

CERTIFIED

ACRS-2721

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10-2270 by J Carroll

ACRS MEETING MINUTES/SUMMARY OF THE
ADVANCED PRESSURIZED WATER REACTORS
SUBCOMMITTEE
SEPTEMBER 20, 1990
BETHESDA, MARYLAND

PURPOSE

The purpose of this Subcommittee meeting was to review the draft SER for the W/SP-90 design. The entire meeting was open to public attendance.

ATTENDEES

J. Carroll, Chairman
I. Catton, Member
W. Kerr, Member
C. Michelson, Member
C. Wylie, Member
P. Shewmon, Member
E. Wilkins, Member
M. El-Zeftawy, Staff
T. Rotella, Staff

NRC

L. Donatelli, NRR
C. Miller, NRR
N. Trehan, NRR
B. Mendelsohn, NRR
R. Correia, NRR
D. Shum, NRR
D. Notley, NRR
H. Brammer, NRR
C. Liang, NRR

WESTINGHOUSE

T. Van De Venne, Westinghouse
E. Burns, Westinghouse
W. Schivley, Westinghouse

OTHERS

Lynn Connor, The NRC
Calendar
Tim Stack, B&WNT
Chong Lewe, NUS
Patrick Harris, Bechtel
Kim Barrett, Bechtel
C. Heitsch, West German
citizen

Meeting Highlights, Agreements, and Requests

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

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2. Dr. C. Miller, NRR Project Director, provided introductory remarks and a history of the staff efforts to date regarding the preliminary design approval (PDA) and the associated recently issued draft Safety Evaluation Report (SER). He indicated that the PDA was originally thought of as being the document which would be referenced in a construction permit. Currently, Westinghouse (W) does not plan to submit a Final Design Approval (FDA). He also stated that the staff efforts on the SP-90 are complete for now.

Dr. Miller commented that the PDA for the W SP/90 has not been drafted but is typically only a few pages in length and would reference the staff's SER and the ACRS letter. He continued with an explanation of the PDA concept and its relationship to 10 CFR part 52.

Dr. Miller stated that the PDA could not be drafted prior to the ACRS October 1990 full committee meeting because the Office of General Council must review the PDA and provide their concurrence. Mr. Miller said the PDA letter would have the condition that final design certification would be based on an FDA review. The draft SER under review by the ACRS is a pre-decisional document and has not been "signed-off" as a final SER. The staff, at this time, was looking for ACRS input.

3. Mr. L. Donatell, NRR, briefed the Subcommittee on the status of open items in the draft SER. The open items summary provided was as follows:

- 17 - Site Specific Issues
- 110 - Information Not in Scope of Application
- 41 - Incomplete Resolution at Close of Review
- 168 - Total

Mr. Donatell went on to state that the PDA is an approval issued under 10 CFR 50 by the NRC deeming a standard preliminary design of a nuclear power plant acceptable for incorporation by reference in individual facility license applications (construction permits and manufacturing licenses only) and providing that the approved design be used and relied on by the staff and the ACRS in their reviews of any such applications. Under 10 CFR 52, the PDA is optional and is not a prerequisite for a final design approval or design certification.

He also stated that a decision on issuance would probably go to the Commission for approval. However, sign-off would come from the Office Director Level, NRR.

Mr. Donatell continued his presentation with a comparison of the PDA versus FDA with respect to design certification, and

a discussion of 10 CFR Part 52, Appendix O, 10 CFR 50.54(f), Conditions of Licenses, and 10 CFR 50.109, Backfitting.

Mr. Donatelli stated that, to date, the ACRS ALWR PWR Subcommittee has discussed with the staff all of the Chapters of the SER.

4. Mr. E. Burns, W, briefed the subcommittee on W's perspective of the RESAR-SP-90 design, the current status, and plans for the future.

Mr. Burns, in response to Mr. Carroll's question regarding the existence of errors of fact in the RESAR/SP-90 design document, stated that clarifications may be needed. W has the same issue on their plate as the NRC staff as to whether to correct these errors or let them remain. Both want to get the SP-90 off their desks. Mr. Burns also stated that a lot of the information provided in the RESAR-SP-90 goes beyond what is needed for a PDA review but some areas of RESAR-SP-90 do not go far enough. Also new issues were added over the years. He stated that the W embarkment on the AP-600 would be on more solid footing if the SP-90 PDA issue was "put to bed."

Mr. Burns' slide presentation included a discussion of the history of the RESAR-SP-90 Licensing Program. He stated the program was started in 1982 and was originally tied to the

Japanese APWR Program. W expects a return of the international and domestic market for large PWRs and wants a NRC approval of the key SP-90 licensing issues. The Japanese current program was stated as being held up by site selection issues.

Mr. Burns further described the RESAR-SP-90 Licensing Program to include the following:

- Comply With Regulatory Requirements Involving Post-TMI (10 CFR 50.34 (f))
- Address Unresolved Safety Issues, Generic Safety Issues as Applicable
- Perform Deterministic Analysis Note: A description of what this was, was not provided in the transcript. However, per telephone conversation with W on October 25, 1990, these analyses were stated as those normally performed for FSAR Chapter 15 accidents and transients including those performed for safe shutdown during design basis events.
- Complete Probabilistic Risk Assessment
 - PDA addressed internal events only
 - FDA to include external events

- Elimination of Pipe Breaks - Mechanistic Pipe Break Concept. This was stated as including certain steam and feed line applications. This concept is not yet fully investigated by W.
- Elimination of Spray Additive from the Containment Spray System
- PRA Applications
 - System/Component selection
 - Failure Mode and Effects Analysis (FEMA)
 - Value Engineering
- Reduction in Operating Basis Earthquake Requirement

Mr. Burns stated that the PDA would provide resolution for some of the Severe Accident Issues in conjunction with the staff's evaluation of the EPRI Utility Requirements Document. The benefits of a PDA issuance from W's perspective include:

- Preserve expenditure of effort invested
- Formalizes agreement reached to date
- Supports U.S. leadership in licensing
- Will be utilized in AP-600 design certification
- Allows NRC staff to focus on Part 52 implementation (i.e., the AP-600 passive design)

Mr. Burns, in response to a question from Mr. Carroll, stated that he believes there is approximately a 90 percent

compliance between the SP-90 design and the EPRI Requirements Document.

In response to Mr. Michelson, Mr. Burns noted that all plant cabinets containing class 1E equipment were designed to have fire monitoring which, in some manner, isolates power to the cabinet in the event of fire monitoring instrument actuation.

5. Mr. T. Van De Venne, W, briefed the subcommittee on the evolutionary light water reactor certification issues. He stated that the goals for the SP-90 design included a core melt frequency of 1×10^{-5} /yr. and a severe release frequency of 1×10^{-6} /yr. "Traditional" source terms were used in the SP-90 dose calculations for Design Basis Accidents. He also discussed SP-90 design considerations in terms of Anticipated Transients Without Scram (ATWS), Mid-Loop Operations, Station Blackout, Fire Protection, and Intersystem LOCA's.

Mr. T. Van De Venne provided summary information and diagrams for Residual Heat Removal (RHR) isolation valve leak testing capabilities, and the Integrated Safeguards System. He also discussed the SP-90 design with respect to hydrogen generation and control, core-concrete interactions, high pressure core melt ejection, containment performance, equipment survivability and the OBE/SSE.

The SP-90 design includes dc powered hydrogen ignitors which maintain post accident hydrogen concentrations in containment below 10 percent. Mr. T. Van De Venne also noted that no specific containment performance goal is included in the SP-90 PDA application. However, containment failure is not expected within 24 hours for the vast majority of severe accident sequences.

To conclude his presentation, Mr. T. Van De Venne discussed the SP-90 design features for In-Service Testing of Pumps and Valves which, as provided, include the capability to perform full flow testing of all safety related pumps at any time.

6. As a result of the Subcommittee's discussion, some of the Subcommittee's members expressed their concerns and comments with regard to the following:

- Mr. Carroll wanted clarification of what the staff expected from the ACRS in terms of a letter since the SER was incomplete with 168 open items.
- Mr. Michelson stated that a letter from the ACRS at this point might be open in nature since the document under consideration is speculative in terms of its use for

design certification. He also stated that the two-step licensing process should result in the same plant design as would result from the PDA/FDA design certification approach.

