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Subject: NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment," Section 16

Enclosure 1 contains the subject partial ESBWR Probabilistic Risk Assessment (PRA) document (Revision 1).

If you have any questions about the information provided here, please let me know.

Sincerely,

Kathy Sedney for

David H. Hinds Manager, ESBWR

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Enclosure:

- 1. **MFN** 06-196 NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment:"
	- \bullet Section 16 Shutdown Risk
- **cc:** WD Beckner USNRC (w/o enclosures) AE Cubbage USNRC (with enclosures) LA Dudes USNRC (w/o enclosures) GB StrambackGE/San Jose (with enclosures) eDRF 0000-0044-8542

ENCLOSURE 1

MFN 06-196

NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment"

• Section 16 – Shutdown Risk

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16 SHUTDOWN RISK

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16 SHUTDOWN RISK

16.1 INTRODUCTION

A detailed PRA is performed to determine the core damage frequency during shutdown. Loss of the Reactor Water Cleanup/Shutdown Cooling System, Loss of Reactor Component Cooling Water System, Loss of Plant Service Water System, and Loss of Preferred Power are all investigated. Additionally, the Core Damage Frequency (CDF) due to drain down of the RPV or Loss of Coolant Accidents (LOCAs) during shutdown is evaluated. Fault trees and event trees are used to determine the shutdown CDF for each event analyzed.

The evaluation encompasses plant operation in shutdown modes. This evaluation addresses conditions for which there is fuel in the RPV. It includes all aspects of the NSSS, the containment, and all systems that support operation of the NSSS and containment.

The scope of the Shutdown PRA is that of a Level 1 PRA. The different accident sequences are classified according to whether the core is damaged or not. The latter are further subdivided into two classes: those without release of radionuclides to the environment and those with release of radionuclides to the environment (either because the containment was open at the time of the initiating event or because the containment cooling failed).

The following subsections discuss the shutdown PRA modeling methodology, data sources, modeling assumptions, and the results of the data analyses for inclusion in the shutdown PRA model. Shutdown PRA analyses during external events are covered in individual sections on external events sections such as Sections 12 (Fire), 13 (Flood) and 14 (High Wind).

16.2 PLANT CONFIGURATIONS IN SHUTDOWN

The differences between the shutdown PRA and the power operation PRA are due to the following:

- **"** Plant operating mode
- **"** Time after shutdown
- RPV and containment status
- Water levels and temperatures
- Fuel location
- Availability of required systems

To develop a suitable shutdown model, multiple bounding plant configurations are defined with similar characteristics in relation to the residual heat, the availability of systems, and the RPV water levels.

The outage plant operating mode is used to define the initial plant condition for individual accident sequence quantification.

Once the outage plant configurations have been defined, the duration of each one is estimated to determine its contribution to the overall calculation of annual core damage frequency. The duration is expressed in hours per refueling outage.

16.2.1 Definition of Plant Shutdown Configurations

The shutdown PRA considers the following outage plant configurations as representative of the possible plant configurations during shutdown.

- Mode 4 (hot shutdown) included in full-power PRA
- **"** Mode **5** (cold shutdown)
- **"** Mode 6-Unflooded (refueling)
- Mode 6-Flooded (refueling)

Figure 16.2-1 displays the duration of the different outage plant configurations considered in the ESBWR Shutdown PRA. The following paragraphs describe each of these configurations, detailing the vessel pressure and temperature conditions, as well as the assumed duration and status of RWCU/SDCS and other decay heat removal systems.

16.2.1.1 Mode 4 -Hot Shutdown

This is the cooldown phase to bring the plant to cold shutdown. The reactor mode switch is in the shutdown position. It begins after control rod insertion is completed. Operation of the reactor mode switch from one position to another bypasses Reactor Protection System trips and channels and automatically alters Neutron Monitoring System trip setpoints in accordance with the reactor conditions implied by the given position of the mode switch (Reference DCD Chapter 7).

Decay and sensible heat are removed through the Main Condenser and/or Isolation Condenser. Approximately one-half hour after control rod insertion, both RWCU/SDCS trains are operating,

with the regenerative heat exchangers bypassed and pumps running at reduced speed to avoid exceeding the RCCWS design cold leg temperature. The control rod drive system is in service to provide makeup water for the reactor coolant contraction.

The duration of this mode is assumed to be 8 hours.

Containment is de-inerted but integrity is maintained during this mode.

The initial RPV pressure and temperature is the same as power operating values. Because the plant configuration during this period is very similar to that existing in full power, the CDF contribution of this mode is modeled as included in the full power PRA.

16.2.L2 Mode 5- Cold Shutdown (After Power Operation)

Cooldown of the reactor coolant is continued during this phase. The reactor mode switch is in the shutdown position. Once in cold shutdown, the heat removal requirements are transferred to the RWCU/SDCS. The Main Condenser and circulating water pumps are removed from service and the use of the isolation condensers is terminated.

Both RWCU/SDCS trains run in parallel, with regenerative heat exchangers bypassed and the pump speed gradually increasing up to the maximum flow rate.

The duration of this mode is assumed to be 88 hours.

Containment is opened at some time during this mode. The drywell head removal operations are expected to begin, and eventually be completed, in this configuration. However, because it is expected that containment will be intact during most of the time spent in this mode, the shutdown PRA assumes containment to be intact for Mode **5.**

Initial RPV conditions in this mode are a pressure of **0.75** MPa (109 psia) and a temperature of 168 **-C** (334 'F).

16.2.L3 Mode 6- Refueling (Unflooded)

As soon as the reactor coolant temperature reaches 49 **'C** (120 'F), reactor head removal operations may start. Prerequisites required to remove reactor head, such as reactor well drain or drywell head removal occur during the cooldown phase.

Decay heat removal is provided by the RWCU/SDCS. At the start of this mode, both trains are expected to be running. Later, only one is required to keep the reactor coolant temperature within limits.

The duration of this mode is assumed to be approximately **59** hours, including the period before refueling and the period after refueling.

In this configuration the reactor head is either removed or not fully tensioned, and the reactor well is not flooded.

RPV is at atmospheric pressure and the temperature is maintained at about 49 °C (120 *F). The reactor vessel is assumed to be open to the reactor building.

16.2..L4 Mode 6- Refueling (Flooded)

The plant enters this configuration after the reactor well flooding is completed.

Decay heat removal is provided by RWCU/SDCS, with only one train running much of the time in this mode. The FAPCS, operating in the reactor well cooling mode, can be used also to cool the reactor. If required, train B of FAPCS can be operated in Reactor Well Cooling mode. In this mode, water from the reactor well is directed to the train B heat exchanger to ensure adequate cooling of the upper layer of the reactor well water. It is expected that FAPCS Train B operates in this mode **8** hours in every refueling outage.

The duration of Mode 6- Flooded is assumed to be approximately 241 hours (10 days).

In this configuration, the reactor head is removed and the reactor well is flooded.

The RPV is at atmospheric pressure and the reactor coolant temperature is maintained between 54 **'C** (150 **F)* and 51 °C (124 'F). The reactor vessel is assumed to be open to the reactor building.

16.2.1.5 Mode 5- Cold Shutdown (Before Power Operation)

After completing the refueling, an additional period in cold shutdown may be required for completing other maintenance activities.

The plant configuration is described in 16.2.1.2, except that initial pressure and temperature of the RPV and decay heat are reduced. One RWCU/SDCS train is running.

This analysis assumes that the duration of this mode is approximately 150 hours per refueling outage.

16.2.2 Mission Time

For the quantification of core damage frequency, the mission time is assumed to be 24 hours. However, the availability of inventories of water and power sufficient to ensure core cooling from 24 h to 72 hours is also considered.

