustness of results. He also indicated that since the BNL study was performed after the completion of the ABWR PRA, this confirmed the thoroughness of the ABWR PRA.

ABWR PRA Methodology

Mr. Visweswaran (GE) discussed the overall methodology for the ABWR PRA, role of the PRA in the ABWR design, specifics of the analyses performed and results in terms of design improvements and goals achieved. He discussed the success criteria for transients, in general, and for ATWS, specifically. The following analyses were discussed: internal events, internal flooding, fire by FIVE, shutdown assessment, and seismic margins approach.

Mr. Visweswaran also discussed the treatment of uncertainties in the ABWR PRA through sensitivity studies and the quantification of data uncertainty. He also discussed the treatment of dependencies in the following areas: support systems, human error and common cause failures. He summarized the ABWR PRA results as follows:

- o The total CDF from internal initiators was calculated to be 1.6E-07/RY.
- o The largest contribution to CDF was station blackout - 71%.
- o The calculated CDF from internal flooding was 2.0E-08/RY.
- o The bounding CDF values for fire were from 4.3E-08 to 8.9E-07/RY.
- o The calculated CDF for shutdown was <<1.0E-07/RY.
- o The seismic margins study gave a HCLPF > 0.60g.
- o The ABWR containment performance meets established goals with wide margin.

Mr. Duncan (GE) discussed the primary uses of the ABWR PRA in the following areas:

- o Assess and improve the design
- o Input to reliability assurance program
- o Identification of important features and operator actions
- o Support improved technical specifications

He presented a listing of fourteen substantive design changes that had resulted from PRA considerations. He indicated that the important plant features study was to identify the most important risk reduction features and he gave examples of each in some seven categories. The examples for severe accidents were: AC-independent water addition systems, lower drywell flooder, and containment overpressure protection system (COPS).

In his concluding remarks, Mr. Duncan indicated that the ABWR met all goals by a large margin for both normal operation and shutdown conditions and that the PRA was very useful in the ABWR certification process.

GE Responses to Subcommittee Questions

In response to questions raised by the Subcommittee early in the meeting, Mr. Power (GE) came to Portland to discuss the following: layout and description of the valves in the Reactor Water Cleanup (RWCU) system, site plan showing the Radwaste Tunnels, and the ABWR response to USI A-45, Decay Heat Removal Requirements. As noted above, GE assigns a low priority to the RWCU system since their calculations indicated that all LOCAs (SBLOCA, LBLOCA and ISLOCA) only contribute about 0.5% to CDF.

Subcommittee Comments, Concerns and Requests

During the meeting, Subcommittee Members and Consultants expressed various comments and concerns as follows (random order):

(1) Mr. Michelson expressed many concerns about the RWCU system and the tunnel arrangement. GE provided an answer to many of these during the last afternoon session.

(2) Drs. Catton, Dhir and Zuber expressed many concerns about the thermal hydraulic aspects of the ABWR. A listing of these concerns has been provided to GE and are summarized as follows:
- o Pool thermal stratification in regard to suppression pool/ wetwell coupling
- o Heat up and retention of heat in the upper drywell and upper drywell head
- o In-vessel FCI assessment using TEXAS-II code
- o Reassessment of simplifications used in DCH analyses
- o Wetwell air space depressurization- flashing concern
- o Steam explosion trigger in cavity by late ejected material
- o Phenomenological models for CCI (NRC and EPRI models)
- o Core level drop/boron mixing during an ATWS

GE was provided a listing of these concerns and will respond to them in writing.

(3) Dr. Corradini expressed some concerns about energetic FCIs which might fail the lower vessel wall and the methodology to handle FCI that might occur if the cavity is flooded before melt ejection. He indicated that quantitative numbers of such events are quite uncertain but that such analyses should be performed.

(4) In response to a question regarding counter current flow models, Dr. Kenton made reference to an article of his, "Combined Natural Convection and Forced Flow Through Small Openings in a Horizontal Partition, With Special reference to Flows in Multicompartment Enclosures," Journal of Heat Transfer, Volume 111, Page 980-987, November 1989. He provided a copy of the paper to Dr. Zuber.

(5) Mr. Michelson expressed a concern about failure of in-core instrument tubes as a source of water to the cavity before molten debris was ejected from the vessel. Ms. Buchholz indicated that it did not appear in the analysis since it would have been screened out due to its low probability. GE agreed to provide the SSAR reference for the description of these tubes.

(6) A general question was raised in regard to the location of the CRD machine during plant operation. This machine could fill an opening in the equipment platform directly below the vessel if it were left in that location during operation and perhaps, lead to debris splashing into the access tunnel. GE provided a response that the machine was moved into the tunnel during plant operation.

