19QC Review of Significant Shutdown Events: Electrical Power and Decay Heat Removal

19QC.1 Review of Significant Shutdown Events

A review was made of operating events involving loss of offsite power (LOOP) and loss of Decay Heat Removal (DHR). These two areas appear to have the greatest potential for causing core damage during shutdown based on past experience. The sources utilized for information on past shutdown events were:

- "Residual Heat Removal Experience Review and Safety Analysis", NSAC-88, March 1986
- "Loss of Vital AC Power and the Residual Heat Removal System during Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990", NUREG-1410, June 1990
- "NRC Staff Evaluation of Shutdown and Low Power Operation", NUREG-1449, March 1992
- Selected INPO Reports (e.g., SOE and SOCR summaries) and NRC Information Notices.

The results of this evaluation are contained in Tables 19QC-1 and 19QC-2 for LOOP and loss of DHR, respectively.

A review of EPRI TR-1003113, "An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989-2000)," provided an additional review of more recent shutdown operating experience. The shutdown operating experience discussed in EPRI TR-1003113 does not identify any new or unique challenges to shutdown safety that were not covered in the ABWR DCD.

19QC.1.1 Summary of DHR Events

Not all of the events discussed in NSAC-88 are contained in Table 19QC-2. Those events that were due to random failures of single components and did not result in loss of DHR or other significant plant effects were not evaluated further. If the single failure resulted in loss of coolant, over-pressurization, flooding, or loss of Shutdown Cooling (SDC) function, the event was included and the applicable ABWR feature to prevent or mitigate the event was discussed.

19QC.1.2 Summary

The results of this review demonstrate that ABWR design includes many features that will prevent or mitigate unacceptable consequences of typical past events.

The main features of the ABWR that will prevent or mitigate shutdown events are:

- Three divisions of ECCS and support systems that are physically and electrically independent

- Two independent offsite power sources
- Four onsite power sources (three emergency diesel generators and one combustion turbine generator)
- Plant configuration to minimize common mode failures due to fire and floods
- Appropriate Technical Specifications and administrative controls to ensure availability of systems during periods of potentially high risk operations
- Several alternate means of DHR if normal systems were to fail or be out of service for maintenance
- Instrumentation availability during shutdown to monitor plant safety status and initiate safety systems when needed

Table 19QC-1 Loss of Offsite Power Precursors

Event	Description	Applicable ABWR Features
Indian Point 1 and Yankee Rowe (11/9/65)	“Great Northeast Blackout”	ABWR has two independent offsite power sources. These are backed up by three physically and electrically separate trains of Class 1E AC power each containing an emergency diesel generator. These are further backed up by a permanent onsite Combustion Turbine Generator (CTG) which is capable of powering any one of the three trains if all three diesels were to fail. The CTG is also capable of supplying power to non-safety busses such that condensate pumps can also be used to provide reactor coolant make up.
Point Beach 1 (2/5/71)	Loss of all transmission lines, failure of three transformer differential relays, causing transformer lockout.	See discussion of Indian Point 2 and Yankee Rowe (11/9/65).
Ginna (3/4/71)	Plant siding fell into 34.5 kV line connecting sole startup transformer.	ABWR has two independent transformers powered by two independent offsite power supplies which reduces the probability of losing offsite power. In the event of losing offsite power, features described under Indian Point 2 and Yankee Rowe (11/9/65) can mitigate the event.
Palisades (9/2/71)	Transmission line fault, isolation breaker failure. Backup relay isolated 345 kV bus.	See discussion of Ginna (3/4/71).
San Onofre 1 (6/7/73)	138 kV auxiliary transformer out for maintenance. Ground fault operated differential relays, de-energizing other auxiliary transformers.	ABWR uses three auxiliary transformers. Each powers one of the three Class 1E and non-1E buses. In addition, a reserve auxiliary transformer is available to power all three Class 1E buses. CTG is also available which can power 1E and non-1E busses without using the auxiliary transformers.
Oconee 1 (1/4/74)	230 kV switchyard isolated, 100 kV offsite source remained energized to supply power to the plant.	The ABWR also has two sources of offsite power.

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Fort Calhoun (3/13/75)	Sole 161 kV backup offsite transmission line out for maintenance. 345 kV output breaker tripped (faulty protective relays), opening remaining connection to offsite power. Offsite power could have been supplied from 345 kV switchyard by opening generator disconnects.	See Indian Point 2 and Yankee Rowe (11/9/65) and Ginna (3/4/71).
Turkey Point 4 (5/16/77)	Loss of Offsite Power (LOOP)	See Indian Point 2 and Yankee Rowe (11/9/65).
Connecticut Yankee (6/26/76)	Protective relays operated when lines were re-energized after service, causing LOOP.	See Indian Point 2 and Yankee Rowe (11/9/65).
Indian Point 2 (7/13/77)	LOOP due to lightning strikes. Emergency Diesel Generators (EDGs) operated.	See Indian Point 2 and Yankee Rowe (11/9/65).
St. Lucie 1 (5/14/78)	Substation switching error.	ABWR has two offsite power sources so probability of one switching error resulting in loss of all offsite power is low. But if it were to occur, mitigation features exist as discussed in Indian Point 2 and Yankee Rowe (11/9/65).
Turkey Point 3 (4/4/79)	Loss of all 7 transmission lines due to weather.	See Indian Point 2 and Yankee Rowe (11/9/65).
Davis Besse (4/19/80)	One EDG out for maintenance. One 13.8 kV bus connected, other energized but not connected. Ground fault on 13.8 kV bus caused loss of non-nuclear instruments. Air was pulled into DHR pump, and pump was stopped by operator. Pump vented and restarted after 2-1/2 hours.	ABWR has two sources of non-1E power. A ground fault on one would not result in loss of all non-1E power. In addition, if all non-1E power were to be lost, no valves connected to the RHR System would automatically cycle and cause loss of NPSH to any RHR pump. Also, the ABWR has three independent (100%) RHR Systems such that loss of one would not result in loss of the ability to remove decay heat.
San Onofre 1 (4/22/80)	Maintenance error caused LOOP.	See St. Lucie 1 (5/14/78).
Prairie Island 1 (7/15/80)	Weather related LOOP.	See Indian Point 2 and Yankee Rowe (11/9/65).
San Onofre 1 (11/22/80)	Maintenance error caused LOOP.	See St. Lucie 1 (5/14/78).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Diablo Canyon 1 (10/16/82)	LOOP caused by brush fire.	See Indian Point 2 and Yankee Rowe (11/9/65).
Farley 2 (10/8/83)	Switchyard breaker failure during refueling.	See St. Lucie 1 (5/14/78).
Palisades (1/8/84)	Deliberate de-energization of offsite power to isolate faulty breaker. One EDG out for maintenance, other available but its service water pump was out for maintenance, and operators failed to recognize this before authorizing work on breaker. Available EDG overheated and was manually tripped.	See Indian Point 2 and Yankee Rowe (11/9/65) and Ginna (3/4/71). ABWR technical specifications require one offsite and one onsite power source be available at all times.
Sequoyah 1 (3/26/84)	Ground short on 500 kV switchyard breaker de-energized transformer. Startup transformer supplied power.	A similar event at an ABWR could be more easily mitigated due to the existence of the CTG and three EDGs.
Yankee Rowe (5/3/84)	One 115 kV line out for maintenance, other energized. Normal supply transformer energized. Temporary fault detection relay caused breakers from normal supply transformer to open.	See Sequoyah 1 (3/26/84).
Salem 1 (6/5/84)	One of three safety buses was out of service for maintenance and one of the batteries in the two remaining safety trains was out of service for replacement. Automatic transfer relay which should have energized this bus was removed and placed in Unit 2 and not replaced in Unit 1, loss of power to two buses resulted in two operable EDGs to start but loss of DC control to one of the trains prevented closing of the EDG output breaker. One EDG did energize one bus but EDG cooling water pump was powered by EDG which lost control power. EDGs ran for two hours without cooling water.	Each of the three ABWR safety trains have separate independent emergency power supplies and support systems so each diesel can supply power to its own cooling water pump. ABWR technical specifications require one offsite and one onsite power supply be available at all times.