16.3 INITIATING **EVENTS**

The purpose of this subsection is to determine the initiating events that challenge the critical safety functions (e.g., heat removal, inventory control) during shutdown operations. A shutdown initiating event is defined as any event that provokes a disturbance in the stable state of the plant and that requires some kind of action to prevent damage to the core.

Section 16.3.1 discusses the shutdown critical safety functions and the shutdown initiating events that challenge these critical safety functions. Section 16.3.2 presents the analysis of the initiating events considered in the Shutdown PRA. Frequencies for initiating events are estimated in Section 16.3.3 and the recovery actions credited are discussed in Section 16.3.4.

16.3.1 Shutdown Critical Safety Functions

The primary critical safety functions accounted for in the Level 1 internal event shutdown evaluation are the following:

- Decay Heat Removal (DHR)
- * Reactor Coolant System Inventory Control

16.3.1.1 Decay Heat Removal

The decay heat removal function during all shutdown modes of operation is provided by the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDCS) operating in shutdown cooling mode. In Mode 6 with the reactor well flooded, the Fuel and Auxiliary Pools Cooling System (FAPCS) may be used as an alternative.

At the beginning of every shutdown period, both RWCU/SDCS trains will be running, with pumps varying their speed in order to meet the cooldown rate objectives. Once in Mode 6, before completing reactor cavity flooding, only one train is required.

If the reactor well is flooded (Mode 6-Flooded), the risk associated to loss of decay heat removal has been judged to be negligible because of the following:

- * In addition to RWCU/SDCS, FAPCS can be aligned to cool the reactor well water, constituting a valid alternative for RWCU/SDCS, thus reducing the probability of losing the decay heat removal function.
- The large amount of water stored above the core assures core cooling during a long period of time. This time would be significantly longer than 24 hours. This long period could be used to establish an adequate path from an external water source to the reactor well. CRD pumps, FAPCS pumps, condensate pumps, or firewater pumps could provide this makeup function. The long period of time available makes it practically certain that sufficient inventory can be supplied.

Therefore, the loss of decay heat removal is not analyzed in detail for the case when the reactor well is flooded (Mode 6-Flooded).

For the other shutdown modes (Modes **5** and 6 with the reactor well unflooded), it is assumed that one RWCU/SDCS train is sufficient to remove decay heat to prevent reactor coolant boiling.

It is assumed that both RWCU/SDCS trains are running, because the time periods in which only one is running occurs when the reactor well is flooded. Consequently, failure of one of the trains is not considered an initiating event.

16.3.1.2 Reactor Coolant System Inventory Control

This critical safety function is defined as maintenance of the RCS inventory at a level sufficient to sustain decay heat removal.

LOCA and RPV draindown events can potentially challenge this critical safety function. They can occur as a result of:

- * Random pipe breaks within the RCS (including breaks related to maintenance or refueling operations).
- **"** Misalignment of systems connected to the RPV.
- **"** Leakage during FMCRD replacement.

16.3.1.2.1 Pipe breaks

Two different cases are analyzed, depending on whether the reactor vessel head is installed or not.

16.3.1.2.1.1 Mode 5

As long as the reactor vessel head is in place with its bolts fully tightened, the accident sequence following a Loss of Coolant Accident is considered included in the Full Power PRA.

The frequency of these events is expected to be lower than at full power, due to the reduced vessel pressure and temperature. For example, the Grand Gulf Shutdown Study (Reference 16-1) reports that the large LOCA frequency for shutdown events is a factor of ten lower than the frequency for the full power case. Also, control rods are fully inserted at the beginning of the sequence, and the reduced pressure and temperature of the reactor coolant, and the lower decay heat level allow for longer times available for adequate action.

It is judged that no specific shutdown LOCA event trees are required for shutdown modes where the reactor vessel is closed.

16.3.1.2.1.2 Mode 6

As long as the RPV level is above Level 3 (L3), it is assumed that RWCU/SDCS provides adequate core cooling. Natural circulation of coolant inside the vessel is not challenged because L3 is above the bottom of the steam separators. As such, any break above L3 does not constitute a shutdown initiating event, as RWCU/SDCS will continue to ensure normal core cooling and the core will remain covered.

However, if RPV level drops to L3, RWCU/SDCS pumps receive a runback signal, slowing down to cleanup mode flow rate. In addition, once water level in the vessel falls below the bottom of the steam separators, natural circulation is not assured and the core cooling function of RWCU/SDCS may not be adequate. As such, breaks below L3 are included in the analysis as shutdown initiating events.

Breaks below L3 are divided into the following two categories: breaks outside containment and inside containment.

Breaks Outside Containment

Breaks outside containment can originate only in RWCU/SDCS piping, as this is the only system that removes reactor coolant from the containment in Mode **6;** the rest of the RPV vessel piping is isolated.

The RWCU/SDCS containment penetrations have redundant and automatic power-operated containment isolation valves that close on signals from the leak detection and isolation system and the reactor protection system. The inboard valves are nitrogen-operated and the outboard valves are air-operated. Both the inboard and outboard valves are solenoid actuated with springto-close provisions. An additional, diverse non-safety isolation of the RWCR/SDCS system provides protection in the event of a break outside containment.

The RWCU/SDCS return lines to the feedwater lines are each provided with redundant check valves in series located in the Main Steam Tunnel. A single-power operated isolation valve in each line is located upstream of the check valves and inside the Reactor Building. The FAPCS and CRDS connections are downstream of the two check valves. A postulated break in the RWCU/SDCS piping system inside the Reactor Building, which would otherwise allow reactor coolant to flow backwards through main feedwater lines and to spill into the Reactor Building, will be isolated by the redundant RWCU/SDCS check valves even if a single failure of one check valve is assumed.

Therefore, the shutdown PRA considers RWCU/SDCS breaks outside containment to be negligible risk contributors and does not analyze them further. This is consistent with the atpower PRA which shows the CDF contribution from RWCU/SDCS breaks outside containment to be negligible.

Breaks Inside Containment

For breaks inside containment, coolant flows through the break to the lower drywell. Decay heat removal is achieved in this case by allowing reactor coolant boiling and then venting the steam to the atmosphere (i.e., the drywell head is removed in Mode 6). To maintain adequate water level in the vessel, a water supply to the vessel is required.

If a break is located below TAF, to reach a safe core cooling condition, it is necessary to flood the drywell and the vessel up to a level above the TAF.

The lower drywell is equipped with a personnel hatch and with an equipment hatch to allow access to the containment for personnel and equipment. These hatches are closed during normal operation, but they may be open during refueling. A manual recovery action to close these two hatches is required for successful drywell flooding (see recovery analysis below).

Two different cases are considered for breaks inside containment below TAF during Mode 6:

* Reactor well flooded (Mode 6-Flooded):

If the reactor well is flooded, the water inventory stored above the core is assumed to be sufficient to flood the drywell and the vessel well above the TAF if the two lower drywell access hatches are closed at the time of the event or they are manually closed before the water level in the drywell reaches the elevation of the hatches

• Reactor well unflooded (Mode 6-Unflooded):

If the well is not flooded, the water inventory stored above the core is assumed to be insufficient to cover the core, and additional coolant supply is required

As discussed previously, only pipe breaks below RPV Level 3 (L3) are considered shutdown initiating events. Therefore, breaks in main steam lines, DPVs, and instrument lines above L3 are not considered as shutdown initiating events.

