August 9, 1993

Docket: 52-001

NOTE TO: Chet Poslusny, DAR, NRR

FROM: Glenn Kelly, SPSB, DSSA, NRR

Glenn Kelly

SUBJECT: UPDATED LIST OF INSIGHTS

I have enclosed a fax I sent to Jack Duncan, GE that seeks to determine if my updated list of PRA insights correctly incorporates, from GE's viewpoint, the insights drawn from the ABWR PRA.

Enclosure: as stated

See attached list

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Table 19.1-A

LIST OF IMPORTANT SAFETY INSIGHTS

Plant-Wide Insights

1) The COL Applicant is to perform a seismic walkdown following the procedures of EPRI NP-6041, revision 1 to insure that the as-built plant matches the assumptions in the ABWR PRA-based seismic margins analysis and to assure that spatial systems interactions do not exist. [ITAAC]

2) The integrity of divisions is a very important assumption in the ABWR PRA. The PRA assumes that no high pressure of high temperature piping lines penetrate walls or floors separating two different safety divisions. Piping penetrations are qualified to the same differential pressure requirements as the walls or floors they penetrate. [ITAAC]

3) To prevent inadvertent spray or dripping from failing equipment, electric motors are all of drip proof design and motor control centers have NEMA Type 4 enclosures. [Tier 2, SSAR Section]

4) The fire analysis assumes that the routing of piping or cable trays during the detailed design phase will confirm with the fire area divisional assignments documented in the fire hazard analysis. [ITAAC]

5) Subsection 9A.5.5 under "Special Cases - Fire Separation for Divisional Electrical Systems" lists the only areas of the plant where there is equipment from more than one safety division in a fire area. These should be the only areas where multiple divisions share the same fire area. [ITAAC]

Combustion Turbine Generator

The combustion turbine generator (CTG), in conjunction with the ac-independent water addition (ACIWA) system, have significantly reduced the estimated frequency of core damage from station blackouts (the dominant contributor to core damage in most BWR PRAs). In the ABWR SSAR, GE indicated that each of the emergency diesel generators (EDGs) and the CTG can be used to power any of the loads identified in the PRA success criteria by manually closing selected breakers (note: EDGs cannot power feedwater pumps). Even if offsite power is lost, the four onsite power sources can be used to power any safety or non-safety bus. This provides significant flexibility which helps reduce the risk from station blackout and selected bus power losses. Procedures must be prepared by the COL applicant to direct this manual transfer of an EDG to a non-safety bus. [COL Action Item]

An important assumption about the CTG is that no plant support systems are needed to start or run the CTG. The CTG starts automatically and safety grade loads are to be added manually. [ITAAC]

AC-Independent Water Addition System

This system is one of the single most important systems in the ABWR from the point of view of prevention and mitigation of severe accidents, since the accidents that have traditionally been identified in BWR PRAs as being the most challenging are station blackout and transients with failure of various ECCS or cooling systems. This system also provides benefits for fires, internal floods, shutdown events, seismic events, and events where containment cooling is lost. It can provide water (as vessel makeup or drywell spray) from a seismic category 1 diesel-driven pump or a fire truck.

The use of the system as a backup source of water to the drywell sprays is perhaps the single-most important feature for reducing the consequences of severe accidents in the ABWR. In this role the system serves to: (1) reduce containment overpressure and delay the time to actuation of COPS, (2) eliminate the potential for drywell overtemperature failure in those events in which debris may be dispersed to the upper drywell, and (3) mitigate the consequences of suppression pool bypass by condensing steam produced in the drywell.

The following are important aspects of the system, as represented in the PRA:
1. a fire protection pump -- seismic category 1, diesel-driven pump (i.e., ac-independent) [ITAAC],
2. connection provided outside of reactor building, which allows a fire truck to be used as a backup to the fire protection pumps [ITAAC],
3. system piping and valves configured to allow fire protection water to be used for either vessel makeup or drywell spray, but not both simultaneously [ITAAC],
4. all valves and controls needed for system operation can be accessed and manually operated in a straight-forward manner and can be operated successfully (including the environment the operator will be in) following an earthquake, internal flood, fire, or internal event [ITAAC],
5. check valves provided to prevent backflow from the reactor coolant system [ITAAC],
6. orifices installed in the associated piping to restrict the injection rates to the vessel and drywell sprays [ITAAC], and
7 seismic Category 1 water supply independent of the suppression pool and the condensate storage tank [ITAAC].