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Connecticut Yankee (8/24/84)	One 115 kV transmission line out for maintenance, one auxiliary transformer out for maintenance. Differential relay opened breakers to remaining auxiliary transformer.	See San Onofre 1 (6/7/73).
Point Beach 2 (10/22/84)	Breaker alignment errors during cross-tie between units caused LOOP.	ABWR design does not allow crossties between plants.
Indian Point 3 (11/16/84)	Object from roof fell onto startup transformer.	See San Onofre 1 (6/7/73) and Ginna (3/4/71).
Turkey Point 3 (4/29/85)	Startup and C transformer were both out of service. Offsite power supplied through main transformer. Relay failure resulted in loss of main transformer and LOOP. One EDG started and loaded its safety bus.	See Indian Point 2 and Yankee Rowe (11/9/65).
Turkey Point 3 (5/17/85)	Brush fire disabled station.	See Indian Point 2 and Yankee Rowe (11/9/65).
Waterford 3 (12/12/85)	Lightning caused loss of preferred offsite power source. Two EDGs started and loaded. Two sources of offsite power were available.	See San Onofre 1 (6/7/73).
Fort Calhoun (3/21/87)	One EDG and alternate offsite power source were out for maintenance, controls for other EDG bypassed to prevent auto-start. Maintenance error tripped offsite power; EDG had to be manually loaded.	See Indian Point 2 and Yankee Rowe (11/9/65) and Palisades (1/8/84).
Yankee Rowe (6/1/87)	Maintenance error caused loss of 2 of 3 safety buses.	See Sequoyah 1 (3/26/84).
McGuire 1 (9/16/87)	One offsite power source and 1 EDG out for maintenance. Test error caused loss of other offsite power source. Remaining EDG started and loaded. Offsite power restored after 25 minutes.	See Indian Point 2 and Yankee Rowe (11/9/65).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Crystal River (10/14/87)	One EDG out for maintenance. Safety buses cross-tied. Maintenance error caused loss of 1 safety bus. Cross-connect breaker then tripped and locked out. Dead bus transfer was required to close one cross-connect breaker. This required shutting the running EDG and resetting the under voltage lockout.	See Indian Point 2 and Yankee Rowe (11/9/65) and San Onofre 1 (6/7/73). Also, ABWR design does not allow safety buses to be cross-tied. Therefore, this event cannot occur in ABWR.
Crystal River (10/15/87)	One safety bus and its EDG out for maintenance. Maintenance error grounded offsite power supply. Remaining EDG started and loaded.	See Indian Point 2 and Yankee Rowe (11/9/65) and San Onofre 1 (6/7/73).
Wolf Creek (10/16/87)	One safety bus and 1 EDG out for maintenance. Error de-energized other bus. EDG output breaker opened and would not close due to anti-pump circuit preventing reclosure once it had been opened after EDG started on under- voltage. DHR lost for 17 minutes.	Anti-pump circuitry has been redesigned in the ABWR to allow closure following breaker trip when required.
Oconee 3 (9/11/88)	All offsite power going through 1 breaker. Maintenance error caused this breaker to open, and it could not be reclosed. No instruments to determine actual level and temperature of water in reactor core region (incore thermocouples not yet reconnected, and no power to RPV level transmitters).	See Indian Point 2 and Yankee Rowe (11/9/65). ABWR RPV water level instruments are powered by batteries and at least two divisions are required to be operable during shutdown to support ECCS automatic initiation functions.
Surry 1 and 2 (4/6/89)	Electrical fault and transformer lockout. This de-energized one safety bus in each unit. Unit 2 EDG started and loaded. Unit 1 EDG control in manual.	See Indian Point 2 and Yankee Rowe (11/9/65) and Point Beach 2 (10/22/84).
Diablo Canyon 1 (3/7/91)	Maintenance error caused power arc and LOOP. EDGs started and loaded.	See Indian Point 2 and Yankee Rowe (11/9/65).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Nine Mile Point (11/17/73)	One transmission line out for maintenance. Maintenance error caused loss of other line.	See Indian Point 2 and Yankee Rowe (11/9/65).
Pilgrim (4/15/74)	Lightning caused loss of all 345 kV lines. 23 kV line remain energized.	See Indian Point 2 and Yankee Rowe (11/9/65).
Pilgrim (5/26/74)	All 345 kV lines de-energized (cause unknown). 23 kV line remained energized.	See Indian Point 2 and Yankee Rowe (11/9/65).
Brunswick 2 (3/26/75)	One train of 230 kV buses for each unit out for maintenance. Relay error caused breakers on all five lines supplying remaining buses to open.	See Indian Point 2 and Yankee Rowe (11/9/65).
Quad Cities 2 (2/13/78)	Reduced voltage on grid caused under-voltage relays to trip breakers on both safety buses. System dispatcher increased grid voltage.	See Indian Point 2 and Yankee Rowe (11/9/65). ABWR has an alarm at 95% of rated voltage (degraded voltage). This gives operator 5 minutes to restore full voltage before offsite breakers would open.
FitzPatrick (3/27/79)	Maintenance error caused LOOP.	See Indian Point 2 and Yankee Rowe (11/9/65).
Browns Ferry 1 and 2 (3/1/80)	Ice storm caused loss of both offsite lines. Power supplied by Unit 3.	See Indian Point 2 and Yankee Rowe (11/9/65).
Monticello (4/27/81)	4.16 kV breaker was racked out under load. Breaker then shorted, causing loss of both safety buses.	See San Onofre 1 (6/7/73).
Quad Cities 1 (6/22/82)	Not really an event: Unit 1 supplied Unit 2 when Unit 2 scrammed.	See Browns Ferry 1 and 2 (3/1/80).
Pilgrim (10/12/82)	Storms failed 345 kV lines. 23 kV remained energized.	See Indian Point 2 and Yankee Rowe (11/9/65).
Brunswick 1 (4/26/83)	One offsite power source out for test. Maintenance error caused loss of second source resulting in LOOP.	See Indian Point 2 and Yankee Rowe (11/9/65).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Fort St. Vrain (5/17/83)	One EDG out for maintenance. 2nd EDG in parallel with offsite power. Storm caused LOOP, and 2nd EDG tripped on overcurrent due to faulty load sequencer and operating non-essential loads.	For the ABWR, the CTG could be used to power one of the safety buses if offsite power was not secure. In event of LOOP from any sources, features described under Indian Point 2 and Yankee Rowe (11/9/65) would mitigate the event.
Pilgrim (8/2/83)	Lightning caused loss of all 345 kV.	See Indian Point 2 and Yankee Rowe (11/9/65).
Oyster Creek (11/14/83)	Fire caused loss of power to 1 startup transformer. Switchyard de-energized to permit cleanup. Main generator disconnect links were removed, which allowed for use of unit transformer if necessary (wasn’t used).	See Ginna (3/4/71) and Sequoyah 1 (3/26/84).
Monticello (6/4/84)	One reserve transformer, 1 safety bus, 1 EDG out for maintenance. Procedure error caused loss of energized bus.	See Indian Point 2 and Yankee Rowe (11/9/65).
Quad Cities 2 (5/7/85)	Unit 2 dedicated EDG out for maintenance. Maintenance error caused LOOP to Unit 2. Unit 1 plus swing EDG powered Unit 2.	See Indian Point 2 and Yankee Rowe (11/9/65) and Browns Ferry 1 and 2 (3/1/80).
Millstone 1 (11/21/85)	Reserve station transformer out for maintenance. EDG out for maintenance. Maintenance error caused loss of 345 kV supply.	See Indian Point 2 and Yankee Rowe (11/9/65).
Peach Bottom 3 (4/13/86)	Explosion and fire in transformer caused loss of 1 startup transformer. Alternate startup transformer supplied power.	In the ABWR design, loss of the preferred offsite power source would result in all three emergency diesels starting and picking up respective 1E buses. Power could be manually transferred to the alternate preferred power source (reserve transformer) if desired depending on offsite power reliability.
Hope Creek (5/2/86)	Two of 4 EDGs out for maintenance. One of 3 offsite line out for maintenance. Inadvertent relay actuation caused LOOP to safety buses.	See Indian Point 2 and Yankee Rowe (11/9/65).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Pilgrim (11/19/86)	Storm failed all 345 kV. 23 kV remained energized.	See Indian Point 2 and Yankee Rowe (11/9/65).
Pilgrim (12/23/86)	One 345 kV out for maintenance. Flashover caused loss of other 345 kV. 23 kV still available.	See Indian Point 2 and Yankee Rowe (11/9/65).
Shoreham (3/18/87)	One of 3 EDGs out for maintenance, 1 safety bus out for maintenance. Current transformers shorted as a safety measure. This unbalanced relays serving both service transformers, but without actuating differential current relays. Three weeks later, condensate pump start caused differential relay trip, opening breakers from service transformer. Automatic fast transfer to reserve service transformer occurred, but unbalance caused it to trip. Two EDGs started and loaded.	See Peach Bottom 3 (4/13/86).
Pilgrim (3/31/87)	One 345 kV ring bus breaker out for maintenance. One 345 kV line lost due to storm. Other line isolated due to resultant breaker openings. 23 kV line still available.	See Indian Point 2 and Yankee Rowe (11/9/65).
Peach Bottom 2 & 3 (7/10/87)	Lightning caused loss of 1 of 2 offsite. This caused loss of 1 startup transformer. Other transformer remained in service.	See Peach Bottom 3 (4/13/86).
Vermont Yankee (8/17/87)	Both startup transformers and 1 of 2 345 kV main generator output breakers out for maintenance. Main generator disconnect links were removed. Unit auxiliary transformer energized by main transformer. Upset on grid caused other output breaker to open, causing LOOP. EDGs started, and backup source was still available.	See Indian Point 2 and Yankee Rowe (11/9/65).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Pilgrim (11/12/87)	23 kV line out of service. Snow failed both 345 kV lines. Startup transformer de-energized due to arcing. EDGs started, and power was restored by removing main generator disconnect links and backfeeding to auxiliary transformer.	See Indian Point 2 and Yankee Rowe (11/9/65).
FitzPatrick (10/31/88)	One 115 kV line out for maintenance. High winds interrupted other 115 kV line. EDGs energized safety buses; efforts were directed at other systems, so shutdown cooling was unavailable for 95 minutes [RCS temperature increased 260 K (10°F)].	See Indian Point 2 and Yankee Rowe (11/9/65).
Nine Mile Point 2 (12/26/88)	One 115 kV line out for maintenance. Current transformer failure caused loss of other line. Out of service line was returned to service and EDGs also started and loaded.	See Indian Point 2 and Yankee Rowe (11/9/65).
Pilgrim (2/21/89)	345 kV lost due to cable failure. 23 kV line available, SBO EDG available. Disconnect links removed for backfeed.	See Indian Point 2 and Yankee Rowe (11/9/65) and Sequoyah 1 (3/26/84).
Browns Ferry 2 (3/9/89)	Bus fault on secondary side of station transformer. EDGs started.	See Ginna (3/4/71) and San Onofre 1 (6/7/73).
Millstone 1 (4/29/89)	Main generator disconnect links removed. Loads had been transferred to station service transformer. Design error in relay of load shed system caused opening of 4.16 kV breakers when reserve station transformer was de-energized. Normal station transformer remained energized.	ABWR undervoltage load shed system will not inadvertently trip 6900 volt loads. ABWR undervoltage relays sense power on bus independent of source.
Browns Ferry 1 (5/5/89)	Ground faults opened breakers from 500 kV switchyard. Offsite power restored to safety buses from 161 kV switchyard through startup transformer.	See Peach Bottom 3 (4/13/86).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
WNP-2 (5/14/89)	One safety bus out for maintenance. Two EDGs out for maintenance. Operator error caused LOOP to other safety buses. EDG started and loaded 1 safety bus.	In the ABWR, the two operable emergency buses could have been energized from either the combustion turbine generator or the alternate preferred offsite reserve transformer.
River Bend (6/13/89)	One of 4 preferred transformers out. Maintenance error tripped 1 preferred transformer, causing loss of power to 1 safety bus. EDG started and loaded. Maintenance error tripped main generator output breakers, causing LOOP to non-safety buses.	See WNP-2 (5/14/89).
Oyster Creek (3/9/91)	One EDG and 1 bus out for maintenance. Routine check revealed other EDG had faulty head gasket which would have caused failure if required. This left plant with only 1 source of power, the startup transformer.	ABWR has two offsite power sources, three diesel generators, and one combustion turbine generator.
Vermont Yankee (4/23/91)	LOOP due to improper maintenance in switchyard. While installing a new battery on non-1E 125 VDC bus, two vital DC buses were cross connected through a battery charger after defeating a mechanical interlock. When the battery charger breaker was opened to install the new battery, a voltage transient was sent through the entire DC control power system which caused both offsite power breakers to trip and lock open.	ABWR procedures do not allow independent vital buses to be cross connected. The multiple sources of onsite and offsite power reduces the need to attempt cross connecting buses. The ABWR has four physically separate and independent 125 VDC systems.
Diablo Canyon Unit 1 (3/7/91)	LOOP caused by boom of mobile crane shorting out 500 kV transformer. Standby startup transformer was out of service for maintenance. The three EDGs started and picked-up vital buses. Offsite power was restored in five hours.	ABWR has two independent preferred sources of offsite power.

