WESTINGHOUSE CLASS 3

SP/90 POSITION

ON

SEVERE ACCIDENT POLICY ISSUES

MARCH 1990

1. EVOLUTIONARY PUBLIC SAFETY GOALS

Staff Position:

As expressed on several occasions over the past few years (e.g. Severe Accident Policy Statement, Standardization Policy Statement, Safety Goal Policy Statement), the USNRC fully expects that any new standard plant design would result in a higher level of severe accident safety than prior designs. This should be achieved by improving safety and by striking a balance between accident prevention and mitigation.

In parallel with the development of the Commission position, the US utilities under the overall ledership of the Electric Power Research Institute (EPRI) has been preparing requirements for future plants, both evolutionary and passive. As part of this effort, the following top level safety goals have been defined:

- o With regard to prevention, the core damage frequency must be less than 10^{-5} per year.
- o With regard to mitigation, the whole body dose at an assumed 0.5 mile boundary must be less than 25 rem for events whose cumulative frequency exceeds 10^{-6} per year.

From the Staff's February 9, 1990 presentation to the ACRS Full Committee it appears that the Staff has concluded that the EPRI goals are consistent with the Commission's Safety Goal Policy Statement, and intends to measure future standard plant designs against those goals.

Westinghouse Position:

Westinghouse endorses the EPRI Safety Goals and will meet them for the SP/90 design. Internal events have been evaluated for the PDA Application. The results of these evaluations, coupled with results obtained from external event evaluations performed for other projects, provide a high degree of assurance that the current SP/90 design will be able to meet these goals. At the FDA stage, Westinghouse will perform a detailed PRA for both internal and external events to demonstrate compliance with the EPRI goals.

SP/90 Resolution:

The SP/90 is in compliance with the resolution proposed by the Staff for the Public Safety Goal issue for evolutionary plants.

2. CONTAINMENT PERFORMANCE

Staff Position:

The Staff is of the opinion that because there are substantial uncertainties in core damage predictions, and because it is very important to maintain defense in depth, it is necessary that the containment boundary serve as a reliable barrier against fission product release for credible severe accident challenges. In order to provide assurance that is indeed the case, a containment performance criteria has been proposed by the Staff.

The first criterion suggested by the staff was probabilistic in nature and required that the probability of failure of the mitigation systems should not exceed 0.1

An alternate deterministic criterion proposed in the Draft SER for Chapter 5 of the EPRI ALWR Requirements Document stated the "...the containment should maintain its role as a reliable leak tight barrier by ensuring that containment stresses do not exceed ASME service level "C" limits for a minimum period of 24 hours following the onset of core damage". This capability should, to the extent practical, be provided by the passive capability of the containment and any related passive design features. The Staff further believes that following this 24-hour period, the containment should continue to provide a barrier against the uncontrolled release of fission products, but that credit may be taken for operator action in that time frame.

Westinghouse Position:

The SP/90 design includes several features to minimize the potential for large fission product releases in the event of a severe accident. These features are primarily aimed at the prevention and/or mitigation of severe accident phenomena which can threaten containment integrity early in a severe accident scenario, such as direct containment heating, core concrete interactions and hydrogen deflagrations. In addition, the SP/90 design includes containment sprays and safety grade fan coolers to prevent containment pressurization for severe accident sequences in which those systems are available. Thus, as shown in the SP/90 severe accident analyses, early containment failure is not credible for the most probable core melt sequences. The only credible containment failure mode identified for the SP/90 design is long term overpressurization for severe accident sequences in which both of the containment heat removal systems are not available. In these sequences, the containment failure is predicted to occur at times greater than 36 hours after the initiation of the event, assuming that no recovery of containment heat removal can be accomplished in this long time period. Application of Accident Management strategies presently being developed as part of the NUMARC Severe Accident Working Group and EPRI Accident Management programs could be effective in preventing containment failures, given the long time period for overpressurization of the containment to occur.

