Probabilistic Risk Assessment for the Transition Phase of Decommissioning of BWR-6 Mark-III Nuclear Power Plant

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Abstract: We discuss the specific risk significance in the extended pre-defueled phase on the basis of the refueling mode operation of a boiling water reactor (BWR) in the decommissioning process, especially under the condition that all spent fuels are still in Reactor Pressure Vessel (RPV). The issue of full-core discharge capability after permanent shutdown during the pre-defueling phase motivated this study on the system risks using a reference plant design of two-unit/BWR-6/Mark-III.

The incorporative influences of the reactor core and the spent fuel pool (SFP) are investigated, and the methodology is systematically developed via the two primary configurations. In this study, the initial scenario of 178-days after permanent shutdown is chosen and consequently two configurations can be defined, the reactor core and the SFP, respectively.

Under the situation of decreasing decay heats after shutdown, the dependence between human operations is assumed to be zero. Furthermore, the WinNUPRA software package is used for the fuel damage sequence quantification.

The relevant initiating events are shown as follows, loss of offsite power (LOOP), loss of coolant accident (LOCA), loss of cooling (LOC), Residual Heat Removal System (RHR) heat exchange tube rupture induced LOCAs (LOR), flow diversion LOCA to suppression pool (SP) via the RHR system (H1), etc.

The quantification results show that H1 and LOR have 75% and 20% contributions to the fuel uncovery frequency (FUF) for the core configuration. On the other hand, the LOOP event contributes the majority of the risk, accounting for 98% of the FUF, for the SFP configuration.

Keywords: PRA, decommissioning, Pre-defueled

1. INTRODUCTION

The pre-defueled (PD) phase is a stage of the transition before decontamination and dismantling stage. The PD phase is similar as refueling outage during low power and shutdown(LPSD). The focus of the study is restricted to fuel damage accident sequences and risk profiles for the reactor and spent fuel pool (SFP) caused by "internal events" which defined by ASME/ANS standard [2][3] under the pre-defueled condition, therefore the internal fire, flooding, seismic, high wind and so on will be considered in the future investment.

There are two models in this study, one is the last cycle of fuel in reactor pressure vessel, and another is the total spent fuel in the spent fuel pool stand on the fuel storage building. The reference type of the nuclear power plant is two-unit/BWR-6/Mark-III. In the period of the PD phase, there is Maintenance Surveillance Cycle(MSC) for one cycle every 18 months, and the MCS is about one term every 3 months. In the PD phase normal operation (18 months), there is no movement of fuel, and the Transfer Canal is isolated by manual valve (which means upper pool and spent fuel pool are separated).

2. METHODOLOGY FOR REACTOR CORE

The Probabilistic Risk Assessment (PRA) model is according with ASME/ANS standard [2][3]. With the unique Plant Operating State(POS) just like refueling outage without fuel transfer and Emergency Core Cooling Systems(ECCSs) maintain in shutdown status. We study internal events with ANS standard [3] part 3: Requirements For Internal Events LPSD PRA. There are 9 technical elements: plant operating state analysis(LPOS), initiating events analysis (LIE), accident sequence analysis (LAS), success criteria (LSC), systems analysis (LSY), human reliability analysis (LHR), data analysis (LDA), quantification analysis (LQU), and LERF analysis (LLE). We will discuss those technical elements one by one and distinguish the difference between refueling outage and PD phase.

2.1. Plant Operating State

As discusses in NUREG/CR-6143[4], a Plant Operating State is defined as "a plant condition for which the status of plant systems (operating, standby unavailable) can be specified which sufficient accuracy to model subsequent accident events." A POS is not identical to a Mode or Operating Condition as define in the technical specifications; however, POSs are defined based on Operating Conditions. The technical specifications define Mode or Operating Conditions (OCs) as follows.

(1) OC 1, Power Operation: Mode Switch in Run, any Temperature
(2) OC 2, Startup: Mode Switch in Startup/Hot Standby, any Temperature
(3) OC 3, Hot Shutdown: Mode Switch in Shutdown, Temperature Greater than 200°F
(4) OC 4, Cold Shutdown: Mode Switch in Shutdown, Temperature 200°F or Lower
(5) OC 5, Refueling: Fuel in Vessel with head Detensioned or Removed, Mode Switch in Shutdown or Refuel, temperature 140°F or Lower

The NUREG/CR-6143 provides a description of the process used to identify and characterize a POS, and it discusses all the POSs analyzed in the screening study [Whitehead, et al.,1991][12]. Using the OCs as a start point, the following seven POSs were defined:

(1) POS 1 consisting of: OC 1 and OC 2 with pressure at rated conditions (about 1000 psig) and thermal power no greater than 15%.
(2) POS 2 consisting of: OC 3 from rated pressure to 500 psig
(3) POS 3 consisting of: OC 3 from 500psig to where RHR/SDC in initiated (about 100 psig)
(4) POS 4 consisting of: OC 3 with the unit on RHR/SDC
(5) POS 5 consisting of: OC 4 (T≦200°F) and OC 5 until the vessel head is off
(6) POS 6 consisting of: OC 5 with the head off and level raised to the steam lines
(7) POS 7 consisting of: OC 5 with the head off, the upper pool filled, and the refueling transfer tube open.

