A Methodology for Spent Fuel Pool Internal Events Level 1 and Level 2 PRA for APR1400

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As considerable inventories of radiological nuclides are typically stored in a spent fuel pool (SFP) near the reactor or in adjacent buildings with relatively less robust confinement structure as compared to the containment building in the case of pressurized water reactors (PWRs), there is an increasing concern for potential release to the environment under accident conditions. The Fukushima accident has also highlighted the importance of SFP risk. Although several studies used to be performed to evaluate the risk associated with SFP operation, there is no set procedure or guidance to carry out Probabilistic Risk Assessment (PRA) for spent fuel pools. This paper presents a new methodological approach that may be applied in developing the SFP PRA for internal events, and describes the process taken to conduct the SFP Level 1 and Level 2 PRA for APR1400 nuclear power plants together with assumptions made and unique aspects of SFP PRA as opposed to Reactor PRA.

I. INTRODUCTION

The purpose of this paper is to provide a general methodology for performing Spent Fuel Pool (SFP) Internal Events Level 1 and Level 2 Probabilistic Risk Assessment (PRA) for all operational modes. The term 'internal events' herein refer to those events that originate in the structures, systems, and components (SSCs) relevant to the spent fuel pool operation, and thereby, have potential for leading to fuel damage and release of radioactive materials to the outside environment.

The major assumptions for the SFP PRA that underlie this study are as follows.

- The SFP PRA will be generally carried out after the Reactor PRA has been performed. Hence, in the SFP PRA, the operating cycle associated with the SFP will be discretized on the basis of the plant operating states (POSs) defined in the Low Power and Shutdown (LPSD) PRA for the Reactor in order to take advantage of availability of the various analyses on the POSs.

- There may be potential interactions between the Reactor and SFP accident progression and risks as indicated in the recent EPRI study[1] on SFP risk assessment integration framework (Ref. 3). However, the scope of the SFP PRA discussed herein is limited to identifying only those initiating events that originate from SFP-related SSCs (due to either equipment failure or human errors), and subsequent accident scenarios.

[1] As to the SFP-Reactor interactions, the EPRI report on SFP-Reactor PRA (Ref. 3) points out that there are basically three different types of interactions between the Reactor and SFP accident progression and risks:

1) SFP events impacting the Reactor/Containment
2) Reactor/Containment events impacting the SFP
3) Common events impacting the Reactor/Containment and SFP simultaneously (e.g., loss of offsite power, electrical bus failure, seismic event, fires, internal flooding).

In PWRs the first type of interaction was judged to be a remote probability, because the PWR SFP is external to containment and generally isolated from the reactor and reactor systems. To analyze the second type, an integrated SFP/Reactor PRA must be performed, which is beyond the scope of this study. Therefore, the primary focus herein will be placed on SFP events that have no significant impact on the Reactor/Containment together with the common events impacting the Reactor/Containment and SFP simultaneously.

II. DESCRIPTION OF SFP DESIGN FACILLITY FOR APR1400

The spent fuel pool of the APR1400[2] is designed to store spent fuel assemblies within the pool in accordance with various design criteria set forth by the U.S. Nuclear Regulatory Commission (NRC), including the general design criteria of 10CFR50 Appendix A and Regulatory Guide 1.13. Hence, the spent fuel pool for APR1400 plants is generally designed as follows:

- With capability to permit appropriate periodic inspection and testing of components important to safety;
- With suitable shielding for radiation protection;
- With appropriate containment, confinement, and filtering systems;
- With a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and
- To prevent significant reduction in fuel storage coolant inventory under accident conditions.

The spent fuel pool for APR1400 is housed in the Auxiliary Building, which contains essential SFP equipment, such as:

- Spent Fuel Pool Cooling and Cleanup (FC) System
- Fuel Tube Transfer System connection between the containment and the SFP
- The pool structures, spent fuel racks, and overhead cranes are designed to Seismic Category I.