Because the issue of containment performance had not been raised by the Staff until relatively late in the SP/90 review, no specific commitments have been made. Another consideration is that the issue of containment performance in long-term overpressurization scenarios is presently considered by the USNRC Staff as part of their review of the EPRI ALWR Requirements Document. Westinghouse is of the opinion that EPRI should continue their leadership role in resolving this issue. Westinghouse also commits to meet in the FDA submittal the intent of any new guidelines or criteria that may result from this resolution process.

SP/90 Resolution:

The Staff stated in their February 9, 1990 presentation to the ACRS Full

Committee that, although Westinghouse has not made a specific commitment with regard to containment performance, the design of the SP/90 is such that the Staff expects the SP/90 to meet the conditional containment failure probability goal of 0.1.

3. HYDROGEN GENERATION AND CONTROL

Staff Position:

It is the Staff position that the likelihood of early containment failure from hydrogen combustion should be reduced. This can either be achieved by providing sufficient containment design margin (free volume, pressure capability) to withstand the effects of hydrogen burns and maintain global and local hydrogen concentrations below detonable limits, or by providing for controlled igniting which maintains global and local hydrogen concentrations below detonable limits, and controls hydrogen burning such that containment integrity is maintained. It is further the Staff's position that a metal-water reaction equivalent to 100 percent of the clad in the active core region should be considered, and that the hydrogen concentration should be maintained below 10 percent.

Westinghouse Position:

The SP/90 plant includes igniters to maintain hydrogen concentrations below 10 percent following a Zr-water reaction equivalent to 100 percent of the cladding in the active fuel region. These hydrogen igniters will be powered from Class 1E DC power sources.

SP/90 Resolution:

In their February 9, 1990 presentation to the ACRS Full Committee, the Staff indicated that the SP/90 design complies with the proposed resolution for this issue.

4. CORE-CONCRETE INTERACTION

Staff Position:

Containment integrity could be breached in the event of a severe accident in which the core melts through the reactor vessel, resulting in interaction between core debris and concrete, which can generate large quantities of hydrogen and other gases. It is the Staff's position that sufficient reactor cavity floor space be provided to enhance debris spreading, and that a method for quenching debris in the reactor cavity be incorporated.

Westinghouse Position:

The SP/90 reactor cavity area of approximately 0.02 square meters per core megawatt thermal output will result in a debris bed height below the 25 cm criterion established for a coolable geometry and will preclude core concrete interactions for all severe accidents in which a water layer can be maintained. In order to ensure that this water layer is indeed present, Westinghouse commits to incorporate a drain from the in-containment emergency water storage tank (EWST) to the reactor cavity in the SP/90 design. This alternate water supply will be remotely actuated from Class 1E DC power supplies by manual operator action as specified in emergency response guidelines.

SP/90 Resolution:

In their February 9, 1990 presentation to the ACRS Full Committee, the Staff indicated that the SP/90 design complies with the proposed resolution for this issue.

5. HIGH PRESSURE CORE MELT EJECTION

Staff Position:

Direct containment heating associated with the ejection of molten core debris under high pressure from the reactor vessel can result in a rapid addition of energy to the containment atmosphere. It is the Staff's position that, pending completion of ongoing research, it is prudent to provide protection against this potential failure mode. This protection should include the following two aspects:

- o Providing a rate of RCS depressurization to preclude molten core ejection and creep rupture of steam generator tubes.

- o Arranging the reactor cavity such that high pressure core debris ejection resulting from reactor vessel failure does not impinge on the containment boundary.

<u>Westinghouse Position</u>:

Direct containment heating has been addressed in the SP/90 design in two independent ways: 1) prevention of high pressure melt ejection scenarios at reactor vessel failure, and 2) prevention of melt ejection from beneath the reactor vessel in the unlikely event of failure of the reactor vessel at high pressure.

The reactor cavity region of the SP/90 plant has been designed to preclude transport of significant core debris from the cavity in the unlikely event of a high pressure melt ejection scenario from the reactor vessel. This is a passive feature involving the geometric configuration of the reactor cavity.