The differences between the LPSD and PD phase are operating states. In the LPSD, there are many POSs mentioned above. The POSs duration are couple hours to several days. The duration of time with refueling and PD phase is quite different. And with refueling outage (**POS 5**), there are many systems will be maintained, On the other side, there is no maintenances in normal operation with PD phase. There is fuel transfer in POS5 converse to no fuel transfer in PD phase normal operating state. In our study, we will focus on the differences between the POS 5 and PD phase.

2.2. Initiating Events analysis

The initiating events with LPSD were reference from NUREG/CR-6143. As go through with initiating events which discuss in NUREG/CR-6143, we assessed all the initiating events and reevaluate the failure mode and effects analysis (FMEA) such as electrical, air, and cooling systems at reference plant. There are 4 types of initiating events at LPSD, loss of coolant accidents (LOCAs), transients, Declay Heat Removal Challenges and Special Events.

The values of initiating events frequencies, these rates are multiplied by fraction of time that the plant is in the POS for which the initiating event applies. Using the same methodology to assess the reference plant's initiating events as Table 1.

Table1. Initiating events list in reference plant in PD phase as normal operation

Initiating Events	Reference Plant PRAID	Description
Transients type:		
Transient with Main Condenser Isolation	T_{1A}	Not Applicable[1]
Transient with Main Steam Isolation Valve Isolation	T_{1B}	Not Applicable[1]
Transient with Main Steam not Isolated	T_2	Not Applicable[1]
Loss of Offsite Power	T_{3G}, T_{3P}, T_{3S}, T_{3W}	NUREG/CR-6890[5]
Transient with Safety Relief Valve Open	T4	Not Applicable[1]
Transient involving loss of Feedwater	T5	Not Applicable[1]
LOCA type:		
Large LOCA at low pressure	A	relevant
intermediate LOCA at low pressure	S_1	relevant
Small LOCA at low pressure	S_2	relevant
Interfacing LOCA	V	Not Applicable

Initiating Events	Reference Plant PRAID	Description
Vessel Rupture	R	relevant
Diversion To Suppression Pool via RHR	H_1	relevant
Diversion To Condenser via RWCU	H_2	Not Applicable
LOCA in connected system(RCIC)	J_1	Not Applicable
LOCA in connected system (RHR)(small)	J_{21}	relevant and assess
LOCA in connected system (RHR)(Rupture)	J_{22}	relevant and assess
Test and maintenance induced LOCA	K	Not Applicable
Decay Heat Removal Challenges		
Isolation of SDC running loop	E_{1B}	Not Applicable[2]
Isolation of SDC common suction line	E_{1T}	Not Applicable[2]
Loss of operating RHR shutdown system	E_{2B}	Not Applicable[2]
Loss of SFPCS	E_{2F}	Not Applicable[2]
Loss of SDC common suction line	E_{2T}	Not Applicable[2]
Loss of operating RHR shutdown system (for long time)	LOR	Add and assess[2]
Special Events		
Rod Withdrawal Error	T_{4A}	Not Applicable
Refueling Accident	T_{4B}	Not Applicable
Instability Event	T_{4C}	Not Applicable
Transient based on Support Systems		
Loss of 4.16kV Bus 1A3	T_{A3}	relevant
Loss of 4.16kV Bus 1A4	T_{A4}	relevant
Loss of 125 VDC: 1RDC	T_{DC}	relevant
Loss of 125 VDC: 1GDD	T_{DD}	relevant
Loss of 480V MCC: 1C3D	T_{3D}	relevant
Loss of 480V MCC: 1C4C	T_{4C}	relevant
Loss of Essential Service Water	T_{CE}	relevant
Loss of Normal Chilled Water	TCH	relevant
Loss of Nuclear Component Cooling Water	TCN	relevant
Loss of Instrument Air	T_{IA}	Not Applicable[3]
Inadvertent Pressurization via Spurious HPCS Actuation	T_{HP}	Not Applicable[3]
Loss of Makeup	T_{LM}	Not Applicable[3]
Inadvertent Overfill via LPCS or LPCI	T_{OF}	Not Applicable[3]
Inadvertent Overpressurization	T_{OP}	Not Applicable[3]
Loss of Recirculation Pump	T_{RPT}	Not Applicable[3]
Inadvertent Open Relief Valve (IORV) at Shutdown	T_{RV}	Not Applicable[3]

1: Transient Events only happen at power operation.
2: Decay Heat Removal challenges as E1B and E1T could recover in two hours, And E2B, E2T and E2F could transfer to another train, As Decay Heat Removal challenges could recover operating train or transfer to another train in a short time, we could screen the initiating events in PD phase. Loss of RHR for a long time (more than 2 hours) is new adding initiating event in PD phase.
3: Each supporting system event in the reference plant has different issue at LPSD. As we discussing in the PD phase, we used FMEA.

