THE 4TH INTERNATIONAL TOPICAL MEETING ON
NUCLEAR THERMAL HYDRAULICS, OPERATIONS AND SAFETY
April 6-8, 1994, Taipei, Taiwan

SYSTEM 80+™ PRA INSIGHTS ON SEVERE ACCIDENT PREVENTION AND MITIGATION

D. J. Finnicum, M. C. Jacob, R. E. Schneider, and R. A. Weston
ABB-Combustion Engineering
1000 Prospect Hill Road
Windsor, Connecticut 06095 USA
(203)285-5881

ABSTRACT

The System 80+ design is ABB-CE's standardized evolutionary Advanced Light Water Reactor (ALWR) design. It incorporates design enhancements based on Probabilistic Risk Assessment (PRA) insights, guidance from the ALWR Utility Requirements Document[1] (URD), and US NRC's Severe Accident Policy[2]. Major severe accident prevention and mitigation design features of the System 80+ design are described. The results of the System 80+ PRA are presented and the insights gained from the PRA sensitivity analyses are discussed. ABB-CE considered defense-in-depth for accident prevention and mitigation early in the design process and used robust design features to ensure that the System 80+ design achieved a low core damage frequency, low containment conditional failure probability, and excellent deterministic containment performance under severe accident conditions and to ensure that the risk was properly allocated among design features and between prevention and mitigation.

INTRODUCTION

ABB-Combustion Engineering Nuclear Systems (ABB-CE), in cooperation with the U. S. Department of Energy, is working to develop and certify a standardized Advanced Light Water Reactor (ALWR) design. The ALWR design objectives include safe and reliable plant operation with systems designed to prevent, and mitigate severe accidents. The System 80+ Standard Design is an evolutionary advancement based upon the existing and proven System 80 Pressurized Water Reactor (PWR) design currently in operation. In developing System 80+, ABB-CE's experience in PWR design was supplemented with PRA techniques so that a systematic procedure for design improvement could be effected. As part of the design enhancements, ABB-CE included the design features specified in the Electric Power Research Institute's (EPRI) ALWR Utility Requirements Document[1], the requirements of 10CFR52 and NRC guidance from emerging policy issues (e.g., SECY-93-087).

Incorporation of design enhancements for severe accident prevention and mitigation has resulted in a plant design that has a core damage frequency two orders of magnitude lower than its predecessor. These design enhancements and their impact on plant safety as measured by core damage frequency and large release frequency are described in the following paragraphs.

SEVERE ACCIDENT PREVENTION AND MITIGATION FEATURES

The goal of the safety system enhancements has been to increase the reliability through diversity and redundancy, improve safety system performance and to minimize operator requirements for performing high stress risk significant operations. Table 1 lists the major preventative and mitigative design features. The following paragraphs provide a brief description for each feature and its impact on System 80+ reliability and risk.

Larger Pressurizer

System 80+ has a larger pressurizer volume as compared to the existing generation of commercial nuclear power plants. This makes the plant response to transients slower and more resilient. The larger pressurizer volume helps maintain a higher pressurizer pressure and water level following a turbine trip. It also helps prevent emptying the pressurizer and uncovering the pressurizer heaters following overcooling transients. For most transient events, the rise in pressurizer pressure will be moderate and consequently the primary safety valves will not be challenged. The larger volume also prevents water level surges that cause liquid or two-phase flow from reaching the primary safety valves following a feedwater line break or a loss of load transient. It also minimizes pressurizer level fluctuations during transient events and increases the margin for a safety injection actuation signal following certain transient events. A larger pressurizer volume also helps to lower the peak pressure that can be reached following an Anticipated Transient Without Scram (ATWS) event.

Larger Secondary Inventory

System 80+ has a larger secondary inventory in the steam generators to make the plant response to transients slower and more resilient. The increased heat transfer area of the steam generators provides a 10% tube plugging margin, which helps increase the availability of the steam generator secondary heat removal. The extent of the increase is limited by Steam Line Break (SLB) effects on containment design and proven manufacturing capability. The increased downcomer volume and the 25% increase in steam generator inventory help reduce fluctuations during transients and increase the time to dry out the steam generators. The time required to dry out the secondary inventory of the steam generators is approximately 50% longer for System 80+ than the dry out time for System 80. The System 80+ steam generators will use thermally treated Inconel 690 for the steam generator tubes. This improved steam generator tube material reduces the potential for steam generator tube degradation.

ShutdownCooling (SCS)/Containment Spray System (CSS)

In addition to their long-term decay heat removal function, the SCS pumps are designed to perform residual heat removal injection and cooling of the In-containment Refueling Water Storage Tank (IRWST). In the residual heat removal injection mode of operation, the SCS is used (in conjunction with the Rapid Depressurization System) as a backup to the Safety Injection System (SIS) to inject borated water into the reactor core. To provide operating flexibility, the design pressure of the SCS for System 80+ is 900 psi versus 600 psi for System 80. The CSS pumps can be used to backup the SCS pumps for improved decay heat removal capabilities. The SCS pumps can also be used as backups to the CSS pumps to perform IRWST cooling during "feed and bleed" operations (beyond design basis events). The two-train redundancy for each of these systems, coupled with the interchangeable SCS and CSS pumps, enhance the availability of these systems.

The Containment Spray System (CSS) is a safety grade system designed to reduce containment pressure and temperature resulting from a main steam line break, loss-of-coolant-accident or a severe accident and to remove iodine and other particulate fission products from the containment atmosphere.

The CSS provides adequate cooling of the containment atmosphere to limit post-design basis accident temperatures and pressures to less than the containment design values.