- Mr. Carroll noted, at the staff's offering of a draft PDA letter, that a draft PDA document provided to the Subcommittee would greatly assist the ACRS in writing a letter on the subject.
- Mr. Carroll expressed concern, that in many places in the SER, the staff concluded that the design is acceptable for the PDA stage of review. The PDA was stated as not being a design certification effort but more of a feedback to W on the SP-90 design. Mr. Michelson shared the same sentiment, as did other Subcommittee Members.
- Mr. Carroll commented that if errors of fact exist as written in the design/SER, then corrections should be made to the document so that a shelf copy used years from now would be accurate.
- Dr. Kerr expressed his belief that the SP-90 draft SER is not up-to-date, is obsolete, and should not be issued in any form. Mr. Kerr stated many examples of out-of-

date designs/approaches exist in the SER and he cited the SP-90 integrated control system design/approval as one such example.

- Dr. Shewmon commented that if a plant were build today based on the criteria cited in the SER for the SP-90 reactor vessel that it would be minimally acceptable.
- Dr. Catton pointed out the SER referenced Appendix K analysis criteria as opposed to Best-Estimate methodology for thermal hydraulic analyses. W shared the same sentiment and said they would use Best-Estimate methodologies today even though the SER refers to Appendix K.
- Mr. Michelson commented regarding the location of the Emergency Diesel Generators and their proximity to the main control room in the SP-90 proposed design.
- Mr. Michelson expressed concern with regard to inadequate fire protection separation in cable spreading areas, the control room, and other places in the plant. He stated the SER was ambiguous in the Fire Protection area. He pointed out that in one instance there is a requirement for a 3 hour fire barrier for Safe Shutdown (SSD)

equipment while another section of the SER specified a 3 feet horizontal and 5 feet vertical requirement for SSD cable trays.

- Mr. Michelson read into the "record" quoting from the SER the conclusion which states, "Westinghouse will be required to provide additional technical or design information as may be required to complete the safety evaluation during the FDA stage of review." The staff concludes that, "open issues identified throughout this report are not of a nature as to prevent issuance of a PDA since they can be resolved during the FDA stage of review without significantly affecting the current plant design." Mr. Michelson did not agree with the staff's conclusion.

- Mr. Michelson questioned W with regard to their apparent use of a common building ventilation system. He pointed out that the staff's description of a common HVAC system for redundant trains is different than what W has presented. W stated that there are two dedicated ventilation systems per train. The reactor external ventilation system handles the ECCS pump areas below the containment sphere and isolation between trains would be provided by redundant dampers. The staff indicated that

this is not what they understood the design to be.

- Mr. Carroll expressed his belief that the examples of technical ambiguity and obsolescence, leads him to feel that W should be given the opportunity to review the draft SER in its current form to effect correctness of content.
- Dr. Shewmon inquired about the vessel wall fluence. He pointed out that a vessel wall irradiation of 1.4×10^{19} neutron/cm² for a 60 year life was presented by W while 1.0×10^{19} neutron/cm² for 40 years (80% usage) was written in the SER (p. 4-11) and that there is a difference between the sources. W stated that the 1.0×10^{19} neutron/cm² for a 60 year life was their design goal. W had not yet seen the staff's number used in the SER.
- Mr. Carroll stated that he expected more discussion in the SER on the new Water Displacement Rods concept used in the SP-90 design.
- Dr. Kerr asked the staff if they made any use of NUREG-1150 during their review of the SP-90 design. The staff could not answer the question but made a note of the

question.

Future Action

The Subcommittee Chairman and members have decided to review a draft PDA letter when it becomes available from the staff. The staff would probably be able to support the subcommittee in time for a November 1990 full committee meeting/letter writing session on the subject PDA. The staff and W would make a final presentation to the ACRS at that time.

- For the next meeting, Mr. Carroll requested a slide from W illustrating the differences between the EPRI Requirements Document and the SP-90 design features. Mr. Carroll also requested the ACRS staff to develop information on the use of alternating current (AC) and direct current (DC) hydrogen ignitors in U.S. nuclear plants.

- The Subcommittee Chairman will brief the full committee during the October 1990 meeting on the current status of the SP-90 review.

NOTE: Additional meeting details can be obtained from a public 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 and Associates, Ltd., 1612 K Street, N.W., Suite 300, Washington, D.C. 20006, (202) 293-3950.

Documents Used

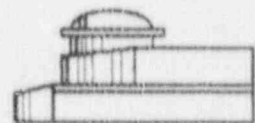
1. Westinghouse SP-90
2. Draft SER on W SP-90 design
3. The slide presentation packages used during the meeting are available from the ACRS.

**ACRS SUBCOMMITTEE ON
ADVANCED PWR'S**

Review of Westinghouse RESAR-SP-90

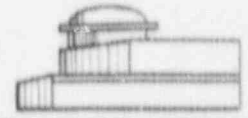
September 20, 1990

Bethesda, Maryland

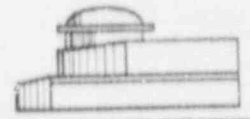




RESAR-SP/90 LICENSING PROGRAM



- Licensing Program Started in 1982
- Design effort was originally tied to the Japanese APWR Program
- Purpose
 - 1) Position Westinghouse for expected market return
 - 2) Obtain NRC approval of key licensing initiatives
- Result in a Preliminary Design Approval (PDA)
 - Safety Evaluation Report

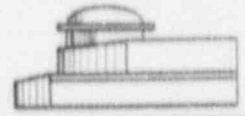


JAPANESE APWR PROGRAM

- 7-Party contract signed August 1982
- \$150M total development costs shared by Japanese utilities, Japanese Government (MITI), MHI and Westinghouse
- Westinghouse/MHI completed the intermediate design for a total plant (including verification testing of major components) in 1987
- Kansai Electric Power announced intent to build first APWR plant in 1988
- APWR site investigations are underway but progress has been adversely impacted following Chernobyl
- Schedule Estimates
 - Final engineering 3-4 years
 - Plant construction 4-5 years



RESAR-SP/90 LICENSING PROGRAM



- **Enhanced Plant Safety**
 - Higher design margins
 - Improved transient performance
 - More reliable safety systems

- **Improved Plant Performance**
 - 90% Average plant availability
 - Higher degree of automation
 - Improved load follow capability
 - Reduced occupational radiation exposure

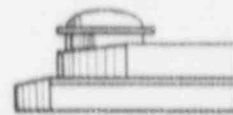
- **Lessons Learned From Existing Designs**
 - Fresh approach to plant layout to minimize effect of common mode failure due to flood, fire and sabotage
 - Separation of safety & normal operation systems

- **Responsive to Current/Future Regulatory Requirements**
 - Unresolved/Generic Safety Issues
 - Severe accident considerations

- **Lower Generating Cost**
 - Plant capital cost
 - Fuel cycle cost



RESAR-SP/90 LICENSING PROGRAM



- **Modular SAR Organized by Systems**
 - Reactor System
 - Primary Safeguards
 - Secondary Safeguards
 - I&C/Control Room
 - Radiation Protection
 - Containment Systems
 - Probabilistic Risk Assessment



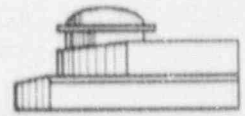
RESAR-SP/90 LICENSING PROGRAM



- Comply With Regulatory Requirements Involving Post-TMI (10 CFR 50.34 (f))
- Address Unresolved Safety Issues, Generic Safety Issues as Applicable
- Perform Deterministic Analyses
- Complete Probabilistic Risk Assessment
 - PDA addressed internal events only
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- Elimination of Pipe Breaks - Mechanistic Pipe Break Concept
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- PRA Application
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 - Failure Mode and Effects Analysis (FMEA)
 - Value Engineering
- Reduction in Operating Basis Earthquake Requirement



RESAR-SP/90 LICENSING PROGRAM



- e **NRC Draft Safety Evaluation Reports**
 - PRA Front End** **PDA Open Issue #107**
(March 1988)

 - Auxiliary Review** **7 Open Issues**
(June 1988)

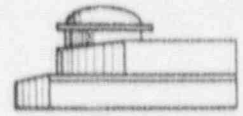
 - Systems Review** **40 PDA Open Issues**
 - **Plant/Reactor/Auxiliary Systems**
 - (March 1989)** **41 PDA Open Issues**
 - **Structural/Mechanical Systems**
 - 26 PDA Open Issues**
 - **Transient Analyses/Single Failure**