2.3. Success Criteria

The target of fuel damage at power operation, we defined fuel temperature over 2200 °F. But in the conserved definition we used in the PD phase, we defined as fuel uncover with coolant inventory. As we defined the success criteria of fuel, we could make sure which system functions need to work. The functions need to work depends on NUREG-2300[6], include the reactivity control, reactor vessel overpressure control, reactor vessel water inventory control, long-term heat removal. As reference plant in PD phase, reactivity is controlled by control rod (all in) and there is safety margin analysis to make sure the reactivity is small enough. Reactor vessel pressure is not an issue when the vessel head is opening. The inventory control and the long-term heat removal still need consideration. The success criteria analysis is evaluated the day started permanent shutdown using a realistic thermal-hydraulics calculation on the heat removal capabilities of the heat exchangers for the essential cooling systems using the reference plant's technical specifications. The reference plant's core decay heat after shutdown about half year is 1.717 MW which is calculated by ASB 9-2[7]. Depends on the core decay heat and loss of cooling system, we can calculate the time before coolant inventory boiling about 66.7 hours, and coolant inventory sustain heating, evaporate to the cavity edge about 567 hours. Compare with decay heat, success criteria with the long term heat removal systems are considering as table 2.

Table 2. long term heat removal systems

System	Flow rate	Capacity*
Spent fuel pool cooling and cleanup system	1300 gpm (single train)	3.985 MW
Residual Heat Removal System	5050 gpm (single train)	40.07 MW

* from reference plant FSAR table 9.1-1 and table 9.1-2.

Spent fuel pool cooling and cleanup system (SFPCCS) and Residual Heat Removal System (RHR), those two systems could do long term heat removal.

2.4. Accident Sequence analysis

For each initiating event could lead to the fuel uncover, the event progress call accident sequence analysis. The accident sequence analysis is assessed by event tree. The development of event trees is conducted through the linking of fault trees. In this approach, the event tree heading distinguishes by function assignment. A function can be a single system or a combination of systems performing the same function. If the function failure with multiple systems, it is termed a "function fault tree". Separate system fault trees are established to evaluate the "function fault tree" causes of each system underneath. The construction of event trees is primarily based on success criteria and is conducted through iterative discussions with power plant operators. During the discussion process of event trees, the power plant's Procedures are also consulted to appropriately simulate the power plant systems and operator responses that could influence the direction of accident sequences.

Assumptions Related to Event Tree Analysis as follow:

(1) The mission time considered for the system function is 24 hours..
(2) Repair of equipment hardware failures is taken into account.
(3) Transitions between different POSs are not considered in the event tree analysis.
(4) Decay heat of POS is calculated based on entry time.
(5) Steam-driven system is not considered as available system.
(6) Primary containment is not isolated.
(7) Spent Fuel Cooling: In the PD phase of the reference plant, based on the operational status and system allocation arrangements, the cooling setup for the reactor is configured to operate with a train of RHR or a train of SFPCCS.
(8) In the POS of PD phase, there are some assumptions with low decay heat.
   a. For certain initiating events, such as failure of the RHR support system, the time required for different POSs to evolve to the extent requiring human intervention varies depending on the decay heat and water level. During these times, as long as these support systems are restored by operators in a timely manner, no adverse effects occur. Evaluation is based on the time to boiling of the reactor water.
   b. During the PD phase, the decay heat is lower, and the plant's ventilation and air conditioning system can assist in removing heat. Therefore, the assumption that the suppression pool needs to be successfully cooled is not considered as long as spent fuel is covered by water.
   c. In this analysis, the allowable time window for electrical power restoration consideration is defined by incorporating the probability of recovery of initiating events before water boiling, after the power restoration time is subjected to data distribution.
   d. The damage caused by steam on instrument is not considered. In the PD phase, due to low decay heat is lower and the presence of a ventilation and air conditioning system in the plant building that can help dissipate heat, there is no sudden generation of large amounts of steam.

Classify and explain each generic event tree based on accidents involving transient (non-LOCA) and LOCA initiating events.

