significant fission product release to the environment due to a dry cavity and unavailable containment sprays.

RC 21 – This category represents those sequences in which the containment fails late with a rupture failure size, the containment spray does not function, and the cavity condition is wet. The release characteristics of this category are the same as those of RC 17, except for the failure size of the containment. This represents the containment failure modes that include a containment failure due to steam overpressurization. In this category, there is no significant fission product release to the environment due to a wet cavity. However, the releases are not scrubbed by the containment sprays.

The summary of the MAAP results (release magnitude and timing) and release categorization (i.e., large release, ~~large early release,~~ or not large release) is presented in Table 19.1-29 through 19.1-30b. Punch List #21

19.1.4.2.2 Results from Level 2 Internal Events PRA for Operations at Power

It should be noted that units for CDF and LRF are expressed in terms of "reactor calendar year" (shortened to "/year" when displayed in the text in this section).

19.1.4.2.2.1 Risk Metrics

Total LRF from internal events is 1.~~1~~0 × 10^{-7}/year. This is well below the NRC goal for LRF below 1 × 10^{-6}/year. Mean value and associated uncertainty distribution can be found in Subsection 19.1.4.2.2.7. R_434-8352(92R)

The conditional containment failure probability (CCFP) from all internal events (at power) in large release sequences is ~~8.4~~9.1x10^{-2}. This meets the NRC goal of no more than approximately 0.1 for CCFP. This CCFP is the conditional probability of a large release (CPLR) for internal events operations at power. R_434-8352(92R)

19.1.4.2.2.2 Internal Events Core Damage Release Category Results

The relative contributions of the release categories to the total STC frequency are shown in Figure 19.1-49. Figure 19.1-50 groups the categories further into no containment failure, large release, and small release. For this analysis, containment failure is defined as the loss of containment integrity within the first 24 hours following vessel failure. Almost the entire small release frequency (SRF) in Figure 19.1-50) is due to basemat melt-through, which would occur much later than 24 hours after core damage, and therefore is not relevant to the conditional containment failure probability (CCFP) calculation. The APR1400 CCFP of ~~0.084~~0.091 was calculated by summing the frequencies of all release classes in which the containment failed within 24 hours after vessel failure or containment integrity was lost due to bypass or isolation failures, and dividing by the total CDF. Therefore, the value thus calculated meets the NRC goal of less than 0.1. R_434-8352(92R)

Approximately ~~49~~71 percent of the LRF for internal events is from STC 1, which are unmitigated, unisolated SGTR releases (both SGTR initiating event and induced SGTR). The next-highest frequency STC is a late rupture with no containment sprays (~~27~~ 12 percent), followed by containment failure (rupture) prior to core damage (~~12~~8 percent), and containment failure (leak) prior to core damage (~~10~~7 percent). ~~Early~~Containment isolation failure with no sprays contributes 0.9 percent to the LRF, and early containment rupture with no sprays contributes ~~1.6~~0.5 percent to the LRF, and containment isolation failure with no spray contributes ~~1.1~~0.9 percent. The remaining STCs have a negligible contribution to the LRF. R_434-8352(92R) R_434-8352(92R)

19.1.4.2.2.3 Significant Sequences and Cutsets S_1A001

The significant LRF cutsets for the internal events Level 2 PRA are illustrated in Table 19.1-31. This table provides the top 100 LRF cutsets, and does not exclude any cutset that contributes over 0.1 percent to the total LRF.

Cutsets that contribute 1 percent or more to large release for internal events are described as follows.

The first ~~six~~ ten cutsets are all pressure-induced SGTRs. The first is an ATWS, in which the high RCS pressure induces the rupture. The next ~~five~~ nine are all main steam line break (downstream of MSIVs) with a failure to close the MSIVs, where the rapid decrease in secondary side pressure creates a large pressure differential across the tubes, inducing the break. The LSSB-D cutsets each have success of safety injection and rapid depressurization (SDR) in their event tree sequence logic, making them conservative in a classification as “large” releases. R_434-8352(92R)

~~LRF cutsets 7 and 8 are a total loss of ESW or CCW, a resulting RCP seal LOCA, and failure to run of the auxiliary charging pump. Containment sprays are unavailable because of the cooling water system failures, and late containment rupture results.~~ R_434-8352(92R)

~~Cutsets 9 through 12 are similar to cutsets 2 through 6, with different CCFs of the MSIVs.~~ R_434-8352(92R)

~~Cutset 13 involves containment failure prior to core damage. It is a medium break LOCA with failure of the containment sprays to provide containment heat removal. The eventual overpressurization of the containment causes the containment failure that is assumed to fail the systems that would prevent core damage (e.g., loss of NPSH for the SIPs).~~LRF cutset 11 is a SGTR initiating event with failure of all AFW and human failures to depressurize the reactor for feed and bleed prior to core damage or for emergency depressurization after core damage. The dependency between the two human actions was considered and is reflected in the cutset.

19.1.4.2.2.4 ~~19.1.4.2.2.3~~ Significant Core Damage End States, Initiating Events, Phenomena, and Basic Events S_1A001

Table 19.1-32 and Figure 19.1-51 present the LRF contribution by internal initiating events. The largest contributor, with ~~27~~ 41 percent, is a steam line break inside the containment (downstream of the MSIVs). This contribution arises because of the steam line break inside containment sequence described in Subsection 19.1.4.2.2.3. The second largest contributor to LRF (16 percent) is SGTR-initiated core damage sequences that result in large releases. The third largest contributor (13 percent) is Medium LOCA sequences involving containment failure before core damage. The ~~second and third~~ fourth and fifth largest contributing initiating events are LOOP and SBO (10 percent), each with failure of containment heat removal and a flooded cavity, leading to late containment overpressurization. General transienst, dominated by ATWS-induced SGTR, is the next largest contributor to LRF, at 7 percent. ~~The fourth largest contributing initiating event is medium break LOCA, with the containment failure before core damage sequences. ATWS-induced SGTR is fifth in frequency.~~ R_434-8352(92R) R_434-8352(92R) R_434-8352(92R)

~~Table 19.1-33 presents the significant plant damage states (PDSs) that contribute to LRF.~~

Table 19.1-34 presents the important basic events to LRF, ranked by RAW. Table 19.1-35 presents the same, ranked by FV. Table 19.1-36 presents the common cause event importance to LRF, ranked by RAW. Table 19.1-37 presents the same, ranked by FV. Table 19.1-38 presents the human action importance, accounting for both pre-initiators and post-initiators, ranked by RAW. Table 19.1-39 presents the same, ranked by FV.