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Nine Mile Point (3/23/92)	LOOP while working on aux. boiler circuitry. Div. I diesel was out for maintenance. Div. II diesel started and loaded. Div. III (HPCS) started but tripped on over temperature due to lack of cooling water. All control room annunciators lost due to loss of A and B UPS.	ABWR offsite power supplies are physically and electrically separated so loss of both is not expected to occur due to common cause failure. Three independent electric divisions (including instrument UPSs) would reduce likelihood of simultaneous failure of all three divisions.

Table 19QC-2 Decay Heat Removal Precursors

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Peach Bottom 3 79-002 January 8, 1979	Mode 4, Cold Shutdown. RHRS in operation on loop 'A'.	A slight reactor water level drop was detected and determined to be caused by leakage through the minimum flow recirculation valve for the 'A' RHR pump (MO-16A). Vessel level was maintained by use of the stay full pressurizing system. Attempts to eliminate the leakage by further closing the minimum flow valve resulted in its failure to the wide open position. This failure caused a loss of coolant to the suppression pool. The loss of vessel water level continued to the point of isolation of the shutdown cooling system on low water level, at which time the water level stabilized. The time required to raise the reactor water level, via the stay full system, clear the RHRS isolation and reestablish shutdown cooling with the 'C' RHRS pump, allowed the coolant to rise to about 366 K (200°F), causing a gaseous release via disassembled RCIC steam isolation valves.	Failure of the minimum flow recirculation valve associated with the 'A' RHRS pump.	ABWR component design and procurement will emphasize fabrication quality and proper maintenance to minimize individual component failures. However, if failure occurs, SDC would be temporarily lost but two other RHR trains would be available to re-establish DHR before any fuel damage occurred. In addition other heat removal systems (e.g., fuel pool cleanup and cooling (FPC), reactor water cleanup) are available for DHR depending on plant conditions. Other makeup sources (e.g., HPCF, feedwater, condensate, AC Independent water addition, CRDS) can be used if no DHR system is available and the reactor coolant begins to boil.
Hatch 1 August 13, 1979	Mode 5, Refueling. RHRS in operation.	The 'B' loop RHRS was placed in service in the shutdown cooling mode and vessel level was observed to be dropping. Valve E11-F004B was determined to be leaking to the suppression pool. A local leak rate test of the RHRS 'B' pump torus suction isolation valve showed the valve to be leaking in excess of specified criteria. Following corrective action, the valve was satisfactorily retested.	None reported.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Oyster Creek 81-038 August 27, 1981 August 28, 1981	Mode 5, Refueling. RHRS system in operation on loop ‘C’. Reactor had been shutdown for 13 days.	This event consists actually of two separate events involving shutdown cooling heat exchanger tube leaks. On August 27, with reactor water temperature at 365 K (197°F), the ‘C’ shutdown cooling heat exchanger developed a tube leak resulting in reactor water leaking into the RBCCW system as indicated by the RBCCW process radiation monitor. About 2 minutes later, reactor water level began to decrease. The decrease occurred over approximately 10 minutes, with an estimated leak rate of 0.025 m^3/s (400 gpm). Reactor vessel water level was recovered by makeup supplied by the feedwater and condensate system. The ‘C’ loop was secured and temperature maintained below 373 K (212°F) by use of the ‘A’ shutdown cooling loop. (Continued on following page)	Circumferential through wall cracks in one tube of the ‘A’ heat exchanger and one tube of the ‘C’ heat exchanger, due to fatigue failure caused by flow induced vibration.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Oyster Creek 81-038 August 27, 1981 August 28, 1981 (continued)		On August 28, another RBCCW process monitor alarm was received and the RBCCW surge tank was reported to be overflowing. The ‘A’ shutdown cooling loop was isolated. The ‘B’ heat exchanger was out of service but was made serviceable in a few hours. Temperature was maintained by increasing flow to the CUW nonregenerative heat exchanger and increasing letdown to the main condenser. Water was pumped back to the reactor using a condensate pump. In addition to CUW and main condenser systems, the isolation condenser and ECCS systems were all available.		
LaSalle 1 82-039 June 9, 1982	Mode 3, Hot Shutdown. Plant cooldown in progress RHR loop ‘A’ being placed in service. (Prior to initial criticality.)	While placing RHR ‘A’ loop in service in the shutdown cooling mode, leakage was discovered at the ‘A’ RHR pump suction line. RHR loop ‘A’ was taken out of service for repairs. Alternate methods of decay heat removal were reactor recirc pumps and inboard main steam line drain with CUW.	Leaking flange on spool piece on ‘A’ RHR pump suction line, caused by thermal growth on heatup and cooldown.	ABWR has three independent RHR loops. Also, the main condenser and CUW are capable of removing decay heat in Mode 3.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 1 82-042 June 11, 1982	Mode 4, Cold Shutdown (Prior to initial criticality).	The unit was in cold shutdown following performance of reactor internals vibration testing. 'B' RHR system was operating in the shutdown cooling mode with all flow bypassing the 'B' RHR heat exchanger to maintain reactor temperature between 333 K (140°F) and 366 K (200°F). The 'A' RHR system was lined up for standby shutdown cooling. The 'A' and 'B' RHR suppression pool suction valves were out of service electrically for repair and the valves were manually closed. No backup means of decay heat removal was available due to the reactor building closed cooling water system being out of service. (No actual decay heat existed.) (Continued on following page)	Personnel did not recognize the potential vessel drain path that existed upon returning the system to a normal lineup from standby operation. The test procedure failed to recognize the current operating status of the RHR system in shutdown cooling. The level instruments tap off the downcomer region where shutdown cooling receives its suction. The Tech Specs were interpreted such that both shutdown cooling loops were required operable with one in operation, and that the idle pump could be out of service for only 2 hours. This was a conservative interpretation but it aggravated the event by imposing an arbitrary time restraint on the test.	ABWR procedures will clearly describe proper operational steps and the technical specifications will be based on minimizing plant risks during normal full power operation and shutdown conditions. The ABWR has three independent RHR systems and SDC is isolated on low RPV level.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 1 82-042 June 11, 1982 (Continued)		Testing of the ‘A’ RHR drywell spray outboard isolation valve was approved and performed in accordance with procedure. After the test was completed, the system was returned to standby operation. The restoration procedure directed the opening of the RHR ‘A’ heat exchanger bypass valve. When this valve was opened, water from the reactor vessel filled the previously drained RHR ‘A’ piping, draining about 11.36 m^3 (3,000 gallons) of water from the vessel. At 31.75 cm (12.5 inches) level, an automatic isolation of the shutdown cooling system occurred. The vessel level was restored, and the ‘B’ RHR loop was verified filled and vented, and shutdown cooling system suction isolation valves reopened. Reactor vessel level again decreased to about 25.4 cm (10 inches) and a second isolation occurred. It was determined that this second isolation resulted from the starting transient and resulting level drop in the downcomer region. Vessel level was again restored; and shutdown cooling unisolated, vented, and restarted; and the ‘A’ RHRs loop determined operable.		

