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ACRS SUBCOMMITTEE MEETING ON THE
ADVANCED PRESSURIZED WATER REACTORS (WAPWR SP/90)
SEPTEMBER 28, 1989
BETHESDA, MARYLAND

Purpose

The purpose of this Subcommittee meeting was to discuss and review the Westinghouse APWR (RESAR SP/90) design.

Attendees

ACRS

J. Carroll, Chairman
I. Catton, Member
C. Michelson, Member
D. Ward, Member
M. El-Zeftawy, Staff

Others

M. Shannon, W
T. Van de Venne, W
D. Sharp, W
S. Stahl, W
W. Shirley, W
P. Trayers, W
R. Lutz, W
D. Noonan, Bechtel
T. Chu, BNL
A. Tingle, BNL
T. Pratt, BNL
L. Rib, AECL

NRC

T. Kenyon, NRR
L. Donatell, NRR
T. Hsia, NRR
C. Y. Li, NRR
D. Notley, NRR
D. Perskino, NRR
H. Vandermolen, RES
P. Niyogi, RES
E. Chelliah, RES
J. Monninger, NRR
D. Shum, NRR

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<u>Meeting Highlights, Agreements, and Requests</u>

1. Mr. Carroll, Subcommittee Chairman, stated the purpose of the Subcommittee meeting and introduced the other ACRS members.

2. Mr. L. Donatell, NRC/NRR - Project Manager, presented the current review status for the <u>W</u>APWR SP/90 design. He stated that the staff is expecting the Commission to establish a new approved priority for the SP/90 preliminary design approval (PDA). So far, the staff has completed one draft SER regarding the PRA analysis (front-end only, March 1988) and two draft SERs (SRP on June 1988 and March 1989). Currently, there are 107 open items that have to be resolved before the PDA is issued. There are additional 53 open items that have to be resolved before the final design approval (FDA) is issued. In addition, there are 99 open items that have to be resolved before the FDA is issued and/or plant specific application.

   Mr. Donatell indicated that there will be two more DSERs that have to be issued. The first one is for the PRA (Back end portion) and expected to be issued in November 1989. The second SER is regarding the USIs/GSIs and severe accidents, and is expected to be issued in April 1990. To finalize the staff's review of the SP/90 design, the staff is requesting three additional ACRS Subcommittee meetings to be held in November 1989, May 1990, and August 1990. The PDA decision will be made approximately in October 1990.

3. Mr. M. Shannon, <u>W</u>/Licensing Manager, described the review status of the NRC safety evaluation of RESAR-SP/90 particularly with respect to the severe accident issues that are currently being discussed within the industry and the NRC. Mr. Shannon indicated that since <u>W</u> is interested in a PDA at the current time, the SP/90 design is not incorporating any additional design features that are being developed with the Japanese until a FDA submittal.

Mr. Shannon indicated that W is not reviewing the SP/90 design versus the EPRI requirements document at the PDA stage. W has responded to the 107 open items in the DSER, and currently finalizing its submittals of Module 2, which deals with USIs, and GSIs.

4. Mr. Van de Venne, W/Engineering Manager, indicated that the SP/90 design could meet up to 98% of the EPRI requirements document. Mr. Van de Venne outlined the plant design features and system reliabilities. The primary systems consist of the following:

   ° Reactor coolant system (RCS) - it includes a reactor vessel with greater internal volume than standard W design. The increased volume of water above the core provides a longer time before core uncovery (e.g.: in case of small LOCA and a loss of secondary cooling).

   ° Core reflood tanks - four tanks with low pressure nitrogen coverage that inject into the RCS vessel assist the HHSI in reflooding the core following a LOCA. These tanks eliminate the need for active low head SI pumps.

   ° Integrated Safeguards System (ISS) - there are four high head pumps that inject through their own RCS vessel connection to provide emergency core cooling for the LOCA events, and provide RCS makeup and boration for all non-LOCA events. Only one of these four pumps is required for small LOCA and feed-and-bleed cooling.

   ° Emergency Water Storage Tank (EWST) - the water supply is located in the basement of the containment. The EWST provides a means to reduce the containment cleanup resulting from discharge from the pressurizer relief tank rupture disc and the hot leg vent path, or the steam generator overfill paths.

- Charging pumps - have substantial RCS makeup capability and are automatically loaded on the emergency diesels in the case of loss of offsite power. However, they are not used to mitigate design basis LOCAs.

- Back-up Seal injection - the CVCS contains a back-up seal injection pump which automatically provides RCP seal cooling in the event of loss of normal seal injection. This pump has its own self-contained diesel generator set.