19.1.4.2.2.5 ~~19.1.4.2.2.4~~ Key Assumptions S_1A001

a. If the MSSVs pass water in an SGTR, they are assumed to be stuck open and there is no means to isolate the ruptured SG. Thus, for the sequences in which RCS pressure control is failed (called ECLDN or SDR), the SG isolation is conservatively assumed to be failed and those sequences are directly considered as containment bypass sequences.

b. The conditional probability of PI-SGTR, given ATWS and MSLB/FWLB sequences without feedwater, is assumed to be 0.027 based on NUREG/CR-6365.

c. For calculation of hydrogen mass generated in a severe accident in the first few hours after core damage, it is assumed that insufficient CCI can occur so that the hydrogen contribution due to CCI is small. This leads to a maximum hydrogen production during the time window concerned, equivalent to 100 percent oxidation of the active cladding.

d. For late hydrogen burns resulting in a "late" containment failure, the hydrogen burn pressures are assumed to result from the AICC process. Those hydrogen burn pressures for various conditions (such as the cavity condition, the operation of PARs, and so on) are estimated by using the MAAP code.

e. For the evaluation of hydrogen burns and detonations in an SBO, it is assumed that an ignition source is available when the burnable condition is established in containment.

f. In the evaluation of induced SGTR during a core melt at high RCS pressure and dry steam generators, the conditional probabilities of tube failure for moderately degraded SG tubes from NUREG-1570 are used. This is considered to be a conservative assumption.

g. In the induced SGTR modeling, it was assumed that any RCP seal LOCA failure under high RCS pressure with a dry SG would result in a cleared RCS loop seal. This is very conservative per the NUREG-1570 guidance and EPRI research, but the conservatism did not significantly impact the LRF.

h. External reactor vessel cooling is conservatively not credited in the baseline Level 2 analysis, but is evaluated in a sensitivity analysis.

o. In the analysis of severe accident-induced SGTR, the induced SGTR is assumed to occur prior to induced hot leg failure.

p. Once core damage occurs, fission products can be released at the design leakage rate, even when containment has not yet failed. The assumed pre-existing containment design leakage is a rate of 0.10 volume-percent per day at the design pressure and temperature. In this analysis, this design leakage was applied to all source term categories including those categories that were assessed as an intact containment release category.

q. In the Level 2 analysis, the model assumes that the ex-vessel core debris coolability for wet cavity condition is 0.5.

r. In the Level 2 analysis, the model assumes that the ignition source inside the containment always exists in both early phase and late phase.

s. In the Level 2 analysis, the model assumes that the probability of low heat transfer rate from ex-vessel core debris to water, which causes an ultimate basemat melt-through, is 0.01.

19.1.4.2.2.6 ~~19.1.4.2.2.5~~ Sensitivity Analysis S_1A001

In the containment performance analysis, several assumptions were made regarding the progression of severe accident phenomena. Sensitivity analyses were performed to assess the potential impact on the Level 2 results due to the potentially significant assumptions. Quantitative sensitivity studies can be performed where propagation of uncertainties is not practical or where the uncertain issues do not readily lend themselves to quantitative treatment.

These analyses assessed the impact of specified assumptions on the containment failure modes and the overall conditional containment failure probability. These analyses involved changing certain conditions or assumptions that are modeled in the CETs/DETs and then requantifying the Level 2 models to ascertain the impact.

These sensitivity analyses also provide significant insights into the dominant containment phenomena in terms of their contribution to LRF and the total containment failure frequency.

Case R9 increases to the induced SGTR conditional probabilities by 20 percent, and results in a very small increase in release frequencies. The LRF increased from 10.0 percent to 10.1 percent, and the total containment failure frequency increased from 17.0 percent to 17.1 percent. This sensitivity demonstrates that the LRF is insensitive to changes in the conditional probability of ISGTR, and a very large change in the conditional probabilities would have to occur in order to significantly affect the APR1400 LRF. R_434-8352(92R)

The remaining sensitivity cases demonstrate that the remaining Level 2 phenomenological uncertainties have a relatively small or no impact on the LRF.

19.1.4.2.2.7 ~~19.1.4.2.2.6~~ Uncertainty Analysis S_1A001

The results of the uncertainty evaluation for the Level 2 internal events LRF are as follows:

5 percent value: 4.~~7~~ 6 $\times 10^{-8}$/year R_434-8352(92R)

Mean value: 1.~~6~~ 2 $\times 10^{-7}$/year

95 percent value: 2.~~9~~ 4 $\times 10^{-7}$/year

19.1.4.2.2.8 ~~19.1.4.2.2.7~~ Risk Insights S_1A001

The sensitivity analyses provide the best insights into the APR1400 Level 2 model. The analyses demonstrate that the LRF is very sensitive to the operation of the ECSBS to prevent long-term containment overpressurization. If ECSBS were not credited to prevent long-term overpressurization, the LRF would rise significantly.

The cavity flood system is important in maintaining an intact containment. Unavailability of the cavity flood system actually causes a ~~slight~~ drop in the LRF because it eliminates the steam overpressure failure, but the total containment failure frequency increases substantially, mainly due to basemat melt-through. R_434-8352(92R)

Finally, the conservative assumption that the SG tubes are in an "average" condition, which is used to define the conditional failure probabilities for induced SGTR, has a ~~significant~~ small to moderate contribution to the baseline LRF. If credit were given to considering the tubes as being "pristine" and well maintained, the LRF would drop ~~significantly~~by approximately 10%. R_434-8352(92R)

19.1.5 Safety Insights from the External Events PRA for Operations at considered credible. Likewise, hot gases entering the turbine building from an adjacent fire compartment via a failed barrier will be directed upwards and out the numerous roof vents; hence, fire spread to the turbine building is not considered credible (Task 11).

n. Due to the size of the containment building (F000-C01), any hot gas layer formed would be near the top of the dome. This is true regardless of whether the fire originated within containment, or entered containment via a failed barrier from the auxiliary building. The top of the dome is well over 45.7 m (150 ft) above the highest cable tray in containment and the highest penetrations to the auxiliary building. Therefore, any hot gas layer formation in the containment would not be located where it is credible to assume: 1) damage to cables within containment, and 2) spread into the auxiliary building. Therefore, potential fire spread scenarios from or to the containment building are not considered credible (Task 11).

o. It is assumed that automatic suppression systems are designed so that, if successfully activated, they will extinguish the fire prior to additional damage beyond the ignition source itself. Hence, if the ignition source is not a fire PRA-credited component, successful operation of the automatic suppression system will result in a general transient (likely a manual trip) with no PRA-credited equipment damaged. If the ignition source is a PRA-credited component, and the automatic suppression system successfully operates, the fire-induced initiator will be dependent upon the ignition source (e.g., fire in dc bus A will result in LODCA initiator), but will only involve the failure of the ignition source. Failure of an automatic suppression system is assumed to result in full room burnout and possible spread to adjacent compartments.

p. Since there is currently no plant for exploratory or confirmatory walkdowns, and since there are currently no post-seismic or post-fire safe shutdown procedures, or fire brigade response procedures, it is assumed that an adequate assessment of seismic-fire interactions can be performed using design and other PRA documents. It is assumed that the COL applicant will re-perform this analysis when the plant is complete, and when the procedures are available.

~~p~~q. Promptly suppressed hotwork fires are assumed to result in a plant trip with no damage to fire PRA-credited equipment or cables (e.g., general transient fire- S_1A001 induced event). The basis for this assumption is that due to the likelihood of manual detection of the fire during its incipient stage, it is unlikely to burn for a sufficient duration and heat release rate (HRR) to damage nearby targets. Furthermore, during hotwork, any nearby equipment or transient combustibles are likely protected by welding blankets or other heat shielding.

~~q~~r. It is assumed that, for all fires, there is damage to at least some of the instrumentation necessary for operator cues in diagnosing plant conditions; however, this damage is limited to the division in which the fire occurred, and it is assumed that the operators will be trained to rely on undamaged instrumentation once the location of the fire is known. S_1A001

~~p~~s. Spurious opening of pressurizer Pilot Operated Safety Relief Valves (POSRVs) due to a fire is not considered, because the power to the POSRVs is removed during normal operation and can only be provided by manual operator action. S_1A001

19.1.5.2.1.3 Analysis Details

Task 1, Plant Boundary and Partitioning, is conducted in two parts. The first activity involves definition of the global plant analysis boundary, which is defined to be the plant protected area and switchyard; however, it does not include all of the licensee-controlled areas. Notable facilities that are located within the licensee-controlled area but not in the global plant analysis boundary include the engineering building, wastewater treatment facility, and sanitary water treatment facility. Miscellaneous support structures and parking lots are also located throughout the licensee-controlled area, but not included in the global plant analysis boundary.