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Grand Gulf N/A April 3, 1983	Mode 4, Cold Shutdown, after initial criticality. RHRS Loop 'B' in Shutdown Cooling.	Loop 'A' of the RHRS was lined up in the LPCI mode, and loop 'B' was lined up in the shutdown cooling mode for a surveillance test. After completion of the test, the operator returned 'B' loop to the LPCI mode, which required shutting the loop 'B' SDC suction valve (F006) and opening the loop 'B' suppression pool suction valve (F004). Since a light bulb was burned out on the open indicator for F006, the operator assumed that F006 was already shut, and opened F004. This opened a flow path from the reactor vessel via the 'B' RHR loop to the suppression pool. Approximately 37.85 m^3 (10,000 gallons) of water drained from the reactor vessel prior to automatic isolation of the RHRS on low water level. The operator attempted to reshut F004 upon receiving a low level alarm, but the valve's MOV breaker tripped.	Operator error; misinterpretation of valve position indication. F006 "fully open" indicator light was not burning, but neither was the "fully shut" indicator. Valve was probably in a partially open position. Reason for F004 MOV breaker trip not explained.	The potential for this operator error has been eliminated in the ABWR design by providing valve interlocks. When RHR system is in the shutdown cooling mode (i.e., taking suction from the RPV), the discharge valves to the suppression pool are interlocked in the closed position to prevent inadvertent draining of the RPV. To realign to the Low Pressure Flooder (LPFL) mode, the suppression pool suction valve cannot be opened until the SDC suction valve is fully closed.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Susquehanna 183-056 April 7, 1983	Mode 3, Hot Shutdown.	During a startup test to determine the capability of the shutdown cooling mode of RHR, the 'A' RHR heat exchanger was valved in causing a rapid temperature decrease. As a result of RPV water volume shrinkage, the RHR automatically isolated on low reactor water level. CRD flow was used to restore level; and MSIVs were opened to decrease the vessel delta-T. CUW was established to stop stratification. RHR loop 'A' was restored, but a valve lineup error caused the pump miniflow valve to bypass RHR flow to the suppression pool, causing a second RHR isolation on low level. Level was restored and RHR reinitiated, but the inventory addition via condensate transfer caused another temperature decrease of 322 K (120^{o}F) in 5 minutes, so the RHR system was isolated a third time to halt the cooldown. The system was restored again, and a fourth short isolation was received when starting the 'B' RHR pump.	Reactor Coolant System shrinkage caused by rapid temperature decrease. Valve lineup error caused loss of inventory to suppression pool.	See LaSalle 1 (6/11/82). RHR valve misalignments are minimized in the ABWR design by mode switches for the five operational RHR modes. Selection of a mode (e.g., SDC causes automatic valve realignments).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 2 N/A August 15, 1983	Cold shutdown. Preoperational testing prior to fuel load.	With the control rod drive system in service and the reactor water cleanup system out of service, reactor water level was being controlled by draining through the RHRS ‘B’ loop to the suppression pool. A new drain path was being established via the ‘A’ RHR loop (F004 and F006). As soon as this new drain path was lined up, the reactor vessel began draining rapidly. The event did not terminate automatically on low RPV water level isolation of RHRS, because the low level isolation signal had been bypassed by transferring control for the RHR shutdown cooling isolation valves to the remote shutdown panel. This was done intentionally to prevent inadvertent isolations of the temporary drain path. The loss of coolant event was terminated by operator action, 81.28 cm (32 inches) above the top of the fuel region (fuel had not yet been loaded).	Using an unusual valve lineup and bypassing automatic safety features.	See LaSalle 1 (6/11/82). The ABWR design has adequate safety features. However, unusual valve lineups and bypassing of safety features should be performed under strict administrative control.
LaSalle 1 83-108 September 1, 1983	Cold shutdown. RHRS operable.	RHRS operable, but shutdown cooling status not stated. RHRS pump ‘A’ minimum flow bypass valve (F064A) stuck open following a test. If shutdown cooling was lined up to loop ‘A’ then a drain path to the suppression pool existed.	Trip fingers which hold the motor operation in handwheel operation were found broken. Valve motor damaged.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 1 83-105 September 14, 1983	Cold shutdown.	RHR logic testing was in progress which required opening most loop B injection and spray valves: drywell spray valves (F016B and F017B), suppression pool spray valve and test return valves (F027B and F024B), and B and C loop injection valves (F042B and F042C). This lineup relied on testable injection check valve F041B to prevent reactor vessel inventory loss via injection valve F042B to the open spray and test return lines. When F042B was opened, reactor vessel inventory was rapidly lost to the drywell and suppression pool because the testable check valve was stuck open. Most of the water lost from the reactor vessel went to the suppression pool. The operator terminated the event after a 1.27-m (50-inch) level drop to about 4.06 m (160 inches) above the top of the active fuel. Total inventory loss was between 18.93 and 37.85 m^3 (5,000 and 10,000 gallons). It should be noted that no automatic isolation feature would have terminated this flow path; however, the LPCI injection line penetration is above the top of the active fuel.	The LPCI injection check valve was stuck open. Inspection of the valve revealed improper maintenance on the valve operator. The valve had been reassembled by lining up the wrong mark on the spline shaft to the air operator gears, which held the check valve 35° open. The packing gland was also too tight to permit full closure.	ABWR component design and procurement will emphasize fabrication quality and proper maintenance to minimize individual component failures. RHR logic testing does not require that RPV isolation rely on a single check valve during RHR logic testing.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Quad Cities 1 1/24/91	Cold shutdown.	The RPV level decreased 35.56 cm (14 inches) in two related events. The shutdown cooling suction valve was stroked as a maintenance check but some vent and drain valves in the loop were also open, when the SDC suction valve was open the RPV drained 12.7 cm (5 inches). The operator isolated SDC to stop the flow but when the loop was returned to service an additional 22.86 cm (9 inches) were drained from the RPV into the partially empty RHR loop.	Operator error in misaligning RHR valves.	ABWR procedures will highlight RHR system valve alignments during maintenance. The keep fill pump and pressure alarm assures a full loop.
Quad cities 2 8/17/87	Cold Shutdown. On shutdown cooling in one RHR loop, reactor water clean up (CUW) system out for maintenance.	After isolating RCW the RPV level began to increase. Operators attempted to reduce level by draining to the suppression pool using the RHR system test return valve [350A (14-inch) valve]. This resulted in rapid decrease in RPV to low level setpoint and an automatic RPV isolation.	Operator error in not following approved procedure for draining the RPV.	ABWR RHR valves are interlocked to prevent SDC suction and injection valves from being open at the same time as the suppression pool return valves.
Fermi 2 3/17/87	Hot Shutdown following loop test, one RHR loop inoperable.	SDC loop was being put in service but normal loop heatup alignment could not be used because one valve would not open [600A (24-inch) testable check valve]. A smaller [25A (1-inch)] valve was used to fill the loop but the normal 100A (4-inch) drain line caused drainage faster than the 25A (1-inch) line could fill the loop. This drained the loop but the operator could not tell. When proper SDC loop temperature was reached the operator opened the SDC suction valve to the RPV and RPV level decreased to the low level setpoint and RPV isolation occurred.	Operator error in placing SDC loop in service using unapproved procedure.	ABWR RHR system keep fill alarm would alert operator to a partially drained loop condition.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Fermi 2 8/2/87	Cold Shutdown. SDC on Division II.	During the process of shifting SDC from Division II to Division I, a RPV low level signal occurred because valves were misaligned resulting in an open flow path to the suppression pool from the RPV.	Operator error in not following proper procedure placing SDC in service.	ABWR RHR suppression pool suction and SDC suction valves are interlocked to prevent inadvertent RPV drainage.
WNP-2 5/7/85	Cold Shutdown in SDC.	While returning from SDC to standby low pressure injection mode of RHR, the operator opened the suppression pool suction valve before the SDC suction valve was fully closed. This opened a drain path from the RPV to the suppression pool resulting in a low RPV and SDC isolation.	Operator error in not knowing that stroke time for each valve is 90–100 seconds.	Suppression pool suction valve cannot be opened until SDC suction valve is fully closed.
Shoreham 7/26/85	Cold Shutdown both RHR loops in SDC mode.	While returning one RHR loop to standby, operator opened suppression pool suction valve while SDC suction valve was partially open (see WNP-2 5/7/85).	See WNP-2 5/7/85.	See WNP-2 5/7/85.
Peach Bottom 2 9/24/85	Cold Shutdown. SDC on ‘A’ RHR loop.	Loop ‘C’ SDC suction valve remained open after previous SDC operation. Loop ‘A’ required a full flow test due to pump problem investigation. SDC ‘A’ isolated and ‘A’ pump aligned to suppression pool for test. This opened path from RPV to suppression pool through ‘C’ SDC suction valve.	Operator error in not knowing status of RHR system valves.	ABWR RHR loops are independent and cross train flow cannot occur.
Riverbend 9/23/85	Cold Shutdown.	While restoring SDC loop to standby, suppression pool suction and SDC suction valves were open at the same time.	See WNP-2 5/7/85.	See WNP-2 5/7/85.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Susquehanna 2 4/27/85	Cold Shutdown.	While placing ‘A’ SDC on line a path was open from the RPV to the main condenser. RPV level dropped 88.9 cm (35 inches) resulting in RPV low level signal and isolation of SDC.	Operator error improper valve lineup.	ABWR procedures will clearly describe proper valve lineups.
Susquehanna 1 5/10/85 5/20/85	Cold Shutdown.	SDC pump miniflow valve failed open allowing water to flow from RPV to suppression pool.	Valve failure.	SDC would isolate on low RPV level.
WNP-2 8/23/84	Cold Shutdown.	While warming up SDC loop, an isolation signal occurred on high SDC flow. Operator did not notice and loop drained to the radwaste system. When operator placed loop in service water drained from RPV into empty SDC loop.	Operator error. SDC loop isolation not alarmed in control room.	The keep fill alarm would alert the operator to a partially drained RHR loop.
LaSalle 1 9/14/83	Cold Shutdown.	RHR loop in test mode with several valves open. Loop check valve depended upon to isolate RPV. Check valve failed open due to misassembly and improper packing gland installation.	Maintenance error.	ABWR RHR system tests would not require all valves be open and rely on check valve to isolate the RPV.
Brunswick 2 9/24/84	Cold Shutdown.	Operator attempted to lower suppression pool level to radwaste but loop was in SDC mode and resulted in water diversion from RPV to radwaste.	Operator error.	RHR system drain to the radwaste system contains two valves in series that automatically close on low RPV level.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Pilgrim 81-064 December 21, 1981	Mode 5, Refueling RHRS in operation. Coolant temperature at 294 K (70°F).	While performing maintenance on a feeder transformer, a live transfer of power was attempted. Mal-operation of a power breaker de-energized a vital instrument panel, causing two shutdown cooling valves (MO-47 and MO-48) to close on receipt of a reactor high pressure isolation signal. The ‘C’ RHRS pump should have tripped immediately when its suction valves shut, but failed to do so. After about 5 hours, when the process computer was returned to service, abnormal heat exchanger temperatures alerted operators to a problem. At this time, the ‘C’ RHRS pump was observed to be running with both suction valves shut. The ‘C’ pump was tripped, the valves opened, and the ‘A’ pump started to restore shutdown cooling.	Electrical contacts in the pump trip logic were corroded to the extent that they seized in the open position. ‘C’ RHRS pump, therefore did not trip when the suction valves left their full open position. Inadequacies in the implementation of administrative controls for shift turnover, valve lineup checks, and board checks aggravated the situation. Extensive maintenance activities distracted operators.	See Peach Bottom 3 (1/8/79) and LaSalle 1 (6/11/82).
Susquehanna 1 83-030 February 16, 1983	Mode 4, Cold Shutdown. RHRS in operation on loop ‘A’.	The RHRS was operating in the shutdown cooling mode. A Division I isolation signal to the inboard isolation valve to the RHRS caused a loss of shutdown cooling. The system was reestablished by resetting the signals. A second occurrence was experienced within an hour.	The Reactor Protection System (RPS) was operating on alternate power supplies while the RPS MG set was undergoing maintenance. Spurious trips of the RPS alternate power supply breakers caused isolation signals.	Loss of power does not cause isolation of SDC in the ABWR design. The safety system logic will only cause isolation if a valid isolation condition existed.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Susquehanna 1 83-060 April 11, 1983	Mode 4, Cold Shutdown. RHR in operation on loop ‘B’.	An RPS actuation caused RHR loop ‘B’ operating in the shutdown cooling mode to isolate. RHR pump ‘D’ tripped twice on attempts to restart. RHR cooling was established again on loop ‘B’ using pump ‘B’.	RPS actuation caused by an inadvertent breaker trip (bumped by a construction worker). The restart trips are believed to be due to a faulty shutdown cooling flow switch.	See Susquehanna 1 (2/16/83).