- Alternate Core Cooling Means - in addition to normal alternate core cooling means (SFWS, EFWS) and their back-up, there are other possibilities. For instance, RCS feed and bleed with charging pumps, RCS depressurization and feed and bleed with RHR pumps, and SG feed by main feedwater or condensate pumps.

The Secondary Systems consist of the following:

- Emergency Feedwater system - contains four pumps (two electric and two turbine driven). Any one of the pumps is sufficient to remove decay heat through S.G.

- Start-up Feedwater System - a single non-safety class pump driven by a 1E motor takes suction from the condenser hot well provides the normal feedwater function following reactor trip.

- Steam Generator overfill protection - each S.G. is provided with an automatic drain system to prevent high S.G. level. The drain path is into the EWST.

The Auxiliary Systems consist of the following:

° Two diesel generators are provided for back-up following a loss of offsite power.

° Component Cooling Water System (CCWS) and Service Water System (SWS) are not interconnected. Therefore for events such as CCWS or SWS pipe breaks, only one subsystem can be affected.

Mr. Van de Venne stated that SP/90 is the U.S. Version of of APWR for Japan. Most changes address U.S. licensing issues such as RV level instrumentation, technical support center, four Class 1E battery sets, redundant EFW storage tanks and single main steam isolation valves. Some changes reflect U.S. siting requirements.

Mr. Van de Venne outlined the PRA (core melt frequency) issues for the SP/90 design. He indicated that the transients initiating frequency is assumed to be 10 per year. Most operating plants approach 3 per year. SP/90 design has certain design features to reduce trips such as full load rejection capability, main generator breaker, enhanced I&C test capability, and no reactor trip following loss of main feedwater pump. For specific events such as station blackout, the SP/90 has the following design features:

° Emergency feedwater system includes two AC and DC independent turbine-drive pumps.

° Chemical and volume control system includes backup seal injection pump powered from small dedicated diesel generator.

° Class 1E batteries are sized for four hours of operation under station blackout conditions.

° Connections are provided between the backup seal injection pump power source and the Class 1E batteries.

° Station blackout coping time is 24 hours. At that time, the emergency feedwater storage tank and spent fuel pit need to be replenished.

° Current SP/90 design exceeds the requirements of Regulatory Guide 1.155.

Mr. Van de Venne indicated that although risk due to station blackout is substantially reduced relative to current plants, it is still the single largest contributor to SP/90 core melt and containment failure frequencies.

For the ATWS event, the SP/90 design incorporates the following features:

° The integrated protection system is highly reliable:

- Two-out-of-four logic

- Continuous on-line testing

- Fail-safe principles.

° Reactor trip switchgear consists of eight breakers arranged in two separate cabinets which can be tested on-line without bypass.

- In addition to the reactor trip switchgear, the motor-generator sets can be tripped from the MCR.

- ATWS mitigation system will generate turbine trip and emergency feedwater actuation signals independent of integrated protection system.

- Moderator temperature coefficient is significantly more negative than for current plants.

- ATWS considerations will be factored into the design of pressurizer safety and relief valves.

- Detailed analyses of ATWS transients will be included in the FDA application to demonstrate compliance with ATWS acceptance criteria.

The ATWS contribution to SP/90 core melt frequency is less than 10^{-7}/yr., and the containment failure following ATWS core melt is highly unlikely.

For the intersystem LOCA, some of the SP/90 design features that prevent and mitigate such an event are:

- RHR suction line is most credible path intersystem LOCA (10^{-6}/yr.).

- RHR isolation valves are included in ISS test header and will be leak tested during startup.

- RHR suction piping design pressure has been increased such that gross failure would not occur even when subjected to RCS operating pressure.

° RHR suction piping is in open connection with the in-containment EWST such that pressure is relieved following failure of RHR isolation valves.

° RHR pumps and piping are arranged to assure sufficient EWST inventory to allow continued core cooling with non-affected ISS subsystems.

Mr. Van de Venne stated that the SP/90 PRA indicate the risk from intersystem LOCA is very low.

Other events such as vessel rupture, loss of cooling, steam generators tube rupture and other internal events have been analyzed in a conservative manner. The total core melt frequency has been reduced to 1.5×10^{-6}/yr. by systematic application of PRA techniques throughout the design process.

No external events PRA has been performed. However, external events have been addressed in the design, e.g.:

<u>Fire</u>

° Three hour fire barriers outside containment.

° Improved separation inside containment.

<u>Flood</u>

° General arrangement limits consequences of storage tank leakage or failure.

° Essential service water piping and valves are segregated with drains to the outside.

° Turbine plant flooding cannot impact nuclear island.

Seismic

° Integral nuclear island basemat.