As a backup, the SP/90 design includes safety grade, a.c. independent pressurizer power operated relief valves that have the capability to reduce the reactor coolant system pressure to less than 200 psig at reactor vessel failure for severe accident scenarios. The initiation of intentional depressurization will be a manual action which will be included in the plant Emergency Operating Procedures.

<u>SP/90 Resolution</u>:

In their February 9, 1990 presentation to the ACRS Full Committee, the Staff indicated that the SP/90 design complies with the proposed resolution for this issue.

6. ABWR CONTAINMENT VENT

This issue is specific to the BWR and PWR Ice Condenser containments and does not apply to the SP/90 design.

7. SOURCE TERM

Staff Position:

Classical source terms (Regulatory Guides 1.3, 1.4 and 1.5, and SRP Section 15.6.3) have been effective in that they have led to plant designs that incorporate a high degree of public safety. However, the Staff recognizes that these source terms contain significant conservatisms and they intend to revise the source terms at some period in the future.

In the meantime, it is the Staff's position that evolutionary plants must comply with 10 CFR 100; however, the Staff will use engineering judgement to allow deviations from classical source terms and will document such deviations in SER's. No modifications to current siting practices are expected, even though source terms may decrease.

Westinghouse Position:

The SP/90 design complies with 10 CFR 100 without deviations. If source terms were to be modified between now and the point in time at which an SP/90 FDA Application would be submitted, Westinghouse would address such modifications in the FDA submittal.

SP/90 Resolution:

The SP/90 is in compliance with current regulations, which are applicable to evolutionary plants under the resolution proposed by Staff.

8. ANTICIPATED TRANSIENTS WITHOUT SCRAM

Staff Position:

This former Unresolved Safety Issue was resolved with the issuance of Rule 10 CFR 50.62. Requirements for currently operating PWR's included diverse scram (except for Westinghouse plants), and diverse actuation of auxiliary feedwater and turbine trips.

It is the Staff's position that future standard plants should include additional enhancements; in particular, it was stated in the Staff's presentation to the ACRS Full Committee on February 9, 1990 that future Westinghouse standard designs should include a diverse scram.

Westinghouse Position:

The SP/90 plant includes the following design features aimed at minimizing the probability of occurrence of an ATWS, and at mitigating the consequences if it occurs.

- The design of the integrated protection system (IPS) is highly reliable. The IPS is based on two-out-of-four logic throughout and features continuous on-line testing. The system contains "fail-safe" features to the extent practical, i.e., it is designed to generate a reactor trip signal when failures occur.

- The reactor trip switchgear consists of eight circuit breakers arranged in a two-out-of-four matrix and located in two separate cabinets. The trip is implemented by undervoltage trip attachments and shunt trip devices on the circuit breakers. To generate a trip, power is interrupted to the undervoltage trip attachment, and the shunt trip attachment is energized; either device will trip the breaker. The eight breaker configuration permits testing of the reactor trip breakers without the use of auxiliary bypass breakers.

- The reactor trip switchgear can be actuated manually from the main control board via reactor trip switches hard wired to the shunt trip and undervoltage coils on each circuit breaker. In addition, it is possible to trip from the main control board the motor-generator sets that provide power for control rod operation.

- An ATWS mitigating system is included in the SP/90 design to generate turbine trip and emergency feedwater start signals independent (including sensors) from the IPS.

- The moderator temperature coefficient (MTC) is significantly more negative than in the case of current plants, typically by a factor of three to four.

- ATWS considerations will be factored into the design of the pressurizer safety valves during the detailed design phase.

- Detailed analyses of limiting ATWS transients will be performed at the FDA stage to demonstrate that ATWS acceptance criteria are met.

The acceptance criteria, as well as the assumptions to be used in the ATWS analysis, will be agreed upon between the USNRC and Westinghouse as part of the process to develop the Licensing Review Basis (LRB) for the SP/90 plant. If the ATWS analyses to be performed during the FDA stage do not demonstrate compliance with the agreed upon acceptance criteria, Westinghouse will consider additional design features to mitigate ATWS transients, including diverse scram.