Figure 1 shows the plant layout of the APR1400 plants especially focusing on the spent fuel pool and the refueling pool. Since fuel movement is performed under water during refueling operation, maintaining water inventory is essential for safety. The figure shows the spent fuel pool, the fuel transfer tube, and the reactor pressure vessel. The fuel transfer tube penetrates the containment building, and the space between the SFP gate shown in the figure and the fuel transfer tube is called a 'fuel transfer canal'. In addition, there is a gate valve on the end of the fuel transfer tube in the auxiliary building side.

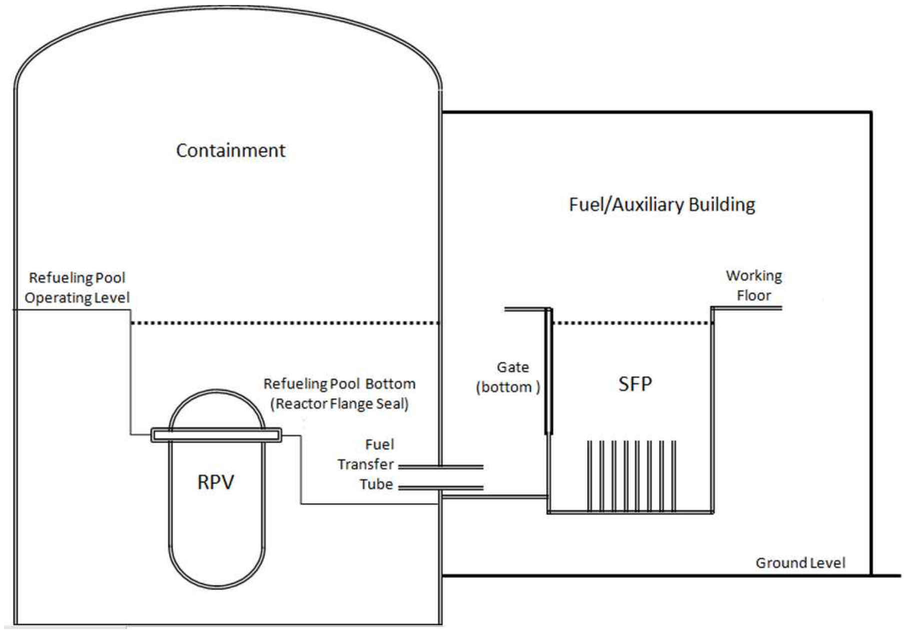

Fig. 1. APR1400 Plant Layout – Spent Fuel Pool and Refueling Pool

[2] The APR1400 (Advanced Power Reactor 1400 [MWe]) is an advanced pressurized water nuclear reactor designed by Korea Electric Power Corporation (KEPCO). Currently, there is one unit in operation (i.e., Shin Kori unit 3), and several units (including four units in the UAE at Barakah) are being under construction.

The spent fuel pool is normally in an isolated state from other nearby facilities such as fuel transfer tube or cask loading pit as shown in the Figure 2. It will be open and hydraulically connected to the refueling pool only during refueling operation. Figure 2 depicts the spent fuel handling area consisting of three areas as the spent fuel pool, the fuel transfer canal, and the spent fuel cask loading pit.