Based on the discussions above, and the line breaks identified in Table 2.3-1, the following line break categories are identified as potential shutdown LOCA initiators:

- * GDCS injection line break: this event degrades the passive inventory control system.
- Feedwater line A break: it is assumed that the break occurs in FW line A because this disables effective injection of LPCI and Firewater.
- **"** Line breaks above TAF other than GDCS injection or Feedwater lines.
- **"** Line breaks below the TAF: closure of the lower DW hatches is required to achieve core cooling.

16.3.1.2.2 RPV draindown events due to misalignments

The ESBWR design has significantly reduced the number of potential RPV draindown pathways due to postulated system misalignment during shutdown conditions.

In particular, as compared to Residual Heat Removal System in current BWRs, the RWCU/SDCS in the ESBWR does not have the potential for diverting RPV inventory to the suppression pool through the SP suction, return, or spray lines. RWCU/SDCS does not provide any drywell spray function, so the potential RPV draindown through drywell spray does not exist either.

In addition, the absence of recirculation lines in the ESBVWR design further reduces the potential RPV draining paths.

The only operating system that has the potential to drain the RPV during this mode of shutdown is RWCU/SDCS. This system is connected to the RPV during shutdown and it is used to discharge excess reactor coolant to the main condenser or to the radwaste system during startup, shutdown and hot standby conditions.

An analysis of this possible path has been conducted. The conclusion of the analysis is that the ESBWR design eliminates most of the risk for potential drain down from this path. Therefore, the contribution to the core damage frequency in shutdown is judged negligible and this scenario is not analyzed further.

16.3.1.2.3 FMCRD replacement

FMCRD replacement is performed in two steps. First, the CRD spool piece is removed, at which time the spindle adaptor seats on the spindle adaptor back seat to prevent any leakage of water from the RPV. Next, the CRDM is withdrawn until the blade back-seats on the guide tube to provide a metal-to-metal contact. This provides a seal preventing the reactor water from draining. The drive can then be removed and replaced. This arrangement for preventing vessel

draining through back-seating of the control blade is the same as the one used in the current BWRs and in the ABWR.

The potential exists for the operator to remove the blade inadvertently, establishing a direct path for draining the RPV. If a crew of operators is replacing one FMCRD drive while another crew is replacing or rotating the blades from the top, and the blade belonging to the removed drive is acting as a plug, and if inadvertently pulled out, a drain path can be established. It is assumed that, as a normal practice, a blind flange is installed during a temporary removal of the drive or a new drive is installed in a short period of time.

According to these assumptions, two operational errors must occur to allow a drain event to occur:

- **"** Failure to install a blind flange
- **"** Failure to recognize that the blade to be pulled out is withdrawn and already decoupled from the drive.

For these reasons, it is judged that the contribution of this initiating event to the core damage frequency is non-significant.

16.3.2 Identification of Initiating Events

The identification of the shutdown initiating events for inclusion in the ESBWR shutdown risk assessment is based on:

- Review of past shutdown PRAs
- * Review of the ESBWR full power PRA initiators
- * Consideration of the ESBWR design and configuration during shutdown

The potential initiator scenarios are described in Section 16.3.1. The resulting list of initiating event types during shutdown (and as a function of critical safety function) is presented in Table 16.3-1.

16.3.3 Frequency of Initiating Events

Initiating event frequencies are quantified based on a review of BWR operating experience as well as ESBWR specific evaluations.

16.3.3.1 Loss of Both Operating R WCU/SDCS Trains

The main components of the system are the pumps, heat exchangers, demineralizers, valves and piping.

RWCU/SDCS is connected to non-safety-related standby AC power (diesel generators) allowing it to perform its reactor cooling function when the preferred power source is not available.

In addition to AC power, RWCU/SDCS requires the operation of the Reactor Component Cooling Water System (RCCWS) and the Plant Service Water System (PSWS) in order to remove decay heat.

The unavailability of the RWCU/SDCS system can occur for the following general reasons:

- **"** Failure of both RWCU/SDCS trains
- **"** Isolation of the RWCU/SDCS, caused by RPV low level (Level 3 causes RWCU/SDCS pump runback and Level 2 causes loss of suction pressure), SLCS initiation, LD&IS signals, high temperature in main steam tunnel or high system flow
- **"** Loss of Preferred Power (LOPP)
- * Loss of RCCWS or PSWS

The Loss of RWCU/SDCS shutdown initiating event is defined by failure of both trains, either due to RWCU/SDCS component failures or by automatic closure of isolation valves. LOPP and Loss of RCCWS/PSWS are modeled as separate shutdown initiating events.

Automatic closure of RWCU/SDCS isolation valves can be initiated by the following signals:

- **"** High RWCU/SDCS flow
- Low reactor water level (level 2)
- **"** High temperature in main steam tunnel
- Initiation of the Standby Liquid Control System
- **"** Leak Detection and Isolation System signals

ESBWR logic design uses four divisions of power backed up by safety-related batteries. Therefore, loss of power to the logic is highly unlikely. Three divisional logic power supply failures are required to initiate the SDC isolation; as such, this RWCU/SDC failure mode is nonsignificant compared to loss of support systems or mechanical failures.

The initiating frequency for Loss of RWCU/SDCS is calculated from the dominant failure mode, common cause failure to run of the two RWCU pumps.

16.3.3.2 Loss of Preferred Power

Loss of Preferred Power (LOPP) may happen as a result of severe weather, grid failures or switchyard faults.

The LOPP shutdown initiator frequency is calculated using the loss of offsite power during shutdown data documented in NUREG/CR-5496. (Reference 16-2)

16.3.3.3 Loss ofRCCWS/PSWS

This initiating event is the loss of the Reactor Component Cooling water System (RCCWS) or the loss of the Plant Service Water System (PSWS) supporting the RWCU/SDCS operating in shutdown cooling mode. This initiating event poses a DHR challenge and renders the RWCU/SDCS system unavailable.

The frequency of this initiator is based on the Loss of PSWS initiating event frequency calculated for the at-power PRA.

16.3.3.4 LOCA in Mode 6

The following LOCA initiators are quantified in the shutdown PRA:

- Break in one of the GDCS injection lines (Mode 6-Unflooded)
- **"** LOCA in FW-A (Mode 6-Unflooded)
- **"** LOCA other than FW or GDCS (Mode 6-Unflooded)
- * LOCA below TAF in RWCW/SDC drain lines (Mode 6-Unflooded; Mode 6-Flooded)
- LOCA below TAF in instrument lines (Mode 6-Unflooded; Mode 6-Flooded)

The shutdown LOCA frequencies are based on the at-power RCS line break frequencies (refer to Table 2.3-1). The at-power RCS line break frequencies are reduced by a factor of 10 for the shutdown PRA to reflect the low pressure and temperature during shutdown conditions. (Reference 16-1)

16.3.4 Recovery Actions

This section documents the calculation of time-dependent, post-initiator recovery probabilities used in the ESBWR Shutdown PRA. The recovery events addressed in this section are those that terminate or mitigate the initiating event before a safety function is challenged. Unlike at fullpower conditions, during shutdown modes extended time can be available to terminate the initiating event. This justifies the quantification of recovery possibilities.

Recovery events are analyzed for the following initiating events:

- * Loss of both operating RWCU/SDCS trains: Operators recover at least one of the two failed trains.
- * Loss of Preferred Power: Offsite AC power recovered.
- * Loss of RCCWS/PSWS: Operators recover the failed equipment.
- LOCA Below TAF (Mode 6): Operators close the two lower drywell hatches if they are open.

The analysis of these events is performed using the BWR industry data from References 16-3, 16-4, 16-5 and 16-6. Industry data is included here based on operating experience in Modes **5** and 6.

The operating experience events in each category are analyzed to determine the time elapsed before the initiating event was terminated. Because of the limited data for extended durations, an additional assumed event with a recovery time of 20 hours is added to all distributions.