 - PRA Back End** **Not issued (Included in "Final" SER)**

 - USIs/GSIs** **Not issued (Included in "Final" SER)**



RESAR-SP/90 LICENSING PROGRAM

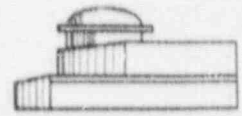


• ACRS/Westinghouse Meetings

- 3/23/82 Subcommittee on Safeguards and Security (Albuquerque)
- 5/5/83 Westinghouse Subcommittee
- 8/10/83 Westinghouse Subcommittee
- 9/25/83 Westinghouse Subcommittee
- 11/6/87 Full-committee - 12 Recommendations (Severe Accident Issues)
- 4/6/88 Advanced PWR Subcommittee - Review of DRAFT SER on Probabilistic Safety Study
- 9/28/89 Advanced PWR Subcommittee - Review of DRAFT SER (Severe Accident Issues)
- 11/3/89 Advanced PWR Subcommittee - Review of DRAFT SER (Chapters 3, 4, 5, 6 & 8)
- 1/10/90 Advanced PWR Subcommittee - Review of DRAFT SER (Chapters 7 & 18)
- 3/6/90 Advanced PWR Subcommittee - Review of DRAFT SER (Chapters 9, 10, 11, 12 and 15)
- 9/20/90 Advanced PWR Subcommittee - Review of Final SER Issues
- 10/5/90 Full Committee



RESAR-SP/90 LICENSING PROGRAM



- **Westinghouse Perspective of PDA**
 - Documents the review that has been completed
 - "Preliminary" NRC Staff safety evaluation of design features
 - "Preliminary" NRC Staff safety evaluation of safety analyses
 - Provides for resolution of some "Severe Accident Issues" after EPRI Utility Requirements Document safety evaluation

- **Benefits of PDA Completion**
 - Preserves expenditure of effort invested
 - Formalizes agreements reached to date
 - Supports US leadership in licensing
 - Will be utilized in AP600 design certification
 - Allows NRC Staff to focus on Part 52 implementation

- **Westinghouse Standard Design Programs Are Responsive to Industry Needs**
 - RESAR-SP/90 and APWR-1000 address near-term need for evolutionary plants, primarily international
 - AP600 is responsive to U.S. market for smaller, simpler plant designs
 - EPRI/Utility Requirements program will establish resolution of generic issues



SP/90 POSITION
ON
EVOLUTIONARY LWR CERTIFICATION ISSUES

T. VAN DE VENNE

PRESENTATION TO THE ACRS SUBCOMMITTEE ON
ADVANCED PRESSURIZED WATER REACTORS

SEPTEMBER 20, 1990



ALWR PUBLIC SAFETY GOAL

- o THE SP/90 PLANT IS DESIGNED TO MEET THE GOALS STATED IN THE "ALWR REQUIREMENTS DOCUMENT"

- . CORE MELT FREQUENCY 1 X 10⁻⁵/YR
- . SEVERE RELEASE FREQUENCY 1 X 10⁻⁶/YR

- o THE PROBABILISTIC SAFETY STUDY FOR INTERNAL EVENTS INCLUDED IN THE SP/90 PDA SUBMITTAL PROVIDES A HIGH DEGREE OF ASSURANCE THAT THESE GOALS WILL BE MET.



SOURCE TERM

- o "TRADITIONAL" SOURCE TERMS HAVE BEEN USED
IN SP/90 DOSE CALCULATIONS FOR DESIGN
BASIS ACCIDENTS.



ATWS

- o THE RELIABILITY OF THE PROTECTION SYSTEM AND THE REACTOR TRIP BREAKERS HAS BEEN IMPROVED RELATIVE TO CURRENT PLANTS.

- o A CONTROL BOARD MANUAL TRIP OF THE ROD CONTROL SYSTEM MOTOR GENERATOR SETS HAS BEEN ADDED.

- o THE VERY NEGATIVE MTC ASSOCIATED WITH THE SP/90 SPECTRAL SHIFT CORE PREVENTS RCS FAILURE DUE TO OVERPRESSURIZATION EXCEPT DURING THE FIRST FEW WEEKS OF CYCLE 1.

- o A DIVERSE ATWS MITIGATION SYSTEM GENERATES TURBINE TRIP AND EMERGENCY FEEDWATER SIGNALS INDEPENDENT OF THE PROTECTION SYSTEM SIMILAR TO CURRENT PLANTS.

- o A DIVERSE SCRAM SYSTEM IS NOT INCLUDED IN THE SP/90 PLANT BECAUSE IT WOULD NOT MEASURABLY REDUCE THE RISK FROM ATWS EVENTS.



MID-LOOP OPERATION

- o THE SP/90 DESIGN INCLUDES FOUR RHR SUBSYSTEMS, EACH OF WHICH IS CAPABLE OF REMOVING DECAY HEAT DURING MID-LOOP OPERATION.
- o WATER LEVEL DURING MID-LOOP OPERATION IS AT LEAST 9 INCHES ABOVE ACTUAL MID-PLANE ELEVATION.
- o WITH VORTEX BREAKER, AIR ENTRAINMENT STARTS TO OCCUR AT APPROXIMATELY 3 INCHES BELOW MID-PLANE ELEVATION, BUT IS LIMITED TO LESS THAN 10%.
- o RHR SUCTION LINES ARE SLOPED CONTINUOUSLY DOWNWARDS TOWARDS RHR PUMPS AND ARE, THEREFORE, SELF-VENTING.
- o RHR PUMP SUCTION LINES PROVIDE ADEQUATE PUMP NPSH AT FULL FLOW ASSUMING SATURATION IN THE HOT LEG.
- o HHSI PUMP WILL BE AVAILABLE DURING MID-LOOP OPERATION FOR EMERGENCY MAKEUP IF REQUIRED.



MID-LOOP OPERATION

- o DEDICATED, REDUNDANT NARROW RANGE LEVEL INSTRUMENTS WITH MCR INDICATION AND ALARM ARE PROVIDED.
- o RANGE OF 'COLD' PRESSURIZER LEVEL INSTRUMENTATION HAS BEEN EXPANDED TO THE BOTTOM OF THE HOT LEG.
- o EACH OF THE FOUR REDUNDANT ISS SUBSYSTEMS INCLUDES RHR FLOW MEASUREMENT AND MAIN CONTROL ROOM INDICATION.
- o REDUNDANT IN-CORE THERMOCOUPLES WILL BE AVAILABLE TO MEASURE CORE EXIT TEMPERATURE DURING MID-LOOP OPERATION.
- o ALL MID-LOOP OPERATIONS CAN BE PERFORMED FROM THE MCR USING THE NORMAL RHR AND CVCS FUNCTIONS.

