19.3 Internal Event Analysis

The information in this section of the reference ABWR DCD, including all subsections and tables, is incorporated by reference with the following departures and supplements.

STD DEP T1 2.4-3

STP DEP T1 5.0-1

STD DEP 2.2-5

STD DEP 8.3-1

STP DEP 9.2-5

STD DEP 10.4-5 (Table 19.3-2)

STD DEP 19.3-1

STD DEP Admin

19.3.1 Frequency of Core Damage

The following site-specific supplement addresses the following departures identified in other sections of the FSAR:

STD DEP T1 2.4-3

STP DEP T1 5.0-1

STD DEP 8.3-1

STP DEP 9.2-5

STD DEP 10.4-5

STD DEP 19.3-1 for evaluation of common cause failures per SSAR 19D8.6

An evaluation was performed to account for departures from the assumptions used in the ABWR SSAR PRA. As part of the evaluation, site-specific information was used to to assess changes in results and insights (Delta-PRA) to confirm continued compliance with the conclusions in the reference ABWR DCD. Details of the changes considered are provided in Section 19.2. In addition, a revised evaluation of common cause failures of various mechanical systems (evaluated as part of a PRA sensitivity study that was documented in ABWR SSAR Section 19D.8.6) was performed as part of this supplemental evaluation to address an issue identified by the NRC regarding the original SSAR evaluation.

19.3.1.1 Accident Initiators

The following site-specific supplement addresses frequency of initiating events.

The total frequency of transient initiators used in these evaluations is based upon a 1985 analysis of operating plant data (Reference 19.3-1). The frequency of transients is a design requirement prescribed in the Advanced Light Water Reactor (ALWR) Requirements Document (Reference 19.3-2). Apportioning of the expected transient frequency by initiating event was done on the basis of historical electrical grid and BWR performance data as described in Subsection 19D.3. In order to verify that the Subsection 19D.3 remains bounding for the STP 3 & 4, loss of offsite power and power recovery data from NUREG/CR-6890 (Reference 19.3-8) was also evaluated in a sensitivity study. Industry composite data in NUREG/CR 6890 for the Electric Reliability Council of Texas (ERCOT) was used, which conservatively bounds the experience for the STP site. This evaluation verified that the overall risk impact of grid events at STP is bounded by the original Subsection 19D analysis.

19.3.1.3 Accident Sequence Analysis

STD DEP 10.4-5

(a) Core Cooling

The capacity of non-safety-related systems, such as the feedwater, condensate booster *and condensate pumps, has been estimated based on the ECCS performance analyses. Non-safety-related systems which contribute to a successful conclusion of the event have been included in the success criteria. The Control Rod Drive (CRD) pumps which have limited capacity have not been included in the success criteria.*

The condensate and condensate booster *pumps are motor-driven pumps and their use depends on the RPV pressure and the availability of makeup water and electrical power. These pumps have higher shut–off heads than the RHR pumps, but still require depressurization before they can be used for core cooling. The source of makeup water for these pumps are the main condenser hotwell and the condensate storage tank. Sufficient makeup water is available to enable these pumps to maintain adequate core cooling for all events except large or medium liquid LOCAs.*

A motor driven feedwater pump is combined in series with a condensate booster and condensate *pump in order to provide a higher pressure system. Therefore, this option also depends on the availability of makeup water and electrical power. Sufficient makeup water is available to enable this series of pumps to maintain adequate core cooling for the small steam LOCA and transient events.*

19.3.1.4 Frequency of Core Damage

The following site-specific supplement addresses the following departures identified in other sections of the FSAR:

STD DEP T1 2.4-3

STP DEP T1 5.0-1

STD DEP 8.3-1

STP DEP 9.2-5

An evaluation was performed to account for departures from the assumptions used in the ABWR SSAR PRA. As part of the evaluation, site-specific information was used to assess changes in results and insights (Delta-PRA) to confirm continued compliance with the conclusions in the reference ABWR DCD.

In some cases (e.g., STP DEP T1 5.01), the departures have the effect of increasing the core damage frequency. In other cases (e.g., STD DEP T1 2.4-3), the departures have the effect of decreasing the core damage frequency. The overall results of the evaluation are bounded by the conclusions of the standard ABWR SSAR.19.3.1.5 Results in Perspective. The net impact of the STP-specific design shows a net decrease in risk as compared to the standard ABWR PRA.

19.3.1.5 Results in Perspective

The following site-specific supplement addresses the following departures identified in other sections of the FSAR:

STD DEP T1 2.4-3

STP DEP T1 5.0-1

STD DEP 8.3-1

STP DEP 9.2-5

STD DEP 19.3-1

The estimated core damage frequencies are extremely low. It is impossible to calculate such low numbers with a high degree of confidence using the PRA models developed here. For example, a number of potential common cause failures of components such as similar ~~pumps and~~ valves have not been included in the fault tree models, on the expectation that such failures are negligible contributors to overall core damage frequency.

The perspectives determined from the original analyses presented in the ABWR DCD remain applicable with respect to the site-specific information and departures summarized above.

19.3.3 Magnitude and Timing of Radioactive Release

STP DEP Admin

(3) ~~(1)~~The time available for offsite evacuation, should it be necessary, is also important. Discussions with several utilities indicate that evacuation of their Emergency Planning Zones (EPZ) can be completed in less than 8 hours, even in the worst weather conditions. Experience has also indicated that ad hoc planning can successfully evacuate a region ~~on~~ in *about 24 hours ~~(Reference 4 of Appendix 6J)~~* (Appendix 6J of Reference 19.3-4).