The PRA performed for the SP/90, as well as PRA's for other plants, indicates that ATWS contributes in only a minor way to core damage frequency, and that its contribution to severe release frequency is negligible. Considering these results, Westinghouse is of the opinion that the SP/90 adequately addresses ATWS issues and that no additional hardware design features are required.

<u>SP/90 Resolution</u>:

Westinghouse recommends that this remain an open item for the SP/90 PDA to be resolved at the FDA stage.

9. MID-LOOP OPERATION

Staff Position:

Loss of decay heat removal function has occurred on a number of occasions in operating plants. In response to these events, the Staff issued Generic Letter 87-12 requesting operating plants to provide information regarding mid-loop operation. Generic Letter 88-17 requested additional information and provided guidance to operating utilities. Subsequent evaluations by the USNRC have indicated that loss of decay heat removal during mid-loop operation may contribute significantly to public risk.

It is the Staff's position that for future standard plants compliance with Generic Letter 88-17 is insufficient, and that additional hardware features should be incorporated into the design to address this issue.

Westinghouse Position:

To ensure its continued availability to perform the residual heat removal function during mid-loop operation, the following features are incorporated in the design of the reactor coolant system (RCS) and the residual heat removal (RHR) portion of the integrated safeguards system (ISS):

- The layout of the RCS hot leg piping and the steam generator channel head is such that installation of the nozzle dams can be performed with an 80% level in the hot leg piping; this is 9.3 inches above the actual mid-plane elevation.

- With the conventional Westinghouse arrangement of a residual heat removal piping connection at 45% from horizontal, it has been calculated that onset of vortexing with attendant air ingestion would occur at a level 3.0 inches below mid-plane elevation. Therefore, during "mid-loop" operation, a margin in excess of 12 inches would exist between normal operating level and the critical level at which RHR pump operation may be impaired due to high levels of air entrainment. While this is a significant improvement relative to current plants, Westinghouse commits to install, in addition, a vortex breaker in each RHR suction nozzle. This vortex breaker consists of a 24-inch long section of 14-inch schedule 140 piping connected in a vertical direction to the bottom of the hot leg piping; the 8-inch RHR suction line is connected to the bottom of this vortex breaker. With a vortex breaker, air ingestion commences at about the same water elevation as with a conventional RHR suction nozzle; however, the amount of air entrainment will remain below 10% unless the hot leg is essentially completely drained. Therefore the potential for RHR pump damage has been reduced substantially.

o Each RHR pump suction line is "self-venting," i.e., it slopes continuously upward from the pump to its connection to the hot leg (vortex breaker). If the pump should stop during mid-loop operation (due to interruption of electric power, for example) any air bubbles present in the pump or suction piping will be vented back up through the suction line to the water surface in the hot leg. This feature provides for re-starting the pump under conditions which automatically assure a flooded suction.

o Separate narrow range level transmitters, calibrated for low temperature conditions, indicate the RCS water level between the bottoms of two hot legs and the tops of the steam generator inlet elbows in the same loops during the approach to and conduct of mid-loop operation. Indication in the main control room and low level alarms are provided.

o The range of the wide range pressurizer level instrumentation used during "cold" operations, has been expanded to the bottom of the hot legs. This provides a continuous level indication in the main control room, transitioning to the range of the two, more accurate, narrow range loop level instruments.

o The RHR pumps will be designed to operate without undergoing cavitation or other adverse effects under conditions of no subcooling in the hot legs. Specifically, definition of design values for "NPSH available," "NPSH required" (by the pump) and the required layout characteristics (elevation difference, pipe routing, etc.) will be coordinated to assure that the RHR pumps can be started and run at their full RHR flowrate even if boiling in the reactor vessel is occurring. This assures that the normal RHR function can be readily used to recover from a temporary loss of cooling.

- A locally mounted flow transmitter in each RHR return header (downstream of the RHR heat exchanger), with readout in the main control room, indicates RHR return flow to the reactor vessel. A low alarm will alert the operator to a decrease in RHR flow in the associated subsystem.