RCIC

RCIC is ac-independent and provides reliable high pressure injection. This makes RCIC particularly important in preventing station blackout from leading to core damage. In addition RCIC is very important for mitigation of control room fires or other emergencies that require the evacuation of the control room. The following capabilities are important for RCIC:

1. RCIC needs to be able to operate for 8 hours following a station blackout (using steam and dc power) and the batteries at the end of 8 hours need to have sufficient power in them to allow for RCS depressurization by the ADS. RCIC pump and turbine are assumed in the PRA to be able to operate for at least eight hours without room coolers. [COL Action Item 19.9.9]
2. For control room fires, the capability for local operation of RCIC outside the control room is very important. [ITAAC (capability to perform)][COL Action Item (existence of procedures)]
3. Sensitivity studies that increased SSC unavailabilities showed that an

increase in RCIC unavailability would cause the greatest increase in estimated core damage frequency of any SSC. RCIC also was found to be the most sensitive system to increased outage time assumptions. [COL Action Item to be included in RAP]

4. The suppression pool temperature up to which RCIC can operate is important for Class II sequences. The ABWR PRA assumes that RCIC can operate up to a suppression pool temperature of 76.7 °C (170 °F). [ITAAC]

Reactor Building Cooling Water (RCW)/Reactor Service Water (RSW)

The RCW and RSW systems are each designed with two parallel loops in each division. Each loop is capable of removing all component heat loads associated with the operation of the ECCS pumps. The parallel loops within each division substantially reduce the estimated core damage frequency. [ITAAC]

Automatic Standby Liquid Control System (SLCS) and Recirculation Pump Trip

The ABWR has a reliable and diverse scram system with both hydraulic and electric run-in capabilities to reduce the probability of an ATWS. SLCS and recirculation pump trip provide backup reactor shutdown capability. Automatic initiation of SLCS avoids the potential for operator error associated with manual SLCS initiation. [ITAAC]

Reactor Building

A flood in the reactor building could fail ECCS equipment and other important equipment. The following are assumptions in the ABWR internal flooding analysis that limit the chances and increase the mitigation capabilities of the ABWR design:

1. The volume of the reactor building corridor on level B3F that surrounds the three ECCS divisions is sufficiently large to handle the biggest break that can occur (water from the suppression pool). [ITAAC]
2. Suppression pool flooding in an ECCS room will reach equilibrium level below the ceiling of the ECCS room in which the flood occurred. [ITAAC]
3. Floor drains direct potential flood waters to rooms where sumps and sump pumps are located. The drain system is sized to withstand the maximum flood rate from a break in the fire water system. Sizing of the drain system is to include provisions for plugging of some drains by debris. [ITAAC]
4. Non-divisional drains will drain to the non-divisional sumps on appropriate floors. [ITAAC]
5. Floor B1F of the reactor building has overfill lines on the non-divisional sumps outside secondary containment. If the sump pumps fail or the flow rate exceeds the sump pump capacity, the lines will direct water to the non-divisional corridor of the first floor (B3F) inside secondary containment. [ITAAC]
6. A water seal in the overfill line is provided to maintain secondary containment integrity. [ITAAC]
7. The ABWR PRA flooding analysis assumes that on the B3F level, all wall and ceiling penetrations are above the maximum water level of all

potential floods. Doors communicating from the ECCS pump rooms to the corridor on the B3F level are water tight doors. [ITAAC]

8. If a flood were to occur during shutdown, some of the ECCS rooms may be open for maintenance. ABWR procedures specify that one safety division will be maintained intact at all times during shutdown. [COL Action Item 19.9.11(10)]

Similarly, a fire in the reactor building could damage important equipment. The smoke control system in secondary containment is important in helping to prevent the migration of smoke and hot gas layers from a faulted division to another. This is accomplished by pressurizing the surrounding areas so that the smoke will be contained. This capability and its adequacy should be confirmed. [ITAAC]

Control Building

Flooding in the control room can lead to core damage. The following design features are important in preventing flooding in the control building:

1. The ABWR internal flooding analysis assumes that flooding of the control building from the UHS cannot be maintained by gravity alone. To limit the consequences of a RSW line break, the RSW system will be designed so that the UHS cannot drain into the Control Building by gravity. [Interface Requirement]
2. To limit the consequences of a RSW line break, there is a maximum of 4000 meters of pipe (2000 each for supply and return) between the UHS and the RCW/RSW room, which can be discharged to the RCW/RSW room following RSW pump trip. [Interface Requirement]
3. Floor drains direct potential flood waters to rooms where sumps and sump pumps are located. The drain system is sized to withstand the maximum flood rate from a break in the fire water system. Sizing of the drain system is to include provisions for plugging of some drains by debris. [Interface Requirement]