The CSS uses the In-Containment Refueling Water Storage Tank (IRWST) and has two independent trains (two containment spray pumps, two containment spray heat exchangers, two independent spray headers, and associated piping, valves, and instrumentation). The pumps and remotely operated valves can be operated from the control room.

The CSS provides sprays of borated water to the containment atmosphere from the upper regions of the containment. The spray flow is provided by the containment spray pumps which take suction from the IRWST. The containment spray pumps start upon the receipt of a Safety Injection Actuation Signal (SIAS). The pumps discharge through the containment spray heat exchangers and the spray header isolation valves to their respective spray nozzle headers, then into the containment atmosphere. Spray flow to the containment spray headers is not provided until a Containment Spray Actuation Signal (CSAS) automatically opens the containment spray header isolation valves. The spray headers are located in the upper part of the containment building to allow the falling spray droplets time to approach thermal equilibrium with the steam-air atmosphere. Condensation of the steam by the falling spray results in a reduction in containment pressure and temperature.

In addition to containment cooling after an SLB or LOCA, the CS pumps and heat exchangers can be manually aligned to provide cooling of the IRWST during post-accident feed and bleed operations when the steam generators are not available to cool the RCS.

The CS pumps are designed to be functionally interchangeable with the Shutdown Cooling System (SCS) pumps. Though not required for normal operation or accident mitigation, interchangeability of the pumps allows backup of the CS pumps and increases the reliability of the containment spray function.

To further increase the reliability of the containment spray function, the containment spray headers are designed to accept spray flow from an external source of water supply via a "tee" connection to the spray line. In case of unavailability of normal containment spray flow, the external source can supply water to the headers, allowing for containment cooling and depressurization of the containment atmosphere.

Multiple Independent Connections to Grid

The System 80+ design includes a main switchyard for incoming and outgoing electric power and a separate and independent backup switchyard (with two safety grade reserve transformers) that is tied to the grid at some distance from the main switchyard. In addition, the System 80+ turbine generator system and the associated buses are designed to runback to maintain hotel load on a loss of grid event. These features, in conjunction with the combustion turbine generator are intended to reduce the frequency of Loss of Offsite Power (LOOP) events and Station Blackout events.

Separate Startup and Emergency Feedwater System

The use of a non-safety related Startup Feedwater System (SFWS) for normal startup and shutdown operations helps reduce the demands on the Emergency Feedwater System (EFWS). In addition, the SFWS provides an independent means of supplying feedwater to the steam generators for removing heat from the Reactor Coolant System (RCS) during emergency conditions when the main feedwater is not available.

Improved Control Room Design

The System 80+ advanced control room design (Nuplex 80+) is intended to improve upon existing control rooms while maintaining their strengths. In that respect it is an evolutionary design that is expected to provide more and better information to the operator than the Standard System 80 design, with corresponding improvements in operator reliability. These improvements include prioritized alarms, parameter processing and validation, mode-dependent alarms, integrated normal/post-accident instrumentation, and hardwired backups to digital safety instrumentation and controls.

Component Cooling Water System

The Component Cooling Water System (CCWS) is a closed-loop system that provides cooling water to remove heat from plant systems, components, and structures. Heat from the CCWS is rejected to the ultimate heat sink through the open-loop Station Service Water System (SSWS). Each of these systems consists of two separate and redundant divisions. Each division contains two pumps: one is normally operating, while the other pump is in standby and starts automatically if the operating pump trips. This configuration eliminates the demand failures of pumps and valves that were found to be significant contributors to risk in the System 80 design with standby CCWS/SSWS configurations.

Facilities Designs

Facilities are designed to provide physical separation of systems or trains of system that perform redundant safety-related functions. This increases the availability of systems due to their protection from failures associated with internal fires, internal floods, sabotage, and similar common-cause failures. This contributes to risk reduction when compared to existing plant designs.

Safety Injection System

The primary function of the Safety Injection System (SIS) is to inject borated water into the RCS for inventory and reactivity control during severe accidents such as Loss of Coolant Accidents (LOCAs) and ATWS. The SIS can be used in conjunction with the Rapid Depressurization System for "feed and bleed" operation as an alternate method for removing decay heat. For continuous long-term post-LOCA (large) cooling of the reactor core, the SIS pumps are realigned to provide simultaneous hot-leg and direct vessel injection to prevent boron crystallization. The following are major evolutionary characteristics of the System 80+ SIS:

- four high-pressure 100% capacity pumps,
- direct vessel injection (pumps take suction from the IRWST and deliver borated water to the reactor vessel downcomer via the DVI lines),
- elimination of need for low pressure pumps,
- elimination of need to realign pump suction to the containment sump,
- capability to test pumps at design flow while operating,
- "feed and bleed" cooling of the RCS (in conjunction with the Rapid Depressurization System for beyond design basis events).

These evolutionary characteristics help reduce the unavailability of the System 80+ SIS to levels below those for existing generation of commercial nuclear power plants. This was achieved by reducing or eliminating several contributors to SIS unavailability. For example: (1) a four-train (as compared to a two-train) SIS, reduces the contribution to the system unavailability that is due to outages for testing, repair and maintenance; (2) the elimination of the low-pressure pumps eliminates the failures to start for these pumps; (3) the elimination of the need to realign the suction of the pumps eliminates the contribution of the failure to do so; (4) the provision for cold-leg DVI increases the time for SIS response during a small break LOCA.

Safety Depressurization System (SDS)

The Safety Depressurization System (SDS) is a multi-purpose dedicated safety system specifically designed to serve important roles in severe accident prevention and mitigation.

The Reactor Coolant Gas Vent (RCGV) function of the SDS provides a safety-grade means of venting non-condensable gases from the pressurizer and the reactor vessel upper head to the Reactor Drain Tank (RDT) during post-accident conditions. In addition, the RCGV provides a safety-grade means to depressurize the RCS in the event that pressurizer Main Spray and Auxiliary Spray systems are unavailable and a means of venting the pressurizer and reactor vessel upper head during pre-refueling and post-refueling operations.