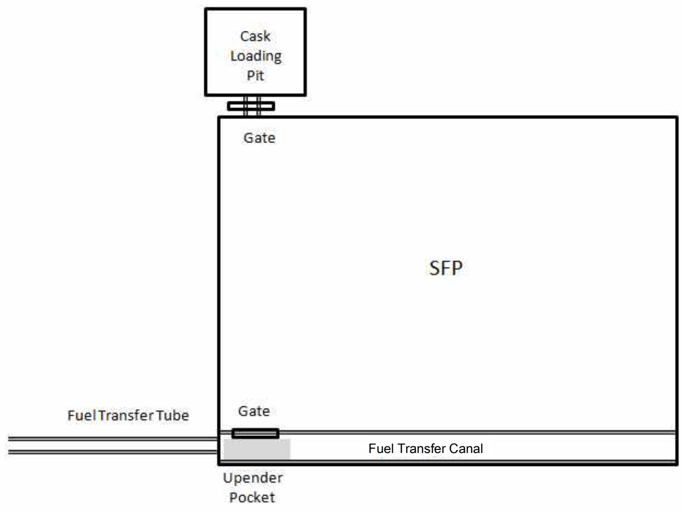

Fig. 2. APR1400 Spent Fuel Pool with Connected Facilities

This study focuses on evaluating the risk associated with the used nuclear fuel contained in the spent fuel pool, as it has significant risk implications due to the large amount of radiological materials or fission products of the fuel assemblies. Storage of fuel assemblies in the spent fuel pool is based on a two-region-storage concept. Total capacity of Region I racks is designed for one full core and one refueling batch. Total capacity of Region II racks is designed for spent fuel assemblies generated for as-designed storage capacity (e.g., 20 years). Therefore, the total spent fuel pool capacity is limited to no more than as-designed maximum fuel assemblies. This design concept of the spent fuel pool will be used in evaluating the thermal hydraulic behaviors in the case of abnormal or accident conditions after making a conservative assumption on the amount of used nuclear fuel in the SFP.

The safety function of removing decay heat generated by the used nuclear fuel is carried out by the Spent Fuel Pool Cooling and Cleanup (FC) System. In addition to the FC system, several makeup sources are available to compensate for evaporative losses or leakage in the pool in case of abnormal or accident situations.

III. APPROACH FOR DEVELOPING THE SFP LEVEL 1 AND LEVEN 2 PRA FOR INTERNAL EVENTS

Ten major technical tasks are defined for the SFP internal events Level 1 and Level 2 PRA for APR1400: 1) Operating Cycle Phases (OCP[3]) Development; 2) Initiating Event Analysis; 3) Accident Sequence Analysis; 4) Success Criteria Analysis; 5) System Analysis; 6) Human Reliability Analysis; 7) Data Analysis; 8) Fuel Damage Frequency Quantification; 9) Spent Fuel Degradation Analysis; and 10) Large Release Frequency Analysis. These tasks are conducted focusing on the internally initiated events in the case of the SFP Internal Events Level 1 / Level 2 PRA, and each task is separately discussed below.

[3] The term 'operating cycle phase (OCP)', which was used in the Level 3 Technical Analysis Approach Plan of the U.S. NRC (Ref. 2), is adopted herein to discretize various configurations of the spent fuel pool and SFP-related structures, systems, and components (SSCs) during the operating cycle of a nuclear power facility. Discretization of the operating cycle into a set of quasi-steady operating cycle phases makes performance of a SFP PRA for the operating cycle practically feasible by allowing logic modeling and necessary calculations to be done using a single decay heat and specific configuration for each phase.

III.A. Task 1: Operating Cycle Phase Development

The objective of the operating cycle phases (OCPs) development is to make discretization of the operating cycle of a nuclear power plant into quasi-steady operating cycle phases, akin to the POSs used in the Low Power and Shutdown (LPSD) PRA for reactor. The overall operating cycle of a nuclear power plant is covered by the defined set of OCPs. The configuration of the structures, systems, and components (SSCs) relevant to SFP safety (e.g., status of fuel transfer tube, fuel pool gate, diesel generator, fuel/auxiliary building venting) and associated physical characteristics (e.g., water level in the RPV refueling cavity; hydraulic coupling of the SFP, the fuel transfer canal, and the RPV refueling cavity) are clearly identified and characterized for each OCP. The major risk contributors of the SFP operation can be adequately found by performing the SFP PRA for each OCP separately, and then, integrating the PRA results.

III.A.1. Use of Plant Operating States Defined in the LPSD PRA

In the LPSD PRA for a nuclear power reactor, the various states that the plant encounters during the low power and shutdown conditions are discretized into a set of several plant operating states (POSs) to facilitate the PRA performance. For instance, 15 POSs were defined in the LPSD PRA for Surry Nuclear Power Plant by BNL (Ref. 4). On the other hand, 13 POSs were defined in the pilot SFP-Reactor PRA by EPRI (Ref. 3).