(7) In regard to multiplexers in the digital control systems, Mr. Michelson expressed a concern about the temperature and water effects from fires, steam or flood. Mr. Power indicated that some of the units (EMS) are located in secondary containment for non-safety informational processing. Others are in the essential equipment rooms external to secondary containment, and those have a controlled environment.

(8) In regard to questions about the COPS system, Ms. Buchholz described the details of the system - a straight vent pipe with two rupture discs and an inert gas filling between them. The inboard disc was intended to rupture at 90 psig while the outboard disc would rupture at just a few psi above atmospheric. She indicated that the COPS system was described in Amendment 32 of the SSAR.

(9) In regard to seismic margins, Mr. Carroll asked GE if they were aware of a recently discovered fact that storage battery plates become brittle in 4-7 years and if such had been factored into their margins analysis. They indicated that it probably wasn't and they would take the matter under consideration.

(10) Mr. Carroll requested that GE provide information in the SSAR on how they addressed GSI A-17, especially in events beyond flooding.

Closing Remarks

In closing, Dr. Kress thanked GE and their consultants for their participation in the meeting, especially their effort to provide quick responses to the questions asked.

FUTURE ACRS ACTION

The Subcommittee agreed to a general overview presentation of the matters discussed for the Full Committee meeting in October 1993.

ACTIONS, AGREEMENTS AND COMMITMENTS

The actions, agreements and commitments that resulted from this meeting are discussed above and are as follows:

(1) GE agreed to provide written responses to the concerns of Dr. Catton.

(2) GE agreed to provide the SSAR reference for the description of the in-core instrument tubes.

(3) Dr. Catton agreed to provide a copy of a PRA article entitled, " Combining Mechanistic Best-Estimate Analysis and Level 1 Probablilistic Risk Assessment," Reliability Engineering and System Safety, Volume 39, 1993.

DOCUMENTS

The review documents for this subcommittee meeting are as follows:

(1) Advanced Boiling Water Reactor (ABWR) Standard Safety Analysis Report, Chapter 19, Response to Severe Accident Policy Statement (thru Amendment 30), **Proprietary Information**

(2) Brookhaven Technical Report, MAAP 3.0B Code Evaluation, J. .. Valente and J. W. Yang, FIN L-1499, October 1992

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Note: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. 20006, (202) 634-3273, or can be purchased from Ann Riley & Associates, 1612 K Street, N.W., Suite 300, Washington, D.C. 20006, (202) 293-3950.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SEVERE ACCIDENTS SUBCOMMITTEE MEETING
SHERATON PORTLAND AIRPORT HOTEL
COLUMBIAN-D ROOM
PORTLAND, OREGON

ABWR SEVERE ACCIDENT/PRA ISSUES

- AGENDA -

Wednesday September 22, 1993	TIME
A. Subcommittee Chairman's Opening Remarks - Tom Kress	8:30 am
B. Introduction - Carol Buchholz (GE)	8:35 am
***** BREAK *****	9:45 am
C. Discussion of Modelling and Phenomena- (Note: The discussion of each issue will include the following as appropriate: o MAAP 3.0B Code o Other Models Used in Analysis o Implications and Resolution of Issue for ABWR - Carol Buchholz, Jeff Gabor and Marc Kenton	
1. Primary and Containment Thermal-Hydraulics	10:00 am
***** LUNCH *****	12:00 - 1:00 pm
C. Modelling and Phenomena Discussion (Continued)	
2. Core Melt Progression and Definition of Onset of Fuel Damage and Hydrogen Generation, Distribution and Control	1:00 pm
3. Fuel Coolant Interactions and Steam Explosions	2:10 pm
***** BREAK *****	4:00 pm
4. Direct Containment Heating	4:15 pm
******** Adjourn ********	5:30 pm

SEVERE ACCIDENTS SUBCOMMITTEE MEETING
ABWR SEVERE ACCIDENT/PRA ISSUES

Thursday, September 23, 1993	Time
Opening Comments - T. Kress, Chairman	8:30 am
C. Modelling and Phenomena Discussion (Continued)	
5. Debris Coolability and Core-Concrete-Interaction (Including Impact on Pedestal and Containment Integrity)	8:35 am
***** BREAK *****	10:50 am
6. Aerosol Generation and Transport	11:10 am
***** LUNCH *****	12:40 - 1:35 pm
C. Modelling and Phenomena Discussion (Continued)	
7. Suppression Pool Carryover	1:35 pm
***** BREAK *****	3:25 pm
D. Overall Containment Performance	3:40 pm
E. Summary of BNL Recommendations (MAAP Code)	4:05 pm
******** Adjourn ********	4:30 pm