It is then assumed that the time to recovery is a random variable following a lognormal probability density distribution. The parameters of the distribution are chosen so that a best fit between the real data and the theoretical curve is achieved. The resulting curve provides the probability of shutdown initiating event recovery as a function of time.

16.3.4.1 Recovery of RWCU/SDCS

The most functionally similar system to RWCU/SDCS in current BWRs is the Residual Heat Removal system.

Events involving the loss of a running RHR pump at BWRs are identified from References 16-3, 16-4 and 16-5, and are shown in Table 16.3-4. However, due to the lack of BWR events

specifying the duration, PWR events (see Reference 16-6) are also used in the recovery analysis, excluding those occurring during reduced inventory conditions. Using the methodology described above, the lognormal distribution parameters m and s are determined to be 2.73 and 2.08, respectively. The resulting recovery probability curve as a function of time for the Loss of RWCU/SDCS initiator is provided in Figure 16.3-1.

The allowable time frame for recovery of a Loss of RWCU/SDCS initiator is defined as the time for RPV level to boil-off and drop to RPV Level 3 (at which point a RWCU/SDCS pump runback signal would be initiated – further hampering recovery of RWCU/SDCS cooling).

For Mode **5,** the available time is 1.2 hours.

For Mode 6-Unflooded, the available time is 1.3 hours.

These times are based on the ABWR thermal hydraulic calculations (Reference 16-7) and are estimated by the time to reach RCS boiling (assuming the initiating event occurs at the start of the outage phase). These estimates are conservative for the following two reasons: 1) the ESBWR RPV is a larger vessel than the ABWR, thus the larger water volume of the ESBWR would take longer to reach boiling; and 2) the additional time for water level boil-off down to L3 is not credited.

The non-recovery probabilities for these two cases are shown in Table 16.3-5.

16.3.4.2 Power Recovery after LOPP

Recovery from this initiating event implies that the offsite AC power has been restored. Following restoration of offsite power, operator action to align the systems participating in the decay heat removal function is assumed to be required.

NUREG/CR-5496 (Reference 16-2) distinguishes two types of LOPP: plant-centered LOPP and external LOPP (severe weather or grid-related). For each type, a different probability density function is given for the time to recover.

For LOPP cases related to plant-centered events, it is conservatively assumed that recovery time probability density distribution corresponds to the type of "maintained events", as reported in NUREG/CR-5496. Using the plant-centered events data, the lognormal distribution parameters m and s are determined to be 3.39 and 1.44, respectively. The resulting recovery curve is provided in Figure 16.3-2.

For external LOPP cases, the recovery time probability density distribution corresponds to the type of "severe weather", as reported in NUREG/CR-5496. This data results in a lognormal distribution with parameters m and s equal to 5.83 and 1.63, respectively. The recovery curve is provided in Figure 16.3-3.

The allowable time frame for recovery is taken to be the same as that for Loss of RWCU/SDCS.

For Modes 4 and 5, the available time is 1.2 hours.

For Mode 6-Unflooded, the available time is 1.3 hours.

The corresponding shutdown LOPP recovery failure probabilities are determined using a weighted average based on the frequencies of the two LOPP cases (plant-centered and external). The resulting LOPP non-recovery probabilities are shown in Table 16.3-5.

16.3.4.3 Recovery of RCCWS/PSWS

The RCCWS and PSWS are used during shutdown to remove decay heat and other heat loads. The main components in the system are pumps and heat exchangers. It is assumed that the time to recover from a Loss of RCCWS/PSWS shutdown initiator follows the same probability density distribution as the time to recover the RWCU/SDCS.

The allowable time frame for recovery is taken to be the same as that for Loss of RWCU/SDCS.

For Mode **5,** the available time is 1.2 hours.

For Mode 6-Unflooded, the available time is 1.3 hours.

The non-recovery probabilities for these two cases are shown in Table **16.3-5.**

16.3.4.4 Close Lower Drywell Hatches

An equipment hatch for removal of equipment during maintenance and an air lock for entry of personnel are provided in both the lower and upper drywell. These access openings are sealed under normal plant operation but are opened when the plant is shut down for refueling. Credit is not given for closing the doors after reactor coolant overflows through the hatches. Therefore, the time available for closing both hatches depends on the volume of the lower drywell under the bottom edge of the hatches, on the size of the break, and on the water level in the vessel or reactor well above the break.

This action is required for the shutdown LOCA below TAF initiators during Mode 6. These LOCAs involve breaks in the RWCU/SDC drain lines and instrument lines.

The flow through the break is assumed constant and equal to $S\sqrt{2gh}$, where S is the break area, h is the height of water above the break, and g is the acceleration due to gravity.

Both break locations, in RWCU/SDCS drain lines and instrument lines, are evaluated. The calculation assumes the reactor well is not flooded at the time of the break. Table 16.3-6 summarizes the calculation.

Detection of the event will be immediate if personnel are present in the lower drywell. If this is not the case, it is assumed that an alarm on drywell sump high level is available in the control room.

Once the event has been detected, the plant operator must correctly diagnose the situation, make the decision to close the hatches, gain access to elevation -6400 mm in the reactor building, and manually close the equipment hatch and the personnel air lock. It is assumed that during the outage, personnel will be continuously located in the area of the doors.

Probabilities of 1.0E-1 and 1.0E-2 are assigned for failing to close the DW hatches for breaks in RWCU/SDCS drain lines and instrument line breaks, respectively.

16.4 EVENT TREES

The shutdown PRA event trees are shown in Figures 16.4-1 through 16.4-13.

The event tree construction takes into account the following aspects:

- Chronological order of system actuation
- **"** Grouping of mitigating systems by safety functions

Descriptions of all event tree headings are provided below, though headings appearing in different event trees are described only once.

The success criteria used in the different events are reported under the description of the event heading.

The accident sequence end state nomenclature is the same as in the full power PRA:

- OK: The core is successfully cooled and the containment is intact. There is no core damage in these events.
- * CD I: The containment is intact when core damage occurs and the RPV is at low pressure.
- * CD II: The containment fails while the core is successfully cooled, leading to subsequent core damage.
- CD III: The containment is intact when core damage occurs and the RPV is at high pressure.
- CD V: The containment is bypassed at the time of core damage.

During Mode 6, when the containment is open, sequences leading to core damage are designated as CD V.

16.4.1 Loss of Decay Heat Removal

Initiators leading to a loss of decay heat removal function are grouped into scenarios occurring during Mode **5** or during Mode 6. Given the three initiating event types leading to loss of DHR scenarios (Loss of RWCU/SDC, Loss of RCCWS/PSWS and Loss of Preferred Power), six shutdown loss of DHR event trees are analyzed.

 $\frac{1}{2}$

16.4.L1 Loss ofDHR in Mode 5

Following loss of the decay heat removal function, pressure and temperature in the RPV gradually increase. Due to reduced initial pressure and temperature during shutdown conditions, the operator has the opportunity to recover the lost function before the pressure reaches the SRV setpoint.

If recovery of decay heat removal is not possible, the isolation condenser (I) function will be initiated on RPV high pressure or low RPV water level (Level 2). This function provides short term as well as the long term core cooling.

If ICS fails, RPV pressure increase leads to SRV opening. Steam generated by decay heat is then discharged to the suppression pool, where it is condensed. As the level inside the RPV decreases, high pressure makeup is required to keep the core covered.

Failure of all of the SRVs to open could be postulated to lead to a vessel rupture scenario. Even if mitigation is still possible, core damage is assumed.

Either the feedwater and condensate pumps or the Control Rod Drive system (CRD) can fulfill the high-pressure makeup function. CRD is automatically initiated in the injection mode on RPV water Level 2.