The Rapid Depressurization (RD) function, or bleed function, provides a manual means of quickly depressurizing the RCS when normal and emergency feedwater (EFW) are unavailable to remove core decay heat through the steam generators. Whenever any event, e.g., a total loss of feedwater (TLOFW) results in a high RCS pressure with a loss of RCS liquid inventory, the SDS rapid depressurization or bleed valves may be opened by the operator, resulting in a controlled rapid depressurization of the RCS. As the RCS pressure decreases, the Safety Injection (SI) pumps start, initiating feed flow to the RCS and restoring the RCS liquid inventory.

The rapid depressurization feature of the SDS also serves an important role in severe accident mitigation. In the event a high pressure meltdown scenario develops and the feed portion of feed and bleed cannot be established due to unavailability of the SI pumps, the SDS can be used to depressurize the RCS to ensure that a High Pressure Melt Ejection (HPME) event does not occur thereby minimizing the potential for direct containment heating (DCH).

The severe core damage depressurization goal is to ensure that the RD can depressurize the RCS from 2500 to 250 psia prior to a reactor vessel melt-through. This is accomplished by designing the SDS rapid depressurization valves for initial bleed flow of approximately 412,000 lbm/hr of steam.

Emergency Feedwater System (EFWS)

The Emergency Feedwater System (EFWS) provides an independent, safety-related means of supplying feedwater to the steam generators during the early phase of secondary heat removal in the event that both the main feedwater and the startup feedwater are lost. The EFWS consists of two divisions, each of which is aligned to deliver feedwater to its respective steam generator. Each division contains a motor-driven train and a turbine-driven train. The steam required to operate the turbine-driven pump is supplied from the associated steam generator to which feedwater is delivered. For station blackout sequences, the turbine-driven trains of the EFWS are available to remove decay heat from the RCS. Because of the redundancy and diversity of the emergency feedwater trains, this system is a significant contributor to risk reduction.

Two Emergency Diesels and Standby Combustion Turbine

Each of the two divisions of class 1E AC power is supplied with emergency standby power from an emergency diesel generator (DG). Each DG is provided with a dedicated 125 VDC battery. The emergency DGs start and load automatically following a LOOP event. In addition to the two emergency DGs, the System 80+ design has an alternate standby onsite AC power source. This is a non-safety combustion turbine power source provided to cope with station blackout scenarios. The alternate power source is independent and diverse from the DGs. The combustion turbine is automatically started and loaded to the Permanent Non-Safety (PNS) bus on loss of site power. The combustion turbine can be manually loaded to power either division of class 1E AC loads when the associated DG is unavailable.

Vital Batteries

Six independent and separate 125 VDC batteries are included in the System 80+ design, in comparison to four batteries for the System 80 design. For System 80+, each battery can supply the continuous emergency load of its own load group for a period of 2 hours. In addition, the batteries provide a station blackout coping capability assuming manual load shedding or the use of a load management program. This permits operation of the instrumentation and control loads associated with the turbine-driven emergency feedwater pumps for a minimum of 8 hours.

In-Containment Refueling Water Storage Tank

Sufficient borated water is stored in the In-Containment Refueling Water Storage Tank (IRWST) to meet all post-accident safety injection pumps and containment spray pumps operation requirements. The volume of borated water is also sufficient to flood the refueling pool during normal refueling operations. The IRWST eliminates the need for switching over from injection mode to recirculation mode during emergency core cooling operations and therefore, eliminates failures associated with the switch-over in existing commercial nuclear power plants. The Primary Safety Valves (PSVs) and the Rapid Depressurization Valves (RDVs) discharge to the IRWST. The IRWST provides steam quenching for PSV or RDV discharges and provides scrubbing of radioactive materials in the discharges. In addition, the IRWST is the source of borated water for cavity flooding at the onset of a severe accident.

Containment Vessel

The System 80+ containment vessel, including all its penetrations, is a low leakage spherical steel shell, inside a reinforced concrete shield building, which is designed to withstand the postulated Loss-of-Coolant-Accident (LOCA) or a Main Steam Line Break (MSLB) while limiting the postulated release of radioactive material to within the requirements of 10 CFR 100[4]. Additionally, the containment and shield building provide a barrier against the release of radioactive materials which may be present in the containment atmosphere following an accident.

The containment spherical shell is 200 feet in diameter and is constructed of steel plate with a wall thickness of one and three-quarter inches. The spherical containment provides 3.34 million cubic feet of net free volume with its internal structures arranged in a manner to (1) protect the steel shell from missile threats, (2) promote mixing throughout the containment atmosphere (see Figure 1), and (3) comfortably accommodate condensable and non-condensable gas releases from design basis and severe accidents.

In severe accident scenarios, the containment vessel is the last fission product barrier protecting the public from potentially large radiation releases. It is therefore of paramount importance to provide a strong robust containment design to meet severe accident internal pressurization challenges. The design basis pressure for the containment is 53 psig. The analyses documented in CESSAR-DC[3] demonstrate that pressures resulting from large break LOCAs or main steam line breaks within the containment will not exceed this design pressure. Calculations also indicate that pressure limits determined in accordance with ASME Service Level C criteria range from 130 psig at an average steel shell temperature of 290 °F to 120 psig at a temperature of 450°F. The median ultimate failure pressure ranges from 188 psia at 150 °F to 160 psia at 450 °F.