Service Water Pump House

Previous PRAs and reliability studies have shown that loss of service water can be an important contributor to core damage. The service water pump house, which is outside the ABWR certification scope, is a building that must be designed to remove the following concerns:

1. Prevent fires or internal floods from impairing multiple safety trains. [ITAAC]
2. Prevent common cause failures such as intake blockage from debris from affecting multiple trains. [ITAAC]

Circulating Water System

Flooding from the circulating water system (an unlimited water supply) can lead to flooding of other buildings that do contain safety related equipment. The following design features help reduce the chances that a circulating water system break will cause core damage:

1. The circulating water system (CWS) has three pumps and each pump has an associated motor operated isolation valve. To limit the consequences of a circulating water system break in the Turbine Building, for cases where the heat sink is at an elevation higher than grade level of the turbine building, an additional isolation valve is installed in each line. [ITAAC]
2. Internal floods are prevented/mitigated in part by automatic actions and operator actions. To prevent flooding of areas surrounding the condenser pit, there are to be water level sensors (two-out-of-four-logic) to alarm to the control room if the water level gets too high in the pit and trip the circulating water and turbine service water pumps and close isolation valves in both systems. [ITAAC]

Turbine Service Water System

Flooding from the turbine service water system (an unlimited water supply) can lead to flooding of other buildings that do contain safety related equipment. The following design features help reduce the chances that a turbine service water system break will cause core damage:

1. The turbine service water system (TSW) has two pumps and each pump has an associated motor operated isolation valve. To limit the consequences of a turbine service water system break in the Turbine Building, for cases where the heat sink is at an elevation higher than grade level of the turbine building, an additional isolation valve is installed in each line. [ITAAC]
2. Internal floods are prevented/mitigated in part by automatic actions and operator actions. To prevent flooding of areas surrounding the condenser pit, there are to be water level sensors (two-out-of-four-logic) to alarm to the control room if the water level gets too high in the pit and trip the turbine service water and circulating water pumps and close isolation valves in both systems. [ITAAC]

Reactor Service Water System

Flooding from the Reactor Service Water (RSW) system (an unlimited water supply) can lead to core damage. The following design features help reduce the chances that a RSW system break will cause core damage:

1. A break in the RSW system can cause a flood in the Control Building that could lead to core damage. For this reason, an anti-siphon capability is installed in the RSW lines to prevent uncontrolled flooding of the Control Building should the RSW isolation valves fail to close on a RSW pipe break. [ITAAC]

2. Water level sensors will be installed in the reactor building cooling water (RCW)/reactor service water (RSW) rooms in the control building. These sensors are used to alert the operators to flooding in the rooms and send signals to trip RSW/RCW pumps and close isolation valves in the affected systems. The sensors are diverse and are arranged in a two-out-of-four logic. [ITAAC]

Reactor Water Cleanup System

The Reactor Water Cleanup (CUW) System provides some benefit in the ABWR PRA by removing decay heat at high pressure. It would only be used in this mode if the containment cooling mode of the RHR system was disabled. [Tier 2, SSAR Section]

The isolation valves in the RWCU system must be capable of isolating against a differential pressure equal to the operating pressure of the reactor coolant system in the event that there is a LOCA in the RWCU [ITAAC].

The reliability of these isolation valves should match the reliability assumed in the ABWR PRA [COL Action Item to include in RAP]. Temperature sensitive equipment in the reactor water cleanup system should be able to remain functional or should be isolated when the CUW system is used as a decay heat removal path at high temperatures. Temperature sensitive equipment such as the resin beds is to be isolated automatically on high water temperature or manually by operator action. The entire CUW system is not to isolate on high temperature of the incoming water [COL Action Item].

Ultimate Heat Sink

The ABWR PRA assumed that the service water system and the ultimate heat sink would work well in tandem to deliver adequate cooling to needed equipment. There was no detailed examination of these systems in the PRA since they are not in the Certification scope. The ultimate heat sink and the Service Water Pump house should be designed in such a manner so that common cause failure of service water is extremely low. A site-specific PRA must be developed by the COL applicant to show that there are no vulnerabilities (e.g., due to debris clogging of the intake, internal or external fires, external or internal floods) in the ultimate heat sink and the Service Water Pump House [Interface Item] [COL Action Item].