It is possible that the suppression pool water level could exceed the maximum allowed temperature limit, and the depressurization of the Reactor Pressure Vessel (RPV) is required. It is considered that this depressurization is performed using only the SRVs, so the containment heat removal function can still be performed by the FAPCS in the suppression pool cooling mode.

If all high pressure makeup alternatives fail, low pressure makeup is required. The opening of two SRVs enables the injection modes of either FAPCS or the Fire Protection System (FPS).

If low pressure injection systems fail after manual depressurization with 2 SRVs, the ADS will actuate and the short term and long term core cooling functions are performed with 2 of 8 lines from the Gravity Driven Cooling System (GDCS), 2 of 3 GDCS pools and the opening of at least one equalizing line. The equalizing line will permit effective RPV flooding with the suppression pool water, as long as at least 4 DPVs have opened. If for some reason GDCS cannot inject into the depressurized reactor, either FAPCS or FPS injection mode can support the short term and long term core cooling functions.

The containment heat removal function is performed, in both cases, by either the PCCS and if this system fails, FAPCS in the suppression pool cooling mode can be aligned. If both PCCS and FAPCS are not effective, the containment vent can be actuated to reduce the pressure in the containment through at least one path to the atmosphere. In the event of failure of the containment vent function, it is assumed that long term core coolant makeup is failed and, as such, such sequences are modeled as leading to core damage.

16.4.1.1.1 Loss of Both RWCU/SDCS Trains (Mode 5)

The event tree for this initiating event is shown in Figure 16.4-1. The description given above corresponds to this case. The event tree headings are described below.

 \mathbf{L}

RW4 - Recovery of **RWCU/SDCS (Mode 5)**

The operator has the opportunity to recover the RWCU/SDCS before any other system actuates or any other safety function is challenged. See Section 16.3 above for the time available and the recovery failure probability.

I - Isolation Condensers

Regardless of initial RPV pressure, if decay heat is not removed, the Isolation Condenser System is initiated automatically on high reactor pressure, or later on RPV water level Level 2.

The I function is able to prevent RPV Level 1 from being reached if:

- The initial RPV water inventory is above Level 3
- There is little or no leakage from the RPV.

The maximum RCPB leak rate within the Technical Specification during full power operation is assumed to be insufficient to decrease the level to the point where an ADS signal occurs, even if high pressure RPV makeup is not established during the sequence mission time. Therefore, failure of the I function due to leaks is considered a low probability.

The success criterion of this function is the operation of at least 3 of 4 ICs during the sequence mission time, and success of the Long Term Containment Cooling System (WT) described later.

M - At Least **1** SRV Open

If the I function fails, the RPV pressure will increase up to the SRVs setpoint. The success criterion for this function is the automatic operation of at least 1 SRV. Failure of this function is conservatively assumed to lead to core damage.

The possibility of a stuck open relief valve is not modeled. As no credit is given for the I function after the opening of an SRV, it is not necessary to assume that all SRVs are closed.

U1CF - High Pressure Injection Systems

Water level can be maintained above RPV Level 1 by the FDW or by the CRD system. FDW needs to be aligned and initiated by the operator while CRD system automatic initiation occurs at RPV Level 2.

Successful FDW is modeled as at least 1 FDW pump and 1 Condensate pump taking suction from the Condenser, which in turn is supplied from the Condensate Storage Tank (CST). If failure of FDW occurs, RPV Level 2 setpoint is reached and CRD injection initiates. Successful CRD injection into the reactor vessel requires 1 CRD pump taking suction from the CST.

XS2 - At Least 2 SRVs Open

If no high pressure injection system is available, it is necessary to depressurize the RPV to allow FAPCS or FPS injection to the RPV.

Success of this function requires the operator to manually open at least 2 SRVs.

The time available to the operator to manually initiate RPV depressurization is defined by the time when RPV level falls below L2 to the time when the ADS system will automatically initiate (i.e., at RPV Level 1).

 $\begin{array}{c} \hline \end{array}$

VLF-Low Pressure Injection Systems

After successful RPV depressurization, either FAPCS or FPS can fulfill the core cooling function when configured in the RPV injection mode. Both systems are manually actuated.

The time available to the operator to manually initiate either of these two systems is defined by the time when RPV pressure has been sufficiently reduced to the time when the ADS system will automatically initiate (i.e., at RPV Level 1).

Success of this function requires the operator to manually align at least 1 FAPCS or FPS train in RPV injection mode.

ADS - At Least 4 DPVs Open Automatically

If RPV water level falls below Level 1, the ADS system automatically initiates.

The success criterion for this function is that at least 4 DPVs automatically open.

VG - Gravity Driven Cooling System

If all the active low pressure injection systems are unavailable after successful RPV depressurization, the passive GDCS system will automatically inject water into the RPV. Short term core cooling is accomplished by the opening of at least 2 GDCS lines and the discharge of at least 2 GDCS pools. One equalizing line must be opened for long term core cooling.

The success criteria for this function are the discharge of at least 2 lines and 2 GDCS pools and the opening of at least 1 equalizing line.

VLFL - Low Pressure Iniection Systems after ADS

This event tree node models initiation of low pressure injection following automatic RPV depressurization by the ADS system. As in the VLF node, either **I** train of FAPCS or FPS operating in the RPV injection mode can fulfill this function. However, the time available to the operator to perform the manually alignment is different in the scenario with automatic ADS. The time available to the operator to perform the alignment is defined by the time when the GDCS fails to initiate to the time when core damage starts.

DL-Vacuum Breaker Closed

When the RPV is depressurized through the DPVs, the vacuum breakers connecting the wetwell and the drywell must remain closed; otherwise the PCCS can not effectively remove heat from the containment.

If a vacuum breaker opens, it is possible to close it or the associated series isolation valve. The time available to close or isolate an open vacuum breaker is defined by the time of RPV depressurization to the time when containment pressure reaches venting pressure.

The success criterion for this heading is that all vacuum breakers remain isolated.

WP - Passive Containment Cooling System

The short term passive containment heat removal function is performed by the PCCS. The PCCS is effective only when the drywell pressure is greater than the wetwell pressure and all the vacuum breakers are closed.

The system is always open to the containment atmosphere and it has no valves that require opening. Failure of this function is the loss of effectiveness of the heat exchangers in removing the decay heat from containment atmosphere (e.g., tube plugging).

The success criterion for this function is the operation of at least 4 heat exchangers.

WT-Long Term Containment Cooling System

This heading models the potential for long term containment heat removal failure.

When PCCS or the ICs are performing the containment heat removal function, long term containment heat removal is achieved via use of all the water in the pools of the upper part of the reactor building.

The success criterion for this function is the full connection of the pools in the reactor building.

WLL - Suppression Pool Cooling after Depressurization

If other containment cooling systems have failed, the decay heat removal function can be accomplished by the FAPCS in the suppression pool cooling mode. This requires manual initiation.

The time available to the operators to initiate suppression pool cooling is defined by the time when PCCS fails to the time when containment pressure reaches the containment venting pressure.

Success of this function requires manual alignment of at least 1 FAPCS train in suppression pool cooling mode.

WC - Containment Overpressure Protection System

If no containment heat removal system is available, the operators can reduce containment pressure by venting from the suppression pool to the stack via the Containment Inerting System.

Success of this function requires manually opening the vent line. The time available to the operator to initiate venting is defined by the time that the CIS setpoint is reach until the time of containment failure.

16.4.1.1.2 Loss of Preferred Power (Mode 5)

The event tree for this initiating event is shown in Figure 16.4-3. The event tree heading descriptions above for loss of DHR in Mode 5 is applicable to this case, except for those systems that are unavailable because they cannot be not powered by the standby diesel generators (i.e., Feedwater and Condensate pumps).