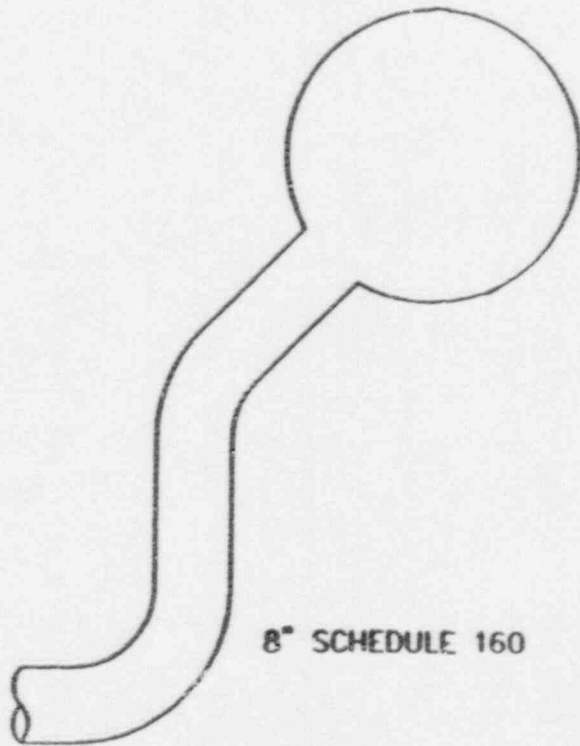
HOT LEG RHR CONNECTION



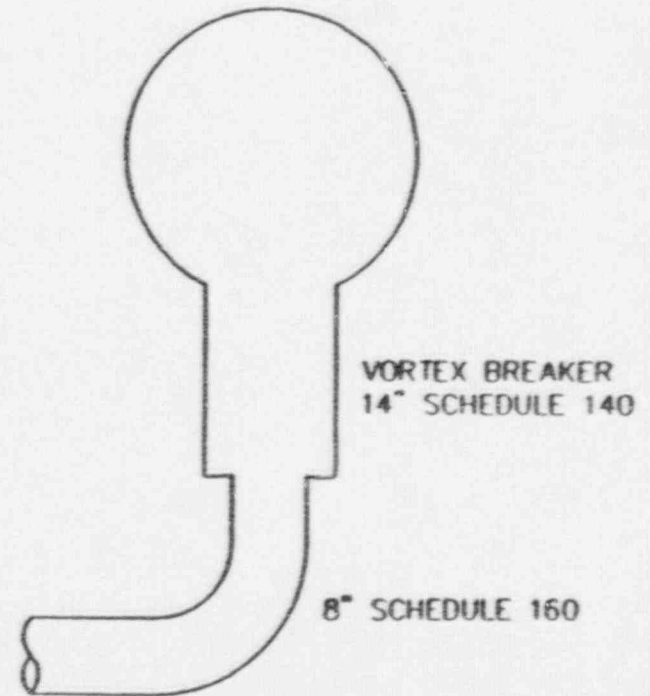
PRESENT SP/90

PDA COMMITMENT

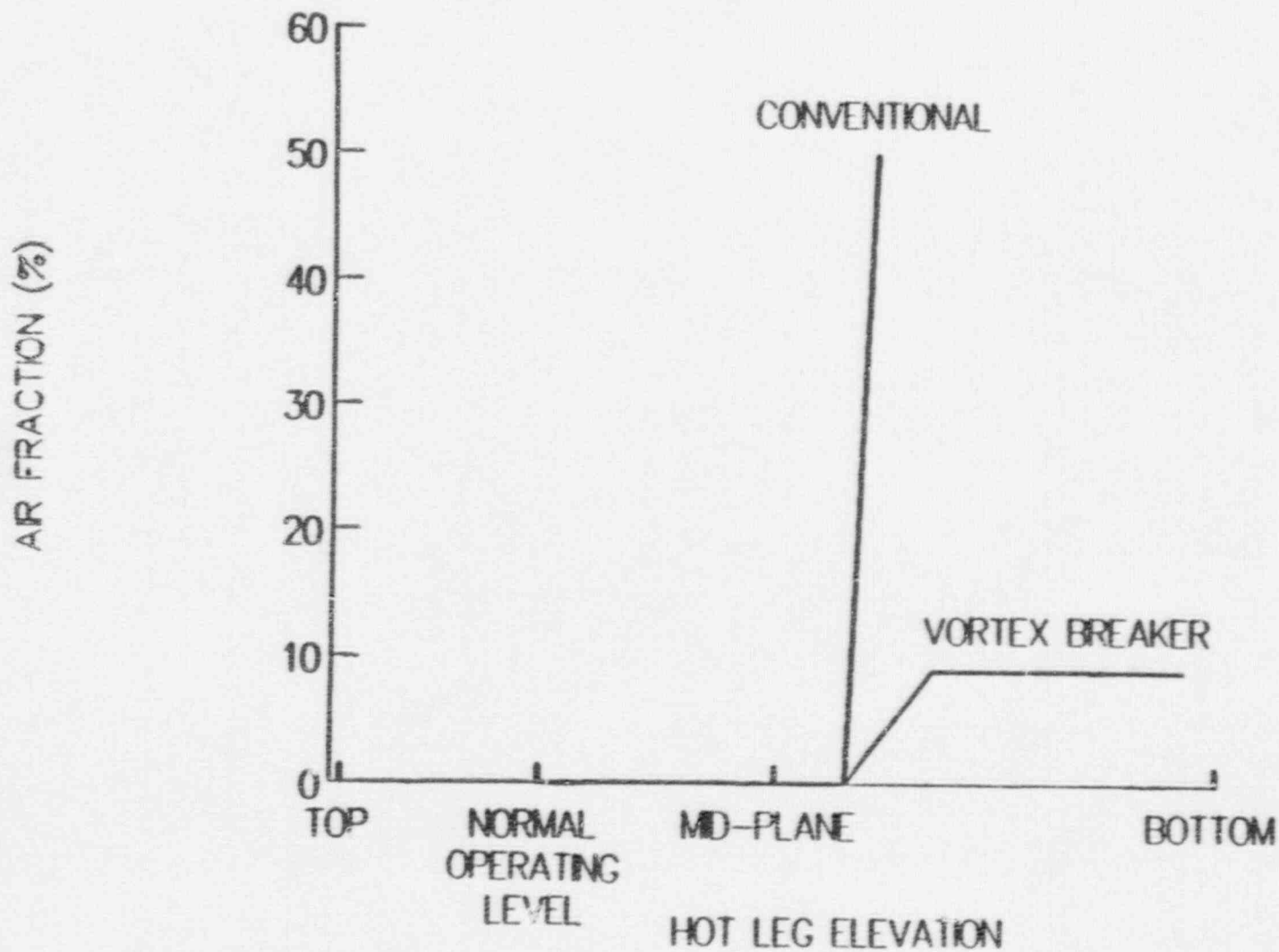
HOT LEG



HOT LEG



RHR SUCTION NOZZLE PERFORMANCE





STATION BLACKOUT

- o FULL LOAD REJECTION CAPABILITY IS INCLUDED.
- o THE ON-SITE POWER SUPPLY CONTAINS AN ALTERNATE AC POWER SOURCE (AAC) WITH SUFFICIENT CAPACITY TO BRING THE PLANT TO COLD SHUTDOWN.
- o THE EMERGENCY FEEDWATER SYSTEM INCLUDES TWO AC/DC INDEPENDENT TURBINE-DRIVEN PUMPS.
- o THE REACTOR COOLANT PUMP SEALS HAVE BEEN UPGRADED SUCH THAT LEAKAGE UNDER STATION BLACKOUT CONDITIONS IS REDUCED BY MORE THAN AN ORDER OF MAGNITUDE RELATIVE TO CURRENT DESIGNS.
- o THE CLASS 1E BATTERIES ARE DESIGNED FOR EIGHT HOURS OF OPERATION UNDER STATION BLACKOUT CONDITIONS.



STATION BLACKOUT

- o SOME OF THE STATION BLACKOUT DESIGN FEATURES ARE NOT ACCOUNTED FOR IN THE SP/90 PROBABILISTIC SAFETY STUDY.

- o EFFORTS ARE UNDERWAY TO EVALUATE THE IMPACT OF THESE FEATURES; PRELIMINARY INDICATIONS ARE THAT THE CORE MELT FREQUENCY CONTRIBUTIONS FROM LOSS OF OFFSITE POWER ($8.1 \times 10^{-7}/\text{YR}$) AND LOSS OF COOLING ($3.4 \times 10^{-7}/\text{YR}$) WILL BE REDUCED BY A FACTOR OF 3 TO 4.