Generic transient event tree

<u>Heading: Initialing Event Recovery</u>

After the initiating event occurs, the time from the state of pool water at ambient temperature, with cavity fully immersed in water, to the time when the reactor coolant is heated by decay heat to boiling (LOOP, RHR and SFPCCS have enough time to recover), or with the added time for the water level to drop to the bottom of the reactor cavity, can all be considered as the time for system recovery (RHR only). Based on thermal hydraulic analysis, the pool water is estimated to boil at approximately 66.7 hours, with the water level dropping to the bottom of the cavity approximately 567 hours after boiling.

Heading: Shutdown Cooling / Alternate Cooling

For initiating events that cannot be recovered from, the backup train or system will be activated. The main failure scenarios considered in this heading are the RHR shutdown cooling mode or SFPCCS to operate failure.

Heading: Emergency Core Cooling System(ECCS) makeup

Once the heat removal function is lost, the residual heat from the reactor core must be removed by other cooling methods, one of which is the ECCS injection. With the reactor vessel head open and the main steam line plugs installed, the safety relief valves are unavailable, disrupting the pathway between the reactor cavity and the suppression pool. The ECCS can inject water to maintain a high water level in the reactor cavity. The main failure scenario considered in this title is the inability of the ECCS to operate automatically or manually to inject water into the reactor cavity.

Heading: Steaming function

Utilizing a lesser amount of water to maintain the fuel in a state covered by water, heat is removed through the boiling of cooling water in the fuel area. In the case of the reactor vessel being open, steam can be directly discharged through the open reactor cavity, but the negative effects of this steam emission require further assessment. Currently, in the PD phase, the decay heat of the core is lower compared to during shutdown, and the ventilation and air conditioning of the building can assist in removing heat. Therefore, as long as there is water covering the nuclear fuel, the assumption that the suppression pool needs to be successfully cooled is not considered.

Heading: Flooding function

The reactor building has entrances and exits for equipment on the 2nd and 7th floors. Currently, only the airlock doors and equipment entrances on the 7th floor are kept open, while those on the 2nd floor remain closed. The reactor and containment vessel are flooded using high-flow fire water (FRW), and the water level in the containment vessel is maintained at an appropriate height. Since fire water is sourced from outside the plant, it can generally raise the water level in the reactor cavity to its maximum. Further heat removal from the containment vessel is required until the cooling water in the reactor cavity reaches saturation temperature.

Generic LOCA event tree

Heading: Initialing Event Isolated

For initiating events related to LOCA, it is necessary to first determine the location and size of the breach and whether the location can be isolated.

Heading: Emergency Core Cooling System(ECCS) makeup

Before the water level drops to the top of the fuel, the fuel must be covered with water. If the LOCA cannot be isolated or if isolation fails, the ECCS must be manually or automatically activated to replenish water and maintain the water level.

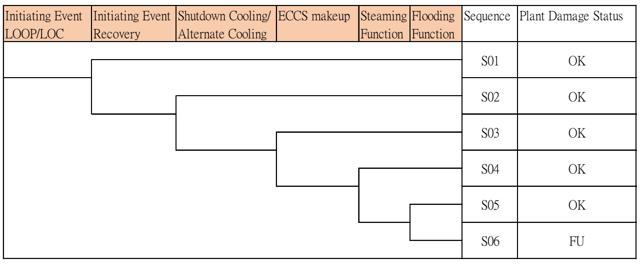

Figure 1. Generic transient event tree

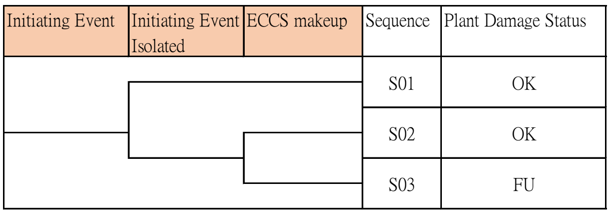

Figure 2. Generic LOCA event tree

2.5. System Analysis

The fault tree model [6] was chosen as the basic system model in this investigation. Independent random component failures and failures of direct-support systems (electric power, cooling, air) that can lead to system failure are modeled in the first stage of fault tree construction. The system fault tree must be simulated according to the decommissioning transition period. Therefore, unlike the fault tree in power operation and shutdown modes, the system fault tree needs to be updated, added, modified, or deleted. As PD phase similar with shutdown mode, we used the shutdown model fault tree. But the difference between the PD phase and shutdown model, we changed the fault trees such as: the maintenance basic event for RHR in shutdown are predictable for a long time, but the maintenance basic event in PD phase is refer to power operation in a short time in PD phase. And the safety related division I and II will maintenance in refueling outage which is not to be in the PD phase. The Power Conversion System, Automatic Depressurization System, Reactor Core Isolation Cooling(RCIC) and so on are not used in PD phase. System fault tree need to review in PD phase.

2.6. Data Analysis

This subsection describes the estimation of six types of parameter values used in the calculation of system failure probabilities and accident sequence frequencies. All the components of a particular type in the reference plant, are assumed to be characterized by a single failure rate or probability for a given failure mode.