Remote Shutdown Panel

1) The ABWR PRA fire analysis found that use of the remote shutdown panel is very important in mitigating fires in the control room. The design of the remote shutdown panel was enhanced by GE adding controls for a fourth SRV (three needed to depressurize, plus one for a single failure). [ITAAC]
2) The ABWR decay heat removal reliability study found that operator actions making use of the remote shutdown panel were important during modes 3, 4, and 5. [COL Action Item (procedures)]

Residual Heat Removal System

The Residual Heat Removal (RHR) system is very important for the removal of decay heat during normal shutdown and in its ECCS function as low pressure core flooder. The following design features and assumptions are important for assuring the RHR system is capable of removing decay heat in various modes and for various accident and transients:

1. An important failure mode for beyond design bases earthquakes is the failure of the RHR heat exchanger in such a manner as to drain the suppression pool. In the ABWR PRA-based seismic margins analysis, the

RHR heat exchanger is assumed to have a HCLPF of 0.7g. [COL Action Item to be added to DRAP to check seismic capacity of equipment]

2. In modes 3, 4, and 5, the permissives and inhibits associated with the RHR Mode switch ensure that valve line ups are correct for most RHR functions, thereby helping to prevent inadvertent diversion of water from the RPV. [ITAAC]
3. The ABWR PRA and the DHR reliability study have shown that it is important for the RHR not to fail as an intersystem LOCA. The RHR system has the capability to withstand normal reactor system pressures without the piping reaching its ultimate capacity. The DHR reliability study indicated that RHR valve interlocks are important in preventing low pressure RHR piping from being inadvertently connected to systems at high pressure. [ITAAC]
4. The ABWR DHR reliability study determined a number of configurations of equipment for modes 3, 4, and 5 such that the estimated core damage frequency from decay heat removal failure conservatively was less than 1 in a million per year. An important assumption in this study was that the three RHR trains would be configured as follows during modes 3, 4, and 5: One loop would be isolated, in standby, and operable with no equipment in maintenance; a second loop would be the operating decay heat removal loop; the third loop would be in maintenance. [COL Action Item]
5. Shutdown cooling piping connects to a nozzle in the RPV at an elevation that is above the top of the active fuel. This reduces the chances of uncovering the core by vessel drain down. [ITAAC]
6. When in the shutdown cooling mode, some operating plants have experienced loss of decay heat removal on loss of power to logic circuits. For the ABWR design, the RHR system does not isolate on loss of logic power. [ITAAC]

<u>High Pressure Core Flood System</u>

(1) HPCF pump B can be operated independently of the essential multiplexing system. This feature is an important factor in reducing the chances of the plant going to core damage since this design should reduce the chance of a common cause failure disabling all ECCS pumps. [ITAAC]

(2) The HPCF pumps will be able to pump water as hot as 171 °C (340 °F). [ITAAC]

<u>Three ECCS Trains</u>

The barrier between each of the three safety divisions in the ABWR is at a minimum a 3 hour fire barrier that also resists internal flood pressures. This design assumption significantly reduces the chance of an internal flood or fire propagating and causing core damage. [ITAAC]

<u>Piping Upgrades to Prevent ISLOCAs</u>

In SECY 93-087 it was recommended that ALWR designers reduce the possibility of a loss of coolant accident outside of containment by confirming that all systems (to the extent practical) and subsystems connected to the reactor coolant system (RCS) can withstand full RCS pressure. Intersystem LOCAs are a concern because many releases associated with them are not contained, held up, or scrubbed, but rather are released directly to the environment. GE has assured that the interfacing systems to the RCS can withstand full RCS pressure. [ITAAC]

Lack of Recirculation Piping

There are no large pipes (i.e., > 2 inches in diameter) that penetrate the ABWR vessel below the level of the core. This has virtually eliminated LOCAs as a severe accident concern for the ABWR. [Tier 2, SSAR Section]

Electrically Driven Control Rod Insertion

In many BWR PRAs, ATWS is a significant contributor to core damage frequency and risk. The diversity (electrically driven) of the fine motion control rod system is important in lowering the estimated core damage frequency for ATWS events for the ABWR. [ITAAC]