As a result, heading UlCF in the Loss of RWCU/SDC event tree is replaced by U1C, and RW4 is removed. The recovery of power is accounted for in the fault tree models.

Also, a new heading, WH, is introduced to account for the RWCU/SDCS restart on diesel generator power after the LOPP event.

The event tree headings UC and WH are described below.

UIC - Control Rod Drive System

In case of failure of the I function, water level can be maintained above RPV Level 1 by the high pressure CRD system. Automatic initiation of the CRD injection mode occurs at RPV Level 2.

Success of CRD for this function is the effective injection into the reactor vessel of 1 CRD pumps taking suction from the CST.

WH - Reactor Water Cleanup/Shutdown Cooling System

After a LOPP event, the RWCU/SDCS pumps are tripped and the decay heat removal function is temporarily unavailable. This heading represents the restart of the two RWCU/SDCS trains relying on the diesel generator power supplies.

The success criterion is that at least one RWCU/SDCS train successfully restarts after a LOPP event and operates during the sequence mission time.

16.4.1.1.3 Loss of RCCWS/PSWS (Mode **5)**

The event tree for this initiating event is shown in Figure 16.4-5. The event tree heading descriptions above apply to this tree as well, except for those systems that are unavailable because they rely on RCCWS/PSWS: Feedwater and Condensate, CRD, and FAPCS.

As a result, headings U1CF, WL and WLL are removed from the event tree; heading VLF is replaced by VF; and heading VLFL is replaced by VFL.

The initiating event recovery heading in this tree is named RC4.

The event tree headings RC4, VF, and VFL are described below.

RC4 - Recovery of RCCWS/PSWS (Mode **5)**

The operator has the opportunity to recover the RCCWS/PSWS before any safety function is challenged. See Section 16-3 above for the time available and the recovery failure probability.

VF - Fire Protection System

After successful RPV depressurization, the FPS can accomplish the core cooling function when configured in the RPV injection mode. FPS must be manually actuated.

The time available to initiate FPS injection is defined by the time when RPV pressure has been sufficiently reduced to the time when the ADS system will automatically initiate (i.e., at RPV Level 1).

Success of this function requires the operator to manually align at least 1 train of FPS in RPV injection mode.

VFL - Fire Protection System after **ADS**

This event tree node models initiation of low pressure injection following automatic RPV depressurization by the ADS system. As in the VF node, 1 train of FPS operating in the RPV injection mode can fulfill this function. However, the time available to the operator to perform the manually alignment is different in the scenario with automatic ADS. The time available to the operator to perform the alignment is defined by the time when the GDCS fails to initiate to the time when core damage starts.

16.4.1.2 Loss of DHR in Mode 6 (Unflooded)

In this case, the RPV is open (i.e., the RPV head has been removed, Mode 6), the containment is also open, and the reactor well is assumed to be drained, so the water level is at the elevation of the vessel flange.

A considerable amount of cool water is above the core, allowing the operator significant time to recover failed equipment or systems before coolant boil-off reduces the water level in the vessel, causing safety system actuations.

However, if the decay heat function cannot be recovered, RPV coolant temperature rises, eventually reaching boiling conditions. Makeup water to the RPV is then needed to prevent core damage. Four systems are allowed for the makeup function. Two of them require manual initiation by the operator: FAPCS in injection mode and FPS in injection mode. The other two automatically initiate: CRD (automatic initiation on L2) and GDCS (automatic initiation on Level 1).

16.4.1.2.1 Loss of Both RWCU/SDCS Trains (Mode 6)

The event tree for this initiating event is shown in Figure 16.4-2. All headings for this event tree have been described previously.

16.4.1.2.2 Loss of Preferred Power (Mode 6)

The event tree for this initiating event is shown in Figure 16.4-4. The descriptions above for loss of DHR in Mode 6 apply to this tree as well, except that some systems are unavailable because they are not powered by the standby diesel generators. All headings for this event tree have been described previously.

16.4.1.2.3 Loss of RCCWS/PSWS (Mode 6)

The event tree for this initiating event is shown in Figure 16.4-6. The descriptions above for loss of DHR apply to this tree as well, except that some systems are unavailable because they rely upon RCCWS/PSWS: CRD and FAPCS. FPS and GDCS remain the only makeup systems available. All headings for this event tree have been described previously.

16.4.2 Loss of Coolant Accidents

16.4.2.1 LOCAs in Mode 6 (Unflooded)

In this mode of operation, all LOCAs are liquid breaks. The evolution of the accident and the systems available for mitigation depend on the break location.

Four break locations are analyzed:

- **"** break in a GDCS injection line
- **"** break in Feedwater line A
- **"** break above TAF other than GDCS or FW
- breaks below TAF.

As soon as the break takes place, the liquid coolant flows into the lower drywell, driven by gravity and hydrostatic pressure. If insufficient coolant makeup is provided to the vessel, water level decreases from the vessel flange down to the break elevation. Only break elevations below L3 analyzed; for breaks above L3, the RWCU/SDCS continues removing the decay heat, and no safety function is directly challenged.

Once the level falls below L3, decay heat removal is lost, as the RWCU/SDCS pumps receive a runback signal, and natural circulation inside the vessel is lost when water level drops below the separators skirts.

It is assumed for breaks above TAF that providing makeup to the vessel and allowing coolant boiling is an effective method for core cooling. CRD, FAPCS, Fire Protection and GDCS are considered for water makeup.

For breaks below TAF, the drywell has to be flooded up to an elevation above TAF to reach a safe core cooling condition. The personnel and equipment access hatches to the lower drywell could be open during shutdown, a recovery action to close these doors is modeled.

16.4.2.1.1 **LOCA** in GDCS Line (Mode 6, Unflooded)

The event tree for this initiating event is shown in Figure 16.4-7. The event heading descriptions above apply to this tree as well. However, the following two new headings are included in this event tree:

" UC - Control Rod Drive System

The CRD system can provide makeup water in case of a loss of coolant accident during Mode 6. CRD injection mode automatic initiation occurs at RPV Level 2.

Success of CRD injection into the RPV in this scenario requires 2 CRD pumps taking suction from the CST.

" VG2 - Gravity Driven Cooling System

If other injection systems are unavailable, the passive GDCS system will inject water into the RPV by gravity.

Success of this function requires at least 1 GDCS injection line, 2 GDCS pools, and opening at least 1 equalizing line.

The fault tree of the GDCS system in this event tree accounts for the GDCS line break.

16.4.2.1.2 LOCA in FDW-A Line (Mode 6, Unflooded)

The event tree for this initiating event is shown in Figure 16.4-8. The event heading descriptions above apply to this tree as well, but no credit is given to the systems using the FDW-A piping to inject water in the vessel (FAPCS and FPS). All event tree headings have been described previously.

16.4.2.1.3 LOCA above TAF other than GDCS or FDW-A (Mode 6, Unflooded)

The event tree for this initiating event is shown in Figure 16.4-9. The event heading descriptions above apply to this tree as well. All event tree headings have been described previously.

16.4.2.1.4 LOCA below TAF in RWCU/SDC Drain Lines (Mode 6, Unflooded)

The event tree for this initiating event is shown in Figure 16.4-10. The event tree descriptions above apply to this tree as well, except the recovery action to close the lower drywell hatches. The remaining event tree headings have been described previously.

RLOC1 - Close the lower drywell hatches

If not closed at the time of the initiating event, the operator must close the lower drywell hatches to prevent flooding of reactor building lower levels. See Section 16.3.4.4 for details on time available for this action and the recovery failure probability.