Meaningful fire analysis within the global plant analysis boundary requires establishment of realistic bounds that describe the expected extent of individual fires. The plant boundary and partitioning task establishes these analysis areas by dividing the global plant analysis boundary into discrete physical analysis units (PAU) or fire compartments. A fire compartment is a well-defined volume within the plant that is expected to substantially contain the effects of fire within the compartment. This volume is typically considered to be a room or clearly distinguishable area of the plant that is separated from other plant areas by substantial construction or other features that would contain the damaging effects of a fire within the compartment. Almost all fire compartments are completely enclosed by 3-hour fire barriers (or equivalent); however, some fire compartments have one or more

#2 –F120-AGAC Unsuppressed Fire in Aux. Bldg. 120' EL. Gen. Access Area "C" R_434-8352(92R)

These fires initiate in the quadrant "C" general access area of the 120 ft elevation of the Aux Bldg. These fires are dominated by spread to F120-AGAD resulting in partial loss of CCW event. Operator failure to trip the RCPs prior to seal damage leads straight to core damage and large ~~early~~ release due to fire damage to equipment in both divisions necessary to mitigate the seal LOCA. The loss of both trains also results in a failure to isolate either train of containment isolation, resulting in containment isolation failure. Fire spread between F120-AGAC and F120-AGAD contribute about 23% to LRF. Punch List #21

#3 - %F120-AGAD-U Unsuppressed Fire in Aux. Bldg. 120' EL. Gen. Access Area "D"

These fires initiate in the quadrant "D" general access area of the 120 ft elevation of the Aux Bldg. These fires are dominated by spread to F120-AGAC resulting in partial loss of CCW event. Operator failure to trip the RCPs prior to seal damage leads straight to core damage and large ~~early~~ release due to fire damage to equipment in both divisions necessary to mitigate the seal LOCA. The loss of both trains also results in a failure to isolate either train of containment isolation, resulting in containment isolation failure. Fire spread between F120-AGAC and F120-AGAD contribute about 23% to LRF. Punch List #21

#~~2~~4 -F157-AMCR-3-4 Safety Console Fire, Supp. Fails, ASD

Scenario F157-AMCR-3-4 involves an unsuppressed safety console (PM05) fire in fire compartment F157-AMCR, the MCR. The MCR analysis assumes the operators have approximately 10 minutes to extinguish the fire before visual obscuration results in the need to evacuate the MCR and perform shutdown from the RSC. The 10-minute time frame is based on the estimated time to reach about 70 percent peak heat release rate for cabinet fires, which have a 12-minute growth period with a t^2 growth profile, based on Appendix G of NUREG/CR-6850 and a review of room effects testing published in NUREG/CR-4527. An estimated CLRP of 0.1 is assumed for ASD from the RSC. Note that due to the lack of fire PRA-credited equipment in the MCR, and use of fiber-optic cable for almost all MCR controls, the resulting initiator is likely a simple transient as no PRA-credited equipment is directly damaged by the fire, and spurious operations resulting in more complicated initiators are unlikely. As decribed previously for the first cutset, a conditional probability of large release (given core damage) of 0.1 is assumed (CLRP of 0.01 given the fire initiating event). R_434-8352(92R)

b. The non-safety related systems, structures, and components (SSCs) are designed such that they will not collapse on or impact the seismic Category I structures containing SSCs (item ~~1~~a above) and will not generate missiles more damaging than the DBT and DBH missiles. R_270-7894(14R2)

19.1.5.4.2 Analysis

The external hazard probabilistic risk assessment (PRA) methodology for currently-operating plants is described in a number of references. Examples include References 2, 4, and 40. The major elements of an external hazard PRA are:

a. Initial screening of external hazards based on a set of qualitative criteria,

b. Bounding analysis for the unscreened~~screened-in~~ external hazards, Punch List #20

c. Detailed PRA for the remaining external hazards.

The initial screening of external hazards is done by first enumerating all potential external hazards that may impact the plant and screen them out using a set of criteria based on magnitude, distance, frequency and severity of the hazard. For an existing plant, these hazards will have been studied during the site selection and plant design. For example, the following USNRC regulatory guides and standard review plan sections provide acceptable criteria for excluding certain external hazards from the design basis of the plant:

a. RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" (Reference 59)

b. RG 1.91, "Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants" (Reference 60)

c. RG 1.115, "Protection Against Low-Trajectory Turbine Missiles" (Reference 61)

d. SRP Section 3.5.1.6, "Aircraft Hazards" (Reference 62)

For the unscreened~~screened-in~~ external hazards, a bounding or demonstrably conservative analysis is done to show either 1) the mean value of the frequency of the design-basis hazard used in the plant design is less than 10^{-5}yr and the conditional core damage probability is less than 10^{-1}, given the occurrence of the design-basis hazard event or 2) the core damage frequency (CDF) from the external hazard is less than 10^{-6}/yr. Punch List #20

For the remaining external hazard, a detailed PRA is performed; the major elements of such a PRA are:

a. Probabilistic external hazard analysis to develop a hazard curve depicting the annual frequency of exceeding different hazard intensities; the uncertainties in the data and model are propagated through the hazard analysis to derive a family of hazard curves with associated weights (or subjective probabilities)

b. Fragility analysis to identify the SSCs that are susceptible to the effects of the external hazard and to determine the plant-specific failure probabilities as a function of the intensity of the hazard.

c. External hazard plant response model to (a) develop a plant response model by modifying the internal events at-power PRA model to include the effects of the external hazard in terms of initiating events and failures caused; (b) quantify this model to provide the conditional core damage probability (CCDP) and conditional large ~~early~~ release probability (CLRP) for each defined external hazard plant damage state and (c) evaluate the unconditional CDF and LRF by integrating the CCDP/CLRP with the frequencies of the plant damage states obtained by combining the results of hazard analysis and fragility analysis. Punch List #21

~~d. While this methodology is equally applicable to new reactors under design certification, some key differences do exist.~~ R_270-7894(14R2)

~~e~~d. Most of the external hazards are location specific. Since the plant site is not known, initial screening of the external hazards cannot be performed. However, the combined operating license (COL) applicant is expected to select the site that meets the enveloping site parameters (DCD Table 2.0-1) and conforms to the NRC regulatory guides and SRP. This practice will provide the basis for screening of many external hazards.

~~f~~e. The plant design will not have progressed to the extent that plant-specific fragilities of SSCs could be evaluated. Therefore bounding or demonstrably conservative analysis of many unscreened~~screened-in~~ external hazards cannot be meaningfully done. R_270-7894(14R2) Punch List #20

The specific guidance in RG 1.76, RG 1.91 and SRP Section 3.5.1.6 are followed in this evaluation.

After an APR1400 units is built at a site, the COL holder should complete the external hazard PRA conforming to RG 1.206; the screening of external hazards such as transportation accidents should be documented.

b. Turbine Missile: The turbine generator layout for the APR1400 is considered to be a favorable orientation and excludes SSCs from low-trajectory turbine missile strikes. This conforms to RG 1.115. DCD Section 3.5.1.3 states that the probability of unacceptable damage resulting from turbine failure is less than 1×10^{-7} per year. Therefore, turbine missile is not considered as a design-basis event. The turbine missile hazard is also screened out in the PRA because of the low probability.