Secondary Containment

The secondary containment consists of the containment shield building and the annulus between the steel containment vessel and the shield building. The containment shield building, which houses the containment vessel and safety-related equipment, is designed to provide biological shielding and external missile protection for the containment vessel and safety-related equipment. It is a reinforced concrete structure consisting of a right cylinder and hemispherical dome. The shield building shares a common foundation base with the nuclear system annex as shown in Figure 1. The Annulus Ventilation System (AVS) provides a mechanism for substantially reducing and/or eliminating unfiltered fission product releases following design basis and severe accidents.

Cavity Flooding System

The function of the Cavity Flooding System (CFS) is to provide a means of flooding the reactor cavity in the event of a severe accident for the purpose of cooling the core debris in the reactor cavity and scrubbing fission product releases. The cavity flooding system is designed (in conjunction with the containment spray system) to provide an inexhaustible continuous supply of water to quench the core debris. The CFS is a manually actuated severe accident mitigation system. The components of the CFS include the In-Containment Refueling Water Storage Tank (IRWST), the Holdup Volume Tank (HVT), the reactor cavity, connecting piping, valves and associated power supplies. A schematic of the CFS is shown in Figure 2. The CFS takes water from the IRWST and directs it to the reactor cavity. The water flows first into the HVT by way of four 12 inch diameter HVT spillways and then into the reactor cavity by way of two 10 inch diameter reactor cavity spillways.

Hydrogen Mitigation System

Large quantities of hydrogen can be generated during the core degradation and melting process associated with a severe accident. While it is unlikely that the hydrogen generated will be sufficient to fail the containment, a Hydrogen Mitigation System (HMS) has been incorporated into the System 80+ design to provide added assurance that hydrogen concentrations will be maintained at non-detonable levels even during the most limiting severe accident. To this end, the Hydrogen Mitigation System (HMS) is designed to accommodate the hydrogen production from 100% fuel clad metal-water reaction and maintain the average containment hydrogen concentration below the 10% limit in accordance with 10 CFR 50.34(f)[7] for a degraded core accident. The HMS also has hydrogen recombiners which ensure that the hydrogen concentration is maintained below 4% during a DBA LOCA in accordance with the requirements of 10CFR50.44.

The HMS is a control room actuated system designed to allow controlled burning of hydrogen at low concentrations in order to preclude hydrogen concentration build-up to detonable levels. The system is designed to prevent the local hydrogen concentration in containment from reaching 10% by volume during a degraded core accident by burning hydrogen throughout the containment as the local concentration reach levels of between 4 to 6 %. Frequent small hydrogen burns will not threaten containment while simultaneously removing the possibility of a significant combustion threat later in the accident.

The HMS consists of two redundant groups of igniters, Group A and Group B. Each group has independent and separate control, power, and igniter locations to ensure adequate coverage within the containment. The igniters are AC powered glow plugs powered directly from step down transformers. The igniter enclosures are designed to protect the glow plugs from water jet impingement and to minimize the temperature rise inside the igniter assembly.

Reactor Cavity Design

The System 80+ reactor cavity is configured to promote retention of and heat removal from the postulated core debris during a severe accident, thus, serving several roles in accident mitigation. Corium retention in the core debris chamber virtually eliminates the potential for significant DCH induced containment loadings. The large cavity floor area allows for spreading of the core debris enhancing its coolability within the reactor cavity region.

The System 80+ cavity includes 32,000 ft^3. of free volume. Large (and well vented) volumes, such as those in System 80+, are not prone to significant pressurization resulting from vessel breach or the corium quench processes. Cavity pressurization analyses performed for System 80+ indicate peak cavity pressure loadings to be less than 100 psid which is within the cavity design basis.

The instrument shaft design serves an important purpose in the severe accident mitigation for System 80+. First, by orienting the instrument shaft vertically and providing only limited gas venting in this path, the possibility of corium carryover is minimized. Analyses provided in Reference 8, Appendix D, suggests that only 10% of entrained corium could be expected to initially be carried upward into the vertical shaft even if the shaft were vented to accommodate significant gas flows. The remainder of the corium not entering the vertical shaft will be captured in a large debris retention chamber located at the base of the instrument shaft.

The System 80+ containment design ensures that actual venting to the upper containment either by the vertical shaft or around the RV flange is small. Thus, steam exits the reactor cavity via a convoluted pathway above the top of the core debris chamber and through louvered vents under the refueling pool. As a consequence, the dominant hot gas and corium carryover pathway will be to the lower portion of the containment where the containment shell is fully protected by the crane wall.

System 80+ has an offset core debris chamber designed to de-entrain and trap the debris ejected during a reactor vessel breach. The core debris chamber and the instrument shaft have been designed such that following a failure of the reactor vessel, high inertia core debris would de-entrain and collect in the debris chamber while the lower inertia steam/hydrogen/air mixture would negotiate a right angle turn and exit the reactor cavity via a convoluted vent path. The chamber has been sized according to ARSAP guidance[8] to hold twice the post-severe accident maximum corium volume. Once deposited in the debris chamber, the debris would be difficult to re-entrain since the retention zone would exhibit a low velocity recirculation flow pattern. Any corium negotiating the 90 degree turn would be de-entrained by the reactor cavity concrete ceilings and seal table structure.

The reactor cavity is configured to spread out the ejected core debris over the floor surface area during a postulated severe accident. The cavity meets the proposed ALWR URD[1] criteria of 0.02 m^2/MWt of surface area below the vessel.

The reactor cavity is designed for 188 psid loading with an American Concrete Institute (ACI) calculated ultimate pressure of 235 psid. This cavity strength is typical of later designed PWR cavities and is relatively robust when compared to the cavity loading expected during a severe accident.