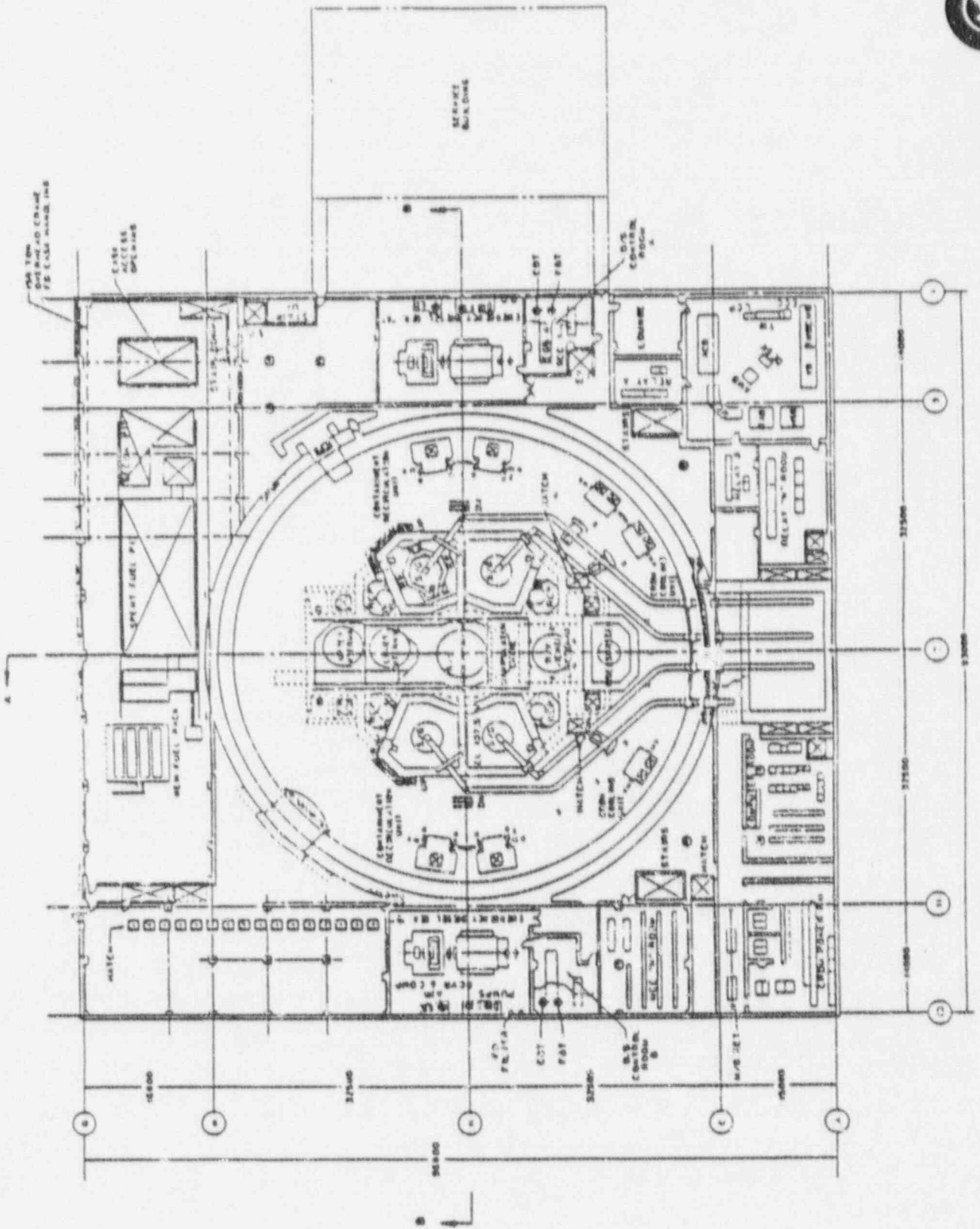
- o A LARGER REDUCTION IN SEVERE RELEASE FREQUENCY IS EXPECTED BECAUSE THE MAJORITY OF COREMELTS FROM THESE EVENTS WILL NOW BE "LATE" RATHER THAN "EARLY."



FIRE PROTECTION
(OUTSIDE CONTAINMENT)

- OUTSIDE CONTAINMENT, 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 REDUNDANT SAFETY AREA IS PROVIDED WITH SEPARATE VENTILATION SYSTEMS.
- THE THREE HOUR FIRE BARRIER BETWEEN THE REDUNDANT SAFETY AREAS IS INTERRUPTED BY PERSONNEL CORRIDORS AND A LIMITED NUMBER OF FIBER-OPTIC DATA LINKS.
- THE MAIN CONTROL ROOM AND MAIN STEAM TUNNEL ARE EXCEPTIONS TO THE ABOVE SEPARATION ARRANGEMENT.

APWR REACTOR BUILDING



ET. 100.0 m



FIRE PROTECTION
MAIN CONTROL ROOM

- o FUNCTIONALITY REQUIREMENTS DICTATE THAT THE MCR BE A SINGLE FIRE AREA.

- o FEATURES ARE INCLUDED IN THE MCR TO
 - . REDUCE THE PROBABILITY OF FIRE INITIATION
 - . REDUCE THE PROBABILITY OF FIRE SPREADING
 - . INCREASE THE PROBABILITY OF FIRE DETECTION
 - . MITIGATE THE EFFECTS OF A FIRE

- o IN CASE OF MCR EVACUATION THE PLANT CAN BE BROUGHT TO COLD SHUTDOWN FROM THE EMERGENCY PANELS.



FIRE PROTECTION
MAIN STEAM TUNNEL

- o MST CONTAINS SAFETY RELATED PORTIONS OF MAIN STEAM AND FEEDWATER LINES.
- o EQUIPMENT IS MOSTLY NOT DIVISION ORIENTED.
- o MST DOES NOT CONTAIN FLAMMABLE MATERIALS.
- o HIGH ENERGY LINE BREAK CONSIDERATIONS MAKE COMPARTMENTALIZATION UNDESIRABLE.
- o EQUIPMENT THAT CAN BE USED FOR SAFE SHUTDOWN IS BACKED UP BY OTHER EQUIPMENT OUTSIDE MST.
- o IN ACCORDANCE WITH DEFENSE-IN-DEPTH PRINCIPLES, SAFE SHUTDOWN EQUIPMENT WILL BE ARRANGED TO THE EXTENT PRACTICAL TO MAINTAIN ONE DIVISION FREE OF FIRE DAMAGE.



FIRE PROTECTION

DESIGN FEATURES (INSIDE CONTAINMENT)

- o CONTAINMENT CONSTITUTES A SINGLE FIRE AREA.

- o 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.

- o FIRE ZONE SEPARATION IS PREFERENTIALLY 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).