19.1.5.4.4 Bounding Analysis for unscreened~~Screened in~~ External Hazards

Punch List #20

Based on previous external hazard PRAs, the following hazards may not be screened using initial screening:

a. Extreme Winds and Tornadoes

b. External Flooding

19.1.5.4.4.1 Extreme Winds and Tornadoes

This group of external hazards includes tornadoes, hurricanes and thunderstorms. All potential sites for APR1400 are exposed to tornadoes and thunderstorms; coastal sites are in addition exposed to hurricanes.

DCD Section 3.3 describes the design basis for tornadoes and hurricanes. The design basis tornado (DBT) is selected as corresponding to Region I of RG 1.76. The maximum windspeed of the DBT is 230 mph and corresponds to a probability of exceedance of 10^{-7} per year. The design basis hurricane (DBH) is selected as having a maximum windspeed of 260 mph and corresponds to a probability of exceedance of 10^{-7} per year at most US coastal sites (except Southern Florida) as specified in RG 1.221. The standard plant structures (i.e., containment building, containment building internal structures, auxiliary building and emergency generator building and diesel oil fuel tank) will be designed to the

19.1.6.1.1 Description of Level 1 Internal Events PRA for Low Power and Shutdown Operations

19.1.6.1.1.1 Methodology

The scope of this analysis included quantitative evaluation of internal events for the LPSD operations. The development of the LPSD PRA includes the following nine major technical tasks.

a. Plant Operating State Development

b. Initiating Events Analysis

c. Accident Sequence Analysis

d. Success Criteria Analysis

e. Systems Analysis

f. Data Analysis

g. Human Reliability Analysis

h. Analysis of Large ~~Early~~ Release Punch List #21

i. Quantification

The draft ANS/ASME LPSD PRA Standard (Reference 9) is used as a guideline for the requirements for these technical tasks. Although the LPSD PRA Standard is still in draft form and has not been endorsed by the NRC, it still provides the best available guideline for identifying potential shutdown concerns. The draft standard was developed to support the analysis of operating reactors and, therefore, some of the specific requirements cannot be met in a design phase PRA. For example, the draft standard requires the review and incorporation of plant-specific operating experience into the PRA, interviews with plant operations and other personnel, plant walkdowns, etc., which cannot be performed for a plant in the design stage.

event to release of radionuclides to the environment. The at-power Level 2 release category evaluation was described in Subsection 19.1.4.2.1.3, and presents the Level 2 definitions of large, small, early and late releases. The LPSD Level 2 evaluation uses the same definitions.

The at-power evaluation delineated 21 release categories. The LPSD release category evaluation is less detailed. The more simplified and conservative LPSD release category evaluation created four release categories. They are:

RC-1-LPSD – The containment is intact and there are no significant releases. Consistent with the at-power Level 2 release categories 9 and 10, releases are modeled as the containment design leakage rate.

RC-2-LPSD – The containment ruptures with large and early releases. The main contributors to this category are containment bypass, CET sequence 31, or failure to close the containment equipment hatch (POS 3B/4A, not shown in the CET diagram). Other contributors are early hydrogen detonation or burn, or ex-vessel steam explosion.

RC-3-LPSD – The containment ruptures with large and late releases. The main contributors are steam overpressurization (CHR failure) and late hydrogen detonation or burn.

RC-4-LPSD – This category represents those sequences in which the containment fails late due to basemat melt-through. In this category, there are significant corium-concrete interaction (CCI) and concrete erosion after RV failure. Since the containment failure occurs below the containment basemat, there is a very small release of airborne fission products to the environment, and the release characteristics of this category are expected to be as an underground water release. The releases of this category are late and small.

The large release frequency (LRF) is the combined frequencies of RC-2-LPSD and RC-3-LPSD.

The summary of the MAAP results (release magnitude and timing) and release categorization (i.e., large release, ~~large early release,~~ or not large release) is presented in Table 19.1-133. Punch List #21

b. For the purposes of identifying the plant-level HCLPF in the LPSD SMA, Mode 5 is used in the quantification. POS in Mode 5 consider the largest set of SSCs in mitigation and the set of SSCs considered is bery similar to that meeded to mitigate core damage or release in other POS and, therefore, is considered representative of the equipment needed to determine the plant-level HCLPF. During Mode 6, two conditions could occur that are significantly different than Mode 5. The first is when there is a high water level in the reactor cavity which is expected during the majority of time spent in Mode 6. With this volume of water, it would take a very long time to heat up and boil off enough water to result in loss of core cooling and subsequent core damage thereby presenting less of a challenge to plant equipment and potentially masking components that could be important to determining the plant-level HCLPF. The second condition that is different from Mode 5 is when nozzle dams are installed. As a result, seismic-related insights from quantifying the SMA for Mode 5 are considered representative for all LPSD POS. R_232-7864(10)

c. The large, medium, and small LOCA scenarios are not considered controlling LOCA events for Mode 5 operation. Seismic non-recoverable LOCA (S-NRLOCA) is considered the controlling LOCA accident in Mode 5. A non-recoverable LOCA is the chemical and volume control system (CVCS) letdown line initiating event which represents a loss of RCS inventory that cause an interruption of cooling from the operating SCS train. Any LOCA originating elsewhere in the NSSS would be a lesser LOCA event similar in response and covered by the S-NRLOCA event tree.

d. The containment hatch is assumed to be open for all scenarios in plant operating states (POS) 3B & 4A and closed for all scenarios in POS 4B – 12A. Since the containment hatch is open for all scenarios in POS 3B & 4A all core damage is assumed to go directly to Large ~~Early~~ Release (L~~E~~R). No level 2 modeling is necessary for POS 3B & 4A level 2 failure since L~~E~~R is assumed and direct. Punch List #21

e. Seismically-induced failure of nozzle dams may affect the timing of accident response for the LPSD SMA. However, the set of equipment needed to respond to such an event will remain unchanged regardless of whether or not the nozzle dams remain installed. That is, the sequence may change from a loss of shutdown cooling to a non-recoverable LOCA. However, because both types of events are evaluated in the LPSD SMA, the plant-level HCLPF will be correctly

g. Failure of the containment exterior is the seventh most significant contributor to core damage. This is basic event SEIS-CTS-EX-FAIL and is assumed to result directly in core damage. The HCLPF for this failure is 0.94g. R_232-7864(10)

h. Failure of the containment interior is the eighth most significant contributor to core damage. This is basic event SEIS-CTS-IN-FAIL and is assumed to result directly in core damage. The HCLPF for this failure is 1.09.

i. Sustained seismic loss of offsite power combined with operator failure to restore SCS is the ninth highest contributor to CDF. The HCLPF for seismic loss of offsite power is 0.09g.