Calculations show that the reactor vessel and the upper cavity could continue to be supported even if the entire lower cavity walls below the corbels were either eroded by corium attack or destroyed by a steam explosion. Reinforcing steel provided between the interface of adjacent walls with the upper reactor cavity wall provide enough resistance through shear-friction to provide this support without relying on support from the lower cavity wall. Alternate calculations have been performed which suggest that a damaged RV can be supported by the four cold legs. Some small vessel motions would be expected. However, these motions would not translate into significant motions of other RCS components.

Missile Protection

During severe accidents, missile protection of the containment shell is primarily accomplished by the use of protective shields and barriers either near the source of the potential missile or in front of the containment shell (such as the crane wall).

The reactor cavity is arranged such that any corium debris leaving the cavity will exit via the door or louvers in the HVAC room above the IRWST pool or via the nozzle cutouts. Corium debris released in these areas will likely interact with either the crane wall or refueling pool wall and ultimately deposit in these areas. In the highly unlikely event that the corium debris is projected upward out of the cavity annulus, the Head Area Cable Tray System (HACTS) above the RV serves to protect the containment shell from a direct corium attack of the containment shell.

Within the reactor cavity, the containment shell is protected from the corium debris by a concrete basemat layer varying from 3 to about 5 feet thick. Assuming flooded cavity conditions, it is conservatively estimated that this thickness of concrete flooring will protect the containment shell from corium debris contact for upwards of 30 hrs at its thinnest point.

Missiles generated via failure of the top head and top head components are considered under the general severe accident category of "in-vessel" or "alpha" mode failure. In these scenarios the upper head or control rod missile will be intercepted by the HACTS located directly above the RV upper head. Thus, consequential damage of the containment steel shell due to failure of a pressurized RV is highly unlikely.

An Ex-Vessel Steam Explosion (EVSE) may occur when corium debris contacts a water pool. While EVSEs are considered plausible, their consequences on containment integrity are insignificant because EVSEs are not expected to be capable of damaging the reactor cavity structures required for support of the RV/RCS. All "in-cavity" structures that may be damaged by such explosions will be confined to non-load bearing structures and thus will not compromise containment integrity.

DETERMINISTIC SEVERE ACCIDENT EVALUATIONS

Bounding deterministic calculations indicate that early containment challenges associated with vessel breach phenomena and hydrogen combustion result in peak loadings below the ASME Service Level C containment pressure limit and hence provide a high degree of confidence that containment integrity can be maintained. Steam explosion loadings were estimated by converting the corium stored energy into an equivalent amount of TNT with an effective impulse loading based on correlations established by Cole[9] for underwater explosions. These assessments suggest that the System 80+ cavity design can withstand impulse loadings associated with a steam explosion involving 5 to 10% of the ejected corium mass without serious damage to the reactor cavity. Further structural analyses confirm that the System 80+ design is sufficiently strong such that even in the unlikely event of a complete loss of load carrying capability of the reactor cavity, a consequential indirect breach of the containment will not occur.

Additional early containment loadings associated with "in-vessel steam explosions" and RV rocket failure were assessed probabilistically. Based on this assessment, these containment failure modes were found to have a negligible contribution to the conditional containment failure probability.

Late containment failure was assessed deterministically using the MAAP 3.0B code. Based on these studies it was concluded that for transients without containment heat removal, containment overpressure failure will not occur until after 50 hours following the initiation of the severe accident scenario. This slow pressurization response allows ample time for the operating staff to establish alternate containment cooling pathways and avert containment failure.

Deterministic basemat melt-through scenarios were performed using CORCON Mod 3. These analyses assumed that 100% of the corium debris was cooled within the reactor cavity as a "layered impermeable media". Under these circumstances, a local below ground penetration of the containment shell will be delayed for more than 24 hours after the onset of core melt. Basemat penetration into the plant extended foundation (soil) will be delayed for upwards of eight days.

PRA RESULTS

The estimated Core Damage Frequency (CDF) attributable to internal events for the System 80+ plant is 1.7E-06 events per year. Conservative scoping estimates were calculated for the CDF for fires and floods. The total CDF, including the scoping estimates for fire and flood and external events, is 2.0E-06 events per year. This is well within the industry goal of 1.0E-05 events per year. Table 2 presents the CDF contributions by initiating events. The CDF of the System 80+ design is a factor of 128 less than that of the System 80 base design. The relative contributions (percent of total) of the various internal events to the total CDF are tabulated in Table 2. For the System 80 design, loss of offsite power and station blackout dominates (46%) the CDF profile. This is followed by the LOCAs (31%) and then transients (14%). The contribution by ATWS is relatively small (6%). For the System 80+ design, the LOCA categories of initiating events dominate (37%) the CDF profile. This is followed by the transient category (34%) of events. The contribution from loss of offsite power (including station blackout) is very small because of the following System 80+ design features:

- multiple independent connections to the grid,
- turbine-generator runback capability to maintain hotel loads,
- alternate standby AC source (combustion turbine), and
- six vital 125 VDC batteries.

The contribution from ATWS is also relatively small.

The results from the containment response (Level 2 PRA) analyses show that the System 80+ containment is robust and capable of accommodating severe accident challenges. The System 80+ Conditional Containment Failure Probability (CCFP) is 0.02. In SECY-90-016, the NRC specified a conditional containment failure probability goal of 0.1. The NRC did not define what constituted containment failure, but they did specify that the applicant could demonstrate that the containment remains intact for 24 hours. Based on this, containment failure, for the purpose of calculating the CCFP, is defined to be a loss of containment integrity within the first 24 hours following vessel failure. The System 80+ CCFP value was calculated by summing the frequencies of all release classes in which the containment failed within 24 hours after vessel failure or containment integrity was lost due to bypass or isolation failures and dividing by the total core damage frequency attributable to internal events and tornado strikes. The value thus calculated is well within the NRC goal.