(i) Component reliability parameters, including failure rates or probabilities of failure on demand;

(ii) Common-cause failure rates;

(iii) Frequencies of initiating events;

(iv) System unavailability due to testing or unscheduled maintenance;

(v) Recovery or repair probabilities (such as loss of RHR cooling, using data from the RHR repair records, then we calculated the time to repair RHR failure probability); and

(vi) Probabilities of special events, such as the recovery of offsite power and diesel generator.

The first three types of parameters, such as component reliability parameters, common-cause failure rates, and frequency of starting events, are developed utilizing the Bayesian approach, by introducing reference plant realistic data. On the other hand, the last three parameter types are analysed as well, such as system unavailability, recovery or repair probabilities, and probabilities of special events.

2.7. Human Reliability Analysis

The HRA's objective is to analyze each of the human actions identified at each stage of systems analysis and the accident sequence analysis, and the study of the progression from core damage to containment failure. The interaction of plant operators with equipment and procedures in the quantification of fuel uncovery frequency is the focus of this research. The methodology divided into three parts: (1) don't know what to do, (2) knowing how to do it but not having enough time to execute, and (3) making operational errors during execution. Each part is paired with the respective analysis method: CBDTM (Cause-Base Decision Tree Method), HCR (Human Cognitive Reliability) for quantifying the probability of completing actions within a limited time, and THERP (Technique for Human Error Rate Prediction) for analyzing operational errors.

Pre-Initiating Event Human Error

The probabilities relating to human errors before the initiating event include: single instrument calibration errors, multiple instrument calibration errors, and human errors such as mispositioning of valves.

Post-Initiating Event Human Error

Using the event response time provided by the thermal hydraulic analysis during the decommissioning transition phase, evaluation is conducted based on human reliability analysis methodologies.

Initiating Events Caused by Human Errors

During refueling outage, both loops of the RHR could be maintained and tested. Therefore, during the shutdown overhaul, there is a higher probability of LOCA(H1) due to human errors. However, unlike NUREG/CR-6143, based on the operational status during the decommissioning transition phase of the reference plant, operators may inadvertently open the wrong loop valve during loop switch operations, leading to reactor coolant water move to suppression pool. The data calculated H1 in NUREG/CR-6143 by data collection, but the data calculated H1 in this analysis by HRA.

2.8. Quantification and Result

We quantify the model by the fault-tree-linking method for the accident sequences are analyzed using via the WinNUPRA [9] software package. As discussing initiating events, setting the success criteria from thermal hydraulic calculation, developing the accident sequence, constructing system fault trees, giving data for model basic events and human reliability, finally we could quantify the result of fuel uncover frequency at PD phase. The result of core FUF is 5.71×10^{-08}/yr. The FUF with the initiating events respective as show in table 3.

Table 3. FUF with the initiating events

Initiating Events PRAID	Initiating Events Description	FUF (/yr)	Percentage (%)
H1	Diversion To Suppression Pool via RHR	4.24×10^{-08}	74.3
LOR	Loss of RHR	1.18×10^{-08}	20.7
R	Vessel Rupture	6.47×10^{-10}	1.1
J22	LOCA in connected system (RHR)(Rupture)	6.44×10^{-10}	1.1
T3G	Loss of Offsite Power (Grid related)	4.66×10^{-10}	0.8
S1	intermediate LOCA at low pressure	4.46×10^{-10}	0.8
T3S	Loss of Offsite Power (Switchyard related)	3.78×10^{-10}	0.7
A	Large LOCA at low pressure	1.66×10^{-10}	0.3
T3P	Loss of Offsite Power (Plant Centered)	2.78×10^{-11}	<0.1
TA4	Loss of 4.16kV Bus 1A4	2.46×10^{-11}	<0.1
T3W	Loss of Offsite Power (Weather related)	1.83×10^{-11}	<0.1
TA3	Loss of 4.16kV Bus 1A3	1.25×10^{-11}	<0.1
TCE	Loss of Essential Service Water	4.32×10^{-14}	0
TCH	Loss of Normal Chilled Water	3.40×10^{-14}	0
J21	LOCA in connected system (RHR)(small)	0*	0
S2	Small LOCA at low pressure	0*	0

*0 means the quantify value small than screen value which means by 10^{-14}/yr

The major consequence form event tree is H1S04 which could see the figure 3. The major consequence is diversion LOCA through RHR to suppression pool. Then water level down to the L-3 which automatic trigger the RHR suction valves isolated success. As reactor water level is low on that time, operator manual start ECCS to make up water injection to the cavity failure. And operator still fail to start and run condensate storage tank transfer pump to keep steaming function. Finally operator fail to use fire water flooding cavity and fuel is going to uncover.