19.1.6.5.2.3.2 Level 2 Risk Insights

The direct to core damage seismic failures were considered to proceed directly from core damage to Large ~~Early~~ Release. This includes seismic failure of: Punch List #21

- compound building
- Turbine building
- Auxiliary building
- Nuclear Island failure
- Reactor Containment building
- Reactor Containment Internals
- Containment Hatch (Open & Stowed) for POS 3B & 4A only
- Reactor Vessel Head (Removed and Stowed) for POS 4B through 12A only

Seismic failure resulting in station blackout and resulting in core damage are also assumed to result in L~~E~~RF. This would include seismic failures of the following equipment that result in core damage: Punch List #21

- Emergency Diesel Generator Building
- Diesel Fuel Oil Tank Building

- Emergency Diesel Generators R_232-7864(10)

In addition, all S-NRLOCA sequences resulting in core damage are considered to proceed directly to core damage as discussed in the assumptions. Also, core damage resulting from seismic loss of offsite power in POS 3B and 4A are also considered to proceed directly to core damage.

The initiating events that required additional modeling for level 2 analysis included seismic failure of I&C for all POS and seismic loss of offsite power for POS 4B through 12A. The only significant failure for the level 2 modeling was the correlated failure of all equipment other than the NSSS. This failure is based on an assumption that the design is for the licensee to develop and that the HCLPF will have a value equal to or exceeding 1.67 times CSDRS.

The top contributors for large ~~early~~ release are the same as the top contributors for core damage: Punch List #21

a. All the seismic failures resulting in core damage that are assumed to proceed directly to large ~~early~~ release are included in the tops contributors for core damage. Punch List #21

b. The only significant seismic failure from the modeling of level 2, correlated failure of all equipment other than the NSSS, is also a top contributor to L~~E~~RF. Punch List #21

These two considerations together cover all of the top contributors to core damage. Therefore, the list of top contributors to Level 2 failures must be the same the top contributors to core damage.

19.1.7 PRA-Related Input to Other Programs and Processes

The APR1400 is expected to perform better than current operating plants in the area of severe accident safety performance, since prevention and mitigation of severe accidents, as shown in Table 19.1-2 and Table 19.1-3, have been addressed during the design stage, taking advantage of PRA results and severe accident analysis. The PRA results indicate that the APR1400 design has a low level of risk and meets the CDF, LRF, and containment performance goals for new-generation PWRs.

the aforementioned configuration control program, a provision that if the flood or fire doors to this designated quadrant must be opened for reasons other than normal ingress/egress, a flood or fire watch must be established for the affected doors. See subsection 19.1.6.3.2.5.

COL 19.1(18) The COL applicant is to describe the uses of PRA in support of licensee programs such as Maintenance Rule implementation during the operational phase. See Subsection 19.1.7.2.

COL 19.1(19) The COL applicant is to describe the uses of PRA in support of licensee programs such as the reactor oversight process during the operational phase. See Subsection 19.1.7.3.

COL 19.1(20) The COL ~~applicant~~holder is to perform the seismic-fire interactions walkdown to confirm a qualitative seismic-fire interaction assessment. Punch List #10

COL 19.1(21) The COL applicant and/or holder is to develop outage procedures to ensure that in fire compartments containing post-seismic or post-fire safe shutdown equipment that: 1) the seismic ruggedness of temporary ignition sources is adequate, or that the duration that these temporary ignition sources are in these areas is minimized, 2) the seismic ruggedness of temporary equipment such as scaffolding in fire compartments containing potential seismic-fire ignition sources, or near fire protection equipment is adequate, and 3) either the duration of activities which could impact manual firefighting is minimized, or alternative firefighting equipment (e.g., pre-stage portable smoke removal equipment, prestage additional firefighting equipment, etc.) is supplied. A_19-51R

COL 19.1(22) The COL applicant and/or holder is to demonstrate that failure of buildings that are not seismic Category I (e.g., turbine building and compound building) does not impact SSCs designed to be seismic Category I. A_19-51R

COL 19.1(23) The COL applicant and/or holder is to ensure that asymmetric conditions due to modeling simplicity will be addressed or properly accounted for when the PRA is used for decision making. A_19-51R

COL 19.1(24) The COL ~~applicant~~holder will demonstrate that maintenance-induced floods are negligible contributors to flood risk when the plant specific data are available. A_19-51R

COL 19.1(25) SAMGs are entered to initiate SI with the core exit thermocouple indicating 1200°F. R_409-8325(28R4)

COL 19.1(26) The COL applicant and/or holder ensures that the fire protection features required for preventing fire-induced damage of the PRA-credited components will be properly incorporated in the cable design. A_19-51R Punch List #11

19.1.10 References

1. ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Revision 1 RA-S-2002), American Society of Mechanical Engineers, April 2008.

2. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008," American Society of Mechanical Engineers, February 2009.

3. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Rev. 2, U.S. Nuclear Regulatory Commission, March 2009.

4. NUREG/CR-2300, "PRA Procedures Guide," U.S. Nuclear Regulatory Commission, January 1983.

5. NUREG/CR-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, December 1990.

6. NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," U.S. Nuclear Regulatory Commission, September 2005.

7. APR1400-E-P-NR-14001-P, "PRA Summary Report," Rev. 0, KHNP.

8. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," U.S. Nuclear Regulatory Commission, Letter issued April 2, 1993 and Staff Requirements Memoranda issued July 21, 1993.

Table 19.1-4 (23 of 27)

No.	Insight	Disposition
Risk Insights from PRA Models		
55	Passive autocatalytic recombiners (PARs) and igniters are normally placed to cope with at-power accidents, but hydrogen control should also be considered for LPSD configurations. The COL applicant should provide reasonable assurance that there is sufficient hydrogen control during a severe accident condition when the RCS is open (e.g., pressurizer manway, etc.).	 COL 13.5(7)
56	Solid state switching devices and electro-mechanical relays resistant to relay chatter are used in the safety I&C platforms. Use of these devices and relays either eliminates or minimizes the mechanical discontinuities associated with mechanical relays at operating reactors.	Subsection 7.4.2.5
57	The COL applicant is to perform a seismic walkdown to provide reasonable assurance that the as-designed and as-built plant conforms to the assumptions in the PRA-based seismic margins analysis and that seismic spatial systems interactions do not exist. Details of the seismic walkdown are to be developed by the COL applicant.	COL 19.1(4)
58	The fire PRA assumes that the fire barrier management procedures used during LPSD will include directions to provide reasonable assurance that breached risk-significant fire barriers can be closed in sufficient time to prevent the spread of fire across the barrier. The procedural direction is to include the use of a fire watch whose duties are commensurate with the risk associated with the barrier. For example, for fire barriers that separate two fire compartments that both contain no equipment or cables necessary to prevent core damage or large ~~early~~ release during LPSD conditions, or have been demonstrated to have low risk significance, there will at least be a roving fire watch to check the barrier during rounds. For fire barriers separating fire compartments that contain equipment or cables necessary to prevent core damage or large ~~early~~ release during LPSD conditions, and have been demonstrated to be risk significant with respect to fire, a permanent fire watch will be established until the barrier is reclosed. In the latter case, the fire barrier management procedure is to direct that hoses or cables that pass through a fire barrier use isolation devices on both sides of a quick-disconnect mechanism that allow for reclosure of the barrier in a timely fashion to re-establish the barrier prior to fire spread across the barrier.	Subsection 19.1.6.3.1.2 COL 19.1(15) COL 19.1(14)

Punch List #21

Punch List #21

R_434-8352(92R)

Table 19.1-48 (1 of 3)