An alternate definition of containment failure is any loss of containment integrity for which there is a release in excess of 25 rems at one half mile from the reactor. Based on this definition, the CCFP for System 80+ is 0.027. This was calculated by dividing the probability of exceeding 25 rem at one half mile from the reactor (see Figure 3) by the total core damage frequency attributable to internal events and tornado strikes. This value is also well within the NRC goal.

The conditional probability for intact containment with design basis leakage only is 0.889. The conditional probability for a very late containment failure (greater than 48 hours after core damage) is 0.074. The conditional probability for late containment failures (8 to 24 hours after core damage) is 0.003. The conditional probability for early containment failure (less than 8 hours after core damage) is 0.011. The conditional failure probability for containment isolation failure is 0.023.

The large release goal specified for System 80+ is that releases in excess of 25 rem at one half mile from the reactor shall have an exceedance frequency of less than 1.0E-06 per year. Figure 3 presents the Cumulative Complementary Distribution Function (CCDF) for whole body dose at 300 meters from the reactor and at one half mile from the reactor. As can be seen from this figure, the probability of having a release in excess of 25 rem at one half mile from the reactor is 5.3E-08/year. This is well within the goal for System 80+. The probability of having a release in excess of 25 rem at 300 meters is 6.2E-08 per year. This demonstrates that System 80+ can meet the large release goal for smaller sites also.

SENSITIVITY ANALYSES AND INSIGHTS

In performing the PRA for the System 80+ design, certain assumptions were made regarding the accident prevention and mitigation features. To assess how some of these assumptions may impact the CDF and the large release frequency, several sensitivity analyses were performed. The major insights obtained from these analyses are listed below.

Core Damage Frequency

1. With the incorporation of system design enhancements, the CDF for System 80+ has decreased by a factor of approximately 128, when compared with the proven System 80 design.

2. Even with such a significant decrease in the CDF, loss of offsite power and station blackout events are no longer the dominant contributors to CDF for the System 80+ design. For most existing light water reactors, loss of offsite power and station blackout events are major contributors to CDF.

3. Because redundancy and diversity is incorporated in the System 80+ design, in general, independent hardware faults are not the major contributors to CDF.

4. Certain systems, such as the Electrical Distribution System, the EFWS, the SIS, and CCWS would adversely affect the overall CDF if the reliability of these systems deteriorated significantly during the operation of the plant.

5. The System 80+ CDF is only somewhat sensitive to control room operator error rates, but is more sensitive to operator actions performed outside the control room during the progression of an accident.

6. The operator actions that are important to risk reduction of the System 80+ include aggressive cooldown of the plant following a steam generator tube rupture or small LOCA event and the initiation of "Feed and Bleed".

7. A failure of the reactor coolant pump (RCP) seals is postulated to occur following loss of cooling to the seals or a station blackout event. The CDF for System 80+ is relatively insensitive to RCP seal failure events because of the diverse dedicated seal cooling system that has been added to System 80+.

8. It has been contended that with complete separation of equipment, improved staff training, and improved maintenance techniques and selection of equipment the potential for common cause failure would be virtually eliminated. The elimination of common cause failure would decrease the CDF of the System 80+ design by a factor of about 8.

Large Release Frequency

1. The System 80+ containment is robust and capable of accommodating severe accident challenges. The containment would remain intact almost 90% of the time.

2. Hydrogen igniters are provided to prevent the build-up of hydrogen inside the containment during a severe accident. However, the frequencies for the various release classes are relatively insensitive to the availability of the hydrogen igniters.

3. The reactor cavity of the System 80+ design is flooded following a severe accident to enable the corium to be cooled, which in turn limits the amount of concrete ablation. Late containment failure releases are somewhat sensitive to the ability to transfer heat from the corium to the cavity water.

4. Although research has indicated that temperature induced creep failure of the Reactor Coolant System (RCS) piping may occur following a severe accident, the System 80+ release classes were found to be insensitive to creep failure of the RCS piping.

SUMMARY AND CONCLUSIONS

The PRA analyses demonstrate that the System 80+ Standard Design has a very low risk as is shown by the low core damage frequency and the very low frequency for large releases. Also, risk is allocated among SSCs and between prevention and mitigation.

System 80+ has been designed to withstand beyond design basis events, with features such as a large containment, a large reactor cavity with thick concrete walls and floors, an in-containment refueling water storage tank for cavity flooding, and a rapid depressurization system for depressurizing the RCS. Therefore, even considering the remote possibility that a core melt condition developes, the System 80+ design is sufficiently robust to ensure that the operating staff has adequate time to mitigate the event progression and minimize radiation releases to the environment.

ABB-CE considered defense-in-depth for accident prevention and mitigation early in the design process and used robust design features to ensure that the System 80+ design achieved a low core damage frequency, low containment conditional failure probability, and excellent deterministic containment performance under severe accident conditions and to ensure that the risk was properly allocated among design features and between prevention and mitigation.

REFERENCES

1. NP-6780-L, Advanced Light Water Reactor (ALWR) Utility Requirements Document, Rev. 3, EPRI-1992.

2. Title 10, Part 52, Code of Federal Regulations.

3. System 80+ Standard Design, CESSAR -DC, ABB-CE Engineering, Amend. Q, June 30, 1993.

4. 10CFR100, "Reactor Site Criteria", U.S. nuclear Regulatory Commission

5. ANSYS Engineering Analysis System User's Manual, G. J. DeSalvo, and J. A. Swanson, Swanson Analysis Systems, Inc.

6. NUREG-75/087; "Standard Review Plan"; U. S. Nuclear Regulatory Commission

7. Title 10, Part 50.34(f), Code of Federal Regulations.

8. DOE/ID-10271, "Prevention of Early Containment Failure due to High Pressure Melt Ejection and Direct Containment Heating for Advanced Light Water Reactors", J. C. Carter, et. al., March 1990.