- The drain down of the RCS to mid-loop operation level and RCS inventory control during mid-loop operation is performed by the operator in the main control room, using the RHR to Chemical and Volume Control System (CVCS) letdown flowpath and normal CVCS functions. This will eliminate the need to coordinate local actions in the containment with the control room operators to control RCS drain down rate and level.

- Procedures will require that one of the four HHSI pump subsystems always will be available for use during mid-loop operations. This will ensure that a backup source of water for restoring RCS inventory is readily available and can be actuated from the main control room.

- At least two incore thermocouples will be available to directly measure the core exit temperature during mid-loop RHR operation. Each of these thermocouples will be on separate instrument electrical channels.

Note that these design features provide the operator in the main control room with all required instrumentation, alarms, and operation controls necessary to adjust, maintain and take any necessary recovery actions for both RCS inventory control and heat removal.

Additionally, during the FDA Application phase, Westinghouse will perform evaluations to examine potential design criteria to establish procedures and administration controls that will reasonably ensure that containment closure will be achieved prior to the time at which a core uncovery could result from a loss of RHR coupled with an inability to initiate alternate cooling or addition of water to the RCS inventory.

In addition to these design features, appropriate operating and emergency procedures will be defined to guide and direct the operator in the proper conduct of mid-loop operation, and to aid detection and correction of off-normal conditions which might occur during such operations.

SP/90 Resolution:

During the February 9, 1990 ACRS Full Committee meeting, the Staff indicated that the SP/90 design complies with the proposed resolution for this issue.

10. INTERSYSTEM LOCA

Staff Position:

Overpressurization of low pressure piping systems due to RCS boundary isolation failure could result in rupture of the low pressure piping outside containment. This may result in a core melt accident with an energetic release outside the containment building which would cause significant offsite radiation release.

It is the Staff's position that designing interfacing systems to withstand full reactor pressure is an acceptable approach. For systems not designed in this manner it is necessary to provide for isolation testing and main control room position indication for RCS isolation valves, and for high pressure alarms for interfacing systems.

Westinghouse Position:

The principal contribution to core melt frequency from interfacing systems LOCA originates from the four residual heat removal (RHR) suction lines of the integrated safeguards system (ISS). These lines connect the RCS hot legs to the RHR pumps suction and therefore penetrate the containment boundary; the low pressure portions of these lines are normally isolated from the RCS by two closed motor operated valves in series. In an interfacing systems LOCA scenario, it is postulated that either one valve is inadvertently left open and that the other one fails, or that both valves fail while the RCS is pressurized, causing failure of the piping outside containment.

The SP/90 design includes the following features specifically aimed at reducing the probability of this scenario:

- The RHR isolation valves have been included in the system provided to allow leak testing of the valves in the lines connected to the RCS during plant startup. Thus, the probability of one of these valves not being fully closed has been eliminated.

- The design pressure of all RHR piping downstream of the RHR isolation valves (including RHR pump casings) has been increased such that no gross failure would occur even when exposed to full RCS operating pressure.

- The RHR piping downstream of the RHR isolation valves is normally in open connection with the Emergency Water Storage Tank (EWST) such that any leakage through these valves is normally directed back into this tank, which is located inside containment.

- Failure of the RHR pump seal could still be postulated; however, the four separated rooms containing the four subsystems of the ISS have been designed for minimum volume such that when the water levels in the EWST and in the pump room are equal, there remains sufficient inventory in the EWST to ensure continued core cooling with the unaffected ISS subsystems.

These features combine to significantly reduce the probability of a core melt in case of leakage from, or failure of, the RHR isolation valves, which has been shown to be the most probable interfacing system LOCA sequence.

<u>SP/90 Resolution</u>:

At the February 9, 1990 ACRS Full Committee meeting, the Staff indicated that the SP/90 design complies with the proposed resolution for this issue.