Grand Gulf 83-069 May 23, 1983	Mode 4, Cold Shutdown. (During initial plant startup phase).	Following electrical maintenance during which some shutdown cooling motor-operated valves were blocked open, power was restored, and the valves were unblocked. The valves isolated as a result of a previously existing isolation signal from the valve isolation logic, causing a loss of both shutdown cooling loops.	The power supply fuses to the isolation logic had not been replaced following completion of a design change.	ABWR solid state logic minimizes use of fuses and logic testing is easier such that these types of operator errors will be reduced.
Grand Gulf 83-119 August 18, 1983	Mode 4, Cold Shutdown. RHRS loop ‘A’ in operation. (During initial plant startup phase.)	Both RHR shutdown cooling loops isolated on two occasions during attempts to start a control room air-conditioning compressor. The systems interaction was due to a common power source to the compressor and to leakage detection logic circuitry, which caused the isolation.	The solid state trip unit for the common 480V trip breaker had failed.	ABWR has three independent (both physically and electrically) RHR systems. No common power supplies between RHR systems exist.
Grand Gulf 83-137 September 1, 1983	Mode 4, Cold Shutdown. RHRS loop ‘A’ in operation. (During initial plant startup phase.)	The RHRS isolated after shifting the RPS power supply to an alternate source. The alternate supply breaker tripped, causing an isolation of shutdown cooling.	The distribution transformer on the unregulated RPS alternate power source was subject to transients.	See Susquehanna 1 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Grand Gulf 83-193 December 27, 1983	Mode 4, Cold Shutdown.	During an instrument surveillance on the isolation logic for shutdown cooling, the outboard suction valve (F008) closed, isolating both loops of the SDC system. The system was returned to service in 49 minutes.	The cause of the isolation was a tip breaking off a minitest clip used for jumpering.	See Grand Gulf (8/18/83). ABWR solid state logic eliminates need for test jumpers. Surveillance is automated to reduce chance of operator error.
Susquehanna 1 83-172 December 30, 1983	Mode 5, 0% Power.	During the Unit 1 - Unit 2 tie-in outage, one of the RPS ‘B’ breakers tripped, closing SDC inboard and outboard isolation valves. Reactor coolant recirculation was established through the fuel pool cooling system.	The cause of the trip was a failed breaker.	See Susquehanna 1 (2/16/83).
Hatch 2 September 19, 1986	Mode 4, Cold Shutdown.	Received a low RPV water level signal while valving out a RPV level indicator. This resulted in a scram signal and isolation of SDC. SDC was restored in 10 minutes.	Personnel error in not placing level transmitter in bypass before valving out detector.	ABWR procedures will clearly specify required maintenance steps and precautions to preclude inadvertent SDC isolation.
Hatch 2 September 21, 1986	Mode 4, Cold Shutdown.	Lost SDC for 1.5 hours due to inadvertent RHR suction valve isolation during a surveillance test.	Surveillance procedure required removal of instrument links instead of jumpering them out. When links were opened, a RHR valve isolation signal was initiated.	ABWR solid state logic does not require the use of jumpers to complete circuit logic checks.
Perry 1 October 24, 1986	Mode 4, Cold Shutdown.	While transferring RPS power to an alternate bus to complete RPS MG set maintenance, a voltage transient occurred which resulted in isolation of SDC.	Inadequate procedure for transferring power between buses.	See Susquehanna 1 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
River Bend 1 October 28, 1986	Mode 4, Cold Shutdown.	SDC valve was inadvertently closed when technician accidently grounded a portion of the valve control circuitry during a surveillance test. The ground caused a blown control circuit fuse which resulted in a valve closure signal.	Personnel error.	See Susquehanna 1 (2/16/83).
Perry 1	Cold Shutdown.	SDC isolated due to loss of power to RPS bus. RPS was being powered by alternate power since MG set was in maintenance.	Voltage fluctuation due to starting one of the plant's circulation water pumps, caused electrical protection devices (EPAs) to trip resulting in loss of power to the RPS.	See Susquehanna 1 (2/16/83).
Clinton 1 January 22, 1987	Mode 4, Cold Shutdown.	While performing a reactor coolant system hydrostatic leak test. An isolation of SDC occurred due to high system pressure.	The breaker controller for the high pressure interlock RHR valve was racked out prior to the test to prevent valve closure. Following the test, the trip function was not reset prior to racking in the breaker. When the breaker was racked in the valve closed due to the locked-in high pressure signal.	See Hatch 2 (9/19/86).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Peach Bottom 2 March 28, 1987	Mode 5, Refueling.	Isolation of SDC occurred during maintenance on emergency bus relays.	Maintenance procedure called for pulling fuses prior to replacement of certain relay coils. When one of the required fuses was pulled, the high pressure RHR interlock coil was de-energized. This resulted in isolation of SDC.	See Susquehanna 1 (2/16/83).
WNP-2 April 21, 1987	Mode 5, Refueling.	SDC isolated when an isolation control relay for a non SDC function was de-energized for maintenance.	The neutral wire for several relays, including the SDC relay, were all connected together. Lifting the neutral to one relay caused a loss of power to all relays with a common neutral.	ABWR solid state is less susceptible to this type of failure. Maintenance bypass does not require the lifting of leads.
Hatch 1 April 22, 1987	Mode 3, Hot Shutdown.	While placing a SDC loop in service, RPV level dropped from 157.5 to 7.6 cm (62 to 3 inches).	SDC loop was only partially full prior to placing in service.	See Hatch 2 (9/19/86).
Hatch 1 June 7, 1987	Mode 5, Refueling.	SDC isolated when power was lost to the RPS bus.	RPS MG set output breaker inadvertently tripped.	See Susquehanna 1 (2/16/83).
Perry 1 July 4, 1987	Mode 4, Cold Shutdown.	SDC isolated when power was removed from the RPS bus for a surveillance test.	Procedure did not recognize the impact on SDC of removing power from the RPS bus.	See Susquehanna 1 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Peach Bottom 2,3 August 16, 1987	Mode 4, Cold Shutdown.	SDC isolation occurred when the normal offsite power supply was lost and a transfer to an alternate source temporarily de-energized electrical buses.	The cause of the loss of offsite power was not included in the report.	See Susquehanna 1 (2/16/83).
Peach Bottom 2 August 28, 1987	Mode 4, Cold Shutdown.	SDC isolated during maintenance on electric circuits.	SDC isolation coil inadvertently de-energized during maintenance.	See Susquehanna 1 (2/16/83) and WNP-2 (4/21/87).
Susquehanna 1 September 13, 1987	Mode 4, Cold Shutdown.	While transferring SDC from the ‘A’ to the ‘C’ RHR pump, SDC isolated.	A spurious high RHR flow signal caused the SDC isolation.	ABWR solid state logic requires two-out-of-four signal to actuate a safety function.
Peach Bottom 2 September 16, 1987	Mode 4, Cold Shutdown.	SDC isolated for 15 minutes.	Loss of power to a MCC.	See Susquehanna 1 (2/16/83).
Perry 1 September 29, 1987	Mode 4, Cold Shutdown.	SDC isolated during a pressure transmitter response time test.	Personnel error in allowing pressure signal from test instrument to exceed SDC high pressure isolation set point.	See Hatch 2 (9/19/86).
Pilgrim October 6, 1987	Mode 5, Refueling.	SDC isolated on loss of power to 480V bus which supplies power to the isolation valve.	Cause for loss of power not reported.	See Susquehanna 1 (2/16/83).
Pilgrim October 15, 1987	Mode 4, Cold Shutdown.	SDC isolated during maintenance on primary containment isolation system.	An incorrect lead was lifted which generated a false high reactor pressure signal.	See WNP-2 (4/21/87).
Susquehanna November 1, 1987	Mode 5, Refueling.	SDC isolated when RPS power supply was transferred between alternate sources.	Momentary loss of RPS power.	See Susquehanna 1 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Grand Gulf November 30, 1987	Mode 5, Refueling.	SDC isolated during maintenance on power buses.	A temporary loss of power occurred when bus was re-energized following maintenance.	See Susquehanna 1 (2/16/83).
Peach Bottom 2 December 6, 1987	Mode 4, Cold Shutdown.	SDC isolated due to initiation of reactor scram signal.	Technician caused a scram signal to be generated during an ATWS logic pressure switch calibration.	See Hatch 2 (9/19/86).
Nine Mile Point 2 February 1, 1988	Mode 4, Cold Shutdown.	SDC isolated during maintenance on RPV level sensor.	Technician caused a pressure surge in the instrument line which resulted in a high RHR system pressure signal to be generated.	See Hatch 2 (9/19/86).
Pilgrim February 2, 1988	Mode 4, Cold Shutdown.	SDC isolation signal generated during maintenance on emergency parameter information computer.	Personnel error during maintenance.	See Hatch 2 (9/19/86).
WNP-2 May 30, 1988	Mode 4, Cold Shutdown.	SDC isolated during refueling outage.	Maintenance personnel pulled wrong set of fuses.	See Grand Gulf (5/23/83) and Susquehanna 1 (2/16/83)
Peach Bottom 2 July 29, 1988	Mode 4, Cold Shutdown.	SDC isolated during maintenance on PCIS logic circuitry.	Inadequate procedure. SDC isolation logic should have been blocked as part of maintenance task.	See Hatch 2 (9/19/83).
Nine Mile Point 2 October 25, 1988	Mode 4, Cold Shutdown.	SDC isolated during modification work on a RPS cabinet.	Technician inadvertently grounded the RPS 24 VDC power supply.	See Susquehanna 1 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
FitzPatrick October 31, 1988	Mode 5, Refueling.	SDC isolated following a loss of two offsite power lines and a 120 VAC UPS.	Loss of RPS power caused SDC isolation.	See Susquehanna 1 (2/16/83).
Peach Bottom 2 December 6, 1987	Mode 4, Cold Shutdown.	SDC isolated due to initiation of reactor scram signal.	Technician caused a scram signal to be generated during an ATWS logic pressure switch calibration.	See Hatch 2 (9/19/86).
Nine Mile Point 2 February 1 1988	Mode 4, Cold Shutdown.	SDC isolated during maintenance on RPV level sensor.	Technician caused a pressure surge in the instrument line which resulted in a high RHR system pressure signal to be generated.	See Hatch 2 (9/19/86).
Pilgrim February 2, 1988	Mode 4, Cold Shutdown.	SDC isolation signal generated during maintenance on emergency parameter information computer.	Personnel error during maintenance.	See Hatch 2 (9/19/86).
WNP-2 May 30, 1988	Mode 4, Cold Shutdown.	SDC isolated during refueling outage.	Maintenance personnel pulled wrong set of fuses.	See Grand Gulf (5/23/83) and Susquehanna 1 (2/16/83)
Peach Bottom 2 July 29, 1988	Mode 4, Cold Shutdown.	SDC isolated during maintenance on PCIS logic circuitry.	Inadequate procedure. SDC isolation logic should have been blocked as part of maintenance task.	See Hatch 2 (9/19/83).
Nine Mile Point 2 October 25, 1988	Mode 4, Cold Shutdown.	SDC isolated during modification work on a RPS cabinet.	Technician inadvertently grounded the RPS 24 VDC power supply.	See Susquehanna 1 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
FitzPatrick October 31, 1988	Mode 5, Refueling.	SDC isolated following a loss of two offsite power lines and a 120 VAC UPS.	Loss of RPS power caused SDC isolation.	See Susquehanna 1 (2/16/83).
FitzPatrick November 9, 1988	Mode 5, Refueling.	SDC pump stopped when SDC isolation valve left its open position.	Momentary loss of power to RPS caused SDC valve to start closing. Interlock of SDC isolation valve and pump caused control breaker to open.	See Susquehanna 1 (2/16/83).
Fermi 2 January 10, 1989	Mode 4, Cold Shutdown.	SDC isolated when Div. 1 ESF power was lost.	Loss of power cause not reported.	See Susquehanna 1 (2/16/83).
Clinton January 10, 1989	Mode 5, Refueling.	SDC isolated during testing of RCIC logic.	While attempting to jumper out the SDC isolation signal, a technician inadvertently grounded the RPV low level circuit. This caused a fuse to blow and SDC to isolate.	See Grand Gulf (5/23/83) and Susquehanna 1 (2/16/83).
Nine Mile Point January 22, 1989	Mode 4, Cold Shutdown.	SDC isolated during a surveillance test of the reactor building high temperature isolation signal.	Test procedure specified the wrong isolation signal be actuated.	See Hatch 2 (9/19/86).
Hope Creek March 1, 1989	Mode 4, Cold Shutdown.	During performance of a surveillance test, the SDC injection valve closed resulting in a loss of SDC.	Procedural error. Leads were lifted to allow completion of RHR logic test without valve actuations. The lead for the RHR injection valve was inadvertently left off the list of leads to be lifted.	See Hatch 2 (9/19/86) and Hatch 2 (9/21/86).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
River Bend March 25, 1989	Mode 5, Refueling.	SDC cooling isolated when 120 VAC divisional logic was de-energized.	Maintenance personnel de-energized logic power to complete work on the reactor plant sampling system.	See Susquehanna 1 (2/16/83).
River Bend March 29, 1989	Mode 5, Refueling.	SDC isolated due to loss of RPS power.	A jumper fell off during installation causing a ground of RPS power and a blown fuse in the RPS power supply.	See Hatch 2 (9/21/86) and Susquehanna 1 (2/1/6/83).
Grand Gulf April 26, 1989	Mode 4, Cold Shutdown.	RHR pump tripped during surveillance test of RCIC trip throttle valve.	Technician lifted DC power lead for RCIC throttle valve but did not realize that the RHR pump “no suction path” trip logic was also on the circuit. When the lead was lifted, the RHR pump tripped.	See Hatch 2 (9/21/86).
River Bend April 27, 1989	Mode 5, Refueling.	SDC isolated during a surveillance test of manual scram function.	Lead became disconnected during test and grounded out the RHR high pressure interlock circuit. This caused the isolation valve to close.	See Hatch 2 (9/21/86).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 1 77-045 July 28, 1977	Mode 3, Hot Shutdown. Plant cooldown in progress. Temperature at 462 K (372°F).	A reactor cooldown was in progress following a scram. With reactor water temperature at 462 K (372°F), preparations were commenced for placing RHRS loop ‘A’ in shutdown cooling. RHRS booster pumps were started in conjunction with the 1B nuclear SW pump. A gasket ruptured on the RHR service water system as it was being placed in shutdown cooling. Water was observed spraying from the overhead of the 6.1-m (20-ft.) elevation in the reactor building. The 1B loop of RHRS was placed in service at 436 K (325°F). When attempting to place the RHRS ‘1B’ loop in shutdown cooling, it was found that the inboard shutdown cooling suction valve would not open, due to a false signal from a pressure switch.	Ruptured flange gasket on RHRSW loop 1A heat exchanger outlet valve, causing spray-induced electrical damage.	See Peach Bottom 3 (1/8/79). ABWR uses analog transmitters instead of pressure switches for actuation circuits, so this type of failure would not occur in the ABWR.