9. Cole, R.H., Underwater Explosions, Princeton University Press, Princeton, New Jersey, 1948.

Table 1 Major System 80+ Preventive and Mitigative Design Features	
TYPE OF FEATURE	DESIGN FEATURE
PREVENTIVE *Transient Prevention*	- Larger pressurizer - Larger secondary inventory in steam generators - High-pressure Shutdown Cooling System (SCS) - functionally interchangeable SCS and Containment Spray System (CSS) pumps - Multiple independent connections to the grid and turbine-generator runback capability - Dedicated startup feedwater system - Improved advanced control room design - Improved Component Cooling Water System (CCWS)/Station Service Water System (SSWS)
Transient Mitigation/ Severe Accident Prevention	- Four train Safety Injection System (SIS) with direct vessel injection - Safety Depressurization System (SDS) - Four train Emergency Feedwater System - Two emergency diesel generators and a standby alternate AC source (combustion turbine) - Six vital batteries - In-containment Refueling Water Storage Tank (IRWST) - Cross-connected CSS and SCS trains - Improved control room design
SEVERE ACCIDENT MITIGATION	- Large spherical steel dual containment - Reactor cavity designed for corium disentrainment - Reactor cavity designed for debris coolability - Cavity Flood System - IRWST and SDS - Hydrogen Mitigation System - Secondary Containment with Annulus Ventilation System - External Connection for Containment Spray System - Missile Protection

TABLE 2 COMPARISON OF CORE DAMAGE FREQUENCY CONTRIBUTIONS BY INITIATING EVENT				
INITIATING EVENT	SYSTEM 80		SYSTEM 80+	
	Core Damage Frequency	% Total	Core Damage Frequency	% Total
Large LOCA	1.8E-06	1.9	1.1E-07	6.6
Medium LOCA	3.6E-06	4.4	3.1E-07	18.5
Small LOCA	9.4E-06	11.6	2.1E-07	12.4
Secondary Side Break	9.0E-07	1.1	2.1E-09	0 .1
Steam Generator Tube Rupture	1.1E-5	12.9	3.0E-07	18.0
Transients	1.2E-05	15.4	5.7E-07	33.7
Loss of Offsite Power (Including SBO with Battery Depletion)	3.8E-05	46.4	2.8E-08	1.8
ATWS	4.8E-06	5.9	4.9E-08	2.9
Interfacing System LOCA	4.5E-09	0.0	5.2E-10	0.0
Vessel Rupture	1.0E-07	0.3	1.0E-07	5.9
TOTAL	8.1E-05	100.0	1.7E-06 [6.3E-07]*	100.0
Total - Without consideration of Common Cause Failure			2.15E-07	

* This value is used for comparison purposes only. It was derived using the same groundrules of the System 80 PRA.