° Water storage tanks inside building.

Sabotage

° Separation between redundant divisions.

° Separation safety/non-safety.

° Single point access control.

5. Mr. T. Chu, Brookhaven National Laboratory (BNL), outlined the status of BNL review of SP/90 probabilistic safety study (PSS) core damage frequency evaluation. BNL performed an independent assessment of the front-end part of the PRA for WSP/90. The scope of the review included:

   ° Internal events only.

   ° Shutdown risk not included.

   ° Requantification of core damage frequency.

   Mr. Chu stated that generally the methodology used by W to model plant systems is typical of current PRA practice. BNL review generally finds that core melt frequency for SP/90 was relatively low, with several open issues remain to be addressed. Specifically:

   ° IPE model.

° Interfacing system LOCA through accumulators and reflood tanks needs to be modeled.

° Partial or total loss of DC power.

° Loss of instrument air.

° Partial or total loss of vital AC.

Preliminary BNL assessment of core damage frequency is 5.98×10^{-6}/yr, compared to W PSS estimate of 1.50×10^{-6}/yr.

6. Mr. T. Pratt, BNL, presented the probabilistic safety study (PSS) for the SP/90 containment and offsite risk evaluation which was performed by BNL. The scope of the BNL review was to provide an independent evaluation of containment performance, fission product release, and offsite consequences for the SP/90 design under severe accident conditions. The BNL evaluation included containment event tree quantification (MARCH and CONTAIN codes), fission product release (STCP and CONTAIN codes) and offsite consequence (CRAC2 and MACCS codes). The BNL review was also based on NUREG-1150 methodology and 59 TOP event questions and 19 release categories.

For the offsite risk evaluation, BNL concluded that if direct containment heating is neglected, SP/90 risk was predicted to be low and if direct containment heating occurs, predicted risk is increased:

° Latent effects and population dose increase slightly.

° Increase in early health effects more noticeable.

- Early health effects only slightly effected by evacuation (because of short warning time if DCH occurs).

- Latent effects and population dose not effected by evacuation assumptions.

Mr. Pratt stated that generally, the methodology used by Westinghouse to model containment performance, fission product release and offsite consequences is consistent with current PRA practice.

The BNL review generally confirmed SP/90 PSS results:

- Longer times to core damage.

- Slower containment pressurization.

- Lower Risk.

Containment Performance uncertain if core melts with primary systems at high pressure; however:

- High pressure sequences relatively low frequency.

- Long times available for operator actions (e.g., depressurization).

7. Mr. Van de Venne described the mid-loop operation for the SP/90 design. The design includes features such as:

   - Water level during mid-loop operation is at least 9 inches above actual mid-plane elevation.

° With vortex breaker, air entrainment starts to occur at approximately 3 inches below mid-plane elevation, but is limited to less than 10%.

° RHR suction lines are sloped continuously downwards towards RHR pumps and are, therefore, self-venting.

° RHR pump suction lines provide adequate pump NPSH at full flow assuming saturation in the hot leg.

° HHSI pump will be available during mid-loop operation for emergency makeup, if required.

All the concerns raised in Generic Letter 88-17 are being adequately addressed.

For the fire protection issue, the design features (outside containment) incorporates the following:

° Redundant divisions of safety related equipment are located in dedicated areas which are separated from each other and from other areas of the plant by three hour fire barriers.

° Each safety area is provided with its own ventilation systems, and piping and cabling interconnections are minimized.

° The main control room and main steam tunnel are exceptions to the above separation arrangement.

For inside containment:

° Containment constitutes a single fire area.

° This single fire area will be subdivided into several fire zones such that loss of one fire zone will not jeopardize the capability to achieve cold shutdown.

° Separation between fire zones will preferentially be based on existing structural walls. Where this is not possible, other methods will be used (e.g., no line of sight exposure, large distance with no intervening combustibles).

8. Mr. R. J. Lutz, Jr., W, briefed the Subcommittee on severe accident containment performance issues for the SP/90 design.

For accident sequences which progress to core melting, depressurization of the reactor coolant system prior to reactor vessel failure can reduce the dynamic challenges to containment integrity at reactor vessel failure. Specifically:

° The containment pressure transient due to RCS blowdown is less severe.

° The potential for and/or the magnitude of direct containment heating is reduced.

The SP/90 design includes pressurizer PORVs of sufficient capacity to depressurize the reactor coolant system to less than 200 psi prior to reactor vessel failure for severe accident sequences.

Depressurization is a backup to the cavity design features to preclude direct containment heating.

Source Terms - The methodology used for the prediction of severe accident source terms (i.e., releases to the environment) is an area of disagreement between the U.S. NRC and the industry.