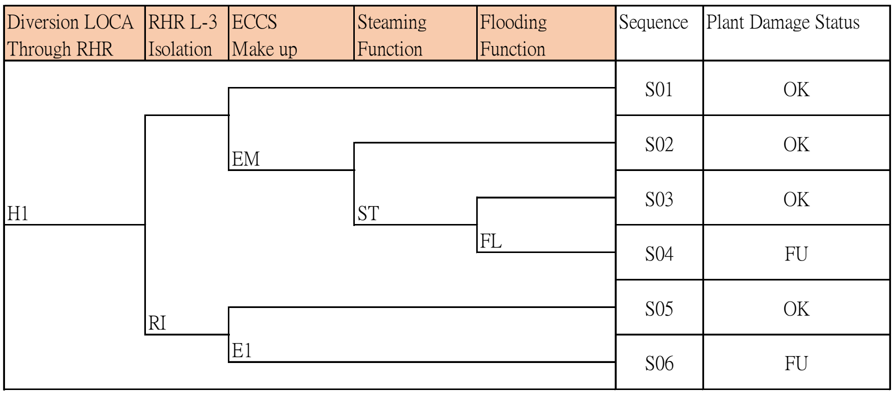

Figure 3. H1 Diversion LOCA Through RHR

The top one minimum cutsets of reactor core in PD phase is AAA-H1×HR-STEMFL: means that initiating event H1 LOCA occurs and operator failure to start and run ECCS injection function (EM), steaming function (ST), and flooding function (FL). Human reliability analysis with consequence has discuss dependency and cutoff value. With reassessment the H1 LOCA initiating event which discuss in 2.7 and human reliability analysis with different time window in PD phase. There is two order less than shutdown model (5.5×10^{-06}). The minimum cutsets is somewhat similar to shutdown model, but the maintenance terms are total different meaning in two models. Compare with NUREG/CR-6143 report, the POS 5 is 59.5% and POS 6 is 37.8% of total CDF. CDF at POS 5(during refueling outage) is 2.1×10^{-6} per year. The major contributor is large LOCA about 23.2%.

3. METHODOLOGY FOR SPENT FUEL IN SPENT FUEL POOL

The Probabilistic Risk Assessment(PRA) model with spent fuel in spent fuel pool is according with NUREG-1738[10]. Although the core fuel has not yet been removed, the spent fuel pool remains an independent system,. The decay heat of the spent fuel in the spent fuel pool can be calculated separately. Additionally, according to the industry decommissioning commitments(IDCS) and staff decommissioning assumptions (SDAs) in NUREG-1738, the reference plant also meets the relevant assumption conditions.

3.1. Initiating Events analysis

According to the NUREG-1738, the spent fuel pool has 9 initiating events list blow:
(1) Seismic event, (2) Cask drop, (3) Loss of offsite power initiated by severe weather, (4) Loss of offsite power from plant-centered and grid-related events, (5) Internal fire, (6) Loss of pool cooling, (7) Loss of coolant inventory, (8) Aircraft impact,(9) Tornado missile.
Items (3) and (4) can be treated the same as LOOP during a general shutdown model; loss of cooling and LOCA are included in the model for discussion. Other events such as internal fire, seismic, aircraft impact, and tornado missile are external events and are not within the scope of this report. Additionally, dry storage operations are not covered by this report; therefore, the handling of casks during dry storage is also not within the scope of this report. The spent fuel pool is discussed in two main categories: transient events and LOCA, as follows:
Transient Events for the Spent Fuel Pool
The only initiating event for transient events is the LOOP. Therefore, if there is LOOP, the spent fuel pool will have no available power, causing the SFPCCS, as well as the subsequent water supply systems, to be inoperable. Without further subsequent actions, the spent fuel in the spent fuel pool could become exposed and potentially melt. Therefore, this initiating event needs to be included for subsequent discussion.
Loss of Cooling Initiating Events for the Spent Fuel Pool
A failure mode and effects analysis (FMEA) is conducted for the plant's support systems, including power, cooling water, plant ventilation, and instrumentation and control systems. LOOP is still the generic initiating events with the transient. It will cause the Loss of cooling initiating events for spent fuel pool. The remaining in-plant power sources, including the 4.16 kV bus, the 480 V load center, and the 480 V motor control center (MCC), all have dual configurations, ensuring that any single failure will not result in a complete loss of cooling functionality for the spent fuel pool. Additionally, a fifth diesel generator is kept as a backup power source for the loss of offsite power. Furthermore, a mobile diesel generator is configured for one unit of the plant. The failure of these two diesel generators will not initiate an event. Therefore, the power-related systems will not cause an initiating event due to a single failure.
In the ventilation of the plant, the fuel building containing the cooling system, exhaust system, and ventilation system for handling tools within the fuel building are included. The cooling system of the fuel building is used to control the temperature of the building during normal operation, while the exhaust system of the fuel building filters the air in the event of an accident and maintains negative pressure in the building, ensuring that the air emitted from the building during an accident is filtered before being released to the outside. Failure of these two systems will not immediately render the equipment and systems in the building unusable, thus not immediately triggering an initiating event.
Regarding the instrumentation and control system, concerning the SFPCCS, which may result in the loss of cooling function for the spent fuel pool, this aspect should be discussed together with the initiating event of the loss of cooling for the spent fuel pool. Additionally, regarding the control station at the spent fuel pool serving as the monitoring instrumentation and control system, there may be situations where the operators at the main control room make incorrect judgments. However, since at least one shift inspection must be carried out for each shift team, and accidents involving the spent fuel pool do not require an immediate response, failures of this type of system are not considered to lead to initiating events.
Finally, during the operation of the spent fuel pool, there is a cooling water replenishment system. If a random failure of an important component within the system occurs during operation, it will trigger an initiating event of interruption in the cooling of the spent fuel pool. At this point, the operators must restore the operation of the SFPCCS or provide cooling water to the spent fuel pool from internal or external water replenishment systems before the water level in the spent fuel pool drops to the top of the stored fuel. Therefore, the systems related to the replenishment of water for the spent fuel pool will not cause initiating events.