11. STATION BLACKOUT

Staff Position:

Station Blackout has been an Unresolved Safety Issue (USI-44) since 1979. A detailed action plan for resolving this issue was published in NUREG-0649, Revision 1. The final evaluation of station blackout accidents at nuclear power plants was performed by the Staff and published in NUREG-1032. In resolving this issue, the Staff performed a regulatory analysis which was documented in NUREG-1109. In June 1988, this USI was finally resolved with the publication of a new rule (53 FR 23203) and Regulatory Guide 1.155, which established new requirements. This resolution applied principally to operating plants; with regard to future standard designs, the Staff appears to go beyond the requirements established as a result of the above resolution in that a diverse alternate AC (AAC) source with full capacity for one safety train and connectability to both divisions will be needed.

Westinghouse Position:

The potential risk contribution of station blackout was recognized from the early days of the SP/90 program, and design features aimed at mitigating the consequences of this sequence were introduced in the design from the beginning. This included an AAC source, albeit one of limited capacity. The SP/90 Probabilistic Safety Study (PSS) demonstrated that the core melt frequency contribution from station blackout was significantly reduced from that observed in current plants; at the same time, station blackout was still responsible for more than half of SP/90 risk due to internal events. In response to the Staff's position as stated at the February 9, 1990 ACRS Full Committee meeting, Westinghouse has expanded its commitment to include a large capacity AAC source in the SP/90 design. The complete list of design features specifically aimed at mitigating the consequences of a station blackout is as follows:

- o The plant includes full load rejection capability which reduces the probability of loss of offsite power.

- The AC power supply system includes a non-Class 1E alternate AC power source (AAC) in addition to the redundant Class 1E emergency diesel generators. The AAC is started automatically on loss of offsite power and can be manually connected to either Class 1E emergency bus; it is sized to allow the plant to be brought to cold shutdown without credit for any other power source.

- The startup feedwater pump will be supplied from the bus that is normally powered by the AAC, and will be started automatically on loss of offsite power.

- The emergency feedwater system (EFWS) includes two turbine-driven emergency feedwater pumps. These pumps are independent of AC and DC power, and the rooms they are located in are cooled in a passive manner. Only one of two pumps is required for decay heat removal.

- The chemical and volume control system (CVCS) contains a backup seal injection pump. This pump takes suction from the spent fuel pit and is connected to the bus that is normally powered by the AAC. The pump is started automatically on loss of normal seal injection.

- The reactor coolant pump (RCP) seals are equipped with an improved O-ring material. In case both seal injection and thermal barrier cooling are lost during a station blackout event, the RCP seals will be subjected to full RCS conditions. Tests have shown that with the new O-ring material RCP seal leakage is reduced to an expected value of 21 gpm per pump. This amount of leakage will not result in core uncovery until beyond 24 hours following initiation of the coincident failure of seal injection and thermal barrier cooling.

- The Class 1E batteries are sized for eight hours of operation under blackout conditions. This assumes normal operation for 2 hours and selective load shedding by the operators thereafter. These batteries can be recharged from the AAC.

- Analyses will be performed at the FDA stage for all rooms containing equipment assumed to operate during station blackout conditions (i.e. loss of both emergency diesel generators and the AAC) including turbine-driven emergency feedwater pumps, Class 1E batteries, inverters, emergency control room, etc... to demonstrate that such equipment will continue to operate for a period of at least 8 hours.

- o Emergency response guidelines will be developed as part of the FDA Application to ensure correct operator action during station blackout. These will cover the operation of the above equipment, as well as any other equipment that may be useful in a station blackout condition.

These features allow the plant to be brought to cold shutdown in case of loss of offsite power in conjunction with failure of both emergency diesel generators to start, assuming that the AAC source successfully operates. In case of station blackout (i.e., loss of the AAC source in addition to failure of both emergency diesel generators to start), the plant can be maintained in a hot standby condition for at least 8 hours.

<u>SP/90 Resolution</u>:

With the additional commitment to incorporate a full capacity alternate AC power source, the SP/90 should be in compliance with the proposed resolution for this issue.

12. FIRE PROTECTION

<u>Staff Position</u>:

Plant fires can be significant contributors to core damage based on PRA's performed at more than a dozen plants. Current fire protection criteria are contained in GDC 3 and 10 CFR 50.48; guidelines for compliance with these criteria are provided in Standard Review Plan 9.5.1 including Branch Technical Position CMEB 9.5-1. It is the Staff's position that more stringent acceptance criteria are required for future standard plant designs as follows:

- o Alternative/dedicated shutdown capability only for control room fires.