Internal Fire PRA LRF Contribution by Top Fire Induced Initiators

Punch List #19

Rank	Fire IE	~~CDF~~LRF	%~~CDF~~LRF	%Cum ~~CDF~~LRF	Description
1	%F157-AMCR-4-4	2.36E-08	12.71%	12.7%	FIRE IN F157-AMCR - TRANSIENT FIRE - UNSUPPRESSED - ASD
2	%F120-AGAC	2.33E-08	12.54%	25.2%	FIRE IN F120-AGAC - GENERAL ACCESS AREA-120' C
3	%F120-AGAD-U	2.01E-08	10.85%	36.1%	FIRE IN F120-AGAD - GENERAL ACCESS AREA-120' D - UNSUPPRESSED
4	%F157-AMCR-3-4	1.51E-08	8.17%	44.3%	FIRE IN F157-AMCR - SAFETY CONSOLE FIRE - UNSUPPRESSED - ASD
5	%F078-AEEB-U	7.95E-09	4.29%	48.5%	FIRE IN F078-AEEB - CLASS 1E SWITCHGEAR 01B ROOM - UNSUPPRESSED
6	%F000-TB-LOOP2	7.95E-09	4.28%	52.8%	FIRE IN F000-TB-LOOP2 - TB FIRES LEADING TO LOOP (SEVERE)
7	%F157-AMCR-1-4	7.57E-09	4.08%	56.9%	FIRE IN F157-AMCR - CCTV FIRE - UNSUPPRESSED - ASD
8	%F157-AMCR-2-4	7.57E-09	4.08%	61.0%	FIRE IN F157-AMCR - FIRE CONTROL PANEL FIRE - UNSUPPRESSED - ASD
9	%F078-A19B-U	7.41E-09	3.99%	65.0%	FIRE IN F078-A19B - CORRIDOR - UNSUPPRESSED
10	%F000-ADGC	4.68E-09	2.52%	67.5%	FIRE IN F000-ADGC - DG01C DIESEL GENERATOR ROOM
11	%F157-AMCR-5-4	4.04E-09	2.18%	69.7%	FIRE IN F157-AMCR - TRANSIENT W/C FIRE - UNSUPPRESSED - ASD
12	%F122-T01-U	3.58E-09	1.93%	71.6%	FIRE IN F122-T01-U - F122-T01 Unsuppressed Fires
13	%F157-AMCR-6-4	3.43E-09	1.85%	73.5%	FIRE IN F157-AMCR - CABLE W/C FIRE - UNSUPPRESSED - ASD

R_434-8352(92R)

Punch List #19

Table 19.1-48 (2 of 3)

Rank	Fire IE	~~CDF~~LRF	%~~CDF~~LRF	%Cum ~~CDF~~LRF	Description
14	%F120-A11B-U	3.10E-09	1.67%	75.1%	FIRE IN F120-A11B - GENERAL ACCESS AREA-120' B - UNSUPPRESSED
15	%F000-AFHL	3.00E-09	1.62%	76.8%	FIRE IN F000-AFHL - FUEL HANDLING LOWER AREA
16	%F000-ADGD-U	2.53E-09	1.36%	78.1%	FIRE IN F000-ADGD - DG01D ROOM - UNSUPPRESSED FIRES
17	%F100-T15-U	2.41E-09	1.30%	79.4%	FIRE IN F100-T15 - SWITCHGEAR RM - UNSUPPRESSED
18	%F073-T11	2.39E-09	1.29%	80.7%	FIRE IN F073-T11 - SWITCHGEAR AREA
19	%F100-AEEB-U	2.29E-09	1.23%	81.9%	FIRE IN F100-AEEB - 480V CLASS 1E MCC 01B RM - UNSUPPRESSED
20	%F100-A08C-U	1.80E-09	0.97%	82.9%	FIRE IN F100-A08C - N1E DC & IP EQUIPMENT RM C - UNSUPPRESSED
21	%F120-A09D	1.76E-09	0.95%	83.9%	FIRE IN F120-A09D - ELECTRICAL PENETRATION ROOM D
22	%F078-AGAC-U	1.72E-09	0.93%	84.8%	FIRE IN F078-AGAC - GENERAL ACCESS AREA-78' C - UNSUPPRESSED
23	%F067-T02-U	1.46E-09	0.79%	85.6%	FIRE IN F067-T02 - UNDERGROUND COMMON TUNNEL - UNSUPPRESSED
24	%F157-ACPX-U	1.42E-09	0.77%	86.3%	FIRE IN F157-ACPX - COMPUTER ROOM - UNSUPPRESSED
25	%F078-A52D-U	1.39E-09	0.75%	87.1%	FIRE IN F078-A52D - 480V N1E MCC RM - UNSUPPRESSED

Table 19.1-48 (3 of 3)

R_434-8352(92R)

Punch List #19

Rank	Fire IE	~~CDF~~LRF	%~~CDF~~LRF	%Cum ~~CDF~~LRF	Description
26	%F157-A01D-U	1.25E-09	0.68%	87.8%	FIRE IN F157-A01D - I & C EQUIP. RM - UNSUPPRESSED
27	%F120-A15B-U	1.25E-09	0.67%	88.4%	FIRE IN F120-A15B - 480V CLASS 1E MCC 03B RM - UNSUPPRESSED
28	%F078-A19A	1.21E-09	0.65%	89.1%	FIRE IN F078-A19A - CORRIDOR
29	%F137-A02D	1.02E-09	0.55%	89.6%	FIRE IN F137-A02D - ELECTRICAL EQUIP. RM
30	%F120-A01D	9.31E-10	0.50%	90.1%	FIRE IN F120-A01D - PIPING CABLE AREA

Table 19.1-50 (1 of 46)

R_434-8352(92R)

Internal Fire PRA Top 100 LRF Cutsets

Rank	Cutset Prob.	Cum. %	BE Prob	Event	Event Description
1	2.36E-08	12.7%	2.36E-06	%F157-AMCR-4-4	FIRE IN F157-AMCR - TRANSIENT FIRE - UNSUPPRESSED - ASD
			1.00E-01	ASD-CDF	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE)
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
2	1.94E-08	23.1%	2.86E-04	%F120-AGAC	FIRE IN F120-AGAC - GENERAL ACCESS AREA-120' C
			5.63E-02	RCOPH-S-RCPTRIP	Operator Fails to Trip RCPs Following Loss of Seal Cooling
			1.20E-03	BF_F120-AGAC_F120-AGAD	BARRIER FAILURE BETWEEN FIRE COMPS F120-AGAC & F120-AGAD
3	1.64E-08	32.0%	2.42E-04	%F120-AGAD-U	FIRE IN F120-AGAD - GENERAL ACCESS AREA-120' D - UNSUPPRESSED
			5.63E-02	RCOPH-S-RCPTRIP	Operator Fails to Trip RCPs Following Loss of Seal Cooling
			1.20E-03	BF_F120-AGAC_F120-AGAD	BARRIER FAILURE BETWEEN FIRE COMPS F120-AGAC & F120-AGAD
4	1.51E-08	40.1%	1.51E-06	%F157-AMCR-3-4	FIRE IN F157-AMCR - SAFETY CONSOLE FIRE - UNSUPPRESSED - ASD
			1.00E-01	ASD-CDF	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE)
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA

Punch List #21

Punch List #21

Table 19.1-50 (2 of 46)

R_434-8352(92R)