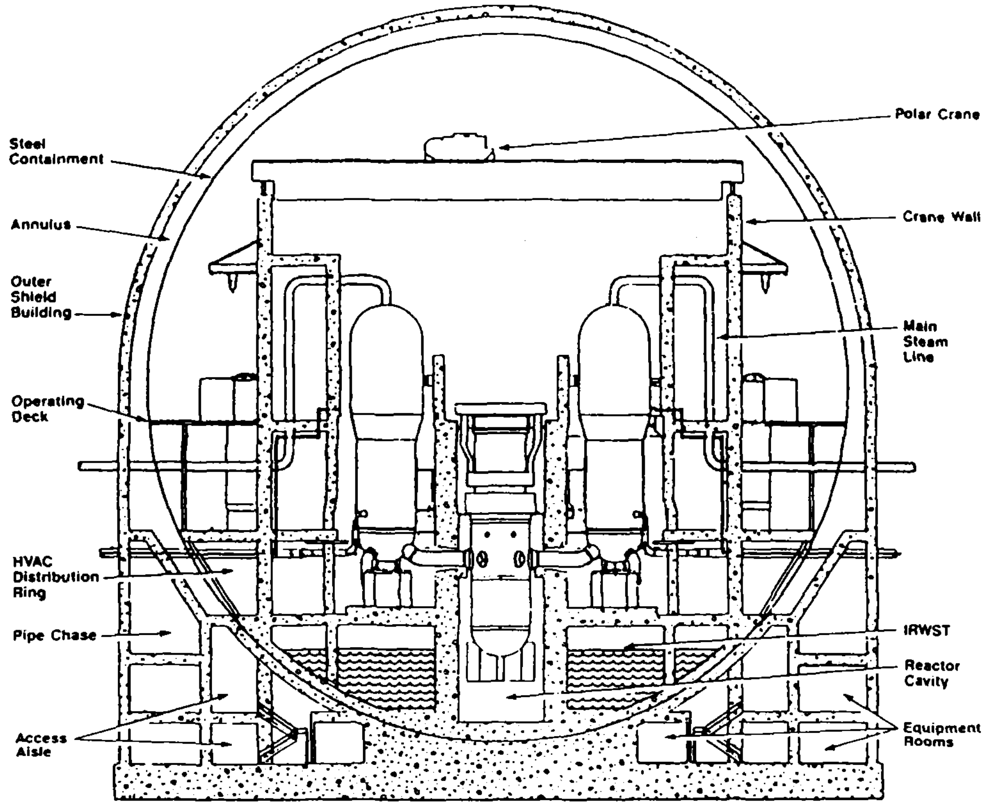

FIGURE 1
Elevation View of System 80+ Containment and Shield Building

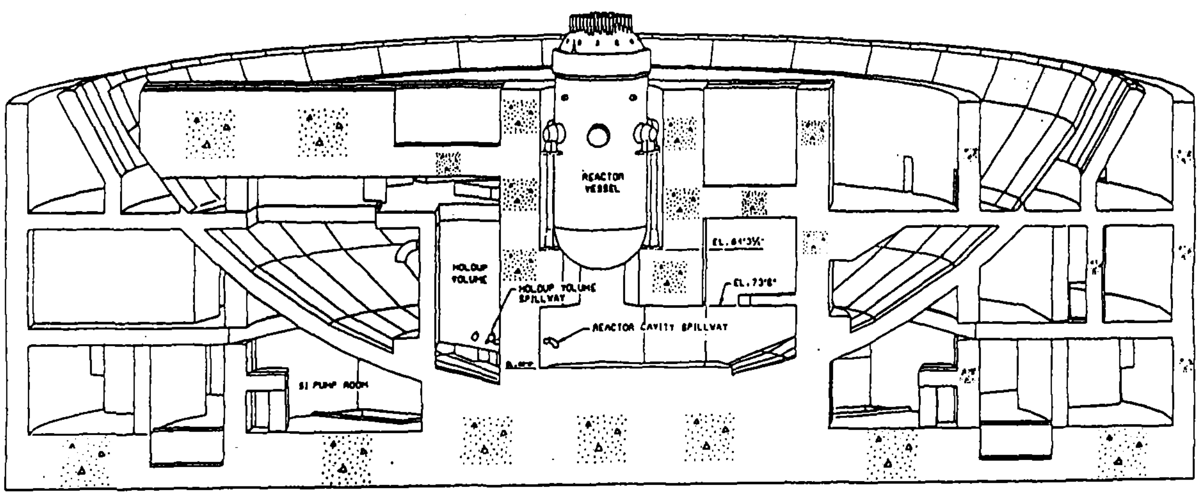

FIGURE 3
IRWST Spillway and Cavity Flooding System

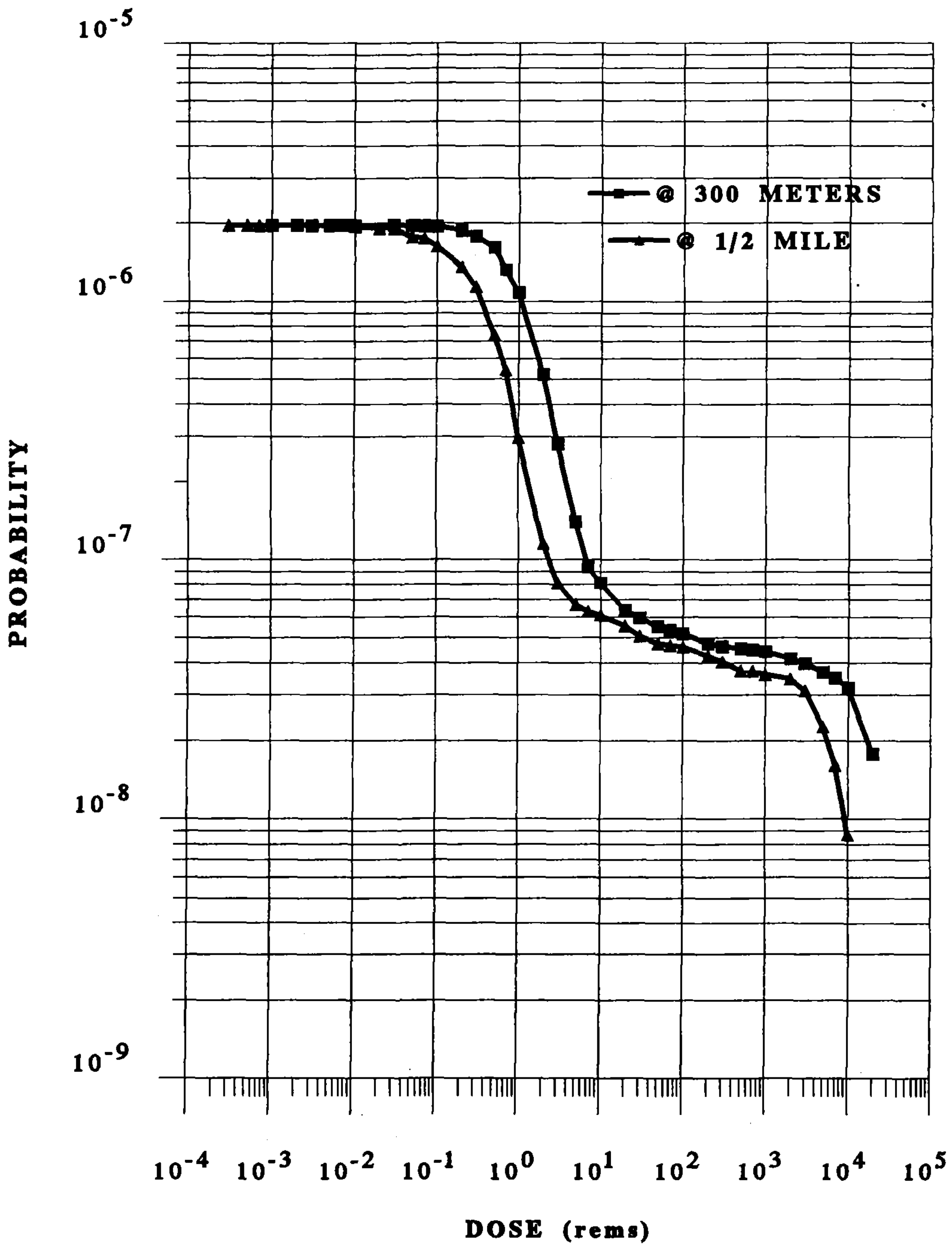

FIGURE 3
Wholebody Dose CCDF at 300 Meters and 1/2 Miles