Electrical Wiring Penetrations

Wiring penetrations between divisions should be rated as three hour fire barriers and should be capable of preventing water/oil from an internal flood from migrating to another division. [ITAAC]

DC Power Supply

The ABWR PRA expects that loss of all dc power will lead to core damage. The ACIWA system is a very important (for severe accidents) low pressure system, and ADS, which is needed for reactor depressurization, requires dc power to operate. In the ABWR design, failure of the batteries during a large seismic event will prevent the diesel generators from starting and loading. Batteries are the only non-building SSC that could, by themselves, decrease the HCLPF of any accident sequence below 0.5g. This would occur if the HCLPF of the batteries were to fall below 0.5g. The dc power supply should be well anchored and carefully designed to handle a design bases 0.3g earthquake. The ABWR PRA-based seismic margins analysis assumed that the HCLPF of the dc power system (batteries and inverter) is 1.1g [COL Action Item].

The emergency batteries provide an important backup to the inverters for providing DC power. For this to be assured, the seismic failure modes of the inverters and their AC supply must not allow an electrical fault to be propagated to the DC busses. The reverse case is also true (the inverters provide backup should the batteries fail). For this to be assured, the seismic failure modes of the batteries must not allow an electrical fault to be propagated to the DC busses. [ITAAC]

Safety System Logic and Control

There are four divisions of self-tested safety system logic and control (SSLC) instrumentation (two-out-of-four logic). The ABWR PRA assumes that this will be a highly reliable configuration to actuate ESF core cooling and heat removal system as well as actuating the CRD scram system for defense against ATWS events. Assumptions about SSLC reliability and redundancy in the PRA substantially reduce the estimated core damage frequency. [COL Action Item to be added to DRAP]

Off-line testing for faults not detected by the continuous self-test feature were judged to be important in the PRA analysis [COL Action Item to be included in RAP].

Fire Truck

The ACIWA makes use of a fire truck connection to provide water if the motor and diesel-driven pumps are unavailable. The PRA assumes the reliability of the fire truck is 0.99. [COL Action Item to include fire truck reliability in DRAP]

Reactor Pressure Vessel Isolation on Low Water Level

The ABWR shutdown reliability study indicated that the isolation of lines connected to the RPV on a low water level signal in modes 3, 4, and 5 prevents uncovering of the fuel for many potential RPV drain down events. [ITAAC]

Operator Check That Watertight Doors Are Dogged

The internal flooding analysis assumes that all watertight doors are closed and dogged to prevent floods from propagating from one area to another. The watertight doors are alarmed to alert the control room operator that a watertight door is open, but will not alarm to indicate that a door is not dogged. To guard against a door being left undogged, operators should check the doors every shift to assure that they are closed and dogged. [COL Action Item]

Suppression Pool Bypass

The suppression pool is an important containment feature for severe accident progression and fission product removal, since releases from the reactor vessel are either directly routed to the pool (e.g., transients with actuation of ADS) or pass through the pool via the drywell-wetwell connecting vents. However, the suppression pool function can be compromised in the ABWR design in the following ways:

- a single failure of a wetwell/drywell vacuum breaker (i.e., a stuck open vacuum breaker), or by excessive leakage of one or more vacuum breakers
- unisolated main steam line breaks
- rupture of the SRV discharge line(s) in the wetwell air space
- inadvertent opening and failure to close sample lines, drywell purge lines, and containment inerting lines
- unisolated LOCAs in the reactor water cleanup and RCIC systems

The following are important to assuring a low risk from wetwell/drywell vacuum breaker bypass, as modelled in the PRA and are to be included in DRAP:

1. a low probability of vacuum breaker leakage (PRA assumes a leakage probability of 0.18 per demand on system)
2. a low probability that the vacuum breakers fail to close (PRA assumes a failure to close probability of about 0.0005 per demand per valve)
3. a high availability of drywell or wetwell sprays (and ACIWA as a backup) to condense steam which bypasses the suppression pool.
4. a position indication switch on each vacuum breaker valve that will indicate the valve to be open should the gap between the disk and seating surface exceed 0.9 cm. (A gap less than 0.9 cm is necessary to

assure credit for aerosol plugging taken in the GE analysis.) [ITAAC]

5. placement and shielding of the vacuum breakers such that pool swell associated with COPS actuation will not impact operation of the valves. [ITAAC]

In addition, it is important to assure that the vacuum breakers are closed. To achieve this control room alarms will be installed to indicate if all the vacuum breakers are closed. (This reduces the potential for suppression pool bypass by assuring that the plant is not operated with a stuck open vacuum breaker, and that pre-existing leakage paths will be limited to small flow areas.) [ITAAC]

The following are important to assuring a low risk from unisolated main steam line breaks:

1. two air-operated, spring close, failed closed isolation valves in each line [ITAAC].
2. automatic MSIV actuation by redundant solenoids through two-out-of-four logic [ITAAC].