Rank	Cutset Prob.	Cum. %	BE Prob	Event	Event Description
5	7.57E-09	44.2%	7.57E-07	%F157-AMCR-1-4	FIRE IN F157-AMCR - CCTV FIRE - UNSUPPRESSED - ASD
			1.00E-01	ASD-CDF	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE)
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
6	7.57E-09	48.3%	7.57E-07	%F157-AMCR-2-4	FIRE IN F157-AMCR - FIRE CONTROL PANEL FIRE - UNSUPPRESSED - ASD
			1.00E-01	ASD-CDF	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE)
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
7	4.04E-09	50.5%	4.04E-07	%F157-AMCR-5-4	FIRE IN F157-AMCR - TRANSIENT W/C FIRE - UNSUPPRESSED - ASD
			1.00E-01	ASD-CDF	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE)
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
8	3.43E-09	52.3%	3.43E-07	%F157-AMCR-6-4	FIRE IN F157-AMCR - CABLE W/C FIRE - UNSUPPRESSED - ASD
			1.00E-01	ASD-CDF	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE)
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
9	1.63E-09	53.2%	1.36E-04	%F078-AEEB-U	FIRE IN F078-AEEB - CLASS 1E SWITCHGEAR 01B ROOM - UNSUPPRESSED
			1.20E-05	PPSO-AP-LC	CCF OF PPS LC APPLICATION SOFTWARE

Punch List #21

Punch List #21

Punch List #21

Punch List #21

Table 19.1-50 (3 of 46)

R_434-8352(92R)

Rank	Cutset Prob.	Cum. %	BE Prob	Event	Event Description
10	8.05E-10	53.7%	9.36E-05	_%F157-ACPX-U	FIRE IN F157-ACPX - COMPUTER ROOM - UNSUPPRESSED
			1.00E-02	_ASD-CDF-MCA	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE) - MC EVENT
			1.00E-01	_ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
			8.60E-03	_BF_F157-ACPX_F157-AMCR	BARRIER FAILURE BETWEEN FIRE COMPS F157-ACPX & F157-AMCR
11	7.00E-10	54.0%	2.86E-04	_%F120-AGAC	FIRE IN F120-AGAC - GENERAL ACCESS AREA-120' C
			5.78E-02	_AFMVC1B-046	AFW ISOL. MOV V046 FAILS TO CLOSE
			3.52E-02	_AFTPR1A-TDP01A	AFW TDP PP01A FAILS TO RUN FOR > 1HR
			1.20E-03	_BF_F120-AGAC_F120-AGAD	BARRIER FAILURE BETWEEN FIRE COMPS F120-AGAC & F120-AGAD
12	7.00E-10	54.4%	2.86E-04	_%F120-AGAC	FIRE IN F120-AGAC - GENERAL ACCESS AREA-120' C
			5.78E-02	_AFMVO1B-046	AFW ISOL. MOV V046 FAILS TO OPEN
			3.52E-02	_AFTPR1A-TDP01A	AFW TDP PP01A FAILS TO RUN FOR > 1HR
			1.20E-03	_BF_F120-AGAC_F120-AGAD	BARRIER FAILURE BETWEEN FIRE COMPS F120-AGAC & F120-AGAD
13	6.63E-10	54.8%	5.52E-05	_%F120-A11B-U	FIRE IN F120-A11B - GENERAL ACCESS AREA-120' B - UNSUPPRESSED
			1.20E-05	_PPSO-AP-LC	CCF OF PPS LC APPLICATION SOFTWARE

Punch List #21

Table 19.1-50 (6 of 46)

R_434-8352(92R)

Rank	Cutset Prob.	Cum. %	BE Prob	Event	Event Description
19	4.61E-10	56.4%	3.52E-04	%F000-TB-LOOP2	FIRE IN F000-TB-LOOP2 - TB FIRES LEADING TO LOOP (SEVERE)
			1.00E-01	L2-PROB-CSRECSBS-NO	MELTSTOP DET: CSRECSBS = NO
			5.00E-01	L2-PROB-DCOOL-YES	DBCOOL DET: DCOOL = YES
			9.99E-01	L2-PROB-ECFEVSE-INT-A	ECF DET: ECFEVSE = INT (CASE A)
			5.23E-01	L2-PROB-LCF-WOB-RUPT	DET LCF: LCFWOB = RUPTURE
			9.00E-01	L2-PROB-RCSFAIL-HLFAIL	RCSFAIL DET - prob that HLFAIL = HLFAIL
			5.57E-05	SXFLP-S-FT0123AB	CCF OF ALL ESW DERIS FILTERS DUE TO PLUGGING
20	4.37E-10	56.7%	3.64E-04	%F137-A02D	FIRE IN F137-A02D - ELECTRICAL EQUIP. RM
			1.00E-02	ASD-CDF-MCA	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE) - MC EVENT
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
			1.20E-03	BF_F137-A02D_F157-AMCR	BARRIER FAILURE BETWEEN FIRE COMPS F137-A02D & F157-AMCR

Punch List #21

Table 19.1-50 (10 of 46)

R_434-8352(92R)

Rank	Cutset Prob.	Cum. %	BE Prob	Event	Event Description
27	2.48E-10	58.0%	3.08E-02	%F000-TB-GTRN	FIRE IN F000-TB-GTR - TB FIRES LEADING TO GTRN
			2.98E-07	I-ATWS-RPMCF	CCF TO SCRAM DUE TO MECHANICAL FAILURES (1HR MT)
			2.70E-02	PI-SGTR	PRESSURE INDUECD SGTR PROBABILITY UNDER LSSB, ATWS, FWLB
28	2.37E-10	58.1%	1.36E-04	%F078-AEEB-U	FIRE IN F078-AEEB - CLASS 1E SWITCHGEAR 01B ROOM - UNSUPPRESSED
			2.50E-02	DGDGR-A-DGA	FAILS TO RUN EMERGENCY DIESEL GENERATOR DG01A
			1.75E-03	NPXHM-M-SAT02M	SAT TR02M UNAVAILABLE DUE TO T&M
			4.00E-02	WOCHM2A-CH02A	ECW CHILLER 02A TRAIN UNAVAILABLE DUE TO T&M
29	2.37E-10	58.3%	2.86E-04	%F120-AGAC	FIRE IN F120-AGAC - GENERAL ACCESS AREA-120' C
			6.89E-04	AFTPKD2-TDP01A/B	2/2 CCF OF AFW TDP PP01/A/B FAILS TO RUN > 1 HR
			1.20E-03	BF_F120-AGAC_F120-AGAD	BARRIER FAILURE BETWEEN FIRE COMPS F120-AGAC & F120-AGAD
30	2.04E-10	58.4%	2.37E-05	%F137-ASTD	FIRE IN F137-ASTD - STAIR
			1.00E-02	ASD-CDF-MCA	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE) - MC EVENT
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
			8.60E-03	BF_F137-ASTD_F157-AMCR	BARRIER FAILURE BETWEEN FIRE COMPS F137-ASTD & F157-AMCR

Punch List #21

Table 19.1-50 (11 of 46)

R_434-8352(92R)