Table 4. Initiating events list in reference plant at spent fuel pool

Initiating Events	Reference Plant PRAID	Description
Transients type:		
Loss of Offsite Power	T_{3G}, T_{3P}, T_{3S}, T_{3W}	NUREG/CR-6890[5]
FMEA type:		

Initiating Events	Reference Plant PRAID	Description
Electrical systems	T_{A3}, T_{A4}, T_{DC}..	Not Applicable
Ventilation systems	$T_{CH...}$	Not Applicable
Instrument and control system	$T_{IC...}$	Not Applicable
Coolant water makeup system	$T_{MU...}$	Not Applicable
Loss of SFPCCS	LOC	Keep

3.2. Success Criteria

The calculated decay heat for the reference plant's spent fuel pool (with 4808 fuel assemblies) is 1.704 MW which is calculated by ASB 9-2[7]. Additionally, calculations have been made for the boiling time of the spent fuel pool water under loss of cooling conditions, estimated to be approximately 59.7 hours (about 2.5 days) until the water level is below the overflow port, approximately 67.1 hours (about 2.8 days) until it is below the bottom of the vortex breaker (used for RHR system water extraction), and approximately 120.4 hours (about 5.0 days) until it reaches the level of the spent fuel pool gate. The time until the water level reaches the bottom of the spent fuel pool gate is estimated to be approximately 399.4 hours (about 16.6 days). success criteria with the long term heat removal systems are considering as the same as table 2.

3.3. Accident Sequence analysis

The accident sequence analysis at the used fuel pool has the same basic assumptions as at the RPV core. However, the accident sequence in the spent fuel pool is described as follows:

Loss of Cooling (LOC) system event tree

At the spent fuel pool, loss of coolant means loss of SFPCCS, and assume that system can not recover.

<u>Heading: RHR cooling</u>

When loss of SFPCCS and can’t be recovery, operator would start RHR spent fool cooling mode to cool the spent fuel pool before water boiling.

<u>Heading: RHR Recovery</u>

If RHR start and run failure, there is a period of time for operator to recover RHR system before water level lower that RHR suction. Here we used the plant specific data analysis for special event.

<u>Heading: Operator Recovery by Onsite sources</u>

Operator recover the water by onsite sources include the emergency (spent fuel) makeup pumps and Condensate Storage (CST) transfer pumps before water level lower than spent fuel pool gate bottom.

LOOP event trees

<u>Heading: LOOP Recovery</u>

LOOP recover before spent fuel pool water boiling. Each type of LOOP recovery data was calculated by plant specific data.

<u>Heading: SFPCCS restart and run</u>

As offsite power recover, operator restart and run the SFPCCS failure.

<u>Heading: RHR cooling</u>

As the same as the LOC heading.

<u>Heading: cooling system recovery</u>

As the same as the LOC heading.

<u>Heading: Operator Recovery By Onsite sources</u>

As the same as the LOC heading. The different between Loop Recovery success or not, there are SFPCCS and CST transfer pumps usable or not.