The following are important to assuring a low risk from rupture of the SRV discharge lines, particularly in seismic events:

1. discharge lines are designed and fabricated to Quality Group C requirements [ITAAC].
2. welds in the airspace region of the wetwell are non-destructively examined to the requirements of ASME Section III, Class 2 [ITAAC]
3. discharge lines are capable of accommodating seismic events at an acceleration level of 0.6g with a high confidence that there is a low probability of failure (HCLPF) [COL Action Item to add to DRAP].

The following is important to assuring a low risk from suppression pool via the sample, drywell purge, and containment inerting lines:

1. lines will be sealed closed during power operation, and under administrative control [COL Action Item]

The following are important to assuring low risk from LOCAs outside containment:

1. redundant and seismically-qualified CUW system isolation valves, qualified to close under postulated break conditions [ITAAC]
2. blowout panels in the RCIC and RWCU divisional areas which prevent overpressurization and impacts on equipment in adjacent areas and other divisions [ITAAC]
3. reliable seating of redundant feedwater, SLC, and ECCS discharge check valves [ITAAC] [COL Action Item to add to DRAP]

Lower Drywell Design

The design of the ABWR lower drywell/reactor cavity is such that there is a low probability that the cavity will be flooded at the time of reactor vessel failure, but a high probability that the cavity will be flooded subsequent to vessel failure. A dry cavity at the time of vessel failure reduces the potential for large ex-vessel steam explosions, whereas the subsequent flooding of the cavity helps minimize the impact of core concrete interactions.

The following ABWR design features are important to assuring a dry cavity at the time of vessel failure:

1. lack of any direct pathways by which water from the upper drywell (e.g., from drywell sprays) can drain to the lower drywell, other than by overflow of the suppression pool, [ITAAC]
2. negligible probability of premature or spurious actuation of the passive flooder valves at temperatures less than 500 F or under differential pressures associated with reactor blowdown and pool hydrodynamic loads [ITAAC on flooder configuration] [COL Action Item to be added to RAP], and
3. a capability to accommodate approximately 2.0 E+6 kg of water in the suppression pool before the pool overflows into the lower drywell. [configuration ITAAC].

The following features are important to assuring reactor pedestal and containment integrity for beyond 24 hours following reactor vessel failure, and to rendering CCI-induced containment failure a relatively insignificant contributor to risk. [configuration ITAAC]

4. a 1.7m thick reactor pedestal capable of withstanding approximately 1.55m of erosion from CCI without loss of structural integrity [ITAAC],
5. the use of basaltic concrete in the floor of the lower drywell, which minimizes the production of non-condensible gases [ITAAC],
6. a sump shield to prevent core debris from entering the lower drywell sump [ITAAC], and
7. the lower drywell flooder system [ITAAC]

Note: The lower drywell flooder system in the ABWR provides a passive means of adding water to the lower drywell following reactor vessel breach. This water would cover the core debris, thereby enhancing debris coolability, cooling the drywell, and providing fission product scrubbing. The passive flooder system is a backup to other means of lower drywell water addition in the ABWR, including: (1) continued water addition through the breached reactor vessel and (2) suppression pool overflow as a result of water addition from water sources outside containment. PRA-based sensitivity studies indicate that the incremental risk reduction offered by the passive flooder is system is minimal. This is because of credit taken in the ABWR for continued water addition using the ACIWA mode of RHR.