Rank	Cutset Prob.	Cum. %	BE Prob	Event	Event Description
31	2.00E-10	58.5%	2.42E-04	%F120-AGAD-U	FIRE IN F120-AGAD - GENERAL ACCESS AREA-120' D - UNSUPPRESSED
			6.89E-04	AFTPKD2-TDP01A/B	2/2 CCF OF AFW TDP PP01/A/B FAILS TO RUN > 1 HR
			1.20E-03	BF_F120-AGAC_F120-AGAD	BARRIER FAILURE BETWEEN FIRE COMPS F120-AGAC & F120-AGAD
32	2.00E-10	58.6%	1.75E-03	%F000-AFHL	FIRE IN F000-AFHL - FUEL HANDLING LOWER AREA
			1.00E-01	L2-PROB-RCSFAIL-ALLSG-DEP	RCSFAIL DET - prob that SGDEPRESS = ALL-DEPRESS
			8.93E-01	L2-PROB-RCSFAIL-NO-SGTR-2D	RCSFAIL DET - prob that PI-SGTR = INTACT (2 SGs depressurized)
			8.89E-01	L2-PROB-RCSFAIL-NOSORV	RCSFAIL DET - probability that RCSSORV = Intact
			5.00E-01	L2-PROB-RCSFAIL-RCSINT	RCSFAIL DET - prob that CSSORV_LATE = RCS-INTACT
			2.40E-01	L2-PROB-TISGTR-C	RCSFAIL DET - prob that TI-SGTR = TI-SGTR (case C)
			1.20E-05	PPSO-AP-LC	CCF OF PPS LC APPLICATION SOFTWARE
33	1.86E-10	58.7%	1.55E-04	%F137-A05D	FIRE IN F137-A05D - PCS RM
			1.00E-02	ASD-CDF-MCA	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE) - MC EVENT
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
			1.20E-03	BF_F137-A05D_F157-AMCR	BARRIER FAILURE BETWEEN FIRE COMPS F137-A05D & F157-AMCR

Punch List #21

Table 19.1-50 (23 of 46)

R_434-8352(92R)

Rank	Cutset Prob.	Cum. %	BE Prob	Event	Event Description
51	1.37E-10	60.2%	1.36E-04	%F078-AEEB-U	FIRE IN F078-AEEB - CLASS 1E SWITCHGEAR 01B ROOM - UNSUPPRESSED
			1.44E-02	DGDGM-A-DGA	DG 01A UNAVAILABLE DUE TO T&M
			1.75E-03	NPXHM-M-SAT02M	SAT TR02M UNAVAILABLE DUE TO T&M
			4.00E-02	WOCHM2A-CH02A	ECW CHILLER 02A TRAIN UNAVAILABLE DUE TO T&M
52	1.30E-10	60.3%	1.51E-05	%F157-A17C	FIRE IN F157-A17C - CORRIDOR
			1.00E-02	ASD-CDF-MCA	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE) - MC EVENT
			1.00E-01	ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA
			8.60E-03	BF_F157-A17C_F157-AMCR	BARRIER FAILURE BETWEEN FIRE COMPS F157-A17C & F157-AMCR
53	1.29E-10	60.4%	2.86E-04	%F120-AGAC	FIRE IN F120-AGAC - GENERAL ACCESS AREA-120' C
			5.78E-02	AFMVC1B-046	AFW ISOL. MOV V046 FAILS TO CLOSE
			6.49E-03	AFTPS1A-TDP01A	AFW TDP PP01A FAILS TO START
			1.20E-03	BF_F120-AGAC_F120-AGAD	BARRIER FAILURE BETWEEN FIRE COMPS F120-AGAC & F120-AGAD

Punch List #21

Table 19.1-57 (3 of 5)

R_434-8352(92R)

Event	Description	RAW
PFHBC1A-SW01A-A2	PCB SW01A-A2 OF 4.16KV SWGR SW01A FAILS TO CLOSE	5
SIMVT-B-303	SI PUMP PP02B/D MINI. FLOW LINE MOV V303 FAILS TO REMAIN OPEN	5
PELXY-A-LX01A-P	FAILURE OF PRIMARY LOOP CONTROLLER LX01A	5
PALXY-C-PA06C-P	PRIMARY LOOP CONTROLLER PA06C FAILS TO RUN	5
VKHVR1A-HV13A	FAILS TO RUN CCW PUMP ROOM CUBICLE COOLER HV13A	4
RC-CSFP-NO-CBO-ISO	COND. FAILURE PROB. OF RCP SEALS GIVEN FAILURE TO ISOLATE CBO WITHIN 20 MIN.	4
CCMPR1A-PP01A	FAILS TO RUN CCW PUMP PP01A	4
PADOY-C-PA06C04	FAILURE OF DIGITAL OUTPUT MODULE PA06C BRANCH 04	4
PADOY-D-PA06C02	FAILURE OF DIGITAL OUTPUT MODULE PA06C BRANCH 02	4
PEDOY-C-LX03C01	FAILURE OF DIGITAL OUTPUT MODULE LX03C BRANCH 01	4
WOMPR1B-PP01B	FAILS TO RUN ECW PUMP 01B	4
PELXY-A-LX05A-P	FAILURE OF PRIMARY LOOP CONTROLLER 745-LX05A	4
ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA	4
ASD-CDF	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE)	4
PFBSY2A-SW01C	BUS FAULT ON 4.16KV SWGR SW01C	4
PGBSY2A-LC01C	BUS FAULT ON 480V LC LC01C	4
PEDOY-A-LX05A03	FAILURE OF DIGITAL OUTPUT MODULE LX05A BRANCH 03	4
PALXY-D-PA06D-P	PRIMARY LOOP CONTROLLER PA06D FAILS TO RUN	4
AFMPM2A-MDP02A	AFW MDP PP02A UNAVAILABLE DUE TO T/M	4
SICVO-B-V101	IRWST RETURN LINE TRAIN B CHECK VALVE V101 FAILS TO OPEN	4
PADOY-D-PA06D01	FAILURE OF DIGITAL OUTPUT MODULE PA06D BRANCH 01	3

Punch List #21

Table 19.1-58 (1 of 5) R_434-8352(92R)

Internal Fire PRA Key Basic Events by FV (LRF)

Event	Description	FV
ASD-L~~E~~RF	Conditional Prob of L~~E~~RF given CD for Alt shutdown in SCA	34.1%
ASD-CDF	FAILURE OF ALTERNATE SHUTDOWN AFTER MCR EVACUATION (CORE DAMAGE)	33.1%
BF_F120-AGAC_F120-AGAD	BARRIER FAILURE BETWEEN FIRE COMPS F120-AGAC & F120-AGAD	23.0%
L2-PROB-CSRECSBS-NO	MELTSTOP DET: CSRECSBS = NO	18.7%
L2-PROB-RCSFAIL-NOSORV	RCSFAIL DET - probability that RCSSORV = Intact	13.0%
L2-PROB-LCF-WOB-RUPT	DET LCF: LCFWOB = RUPTURE	11.4%
L2-PROB-RCSFAIL-NOSG-DEP	RCSFAIL DET - prob that SGDEPRESS = NO-DEPRESS	10.6%
L2-PROB-RCSFAIL-NOSGSORV	RCSFAIL DET - prob that SGSORV = NO-DEPRESS	10.3%
L2-PROB-ECFEVSE-INT-A	ECF DET: ECFEVSE = INT (CASE A)	9.4%
L2-PROB-RCSFAIL-RCSINT	RCSFAIL DET - prob that CSSORV_LATE = RCS-INTACT	9.2%
L2-PROB-RCSFAIL-HLFAIL	RCSFAIL DET - prob that HLFAIL = HLFAIL	8.4%
L2-PROB-RCSFAIL-1-2-LK	RCSFAIL DET - prob that SG-LEAK = 1OF2-LEAK	6.8%
L2-PROB-DCOOL-NO	DBCOOL DET: DCOOL = NO	5.7%
L2-PROB-DCOOL-YES	DBCOOL DET: DCOOL = YES	5.7%
AFTPR1A-TDP01A	AFW TDP PP01A FAILS TO RUN FOR > 1HR	4.9%

Punch List #21