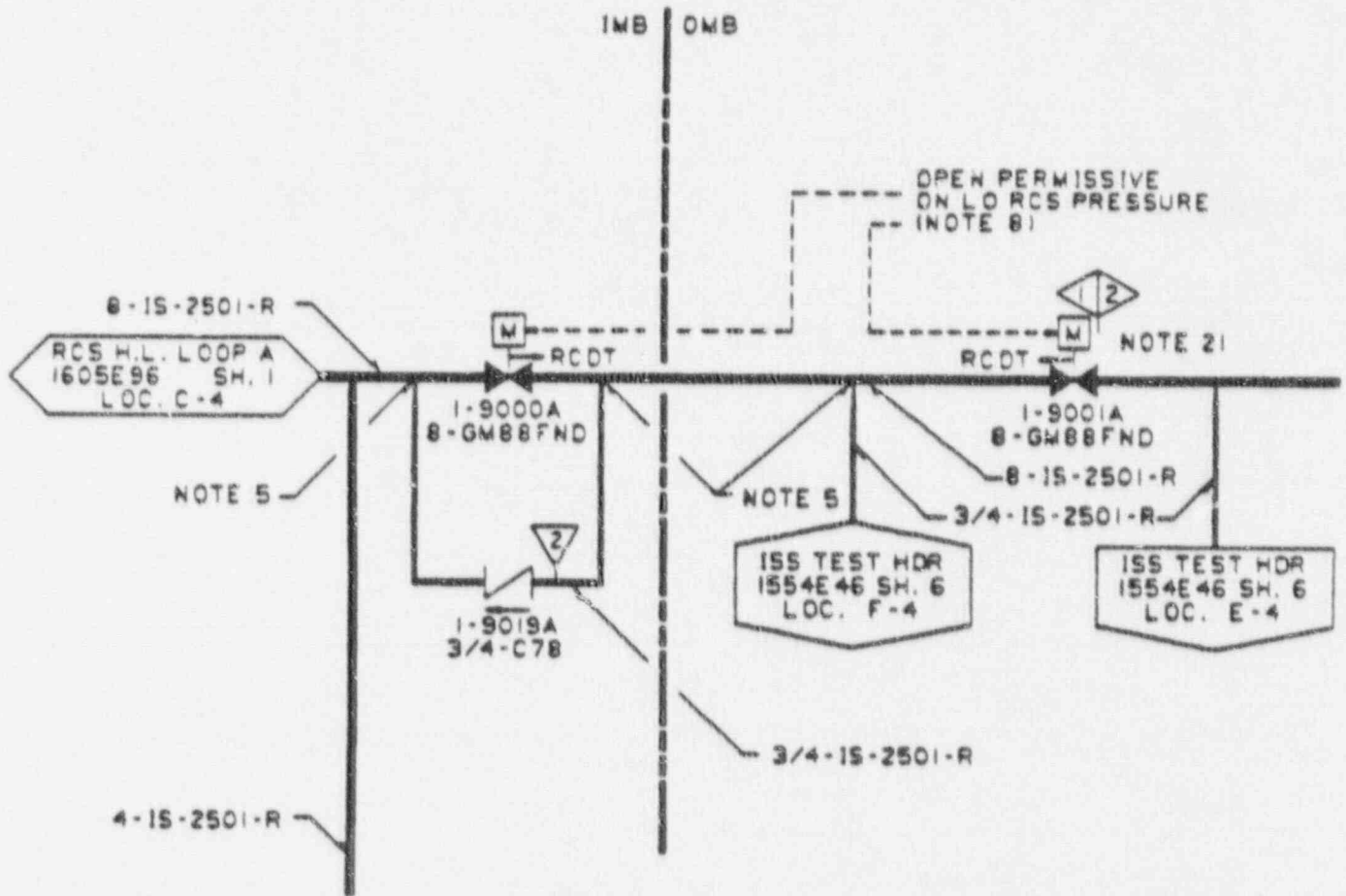


INTERSYSTEM LOCA

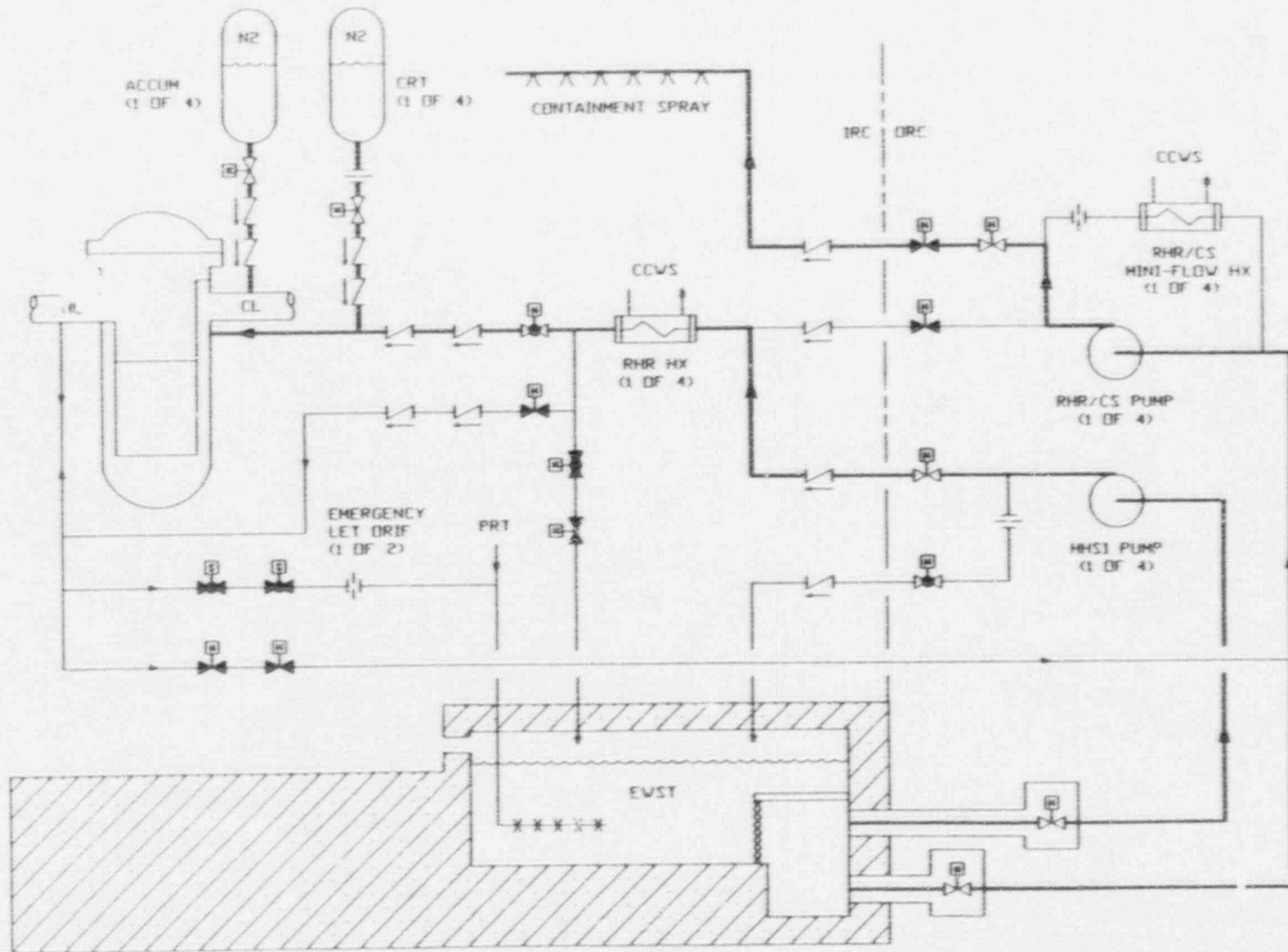
- o RHR ISOLATION VALVES ARE INCLUDED IN ISS TEST HEADER AND WILL BE LEAK TESTED DURING STARTUP.
- o RHR SUCTION PIPING DESIGN PRESSURE HAS BEEN INCREASED SUCH THAT GROSS FAILURE WILL NOT OCCUR EVEN WHEN SUBJECTED TO RCS OPERATING PRESSURE.
- o RHR SUCTION PIPING IS IN OPEN CONNECTION WITH THE IN-CONTAINMENT EWST SUCH THAT PRESSURE IS RELIEVED FOLLOWING LEAKAGE THROUGH RHR ISOLATION VALVES.
- o RHR PUMPS AND PIPING ARE ARRANGED TO ASSURE SUFFICIENT EWST INVENTORY TO ALLOW CONTINUED CORE COOLING WITH NON-AFFECTED ISS SUBSYSTEMS.
- o OTHER RCS CONNECTIONS ARE MUCH LESS LIKELY TO LEAD TO AN INTERSYSTEM LOCA AND WOULD MOREOVER RESULT IN SMALLER CONSEQUENCES.