SEVERE ACCIDENTS SUBCOMMITTEE MEETING
ABWR SEVERE ACCIDENT/PRA ISSUES

Friday, September 24, 1993

Opening Comments - T. Kress, Chairman	8:30 am
F. Probabilistic Safety Analysis Topics - S. Visweswaran, Jack Duncan	
1. Overall Methodology	8:35 am
***** BREAK *****	10:05-10:20 am
1. Overall Methodology (Continued)	10:20 am
2. Treatment of Common Cause Failures and Human Error Modelling	11:30 am
***** LUNCH *****	12:15 - 1:15 pm
3. Treatment of Uncertainties	1:15 pm
4. Summary of Results	1:45 pm
5. Use of the PSA in the ABWR Design and Important Features Identified by the PSA	2:25 pm
G. Discussion of Additional Information Provided by GE	3:15 pm
H. Subcommittee Summary Discussion and Planning for October 7-8, 1993 Full Committee Discussion	4:10 pm
******** Adjourn ********	4:30 pm

GE Nuclear Energy

ABWR Severe Accident Review

Presentation to ACRS Subcommittee on Severe Accidents

C. E. Buchholz
ABWR Engineering

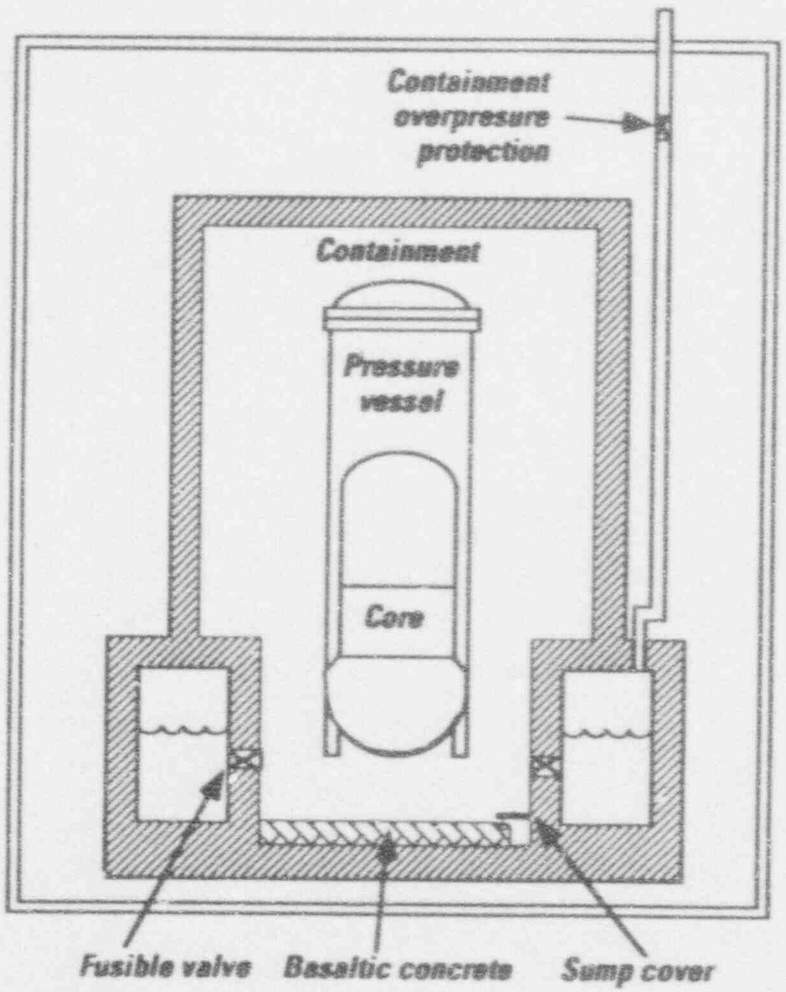
M. A. Kenton and J. R. Gabor
Gabor, Kenton and Associates (ARSAP)

September 22, 1993

ABWR features which improve core damage mitigation

- Suppression pool
- Passive lower drywell flooder
- Containment overpressure protection system
- Firewater addition system
- Increased pressure capability of lower drywell head
- Basaltic concrete specified for the lower drywell
- Protection for lower drywell sumps
- Improved accident management strategies
  - Containment flooding procedures
  - Optimized use of water injection systems
  - Considerations for containment overpressure protection system reclosure