- o Safe shutdown capability required for a fire in any other fire area without reliance on any equipment in that area or re-entry into that area for repairs and/or operator actions.

- o For fire areas in the containment, use of a redundant shutdown train in the same area can be allowed in accordance with applicable BTP acceptance criteria and plant specific reviews.

- o As a minimum, migration of smoke, hot gases or fire suppressant into other applicable fire areas must be minimized by design to prevent any adverse impact on safe shutdown capability including operator actions.

Westinghouse Position:

Enhanced fire protection has been one of the goals of the SP/90 design. Since the initial submittal of the RESAR-SP/90 PDA Application, Westinghouse has had discussions with the Staff to ensure acceptability of the design. As a result of these discussions, the following physical separation philosophy has been defined:

- o Within the Nuclear Power Block, redundant divisions of safety related equipment outside containment are located in redundant safety areas which are separated from each other and from other areas in the plant by fire barriers with a minimum fire resistance rating of three hours. This degree of separation is in some cases not required, since there are non-safety related components and systems (e.g., startup feedwater system and charging pumps) in other fire areas that could be relied upon to achieve safe shutdown following a fire affecting safety-related equipment; however, such separation addresses other external events (e.g., flooding and sabotage) and greatly simplifies the design and analysis of the safety-related systems and is, therefore, cost effective. Each redundant safety area is further subdivided by internal fire barriers in order to separate components which present a fire hazard to other components or cable concentrations within the same area. For example, each diesel generator is separated from the remainder of its associated safety area.

- o Exceptions to the use of three-hour fire barriers outside containment are only made in two cases where physical separation conflicts with other requirements and where the equipment is not clearly division oriented, i.e. the main control room and the main steam tunnel.

- o Outside containment, safe shutdown equipment is, with the two exceptions noted above, division oriented, i.e. it can be clearly associated with either Division A or Division B. This is not the case inside containment where only a few components fall into this category. In addition, separation by three-hour fire barriers inside containment is not practical because all compartments need to be in communication with each other in order to relieve pressure following a high energy line break. For this reason, the containment is considered to be a single fire area. This single area will be subdivided into fire zones; these fire zones will be defined such that loss of one fire zone will not result in loss of safe shutdown capability. Separation between fire zones will preferentially be by 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, etc. Fire zones and separation will be agreed upon by Westinghouse and Staff during the FDA Application process.

This philosophy is implemented in the SP/90 design in a conservative manner. For example, each redundant safety area is provided with its own separate ventilation system, rather than relying on three-hour fire dampers for isolation. Also, the number of other penetrations (i.e., non-HVAC) between redundant safety areas has been reduced to the extent practical.

SP/90 Resolution:

The above SP/90 position is intended to address the Staff's position; however, at the February 9, 1990 ACRS Full Committee meeting, the Staff indicated that none of the ALWR applicants is yet in strict adherence to that position.

13. SEVERE ACCIDENT MITIGATION EQUIPMENT SURVIVABILITY

Staff Position:

Safety related equipment used in the mitigation of design basis accidents (DBA) is subject to a comprehensive set of criteria related to redundancy, diversity, environmental qualification, quality assurance, etc., in order to ensure that they will perform their intended functions in case of need. Severe accidents are not intended to be designated as DBA's, and as such, credit may be taken for non-safety equipment that is not designed to the above criteria; however, it is the Staff's position that equipment incorporated for severe accident mitigation must also provide reasonable assurance that they will perform their intended functions in the severe accident environment for the time span for which they are needed. Specific guidance as to the activities required is provided in Appendix A to Regulatory Guide 1.155, "Station Blackout" which similarly is concerned with equipment used to mitigate non-DBA events.

Westinghouse Position:

Survivability of equipment intended to mitigate severe accidents is a new issue that was not previously raised during the RESAR-SP/90 PDA application. Westinghouse commits to identify equipment used in the mitigation of severe accidents and to define a program to ensure that such equipment will perform reliably under severe accident conditions; this commitment will be implemented in the FDA Application.