Based on the forgoing, four time frames were selected in determining the time of fission product release, either via the rupture disk or directly from the drywell. Table 19.3-6 summarizes the results which were obtained by using the probabilities ~~given in Table 19D.5-3~~ summarized in Table 19.3-4 *and assigning them to a time and mode of release based on the accident analysis contained in Subsection 19E.2.2.*

19.3.4 Consequence of Radioactive Release

STD DEP 2.2-5

The evaluation for consequences of potential radioactive releases was performed using the ~~CRAC-2~~ MACCS2 *computer code* (Reference 19.3-9) *as is detailed in Subsection 19E.3. Based upon the evaluation of plant performance, accident classes were defined in terms of their associated release characteristics and fission product releases. Each accident class was then evaluated by the ~~CRAC-2~~* MACCS2 *code at ~~five sites~~*STP*~~, one representing each major geographical region of the United States. Each site was chosen as representative of its geographical region based upon meteorological calculations and was further defined as average in terms of population density for that geographical region.~~ The results ~~for the five sites were averaged and~~* were *compared to three goals, two based upon the NRC safety goal policy of minimizing risk to an individual and the public near a plant, and the third based upon an industry goal of minimizing the dose close to the plant. The results of this study show that ~~the ABWR Standard Plant~~*STP 3 & 4 *satisfies these goals* and that the results of the ABWR DCD analysis using the CRAC2 code are bounding.

19.3.5 References

The following site-specific supplement includes new references.

19.3-8 Reevaluation of Station Blackout Risk at Nuclear Power Plants: Analysis of Loss of Offsite Power Events: 1968 – 2004, NUREG/CR-6890, US Nuclear Regulatory Commission, December 2005.

19.3-9 Code Manual for MACCS2, Users Guide, NUREG/CR-6613, May 1998.

Table 19.3-2 Success Criteria to Prevent Initial Core Damage for Transient and LOCA Events With RPS Scram

Event	Success Criteria
CORE COOLING:	
Large Liquid LOCA [>=278.7 cm^2 (0.3 ft^2)]	HPCF-B or C or LPFL(1) – A or B or C
Large Steam LOCA [>=278.7 cm^2 (0.3 ft^2)]	HPCF-B or C or LPFL(1) – A or B or C or 1 Condensate Pump + 1 Condensate Transfer Pump(2)
Medium Liquid LOCA [>=278.7 cm^2 (0.3 ft2) >5.063 cm^2 (0.00545 ft^2)]	HPCF-B or C or ADS3(3) + LPFL(1) – A or B or C
Small Liquid LOCA [>=5.063 cm^2 (0.00545 ft^2)]	RCIC(4) or HPCF-B or C or ADS3(3) + LPFL(1) – A or B or C or ADS3(3,5) + 1 Condensate Pump + 1 Condensate Booster Pump + 1 Condensate Transfer Pump(2)
All Transients (including IORV)	RCIC(4) or HPCF-B or C or 1 Feedwater Pump + 1 Condensate Booster Pump + 1 Condensate Pump + 1 Condensate Transfer Pump(2) or ADS3(3) + LPFL(1) – A or B or C or ADS3(3,5) + 1 Condensate Pump + 1 Condensate Booster Pump + 1 Condensate Transfer Pump(2) or ADS8(6) + 1 Firewater Addition System Pump

Table 19.3-2 Success Criteria to Prevent Initial Core Damage for Transient and LOCA Events With RPS Scram

Event	Success Criteria
CORE COOLING (Cont.) Small Steam LOCA [<278.7 cm^2 (0.3 ft^2)]	HPCF-B or C or 1 Feedwater Pump + 1 Condensate Booster Pump + 1 Condensate Pump + 1 Condensate Transfer Pump(2) or ADS3(3) + LPFL(1) – A or B or C or ADS3(3,5) + 1 Condensate Pump + 1 Condensate Booster Pump + 1 Condensate Transfer Pump(2)
LONG-TERM HEAT REMOVAL:	
All Transients or Small Liquid LOCA	RHR-A or B or C(7) or Normal Heat Removal(8) or CUW(9)
All Steam LOCAs or IORV or Liquid LOCA (Large or Medium)	RHR-A or B or C(7) or Normal Heat Removal(8)
PRESSURE RELIEF:	
Isolation Events	6 Safety/Relief Valves
Non-Isolation Events	3 Turbine Bypass Valves or 2 Turbine Bypass Valves + 2 Safety/Relief Valve or 1 Turbine Bypass Valve + 4 Safety/Relief Valves or 6 Safety/Relief Valves

Notes:

(1) The term “LPFL” refers to the low pressure core flooding mode of the Residual Heat Removal (RHR) System.

(2) The condensate pumps take suction from the hotwell which is a limited water source. Therefore, if the MSIVs are not open, a condensate transfer pump is necessary to pump water from the condensate storage tank to the hotwell in order to replenish the water in the hotwell.

(3) The term “ADS3” implies that at least 3 automatic depressurization valves are automatically actuated on low level and high drywell pressure or the same number of SRVs are manually opened when the ADS would have actuated. For transients, the high drywell pressure signal is not present.