Containment Ultimate Pressure Capacity

The ultimate pressure capacity of the ABWR containment is limited by the drywell head, whose failure mode is plastic yield of the torispherical dome. Subsequent to the original SSAR submittal, GE increased the ultimate pressure capability of the drywell head from 100 psig to 134 psig, and increased the COPS setpoint from the original value of 80 psig to the final value of 90 psig. The strengthening of the drywell head increases the ability of the containment to withstand rapid pressurization events, such as direct containment heating, without loss of structural integrity, and provides additional margin between the COPS setpoint and the drywell failure pressure, thereby reducing the potential for drywell failure prior to COPS actuation. The drywell head is the limiting component in the containment pressure boundary during slow overpressure events. [Tier 2, SSAR Section]

Containment Overpressure Protection System (COPS)

COPS is part of the atmospheric control system in the ABWR, and consists of a pair of rupture disks installed in a 10-inch diameter line which connects the wetwell airspace to the stack. COPS provides for a scrubbed release path in the event that containment pressure cannot be maintained below the structural limit of the containment. Without this system, late containment overpressure failures would be expected to occur in the drywell, resulting in unscrubbed releases. COPS provides a significant benefit by reducing the source terms for late releases, and minimizing the potential for containment-failure-induced loss of core cooling (e.g., in Class II sequences). The following are important features of the system, as modelled in the PRA:

1. rupture disk actuation at 90 psig +/- 5% [Tier 2, SSAR Section]
2. piping (and disk) designed to flow steam at a rate equivalent to 2% reactor power, and accommodate peak pressure loads associated with system actuation [ITAAC]
3. no normally-closed or automatic isolation valves in vent path [Tier 2, SSAR Section]
4. two normally-open, fail-open isolation valves in the vent path, manually operated from the control room, with key-lock switches [Tier 2, SSAR Section]
5. capability of related isolation valves to close against full vent pressure [Tier 2, SSAR Section].

Containment Inerting System

Because the ABWR containment will be inerted during power operation [ITAAC], hydrogen combustion is not considered to be an important containment challenge, and was not modelled in the PRA.

To assure the validity of this treatment, strict controls must be placed on the period of time that the reactor can be operated with the containment de-inerted [Technical Specifications].

Direct Containment Heating (DCH)

DCH is the only severe accident phenomena that represents a significant challenge to containment integrity (5% probability of containment failure given reactor vessel failure at high pressure). The impact of DCH is "controlled" in ABWR by reducing the frequency of high pressure reactor vessel failure using ADS (30% of vessel failures). The following aspects of ADS should be assured by ITAAC and RAP:

1. reliability/availability consistent with Level 1 PRA assumptions [DRAP],
2. no dependency on ac-power [ITAAC],
3. availability of sufficient DC power to actuate ADS in a long term station blackout (following loss of RCIC due to battery depletion) [Tier 2, SSAR Section][COL Action to add to DRAP]

There are no specific ABWR containment design feature to deal with DCH loads other than the general arrangement of the drywell and wetwell, and connecting vents, which provide for a series of 90-degree bends that debris must traverse in order to reach the upper drywell. [configuration ITAAC]

Important Human Actions

Human actions with high risk impact for the ABWR were identified based on the PRA and supporting analyses. Section 19D.7 of the SSAR includes a listing of these actions, classified into three categories corresponding to the COL-actions necessary to assure the validity of the PRA treatment of the action: (1) critical tasks, (2) maintenance items, and (3) COL procedures and planning.

1. The items identified as "critical tasks" in 19D.7, as well as actions to recover emergency diesels, have the greatest impact on core damage frequency and risk for the ABWR. Accordingly:
   - these actions are to be addressed by the COL-applicant as part of the detailed design of human-system interfaces
   - the following will be provided for each action:
     1. clear unambiguous indication of conditions requiring the action
     2. the operator must have the capability to perform the action in a straight forward manner
     3. the operator must have clear written operating procedures regarding the actions to be taken
     4. the operator must have thorough training in the conditions requiring the action.

   [COL Action Item]
2. The probability of miscalibrating single and multiple sensors was assigned very low values on the basis that the COL-applicant would incorporate a special procedure governing calibration activities. At a minimum, the COL-applicants maintenance procedures for sensor calibration should require that whenever a sensor is found to be out-of-tolerance, before the sensor is recalibrated, the calibration instrument is first checked or an alternate instrument is used to confirm the condition.

   [COL Action Item]
3. For items identified as "COL Procedures and Planning" items, the COL-applicant is to develop procedures to assure that these actions can be effectively implemented. [COL Action Item]

Importance/Uncertainty Analyses

Examination of the top ten events contributing to uncertainties in the estimate of the ABWR core damage frequency (CDF) revealed that nine of these events were identified by importance analyses as leading contributors to CDF.

The highest contributor to uncertainties in the CDF as well as the CDF estimate was RCIC test and maintenance. The remaining top contributors to uncertainties (and CDF) are listed in SSAR Table 19D.10-5. [COL Action Item to add to ORAP]