Brunswick 2 78-036 April 3, 1978	Mode 3, Hot Shutdown. Plant cooldown in progress.	After a reactor shutdown, while establishing shutdown cooling, the shutdown cooling outboard suction valve (F008) would not open remotely. Valve was opened manually and reactor placed in cold shutdown.	Electromechanical brake on valve operator failed, causing valve to bind and the motor operator to draw excessive current when energized.	See Peach Bottom 3 (1/8/79). The current level of the ABWR design does not generally address detail component features. But it is expected that as is the case for operating plants, MOVs will include handwheels to mitigate events such as this.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 2 78-052 June 3, 1978	Mode 3, Hot Shutdown. Plant cooldown in progress.	During normal shutdown and cooldown, RHRS shutdown cooling valve located inside the containment (F009) would not open from the control room. This valve must be opened before the reactor can be placed in cold shutdown. Entry into the drywell via the personnel air lock was unsuccessful. Entry into the drywell was made through the CRD hatch and the RHRS valve was manually opened.	Cause for valve failure not reported. Personnel air lock inner door would not open due to sticky gaskets, caused by large amount of compressive force applied to gaskets by strongback installed 2 days earlier for test. Strongback removed on day of event.	See Peach Bottom 3 (1/8/79).
Brunswick 2 78-074 November 12, 1978	Mode 3, Hot Shutdown.	Reactor steam dome high pressure switch would not reset and would not allow RHRS valve (F008) to open for shutdown cooling at a reactor pressure of 0.80 MPa.	Sticking microswitch caused instrument failure.	See Brunswick 1 (7/28/77).
Brunswick 2 81-019 February 14, 1981	Mode 3, Hot Shutdown. Plant cooldown in progress.	Following a reactor shutdown, while attempting to place RHRS shutdown cooling into service, the RHR supply inboard isolation valve (F009) would not open electrically. Burned motor windings prevented the valve motor from opening the valve. Valve was manually opened and RHRS shutdown cooling placed in service. Cold shutdown reached 8 hours after opening valve.	Thorough investigation revealed no cause for failed motor windings.	See Peach Bottom 3 (1/8/79).
Brunswick 2 81-070 July 18, 1981	Mode 3, Hot Shutdown. Plant cooldown in progress.	While attempting to place RHRS shutdown cooling into service, RHRS shutdown cooling supply inboard isolation valve (F009) would not open on a remote signal. Valve was manually opened, RHRS shutdown cooling placed in service and cold shutdown achieved in 8 hours.	Loose fastener on one of the overcurrent devices in the valve motor breaker, resulting in an overcurrent condition on two of the motor phases, tripping the breaker.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 1 82-034 June 5, 1982	Mode 3, Hot Shutdown at 380 K (225°F). (During initial plant startup phase.)	When lining up for shutdown cooling operation, the RHR shutdown cooling isolation valve (F009) would not open due to an isolated RHR pump suction flow switch.	Flow switch had been isolated to perform calibration check; maintenance tech failed to unisolate instrument after test.	See Brunswick 1 (7/28/77).
Monticello 82-009 September 2, 1982	Mode 3, Hot Shutdown. Plant cooldown in progress.	During startup of shutdown cooling for a refueling outage, the RHRS outboard shutdown cooling isolation valve (MO-2030) motor failed.	Relaxing torque switch problem, which caused continuous close signal to jam the valve gate into the seat.	See Peach Bottom 3 (1/8/79).
LaSalle 1 83-142 November 4, 1983	Mode 3, Hot Shutdown.	The RHR shutdown cooling suction inboard isolation valve (F009) could not be opened either by the motor operator or manually. The unit was shutting down for planned maintenance.	During the last operating period, the valve was manually seated to stop leakage. With the plant at lower temperature, the valve would not open. Failure was attributed to high differential temperatures resulting in thermal contraction and pinching of the disk wedge into the valve seat.	See LaSalle 1 (6/11/82). ABWR has 3 RHR systems. One of the two remaining SDC loops would be available to bring the plant to cold shutdown.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Browns Ferry 1 84-012 February 14, 1984	Mode 3, hot shutdown. Plant cooldown to cold shutdown in progress.	While cooling down to cold shutdown following a manual scram, the inboard RHR shutdown cooling isolation valve (FCV-1-74-78) failed to open, making it impossible to achieve cold shutdown using normal shutdown cooling. An ALERT was declared, and the plant brought to cold shutdown through continued normal cooldown to the main condenser, and the use of control rod drive pumps and RWCUS as alternate inventory addition and heat removal systems. Since the stuck shut suction valve was inside containment, a containment entry was necessary to open the valve manually. It took approximately five hours to de-inert the drywell to permit entry, and another fours hours to open the stuck valve and establish shutdown cooling, after which the ALERT was cancelled. Additional alternate means of heat removal were available.	'B' phase winding of motor operator had failed. Apparently the gate had stuck in the valve seat and the motor could not generate enough torque to open the valve. Further investigation revealed that the 'close' torque switch setting was set higher than the manufacturer's recommended value (2.5 vice 2.0). This over-tightening probably contributed to the stuck valve.	See LaSalle 1 (11/4/83) and Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Dresden 3 May, 1978	Mode 4, Cold Shutdown. RHRS in operation at 344 K (160°F).	An inadvertent heatup and pressurization was caused by a valve lineup error during containment leak rate testing. About 18 hours after reaching test pressure, reactor vessel flange temperature was discovered to be at approximately 422 K (300°F) and increasing. One loop of shutdown cooling was in service recording a temperature of approximately 344 K (160°F). The RHRS heat exchanger shell temperature and vessel flange temperature should have been equal. Investigation revealed that the recirc pumps were off and recirc loop suction and discharge valves were open. This lineup resulted in the majority of RHRS flow circulating through the recirc loop and not the core. The vessel heatup and pressurization caused a temperature and pressure increase in the drywell. The computer program used to calculate the containment leak rate was using shutdown cooling temperature to indicate conditions inside the vessel. The computer misinterpreted vessel conditions and concluded there was a large inleakage condition.	Valve lineup error. Post maintenance testing of a recirc pump MG set required a recirc pump test run. The motors were uncoupled from the recirc pumps for the test. The motors would not start because pump/ valve interlocks gave a trip signal to the pump motor since the suction and discharge valves were closed. Consequently, maintenance personnel opened the valves to perform the test. This permitted shutdown cooling flow to bypass the core via the recirc loop, causing the inadvertent heatup and pressurization.	See LaSalle 1 (6/11/82). ABWR does not have external recirc pumps or valves. Reactor internal pumps (RIPs) supply recirc flow so this event could not occur in the ABWR.
Hatch 1 80-057 May 25, 1980	Mode 4, Cold Shutdown. RHRS in operation.	With the reactor in the shutdown mode during testing, the shutdown cooling suction valve for the ‘B’ RHRS pump (F006B) failed to open. The ‘B’ pump was declared inoperable. Since the ‘A’ division of RHRS was out for maintenance, both pumps in the ‘B’ division were required to be operable.	Faulty auxiliary contact block. The normally closed relay contact was found stuck in the open position.	ABWR has three RHR loops, failure of loop ‘B’ with loop ‘A’ in maintenance could be mitigated by using loop ‘C’. The CUW system, FPC, and main condenser can also be used for DHR under certain plant conditions.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Dresden 3 80-047 December 21, 1980	Mode 4, Cold Shutdown. RHRS system in operation.	Shortly after achieving cold shutdown, with recirc pumps off, CUW system isolated, and with one loop of shutdown cooling system in operation, it was noted that reactor vessel pressure was 1.136 MPa while recirc loop temperature was 341 K (155°F). Primary containment integrity specifications had been violated and both the HPCI and isolation condenser systems were out of service. A second shutdown cooling loop was placed in operation to achieve greater vessel flow, and to eliminate temperature stratification. When the mixing occurred, recirc loop temperature temporarily exceeded 373 K (212°F). Pressure and temperatures were reduced when the second loop was placed into service. The reactor pressure was above 0.72 MPa for about 1.25 hours.	Procedures were inadequate to address temperature stratification in reactor vessel with recirc pumps off and low shutdown cooling flow. Analysis in NSAC-27 also indicated a lower than normal reactor vessel water level contributed to the event by precluding core natural circulation.	See LaSalle 1 (6/11/82).
Dresden 2 83-052 June 21, 1983	Mode 3, hot shutdown. Plant cooldown in progress.	During preparation for placing shutdown cooling in service, a shutdown cooling return valve (MO-5A) failed to open.	The valve stem packing leakage from a nearby valve shorted out the valve operator motor.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 1 83-096 August 24, 1983	Mode 4, Cold Shutdown. RHRS loop ‘B’ in shutdown cooling operation.	During shutdown cooling operation on RHRS loop ‘B’, the ‘B’ heat exchanger discharge valve (F003B) failed to open. Most or all RHR flow was allowed to bypass the heat exchanger, and the heat exchanger outlet temperature increased from 333 to 359 K (139 to 186°F) over a three hour period. The inlet temperatures similarly increased. After three hours of attempts to open the shut valve, the ‘B’ RHR loop was secured and the ‘A’ loop started. ‘B’ loop temperature indication had not been accurate because of low flow conditions and temperature element placement, so actual reactor coolant temperature was higher. ‘A’ loop heat exchanger inlet temperature reached 370 K (207°F) (violating cold shutdown limits). The RPV head drain indicated a maximum temperature of 378 K (220°F).	The ‘B’ heat exchanger valve breaker was defective. The measured ‘B’ heat exchanger temperatures were concluded to be inaccurate due to temperature element location.	Placement of RHRS temperature detectors accurately reflect RCS temperatures if proper flow rates exist. See Peach Bottom 3 (1/8/79) for discussion of component quality and redundancy of DHR capability.
LaSalle 1 83-147 November 12, 1983	Mode 4, Cold Shutdown.	The ‘B’ RHR heat exchanger outlet valve (F003B) failed to open either by the motor operator or manually. The ‘A’ loop of RHRS was operable to control decay heat, but one of the two RHR SW pumps cooling the ‘A’ loop was inoperable.	It is believed that the valve became inoperable in the closed position due to water trapped in the body/bonnet cavity above the disk/seat ring seals. The cavity does not have a mechanism to vent entrapped water.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Hatch 1 79-050 July 25, 1979	Mode 4, Cold Shutdown. RHRS in operation on loop ‘A’.	While in the shutdown cooling mode, the 1C RHRS pump was found to have an excessive leak at the mechanical seal. The pump was removed from service to repair the seal. Both RHR pumps in the ‘B’ RHRS loop were out of service for hanger repairs. The 1C RHRS pump was returned to the shutdown cooling mode, the 1C RHRS pump was found to have an excessive leak at the mechanical seal. The pump was removed from service to repair the seal. The 1C RHRS pump was returned to service on July 27, 1979.	Ruptured seal in 1C RHRS pump.	ABWR technical specifications will be based on risk associated with shutdown mode and decay heat loads. Under certain conditions to minimize risk, at least two divisions of RHR or multiple alternate methods of DHR will be required to be operable.
Hatch 1 79-051 July 26, 1979	Mode 4, Cold Shutdown. RHRS in operation.	While performing design changes, control power cables to the RHRS outboard isolation valve (F008) were disconnected and cut with the valve in the open position. The inboard isolation valve (F009) had been made inoperable to allow modifications to be made to it. One of these valves is required for isolation of both divisions of the RHRS.	Personnel error in making the modifications to the operable RHRS isolation valve instead of the inoperable valve.	The three ABWR RHR systems are independent of each other. No common components, outside the RPV, exist which would impact more than one RHR division.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 2 80-107 December 8, 1980 80-112 December 9, 1980	Mode 4, Cold Shutdown. RCS temperature at 347 K (165°F). RHRS in operation on 'A' loop.	On December 7, RHRSW was secured in the 'A' loop to repair a leak on a 2.54 cm (1 inch) pipe to the RHRSW radiation monitor. Shutdown cooling was lined up to the 'A' loop with an RHRS pump running (to recirc the vessel water volume without heat removal). Both reactor recirc pumps were secured. 45 minutes were estimated to complete SW repairs. However, repairs were completed in 3 hours. RCS temperature at this time approached 373 K (212°F) with a local maximum of 376 K (217°F). The reactor head vents were open with atmospheric pressure in the vessel. SW was restored and shutdown cooling was initiated. Primary coolant temperature decreased to normal levels approximately 30 minutes after repairs were complete. Shutdown cooling was not lined up in loop 'B' because it was expected that loop 'A' would be back in service prior to approaching 373 K (212°F), and because there were possible leaks on a room cooler and inoperative 'B' loop pump suction valve motors. (Continued on following page)	In both events, maintenance was not completed in expected time. In the first event, loop B was available but not used, due to potential leaks on a room cooler and the requirement for manual valve operation due to inoperative pumps suction valve motors. In the second event, securing RHRS pumps while maintenance was in progress caused loss of representative temperature indications due to low flow and lack of vessel recirculation. Control room operators did not recognize the heat up rate. Failure to plan and promptly implement contingency plans for the possibility of unexpected delays in maintenance also contributed to the problem.	ABWR has three divisions of RHR. In this case, loop 'C' could have been used. See LaSalle 1 (6/11/82) for discussion of ABWR procedures.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 2 80-107 December 8, 1980 80-112 December 9, 1980 (Continued)		On the next day, the conventional and nuclear SW systems were secured to repair the 2A conventional SW pump discharge check valve. RCS temperature was initially <322 K (<120°F). Approximately 2 hours later, RHRS pumps were secured to reduce coolant heat input from the pumps. Approximately 4.5 hours later when the system was restored, the average RCS temperature was over 373 K (212°F) with a local maximum of 398 K (256°F). Again, vessel head vents were open during the event.		