The SP/90 severe accident source terms were calculated using the MAAP 2.0B computer code. The source terms predicted by MAAP 2.0B are considered to be realistic, based on best estimate methodology.

Hydrogen Control - For Accident sequences which progress to core melting, the accumulation of hydrogen in the containment may lead to flammability conditions which can challenge the containment integrity. Specifically; Mr. Lutz indicated that:

° Per 10 CFR 50.34(f), the containment design shall be capable of withstanding a hydrogen burn which involves a hydrogen inventory from reaction of 75% of the core active cladding zirconium inventory.

° The hydrogen concentration in the containment should remain below the limits at which transition to detonation would occur.

The containment design in the SP/90 is capable of withstanding a hydrogen burn resulting from a hydrogen inventory equivalent to the reaction of 75% of the core zirconium inventory.

The containment volume is large enough so that even in the event of hydrogen generation equivalent to the reaction of 75% of the core zirconium inventory, the containment hydrogen would not exceed 15 V/O.

Although not required by the analysis results, hydrogen igniters have been included in the SP/90 design specifications.

Containment Venting - For the SP/90 design, the analyses indicate that no containment failure is predicted during the first two days following a severe accident; this is considered to be a sufficient time period to initiate accident management strategies to prevent ultimate containment failure. Thus, the need for containment venting is precluded.

Containment Performance - For the SP/90 design, best estimate analyses predict that the containment integrity will be maintained for at least 2 days for all severe accident sequences. This is a sufficient time period to initiate accident management strategies to prevent containment failure for any severe accident sequence.

The capability to maintain containment integrity for 2 days following a severe accident is not sensitive to reasonable uncertainties in the dominant severe accident phenomena.

Core/Concrete Interaction - For the SP/90 design, the reactor cavity is designed to preclude core/concrete interactions for all severe accidents in which a water layer can be maintained in the reactor cavity.

The plant is designed to preclude cavity dryout by providing an alternate water supply from the EWST.

Mr. Van de Venne summarized at the end of Mr Lutz's presentation by stating that the:

° Current SP/90 design is expected to meet nuclear industry safety goals stated in EPRI ALWR requirements document;

- 10^{-5} per year core melt frequency.

- 10^{-6} per year severe release frequency.

° If detailed PRA analyses to be performed at the FDA stage indicate that the SP/90 design does not meet the industry goals, additional features will be evaluated and incorporated to the extent required to meet the goals.

9. As a result of the Subcommittee discussion, some of the Subcommittee's members expressed some concerns in regard to the following:

° Dr. Catton expressed some concern that neither the NRC staff nor W has carefully analyzed the flow instabilities and vibration in the steam generators to investigate any fluid structural interaction problems.

° Mr. Michelson commented that another subcommittee meeting is needed to discuss the safety-related systems before finalizing the PRA analysis.

° Mr. Carroll indicated that the interaction between W and EPRI regarding the compliance with the EPRI requirements document is not very clear at the PDA stage. Mr. Michelson agreed.

° Mr. Carroll questioned the significance of the PDA approach and asked W representatives if there is any funded plans for final design approval that exist at the present time. W's response was no, and especially in the U.S.

° Mr. Michelson advised the APWR Subcommittee to review W Modules (1 through 16) chapter by chapter.

° Dr. Catton indicated that throughout the SP/90 design; the check valves performance, piping and location was not analyzed carefully.

° Dr. Catton questioned the depressurization process for the SP/90 design and advised W to study the work that was performed by Mr. Aubrangeli. Dr. Catton indicated that W may benefit from depressurizing sooner and faster.

° Dr. Catton questioned the use of the ill-documented MAAP computer code for calculation in the severe accident source term.

° Mr. Carroll indicated that there is an error in the W's presentation of the hydrogen control. According to 10 CFR Part 50.34(f); H_2 concentration should not exceed 10% during and following an accident that releases an equivalent amount of H_2 as would be generated from a 100% fuel-clad metal water reaction/and not 75% as claimed by W.

° Mr. Ward questioned W philosophy for the hydrogen control issue to install the igniters.

° Mr. Michelson expressed some concern regarding the lack of data base to support the final results of the PRA analysis, especially for accident conditions.

° Mr. Michelson expressed some concern regarding the definition of PDA and FDA in regard to the design and what kind of study and analysis will be included under each definition.

Future Action

The Subcommittee Chairman and Members decided to conduct another Subcommittee meeting on November 3, 1989, to continue discussion of the subject matter.

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<u>NOTE</u>: 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 Heritage Reporting Corporation, 1220 L Street, N.W., Suite 600, Washington, D.C. 20005, (202) 628-4888.