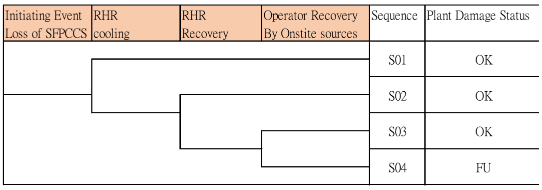

Figure 4. Loss of Cooling system event tree

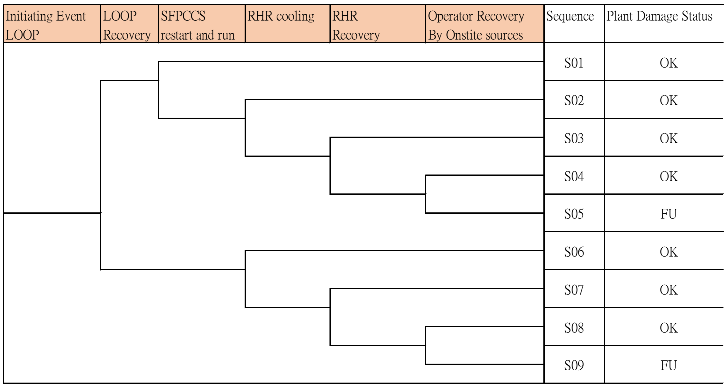

Figure 5. Generic transient event tree

3.4. System Analysis

There is no spent fuel pool model in the reference plant, we assess a new model for spent fuel pool. Therefore, we add analysis SFPCCS, spent fuel emergency makeup, CST transfer pumps to spent, RHR cooling for spent fuel pool, and associate supporting systems, etc.

3.5. Data Analysis

Among the six types of failure data, the component failure data, common cause failure data, initiating event data, and system testing and maintenance unavailability data are used as described in Section 2.6. The recovery data and special data, however, are related to success criteria and are derived from thermal hydraulics analysis results or plant specific data analysis.

3.6. Human Reliability Analysis

We used the same methodology for analysis spent fuel pool model. The difference between the RPV and spent fuel pool are the thermal hydraulic calculation within time window.

3.7. Quantification and Result

We quantify the model as the same as RPV side. The result of spent fuel pool's FUF is 3.12×10^{-08}/yr. The FUF with the initiating events respective as show in table 5.

table 5. FUF with the initiating events at spent fuel pool

Initiating Events PRAID	Initiating Events Description	FUF (/yr)	Percentage (%)
T3SP	Loss of Offsite Power (Switchyard related)	1.51×10^{-08}	48.5
T3GP	Loss of Offsite Power (Grid related)	1.13×10^{-08}	36.1
T3WP	Loss of Offsite Power (Weather related)	2.35×10^{-09}	7.6
T3PP	Loss of Offsite Power (Plant Centered)	1.98×10^{-09}	6.4
LOC	Loss of cooling (SFPCCS)	4.58×10^{-10}	1.5

The top minimum cutsets of spent fuel pool in PD phase is AAA-T3SP×AAB-RECOV-S×AAR-SDR3×HR-SD7/MK1: means that initiating event T3SP occurs and offsite power not recovery, RHR spent fuel pool cooling mode failure and could not recover(AAR-SD3), and operator fails to start and run RHR system and makeup systems(HR-SD7/MK1).
The result from NUREG-1738 boil down sequences list as blow:

table 6. Frequency of Boil Down Leading to Spent Fuel Uncovery

Initiating Events Description	FUF (/yr)
Loss of Offsite Power - Severe Weather	1.1×10^{-07}
Loss of Offsite Power – plant centered and Grid related events	2.9×10^{-08}
Internal fire	2.3×10^{-08}

Initiating Events Description	FUF (/yr)
Loss of pool cooling	1.4×10^{-08}
Loss of coolant inventory	3.0×10^{-09}

The LOOP in NUREG-1738 is accounting for about 77%. LOC is accounting for about 7.7%.

4. CONCLUSION

The risk assessment model for the pre-defuel phase of decommissioning transition stage has been relatively underexplored. This report conducts a standard PRA analysis for a reference BWR-6/Mark-III plant. The analysis not only examines the nuclear fuel in the RPV but also the nuclear fuel in the spent fuel pool. Different reference reports were used for each part. The reactor core analysis refers to the NUREG/CR-6143 refueling outage during shutdown, while the spent fuel pool analysis uses the model from NUREG-1738.

The model for the reactor core differs from the shutdown mode by about two orders of magnitude. The main differences lie in the redefinition of initiating events and the fact that the available time for thermal-hydraulics analysis is much greater during the decommissioning transition phase than during a normal shutdown. As a result, the effectiveness of systems for recovering from initiating events is taken into account, and the probability of system recovery failure is lower. Human reliability analysis also shows a decrease in failure probability as the available time increases.

The risk associated with the spent fuel pool is not significantly different from that in NUREG-1738. The loss of offsite power remains the main contributor to risk.

This analysis provides a risk assessment model for the early phase of the decommissioning transition for a reference BWR-6/Mark-III plant, offering a risk profile and risk proportions for each initiating event during this phase. Although the risk during the decommissioning transition phase is low, the risk insights can inform the normal operation of the plant in PD phase.

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