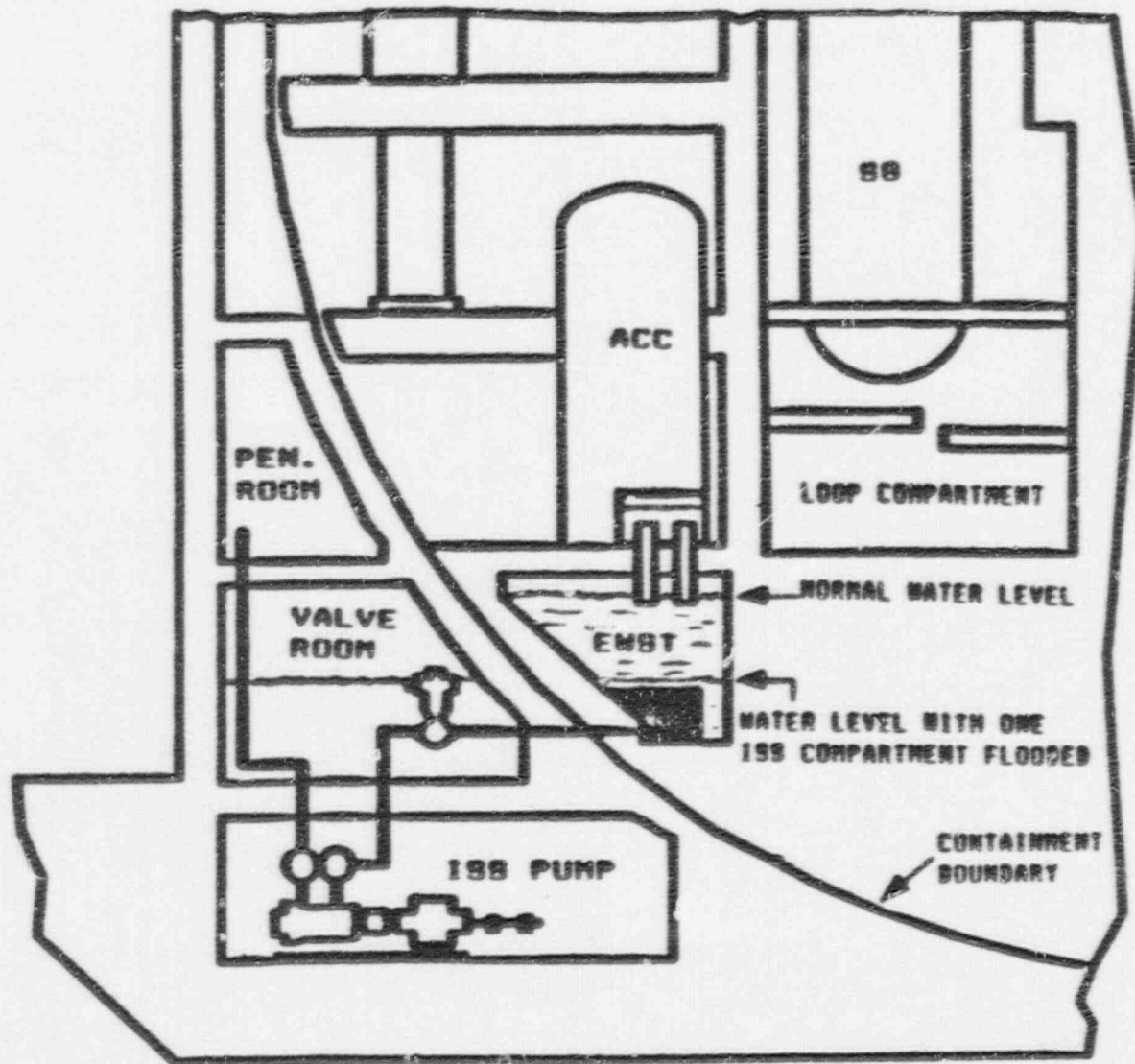


RHR ISOLATION VALVES LEAK TESTING CAPABILITY



APWR - INTEGRATED SAFEGUARDS SYSTEM





INTEGRATED SAFEGUARDS SYSTEM COMPARTMENT ARRANGEMENT





HYDROGEN GENERATION AND CONTROL

- o IGNITERS ARE PROVIDED IN ORDER TO MAINTAIN THE HYDROGEN CONCENTRATION BELOW 10 PERCENT ASSUMING A REACTION OF WATER WITH 100 PERCENT OF THE CLADDING SURROUNDING THE ACTIVE FUEL.



CORE-CONCRETE INTERACTION

- o THE REACTOR CAVITY FLOOR AREA IS APPROXIMATELY EQUAL TO 0.02 M²/MWTM.

- o THE CONTENTS OF THE IN-CONTAINMENT EMERGENCY WATER STORAGE TANK CAN BE DRAINED TO THE REACTOR CAVITY BY REMOTE OPERATOR ACTION.



HIGH PRESSURE CORE MELT EJECTION

- o THE REACTOR CAVITY LAYOUT IS DESIGNED TO MAXIMIZE THE PROBABILITY OF CONTAINING CORE DEBRIS.

- o AC INDEPENDENT RCS DEPRESSURIZATION IS PROVIDED BY THE SAFETY GRADE PRESSURIZER POWER OPERATED RELIEF VALVES.



CONTAINMENT PERFORMANCE

- o NO SPECIFIC CONTAINMENT PERFORMANCE GOAL IS INCLUDED IN SP/90 PDA APPLICATION.

- o NO CONTAINMENT FAILURE IS EXPECTED WITHIN 24 HOURS FOR THE VAST MAJORITY OF SEVERE ACCIDENT SEQUENCES.

- o IT IS EXPECTED THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY WILL BE LESS THAN 0.1 ASSUMING REASONABLE CREDIT FOR RECOVERY ACTIONS.



EQUIPMENT SURVIVABILITY

- o WESTINGHOUSE AGREES THAT FEATURES PROVIDED IN THE SP/90 PLANT FOR SEVERE ACCIDENT PROTECTION ONLY NEED NOT BE SAFETY GRADE.

- o AN EFFORT IS UNDERWAY TO DEFINE AN APPROPRIATE CLASSIFICATION FOR ALL NUCLEAR PLANT EQUIPMENT; THE RESULTS OF THIS WILL BE SUBMITTED AS PART OF THE AP600 FDA SUBMITTAL.



OBE/SSE

- o WESTINGHOUSE AGREES THAT THE OBE SHOULD NOT GOVERN SEISMIC DESIGN.

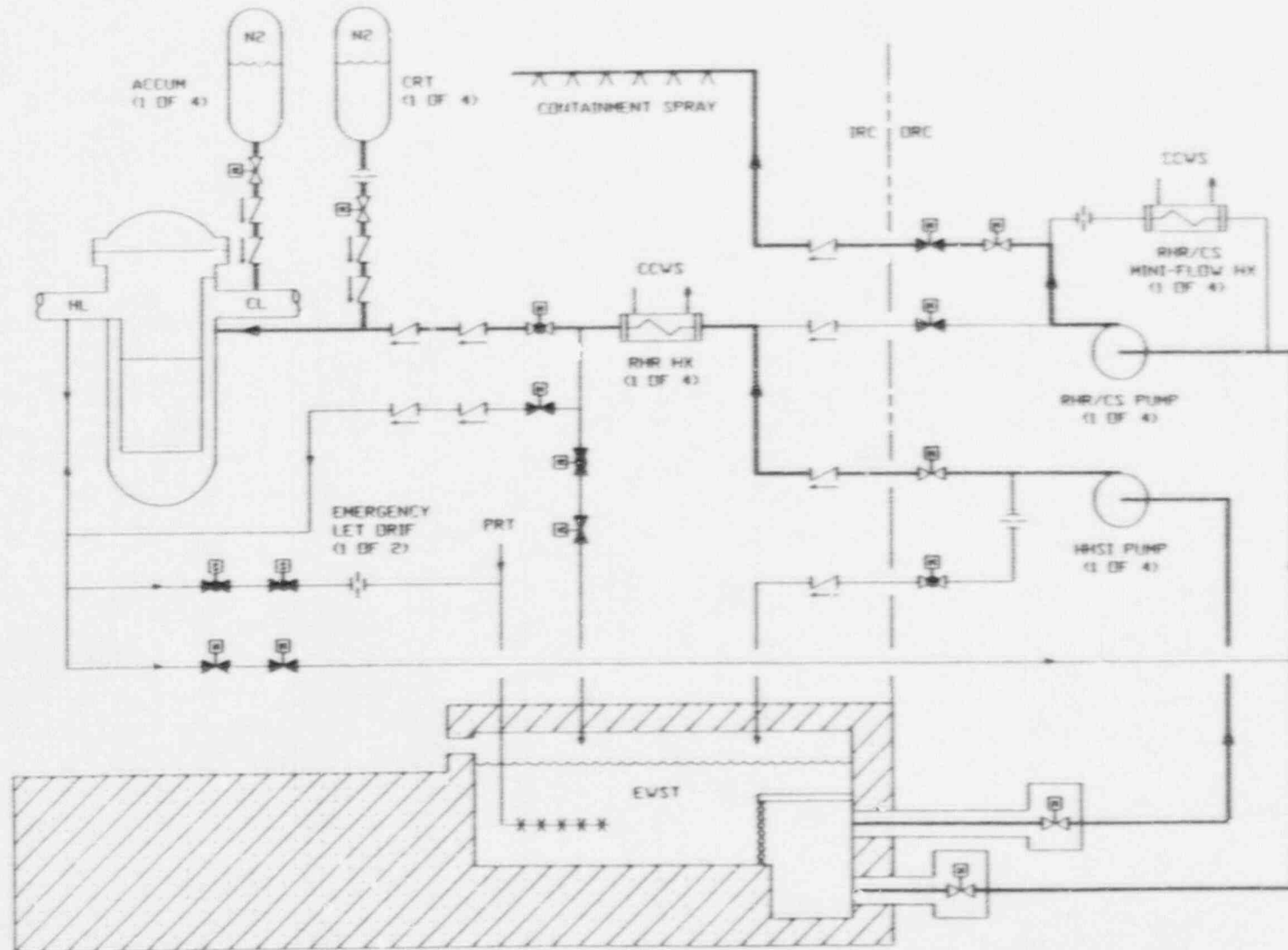
- o THE PROPOSED SP/90 VALUES (0.3 G SSE, 0.1 G OBE) WOULD RESULT IN THE SSE BEING LIMITING.