16.4.2.1.5 LOCA below **TAF** in Instrument Lines (Mode 6, Unflooded)

The event tree for this initiating event is shown in Figure 16.4-11. The description is the same as the preceding case, except that heading RLOC2 is used instead of RLOC1 to account for the longer time available to the operator to close the drywell hatches.

16.4.2.2 *LOCAs in Mode 6 (Flooded)*

Breaks below **TAF** are analyzed for LOCAs in Mode 6-Flooded. Two break locations are analyzed:

- **"** LOCA below TAF in RWCU.SDC Drain Lines
- **"** LOCA below TAF in Instrument Lines.

The event trees for these two cases are provided in Figures 16.6-12 and 16.6-13, respectively. Each event tree has a single top event modeling failure to close the lower drywell hatches. The failure probabilities for this node are the same as that described previously. Failure to close the lower drywell hatch is modeled as directly leading to a core damage scenario. The scenario with successful closure of the lower drywell hatches does not result in a core damage end state (and no additional top events are questioned) because sufficient water exists above the break to flood the containment above **TAF.**

16.5 SYSTEM ANALYSIS

This section describes the fault trees used in the Shutdown PRA evaluation. The unavailability of a system to perform its safety function on demand is evaluated by fault tree analysis.

The necessary fault trees are identified following construction of the event trees. These fault trees represent the nodes included in the event trees.

Maximum use is made of the fault trees developed for the Full Power PRA. Potential differences between the full power and the shutdown fault tree models may result from:

- * Differences in maintenance unavailabilities
- Differences in success criteria between full power and shutdown condition
- **"** Differences in initial system configuration between full power and shutdown condition
- **"** Differences in human actions

Differences in maintenance during full power and shutdown are addressed by modifying the maintenance unavailability probabilities in the shutdown fault trees. Table 16.5-1 summarizes the maintenance assumptions made for all frontline and support systems used in the shutdown event trees.

The following paragraphs discuss the cases where modifications were made to the full power fault tree logic to reflect the shutdown conditions.

16.5.1 Reactor Water Cleanup **/** Shutdown Cooling System

A fault tree is required for heading WH in the LOPP event trees. The function to be modeled is the restart of at least one train after a LOPP event, including the start up sequence of the standby diesel generators. This is the same situation as in full power. However, contrary to the full power case, the system is initially operating in the shutdown cooling mode, not in cleanup mode. This means that failures related to the alignment of the system (i.e., failure to open of several MOVs) are removed from the fault tree.

Maintenance is expected to be performed mostly during full power operation, when only one train is operating. Nevertheless, the same maintenance unavailability as used for the full power PRA is conservatively used in the shutdown evaluation.

16.5.2 Feedwater and Condensate

Feedwater and condensate unavailabilities due to maintenance are expected to be higher during shutdown than at full power, as these systems are not required during shutdown.

The initial configuration in shutdown of these systems is with all equipment in standby. This differs from the full power case where the system is running.

A human action to align and initiate the feedwater and condensate pumps to enable water injection to the vessel is required in the shutdown case.

16.5.3 Control Rod Drive System

It is assumed that during the entire shutdown period a single CRD pump is running (providing purge flow to FMCRD and/or the RWCU/SDC pumps) and the second pump is in standby, which is the same initial configuration as at full power.

Additionally, it is assumed that automatic initiation of CRD injection mode on RPV Level 2 is available during shutdown.

16.5.4 Gravity Driven Cooling System

The same initial configuration is valid for full power and for shutdown. However, it is possible that one GDCS pool and one equalizing line could be out of service for maintenance during Mode 6.

16.5.5 Isolation Condenser System

It has been assumed that Technical Specification requirements on ICS availability are the same during Mode 5 as during full power operation. Therefore, the same maintenance unavailabilities are applicable. The only difference in the fault tree model is that automatic ICS initiation signals upon MSIV closure are not available during Mode 5.

16.5.6 High-Pressure Nitrogen Supply System

During Mode 6, low pressure nitrogen loads inside the containment are fed by instrument air instead of nitrogen. As such, for accident sequences initiated during Mode 6, the top event 'Loss of nitrogen to low pressure users inside PC' is replaced by 'Loss of instrument air system'.

16.6 QUANTIFICATION RESULTS

The shutdown accident sequence analysis models the impacts on the following two critical safety functions during shutdown:

- **"** Decay Heat Removal (DHR)
- **"** Reactor Coolant System Inventory Control

Initiating event types, and associated frequencies, are identified that challenge the above critical safety functions (refer to Section 16.3). Event trees are developed specific to the shutdown configurations (refer to Section 16.4) and the system fault tree analysis is based on at-power fault tree models (refer to Section 16.5). The model development and quantification is performed in the CAFTA code. The quantification is performed at a truncation limit of 1E-14/yr.

The core damage frequency results of the ESBWR shutdown risk analysis are summarized in the following tables:

- Shutdown CDF by Initiating Event and Operating Mode (Table 16.6-1)
- **"** Shutdown CDF by Accident Class (Table 16.6-2)

As can be seen from these tables, the shutdown CDF is estimated at 5.56E-09/yr.

The top 200 cutsets for the shutdown CDF are provided in Table 16.6-3.

The risk importance measures for the shutdown CDF are provided in Table 16.6-4.

All evaluated shutdown core damage events are assumed to result in a large release because of the potential for the containment being open during the outage. CCFP is not affected because the containment is not being used as a mitigating system during shutdown.

Maintenance Unavailability Sensitivity

As discussed in Section 16.5, maintenance unavailability was addressed in this analysis be maintaining the maintenance unavailability basic events in the fault trees for all systems, and by adding additional maintenance unavailability basic event probabilities for selected systems.

Quantification sensitivities were performed to address the impact of an alternative modeling approach for treating shutdown maintenance activities. These sensitivities assume a particular train of equipment (Train B selected) is out of service with a 1.0 probability during the entire outage, and the maintenance terms for all other systems are maintained at their shutdown PRA base values.

Multiple sensitivity cases are quantified and the results are summarized in Table 16.6-5. It shows that ESBWR shutdown risk is insensitive to individual system trains being taken out of service. If multiple systems taken out of service for test and maintenance at the same time during shutdown, it has the potential to influence the CDF results, but it is expected that the durations of these combinations would be short and under control of the license holder's outage risk management program. These types of conditions could be searched and minimized in the shutdown program at the plant. The sensitivity to the change in CDF due to multiple systems taken out of service at the same time for test and maintenance during shutdown is also shown in Table 16.6-5. The CDF values are well away from any risk significance threshold.

16.7 INSIGHTS FROM **SHUTDOWN** PRA

By far, the greatest contribution to shutdown risk comes from breaks in lines connected to the vessel below TAF that occur in Mode 6. In this mode, the lower drywell equipment hatch or personnel hatch is likely to be open to facilitate work in the lower drywell. Although the frequency of these events is very low, there is only one method for mitigation – manual closure of the hatch(es).

In order to minimize the risk from these scenarios, refueling outages must be conducted in a judicious manner. Whenever the hatches are open, procedures shall require personnel to be available and in close proximity to the hatches, with the purpose of providing fast closure of the containment in the event of a water leak. Other measures can be taken, including temporary installation of equipment to aid in closing the hatch or to minimize the flooding rate in the lower drywell.

The next largest contribution to shutdown risk is due to loss of preferred power (LOPP) initiated scenarios.

The contribution from LOPP initiated scenarios is due in part to the need for electric power for alignment of FPS injection to the RPV. Electric power is required to reposition a valve in the injection piping. However, LOPP scenarios are slowly developing events because of the mass of water that must boil away prior to core uncovery and damage. Thermal-hydraulic calculations show that the core will not begin to uncover before approximately 23 hours following the loss of power, allowing significant time for recovery of offsite power that will enable the operators to provide injection to reflood the core.