Peach Bottom 2 81-031 May 18, 1981	Mode 4, Cold Shutdown. RHRS in operation.	With the unit shutdown for maintenance, shutdown cooling was secured to permit maintenance of a shutdown cooling suction isolation valve. RCS temperature exceeded 373 K (212°F) before cooling was reestablished. Temperature exceeded 373 K (212°F) for about 2.5 hours. Primary containment integrity requirements were not met during this period.	Lack of timely coordination between operations and maintenance personnel.	See Brunswick 2 (12/8/80).
Hatch 2 82-030 April 20, 1982	Mode 5, Refueling. RHRS in operation on loop ‘A’.	The ‘A’ loop flow indicators for both RHRS and RHRSW systems were noticed to be inoperable. Investigation revealed that the indicators and controller for the RHRSW heat exchanger pressure control valve were de-energized. The ‘A’ loop RHRS and ‘A’ RHRSWS were declared inoperable and fuel movement was suspended.	Sliding links were opened by maintenance personnel while performing a wiring change.	See LaSalle 1 (6/9/82 and 6/11/82).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Hatch 2 82-042 April 27, 1982	Mode 4, Cold Shutdown, with 'A' loop of shutdown cooling service.	The RHR and RHRSW flow indicator for the 'A' loop in shutdown cooling were inoperable. The 'A' loop was declared inoperable. The 'B' loop was already inoperable for the leak rate testing.	The spring clips on the fuse block energizing the 'A' loop RHR and RHRSW flow indicators were loose.	See Brunswick 2 (12/8/80) and Hatch 1 (7/25/79).
Browns Ferry 1 77-003 January 4, 1977	Mode 4, Cold Shutdown. RHRS in operation on loop 'A'.	The radiation monitor on the RHRSW discharge line from 1A RHRS heat exchanger showed an increasing radiation level, approximately 1 hour after being placed in service. Heat exchanger service water effluent was sampled and found to be in excess of release limits. The 1C RHRS heat exchanger was then placed in service, approximately 5 hours after the initial radiation alarm.	Leaking inner head gasket in heat exchanger, due to loose stud bolts. Delay in leak isolation due to failure to acknowledge alarm, and communications misunderstanding over the actual release rate occurring.	See LaSalle 1 (6/9/82 and 6/11/82).
Brunswick 2 80-030 April 12, 1980	Mode 4, Cold Shutdown. RHRS in operation.	During inspections, the 2B RHRS heat exchanger baffle plate was found to be partially buckled near the bottom where it fitted into the groove of the channel cover. The plate was 21.59 cm (8.5 inches) off-center, and welds up each side were pulled loose within the waterbox. Approximately 20 - 25-cm (8 - 10-inch) thick accumulation of marine growth shells were found in the inlet side of 2B heat exchanger waterbox, and about the same in 2A heat exchanger inlet waterbox, although the 2A baffle plate was not damaged. The buckling created a service water bypass flow path from the heat exchanger inlet to outlet bypassing the tubes.	Excessive differential pressure across the baffle plate due to an accumulation of marine growth shells in the heat exchanger.	ABWR procedures to minimize marine growth have been modified to ensure this type of event does not occur. Intermediate RCW system loop provides clean water to the RHR heat exchanger. Also, alternate methods of DHR such as the main condenser, FPC, and CUW can be used under certain plant conditions.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 1 81-032 April 19, 1981	Mode 4, Cold Shutdown. RHRS in operation.	During inspection of the 1B RHRS heat exchanger, it was found that the heat exchanger baffle plate was displaced about 23 cm (9 inches), creating a direct SW flow path from inlet to outlet, bypassing the tubes. During repair of the 1B heat exchanger, a loss of cooling was experienced immediately following the starting of a second RHRSW pump on the 1A heat exchanger. Alternate cooling was established with the RHRS system by flow from the vessel, through the fuel pool coolers and the CST. Vessel temperature remained below 350 K (170°F). The 1A heat exchanger was also found to have a displaced baffle plate.	Failure of plate welds, resulting from excessive differential pressure across the plate. Excessive differential pressure attributed to blockage of the tubes by marine shells accumulating in the heat exchanger. The SW chlorination system had been out of service for an extended period.	See Brunswick 2 (4/12/80).
Brunswick 2 81-049 May 6, 1981	Mode 1, 76% power.	As a result of problems with Unit 1 RHRS heat exchangers, a special inspection of Unit 2 RHR HXs was conducted at power. Heat exchanger 2B was damaged and plugged by marine shell buildup. The divider plate was found buckled about 7.6 cm (3 inches). (It had been replaced in 1980—see LER 80-030 above.) The heat exchanger had blocked and obstructed tubes. Heat exchanger 2A was undamaged with no divider plate buckling, but was substantially blocked by shells.	Same cause as for Unit 1 above.	See Brunswick 2 (4/12/80).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Browns Ferry 1/2/3 81-047 August 22, 1981	Units 1 & 3, Mode 4, Cold Shutdown. Unit 2, Condition 1, 91% Power.	The A2 RHR service water/EECW pump discharge line air vent valve failed, resulting in the flooding of ‘A’ RHRS service water/EECW pump room to a depth of approximately 198 cm (6 1/2 ft), rendering A1, A2, and A3 RHRSW/EECW pumps inoperable. Consequently, the ‘A’ RHRS heat exchangers for the 3 units became inoperable. (The RHRSW/EECW system is common to all three units.)	The ‘A2’ pump discharge air vent valve failed to seal because of a broken float guide, causing the float to misalign with the seat.	ABWR is a single unit design and failures will not propagate to other plants. If more than one ABWR is at a site, cross connected systems between units will not be allowed. In addition, ECCS divisional rooms contain water tight doors such that flooding would be contained within the room and only affect one division. Floods in other reactor building rooms are mitigated by raised sills, floor drains, and operator action in response to flood alarms.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Browns Ferry 3 83-004 January 16, 1983	Mode 4, Cold Shutdown RHRS in operation on loop 'B'.	RHR heat exchanger 3D leaked reactor coolant into the RHRS service water in excess of Technical Specification limits. The 'D' heat exchanger and pump were removed from service and the 'B' heat exchanger and pump placed in service. Approximately 8 hours later an alarm was received on the SW effluent monitor. The 'B' heat exchanger and pump were removed from service. (The 'A' and 'C' heat exchanger and pumps were inoperable due to a bent stem on their common injection valve.) Thus there was a complete loss of RHR shutdown cooling capability. The RCS temperature increased from 360 to 373 K (188 to 211°F) in approximately 45 minutes. Reactor heat removal was provided by steaming to the main condenser, and by coolant makeup from the CRD and CUW systems.	Twelve dented tubes were found in the 'D' heat exchanger. One of the dented tubes was leaking. The 'B' heat exchanger did not actually leak, but had to be isolated until it could be confirmed to not be leaking. The 3B and 3D heat exchangers share a common radiation monitor.	ABWR has three independent RHR loops. The probability of losing all three loops due to component failures is very low. Even if loss of all RHR were to occur, DHR could be completed using the main condenser, CUW, FPC, CRD, HPCF, condensate, or fire protection water systems depending on plant conditions.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Peach Bottom 3 81-014 September 2, 1981	Mode 4, Cold Shutdown.	While in cold shutdown near the end of an extended refueling outage, difficulty in maintaining reactor water level was encountered when the control rod drive water system was removed from service to support plant testing. Vessel level decreased to about 152 cm (60 inches). A feedwater inlet valve was then opened slightly to supply makeup water to the vessel. (Pumping source not stated in LER, and uncertain because turbine driven feed pumps are unusable in cold shutdown. Source was probably condensate pump.) Vessel level was recovered to 229 cm (90 inches) and the feedwater valve closed. Leakage through the valve occurred and level increased above the main steam line nozzles. As a result of the loss of the reactor vent path (main steam lines to condenser), the reactor pressurized to about 0.322 MPa for about 35 minutes (MSIVs assumed to be shut to prevent flooding of steam lines). To decrease vessel level, water was transferred from the vessel to the torus. Later during an attempt to obtain a tight shutoff of the feedwater inlet valve, the reactor was again pressurized to about 0.646 MPa for 30 minutes. The reactor head vents were opened to depressurize the vessel.	Operator failed to recognize that he was losing primary system inventory when the CRD water system was removed from service. Incomplete closure of feedwater inlet valve MO-3-2-29B caused level increase above main steam line nozzles and subsequent pressurization.	See LaSalle 1 (6/11/82).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Browns Ferry 2 83-005 February 16, 1983	Mode 5, Refueling.	During preparation for a containment integrated leak rate test (ILRT), a spurious low reactor vessel water level signal was initiated, apparently due to improper operation of a high drywell pressure switch drain. The combination of low water level and high drywell pressure signals started four core spray pumps, four RHR pumps, and eight diesel generators. The RHR system was secured before injection into the vessel occurred. However, a total of 167 m^3 (44,000 gallons) of water were injected into the vessel from the torus via the core spray system, which caused spillage into the drywell sumps via an open head vent, and put some water into the steam lines. The vessel head was in place with the head fastening nuts not installed.	Cause not reported in LER.	See LaSalle 1 (6/11/82). ABWR has complete divisional separation in a two-out-of-four logic network that prevents spurious initiation signals from single event errors.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Peach Bottom 3 83-007 March 3, 1983	Mode 5, Refueling.	During a refueling outage, an inadvertent initiation of two RHR pumps in the LPCI mode caused an injection of 246 m^3 (65,000 gallons) of water from the torus into the reactor vessel. Since the unit was in refueling with the reactor cavity flooded, most of the water overflowed onto the fuel floor, and down the main hatchway to El. 41,150 mm (135 feet), where approximately 0.189 m^3 (50 gallons) flowed out the building under the railroad door and into the storm drain system. The initiation was a false low water level signal which was present for less than 3.5 seconds. The signal started all operable diesel generators, tripped and isolated recirc pumps, tripped HPSW pumps, and started 2 RHR pumps. Diesel generator starts and the large number of spurious alarms distracted operators from verifying reactor water level until about 4 minutes after actuation, at which time the pumps were tripped and injection valves closed. Personnel exited the area, and no personnel exposures resulted from the flooding. The total dose associated with the subsequent cleanup effort was less than 0.02 person-Sievert. Total release was estimated at 11.69 megabecquerel.	The spurious low water level signal was caused by a pressure surge in the reference leg of the 2B Yarway instrumentation loop during surveillance testing.	See LaSalle 1 (6/11/82) and Browns Ferry 2 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Vermont Yankee March 9, 1989	Mode 5, Cold Shutdown. RHRS in operation on loop ‘B’.	With loop ‘B’ of RHR in SDC mode and loop ‘A’ out of service for maintenance, ‘A’ and ‘C’ RHR pump motor breakers were racked out for maintenance. System logic then causes the mini-flow valves for these pumps to open. Following maintenance, the ‘A’ and ‘C’ SDC suction valves were manually stroked open per procedure. This opened a drain path from the RPV to the suppression pool. Reactor cavity level dropped approximately 554 to 183 cm (218 to 72 inches) above top of active fuel.	Improper use of procedures.	In the ABWR design, racking out the RHR pump breakers does not result in the mini-flow valves opening. See LaSalle 1 (6/11/82) for discussion of ABWR procedures.
Susquehanna 1 February 3, 1990	Mode 4, cold Shutdown. RHRS in SDC mode using loop ‘A’.	With reactor coolant temperature approximately 264 K (125°F), the RHR system was removed from service to perform a test of the RPS electrical protection assembly (EPA) breakers. RHR must be secured during this test because opening the EPA breakers causes isolation of SDC. Following testing, difficulty was experienced in closing some of the EPA breakers to energize the RPS. This delayed reestablishing SDC. Reactor coolant temperature increased to 396 K (253°F) and pressure increased to 0.232 MPa before SDC was restored.	Excessive time to complete maintenance.	Alternate means of DHR could be used including main condenser, venting steam to the suppression pool through SRVs and suppression pool cooling. See Susquehanna 1 (2/16/83).
Quad Cities 2 4/2/92	Mode 4, Cold Shutdown.	SDC was lost for two hours and twenty minutes due to loss of power to 1E buses.	Inadvertent actuation of fire protection deluge system.	See Susquehanna 1 (2/16/83)