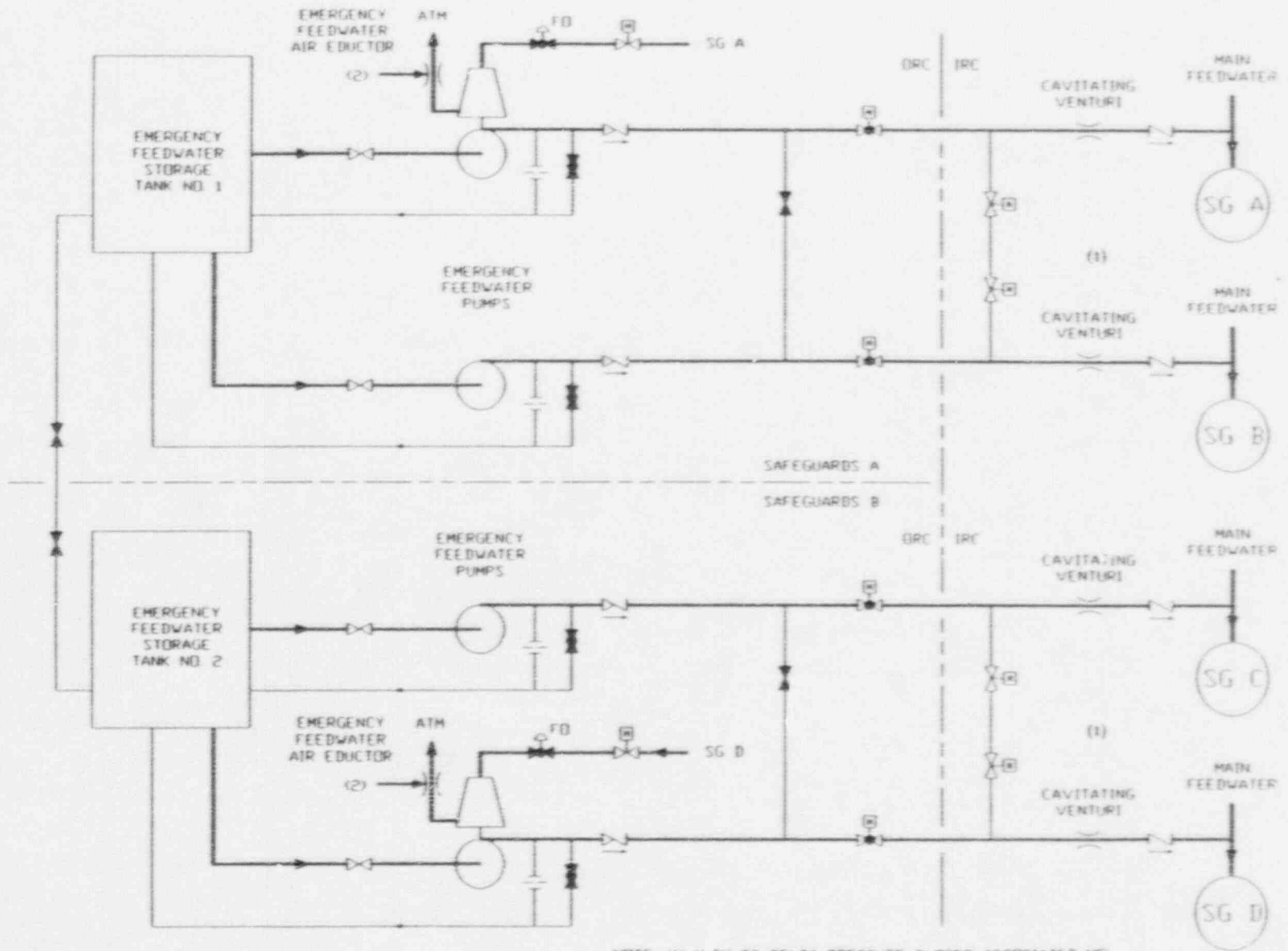
IN-SERVICE TESTING OF PUMPS AND VALVES

- o CAPABILITY IS INCLUDED IN THE DESIGN TO PERFORM FULL FLOW TESTING OF ALL SAFETY RELATED PUMPS AT ANY TIME.
- o THE ECC AND EFW SYSTEMS ARE PERMANENTLY ALIGNED TO PERFORM THEIR SAFETY FUNCTION WITHOUT NEED FOR MOV REALIGNMENT.
- o ALL ECCS AND EFWS CHECK VALVES CAN BE TESTED AT FULL FLOW DURING PLANT SHUTDOWN.
- o A PERMANENTLY INSTALLED SYSTEM IS PROVIDED TO ALLOW LEAK TESTING OF RHR LETDOWN MOV'S AND ECCS INJECTION CHECK VALVES DURING PLANT STARTUP.
- o INSTRUMENTATION TO DETECT EFW CHECK VALVE BACKLEAKAGE IS PROVIDED AND EFW PUMPS HAVE INDIVIDUAL SUCTION LINES TO LIMIT THE EFFECTS OF ANY LEAKAGE.
- o HIGH ΔP MOV'S SHOULD BE QUALIFIED PRIOR TO INSTALLATION SINCE IN-SITU TESTING AT OPERATING CONDITIONS APPEARS TO BE UNDESIRABLE (PRESSURIZER PORV'S, EMERGENCY LETDOWN VALVES).

APWR - INTEGRATED SAFEGUARDS SYSTEM



APWR - EMERGENCY FEEDWATER SYSTEM



NOTE: (1) 14GH SG DELTA PRESSURE CLOSES ASSOCIATED MP
 (2) EDUCTOR REMOVES WARM AIR FROM T/D PUMP A



NRR STAFF PRESENTATION TO THE
ACRS

SUBJECT: RESAR SP/90

DATE: September 20, 1990

PRESENTER: Loren F. Donatell

PRESENTER'S TITLE/BRANCH/DIV.: Project Manager
Standardization Project Directorate
Division of Reactor Projects - III,
IV, V, and Special Projects
Office of Nuclear Reactor Regulation

PRESENTER'S NRC TEL. NO.: (301) 492-1141

SUBCOMMITTEE: Advanced Pressurized Water Reactors

CURRENT REVIEW STATUS

WESTINGHOUSE APPLICATION FOR A PDA	OCTOBER 24, 1983
RESAR SP/90 SUBMITTED	OCTOBER 24, 1983 to MARCH 9, 1987
DSER PRA "FRONTEND"	MARCH 1988
ACRS SUBCOMMITTEE	APRIL 1988
DSER - SRP	JUNE 1988
DSER - SRP	MARCH 1989
WESTINGHOUSE RESPONDED TO OPEN ITEMS	JUNE-SEPTEMBER 1989
ACRS SUBCOMMITTEE	SEPTEMBER 1989
WESTINGHOUSE SUBMITTED AMENDED USIs/GSIs	OCTOBER 1989
ACRS SUBCOMMITTEE	NOVEMBER 1989
ACRS SUBCOMMITTEE	JANUARY 1990
ACRS SUBCOMMITTEE	MARCH 1990
ACRS SUBCOMMITTEE	SEPTEMBER 20, 1990
ACRS FULL COMMITTEE	OCTOBER 5, 1990

OPEN ITEMS

SITE SPECIFIC

17 ITEMS

INFORMATION NOT IN SCOPE OF APPLICATION

110 ITEMS

INCOMPLETE RESOLUTION AT CLOSE OF REVIEW

41 ITEMS

TOTAL

168 ITEMS

COMPLETE PDA REVIEW

items to be accomplished

ACRS SUBCOMMITTEE

SEPTEMBER 1990

Re: FINAL SER

ACRS FULL COMMITTEE

OCTOBER 1990

Re: FINAL SER AND TO REQUEST LETTER

NRC ISSUES FINAL SER

OCTOBER 1990

PDA DECISION

OCTOBER 1990

The preliminary design approval (PDA) is an approval issued by the NRC deeming a standard preliminary design of a nuclear power plant acceptable for incorporation by reference in individual facility license applications (construction permits and manufacturing licenses only) and providing that the approved design be used and relied on by the staff and the Advisory Committee on Reactor Safeguards (ACRS) in their reviews of any such applications. The PDA is optional and is not a prerequisite for a final design approval or design certification.

10CFR52 Appendix O

- o Staff determination of acceptability
 - o Subject to such conditions as may be appropriate

- o Utilized by and relied upon by the staff and the ACRS
 - o Any facility license application referencing an approved design
 - o Unless there exists significant new information which substantially affects the earlier determination or other good cause

- o Information requests regarding an approved design
 - o Evaluated prior to issuance to ensure justified
 - o Approved by the EDO
 - o In accordance with 10CFR50.54(f)

10CFR50.54(f)

(Conditions of Licenses)

- o The licensee upon request of the Commission
 - o Submit written statements under oath or affirmation

- o The Commission will:
 - o Determine if license should be modified, suspended or revoked
 - o Verify licensee compliance with the current licensing basis

- o The reasons for information requests must be developed before issuance
 - o Assure burden on respondents is justified

- o Requests must be approved by the EDO

10CFR50.109

(Backfitting)

- o Applicable after issuance of a design approval
- o Exceptions - 10CFR50.109(a)(4)
 - o Modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee; or
 - o Regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security; or
 - o Regulatory action involves defining or redefining what level of protection to the public health and safety or common defense and security should be regarded as adequate