16.8 CONCLUSIONS

The main conclusion that can be drawn from the ESBWR shutdown risk analysis is that the ESBWR containment and reactor well systems provide a robust, passive means for preventing shutdown core damage events. The key risk insight is that shutdown process should provide assurance that the equipment and personnel hatches in the lower drywell can be isolated in the event of a leak.

16.9 REFERENCES

- 16-1. Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1, NUREG/CR-6143, Vol. 2, Part 1 A, June 1994
- **16-2.** Evaluation of Loss of Off-site Power Events at NPPs: 1980-1996, NUREG/CR-5496, November 1996
- 16-3. Residual Heat Removal Experience Review and Safety Analysis: Boiling Water Reactors, NSAC-88, March 1986
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- 16-5. Operational Data Analysis of Shutdown and Low Power Licensee Event Reports, AEOD/S93-05, April 1993
- 16-6. Residual Heat Removal Experience Review and Safety Analysis: Pressurized Water Reactors, NSAC-52, January 1983
- 16-7. ABWR Standard Safety Analysis Report, 23A6100, August 1996

Shutdown Initiating Events Challenging Critical Safety Functions

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Table 16.3-2

ESBWR Shutdown PRA Initiating Event Types

Notes:

- (1) RWCU pump CCF failure to run on a per hour basis.
- (2) NUREG/CR-5496 (Reference 16-2) shutdown loss of offsite power frequency on a per hour basis.
- (3) Loss of PSWS initiator frequency (refer to Table 2.3-3) on a per hour basis.
- (4) Shutdown LOCA frequencies based on at-power RCS line break frequencies (refer to Table 2.3-1), and reduced by a factor of 10 to reflect the low pressure and temperature during shutdown conditions. Expressed on a per hour basis.
- *(5)* Based on lines 'g' of Table 2.3-1.
- (6) Based on lines 'd' of Table 2.3-1.
- (7) Based on lines 'e', 'f', 'gl', 'i', and 'j2' of Table 2.3-1.
- (8) Based on lines 'h' of Table 2.3-1.
- (9) Based on lines 'jl' of Table 2.3-1.

Table 16.3-3

ESBWR Shutdown PRA Initiating Event Frequencies

Notes:

- (1) As discussed in Section 16.2, the time in each Operating Mode is assumed to be as follows: Mode 5, 238 hrs; Mode 6-Unflooded, 59 hrs; and Mode 6-Flooded, 241 hrs.
- (2) The shutdown initiating event frequencies per year are calculated using the hourly frequencies of Table 16.3-1 and multiplying by the hours in the corresponding Operating Mode. An additional 0.5 factor is applied given that shutdown is expected to occur every two years.
Table 16.3-4

BWR and PWR Loss of Running RHR Pump Events During Shutdown

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Table 16.3-5

Recovery Actions Failure Probabilities

Table **16.3-6**

Time Available to Close Lower DW Hatches

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Note:

(1) This is the time available to close the lower DW hatches (if open) before flood level in containment reaches the bottom elevation of the hatch opening.

Table **16.5-1**

System Maintenance Unavailabilities During Shutdown

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Table 16.5-1

System Maintenance Unavailabilities During Shutdown

Table **16.6-1**

Shutdown **CDF by** Initiating Event and **by** Mode of Operation

Note:

(1) No cutsets survived the quantification truncation limit of 1E-14/yr.

Table 16.6-2

Shutdown CDF by Accident Class

Notes:

(1) No cutsets survived the quantification truncation limit of 1E-14/yr.

(2) During Mode 6, when the containment is open, sequences leading to core damage are designated as CD V.

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Table 16.6-3

Internal Events Shutdown PRA Cutset Report

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Table 16.6-3

Internal Events Shutdown PRA Cutset Report

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Table 16.6-3

Internal Events Shutdown PRA Cutset Report

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Table **16.6-3**

Internal Events Shutdown PRA Cutset Report

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Table **16.6-3**

Internal Events Shutdown PRA Cutset Report

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Table 16.6-3

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Table 16.6-3

Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Table **16.6-3**

Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Internal Events Shutdown PRA Cutset Report

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Table 16.64

Internal Events Shutdown PRA Importance Measure Report

16.9-43

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 $\mathcal{O}(2\pi\log n)$

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Table 16.6-4

Internal Events Shutdown PRA Importance Measure Report

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Table 16.6-4

Internal Events Shutdown PRA Importance Measure Report

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Table 16.6-4

Internal Events Shutdown PRA Importance Measure Report

16.9-46

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Internal Events Shutdown PRA Importance Measure Report

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Internal Events Shutdown PRA Importance Measure Report

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Internal Events Shutdown PRA Importance Measure Report

Table 16.6-4

Internal Events Shutdown PRA Importance Measure Report

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Table 16.6-5

Maintenance Sensitivity Quantifications for Shutdown PRA

Notes:

(1) Equipment assumed out of service is assigned maintenance basic probabilities of 1.0. The maintenance terms for all other equipment are maintained at their shutdown PRA base values.

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(2) The equipment out of service assumptions in these sensitivities are applied to all shutdown modes.

Figure 16.2-1. ESBWR Refuel Outage Plan for Shutdown PRA

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Figure 16.3-1. RWCU/SDCS Recovery Probability (Cumulative)

Figure 16.3-2. "Plant-Centered" LOPP Recovery Probability (Cumulative)

16.9-54

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Figure 16.3-3. "External" LOOP Recovery Probability (Cumulative)

16.9-55

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Figure 16.4-1a. Loss of Both RWCU/SDCS Trains (Mode 5) - RWCU5A

16.9-56

Figure 16.4-lb. Loss of Both RWCU/SDCS Trains (Mode 5) - RWCU5B

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Figure 16.4-2. Loss of Both RWCU/SDCS Trains (Mode 6-Unflooded) - RWCU6

16.9-58

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 $\label{eq:3} \mathcal{L}_{\text{max}} = \mathcal{L}_{\text{max}} + \mathcal{L}_{\text{max}}$

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Figure 16.4-3a. Loss of Preferred Power (Mode 5) - LOPP5A

16.9-59

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Figure 16.4-3b. Loss of Preferred Power (Mode 5) - LOPP5B

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Figure 16.4-4. Loss of Preferred Power (Mode 6-Unflooded) - LOPP6

16.9-61

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Figure 16.4-5a. Loss of RCCWS/PSWS (Mode 5) - SW5A

16.9-62

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Figure **16.4-5b.** Loss of RCCWS/PSWS (Mode **5)** -SW5B

16.9-63

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Figure 16.4-6. Loss of RCCWS/PSWS (Mode 6-Unflooded) - SW6

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Figure 16.4-7. LOCA in GDCS Line (Mode 6-Unflooded) - LGDCS

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Figure 16.4-8. LOCA in FW-A (Mode 6-Unflooded) - LFWA

16.9-66

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Figure 16.4-9. LOCA Other Than FW or GDCS (Mode 6-Unflooded) - LOTHER

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Figure 16.4-10. LOCA Below TAF in RWCU Drain Lines (Mode 6-Unflooded) - LBTAF1

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Figure 16.4-11. LOCA Below TAF in Instrument Lines (Mode 6-Unflooded) - LBTAF2

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Figure 16.4-12. LOCA Below TAF in RWCU/SDC Drain Lines (Mode 6-Flooded) - LBTAF1F

16.9-70

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Figure 16.4-13. LOCA Below TAF in Instrument Lines (Mode 6-Flooded) - LBTAF2F

16.9-71

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