ABWR Severe Accident Mitigation



ABWR passive features which mitigate severe accidents

- Inerted containment
- Lower drywell flood capability
- Lower drywell special concrete and sump protection
- Suppression pool - fission products scrubbing and retention
- Containment overpressure protection

Level 2 / Severe Accident Analysis

- *Containment systems*
  - *Important features identified*
  - *Independence from systems which must fail to lead to core damage was examined*
- *Basic containment performance characteristics determined*
- *Accident management strategies and emergency procedure guidelines developed*
- *Phenomenological Uncertainty studies*
  - *Survey for key phenomena / uncertainties*
  - *Eliminate those phenomena precluded by design*
  - *Perform sensitivity analysis to determine important phenomena*
  - *Perform detailed uncertainty analysis for important phenomena*
- *Containment event trees developed to examine both system behavior and severe accident phenomena*

Level 3 Analysis

- *The CRAC II code was used to calculate atmospheric transport and consequence analysis*
- *Median US site*
  - *Weather*
  - *Demographics*
- *Evacuation scheme used similar to 1150*
  - *95% of people begin to evacuate an hour after being given notice*
  - *Remaining 5% do not evacuate*
- *Dose calculations were performed with no credit for sheltering or shielding*
- *Health physics performed per International Committee on Radiation Protection 30*

Probabilistic methodology

- *Containment event trees used to assess conditional event probabilities for major containment performance characteristics*
  - *Containment heat removal availability*
  - *Continued core concrete interaction*
  - *Initiation of firewater addition system*
- *Branch probabilities reflect several types of probabilities*
  - *System unavailabilities*
  - *Operator actions*
  - *Phenomenological uncertainties*
- *Branch probability values for systems and phenomena were calculated using Decomposition Event Trees (DETs)*

GE Nuclear Energy

Summary of Containment Performance

Presentation to ACRS Subcommittee on Severe Accidents

Carol E. Buchholz
ABWR Programs

September 22-24, 1993

Containment performance goals

- *Conditional Containment Failure Probability: CCFP < 0.1*
  - *Large Release Definition (More than 25 rem): CCFP = 0.002*
  - *Structural Integrity Definition: CCFP = 0.005*
- *Prevent fission product release before 24 hours for dominant sequences*
  - *Time of release for dominant sequences ≈ 31 hours*
  - *Conditional probability of events with relesase before 24 hours: 0.004*
- *Individual Risk Goal < 3.9E-7*
  - *Result for ABWR = 1E-13*
- *Societal Risk Goal < 1.7E-6*
  - *Result for ABWR = 8E-13*

ABWR satisfies all goals with a large margin

Summary of ABWR PRA Results

Presentation to the ACRS Subcommittee on Severe Accidents

S. Visweswaran
GE Nuclear Energy
September 24, 1993

CDF by Type of Event

Type of Event	CDF (per reactor-year)
• Transients	3.7E-08
• Loss of offsite power	1.2E-07
• LOCAs	6.9E-10
• ATWS	2.7E-10
Total	1.6E-07

Dominant Accident Sequences

- Station blackout (SBO) with failure of reactor core isolation cooling (RCIC) and failure to recover offsite or emergency power prior to core melt
- Transients with loss of feedwater followed by failure of high pressure injection systems and failure to depressurize the reactor prior to core melt
- Loss of offsite power with failure of high pressure injection systems and failure to depressurize the reactor prior to core melt

Conclusions

- ABWR is designed to mitigate internal floods
- Total core damage frequency for internal floods is 2E-8 per reactor-year
- Most potential floods will be automatically terminated by level sensors (turbine and control buildings)
- Floods in the reactor building can all be contained in the ECCS rooms or the corridor of the first floor
- Operator action not required to terminate floods although timely operator action can limit potential flood damage
- ABWR can safely be shutdown for all postulated internal floods

Internal floods have negligible contribution to ABWR core damage frequency

Primary Features Contributing to High Seismic Margins

- Many ABWR design features provide protection against core damage events
  - High HCLPF for safety building structures and DC power
  - Multiple diverse systems for reactivity control, core cooling and containment cooling
  - Fire water injection (with valves having manual operation capability)

Overall Containment Performance

- Conditional containment failure probability goal of 0.1 established in the ABWR Licensing Review Basis
- Potential for containment failure modes examined for all core damage sequences
- Two definitions for containment failure considered
  - Large release (assumed to be greater than 25 rem at the site boundary)
    CCFP = 0.002
  - Pressure Integrity ("Integrity as a pressure boundary can no longer be controlled" – SECY-90-016)
    CCFP = 0.005

ABWR meets established goal with wide margin