SP/90 Resolution:

Westinghouse recommends that this remain an open item to be resolved at the FDA stage.

14. OPERABILITY OF PUMPS AND VALVES

Staff Position:

Periodic testing in accordance with Section XI of the ASME Code is required to ensure operability of safety related pumps and valves to ensure reliable operation in case of need. During the ACRS Full Committee meeting of February 9, 1990, the Staff characterized the current program as "wholly inadequate." The NRC has issued Generic Letters 89-04 and 89-10, and has proposed rulemaking to extend in-service testing beyond code components and to demonstrate capability to perform safety functions. It is the Staff's position that future standard plants must incorporate enhanced provisions for in-service testing, for example:

- o Testing of safety related non-ASME components.
- o Full flow testing capability for pumps and check valves.
- o Testing of MOV's under design basis differential pressure.
- o Use of non-intrusive techniques.
- o Program to determine appropriate frequency for disassembly and inspection of pumps and valves.

<u>Westinghouse Position</u>:

The SP/90 Integrated Safeguards (emergency core cooling, containment spray, and residual heat removal functions) and Emergency Feedwater Systems contain features aimed at eliminating many of the problems encountered in current plants, in particular:

- o Pump miniflows are sized to allow continuous operation without damage.
- o Each pump can be tested over its full operational range with the plant at power by means of specially installed test lines.
- o The emergency core cooling and emergency feedwater systems are permanently aligned to perform their safety function without reliance on MOV's to change position.

- o ECCS injection lines including redundant check valves cannot be tested with the plant at power because the shutoff head of the HHSI pumps is below RCS pressure; however, these lines are at full flow conditions as part of normal RHR operation.

- o EFWS injection lines including redundant check valves could in principle be tested at power; however, such testing is undesirable because of the thermal transients it induces. Full flow testing at shutdown is possible.

- o Motor operated valves (MOVs), with the exception of the residual heat removal suction isolation valves, can be stroked with the plant at power.

- o A permanently installed system is provided to allow leak testing of valves isolating the RCS from low pressure systems during each plant startup.

- o EFW pumps are provided with individual suction lines in order to eliminate the possibility of steam binding in more than one pump as a result of backleakage through the series of check valves in one pump discharge line; temperature instrumentation is also provided to detect instances of such backleakage.

In addition, the FDA Application will include the following:

- o Frequency of pump and valve testing as well as disassembly and inspection.

- o Description of pump and valve diagnostic systems.

- o Information on pump and valve prototype or in-situ testing.

<u>SP/90 Resolution</u>:

Westinghouse recommends that this remain, at least in part, an open item to be resolved at the FDA stage.

15. SEISMIC DESIGN

Staff Position:

Presently, the magnitude of the Operating Basis Earthquake (OBE) is at least half that of the Safe Shutdown Earthquake (SSE) as defined in 10 CFR 100. This, coupled with the fact that OBE stresses should be within normal allowables, has resulted in the OBE governing the seismic design, and has caused structures and systems to be greatly stiffened. This stiffening may be detrimental to actual plant conditions. It is the Staff's position as expressed in the February 9, 1990 ACRS Full Committee meeting that coupling of OBE and SSE is no longer required and that SSE should be governing the seismic design of future standard plants. In fact, in response to an ACRS question, the possibility of eliminating OBE altogether is favor of an "inspection earthquake" was not ruled out altogether.

Westinghouse Position:

In the RESAR-SP/90 PDA application, Westinghouse had proposed that the OBE be one-third of SSE; this would in most cases result in the SSE governing the seismic design. Since that time, EPRI has made a similar proposal as part of the ALWR Requirements Document; the review of this document is still in progress.

SP/90 Position:

Westinghouse and the Staff have agreed that the review of the EPRI ALWR Requirements Document is the appropriate vehicle for resolving this issue. With regard to SP/90, this will, therefore, remain an open item to be resolved at the FDA stage.