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Washington Nuclear Plant 2 May 1, 1988	Cold Shutdown with RHR ‘B’ in SDC mode.	Operators were in the process of changing the operating SDC loop from ‘B’ to ‘A’. The procedure called for closing the loop ‘B’ SDC suction valve and then open the loop ‘B’ suppression pool suction valve. The operator did not wait until the SDC suction valve completely closed before opening the suppression pool suction valve. The stroke time on each of these valves is 120 seconds. Both valves were partially open for 40 seconds and resulted in about 37.85 m^3 (10,000 gallons) of water draining from the reactor cavity to the suppression pool. Draindown was automatically terminated on low RPV level when SDC was isolated.	Improper operator action.	The ABWR suppression pool suction valve cannot be opened until the SDC suction valve is fully closed.
River Bend April 19, 1989	Cold Shutdown.	Work was being performed on the standby service water (SSW) supply and return valves. As these valves are unisolatable, freeze seals were being used to isolate the valves. One of the freeze seals failed and caused approximately 56.78 m^3 (15,000 gallons) of water to flood the Division II ECCS power supply room. Electrical faults resulted in loss of power to RPS bus ‘B’. This caused containment isolation and loss of SDC.	Improper freeze seal implementation.	ABWR freeze seal procedures will include adequate administrative controls to minimize freeze seal failures. Analysis have been completed to ensure that flooding in the ABWR will not result in loss of ECCS or RPS power supplies.