Since the definition and analysis of plant operating states is one of the most important elements in the LPSD PRA, considerable analysis is typically performed for the defined POSs, including: 1) identification of specific plant configurations (e.g., primary system water level; hydraulic coupling of the SFP and reactor refueling cavity due to opening of the fuel transfer canal; status of RPV head, containment, reactor cavity, and fuel pool gate); 2) estimation of the duration of each POS based on a representative plant outage; and 3) investigation on the safety function requirements success criteria along with the availability of equipment potentially affecting LPSD risks (e.g., shutdown cooling system, fuel building venting, diesel generator, unit auxiliary transformer) for each POS.

Therefore, the operating cycle phases for the SFP PRA may be defined based on the plant operating states as determined in the LPSD PRA, provided that the results of the LPSD PRA are available before performing the SFP PRA. In order to make use of the POS-specific information, OCPs will be defined in such a manner that several POSs are merged into an OCP.

III.A.2. Special Considerations in SFP Operating Cycle Phases Development

The more OCPs are defined, the greater modeling efforts will be needed in the PRA process although the PRA could be performed more accurately than otherwise possible. Hence, for efficient performance of the SFP PRA, an appropriate number of quasi-steady operating cycle phases need to be defined. In defining operating cycle phases for the SFP PRA, the following need to be taken into account, among others:

- The operating cycle phases must be defined in such a manner that the overall risk of the SFP operation can be appropriately evaluated.
- All the typical technical elements of a PRA (e.g., initiating event analysis, accident sequence analysis, fuel damage frequency quantification, large release frequency quantification) will be performed for each OCP, and then, the risk evaluation results for each OCP combined to result in the integrated risk insights for SFP.
- The SFP-related SSC configurations, safety function success criteria, decay heat of spent fuel, and risk-significant physical characteristics (e.g., status of the pool gate or fuel handling operations) will be determined or assumed uniquely for each OCP.
- The SFP PRA can be carried out with an initial set of OCPs, and then, further discretization of the defined OCPs may be found necessary during the process of PRA performance. In such a case, the PRA could be modified to increase resolution in the accident sequence modeling for the OCPs that have been further discretized.

III.A.3. SFP Operating Cycle Phases for APR1400

Table 1 shows an example of the OCP development for the APR1400 SFP PRA. The each OCP duration can be determined in consideration of the durations of the corresponding POSs as shown in the Table 2, which shows an example of POSs grouping in the LPSD PRA for APR1400. The total plant operating states of POSs 0 to 15 correspond to the six OCPs. As a result, the six OCPs cover the whole plant operating status including at-power and LPSD operation.

Table 1. An Example of Operating Cycle Phases Development for APR1400 SFP PRA

OCP	Corresponding POSs	Description	Modeling Considerations for SFP PRA
OCP 1	POS 0 ~ POS 3A	• From power operation to hot shutdown (Technical Specifications Mode 1 to 4) • Safety systems status almost identical to the power operation state as per Technical Specifications • Both diesel generators (DGs) are on standby in OCP1 • In POS 0 to POS 2 the main transformer is running and the standby auxiliary transformer (SAT) is on standby, while in POS 3A the main transformer is put on maintenance and the SAT is running. However, this minor difference in the plant state is ignored due to insignificant impact on the plant risk.	
OCP 2	POS 3B ~ POS 5	• Plant in cold shutdown (Tech Spec Mode 5) • Reduced RCS inventory operation performed in POS 5 • One DG is on standby, but the other in maintenance in OCP2	
OCP 3	POS 6	• Initial period of refueling (Tech Spec Mode 6) • Fill for refueling • RCS isolated from the spent fuel pool • One DG is on standby, but the other in maintenance in OCP3	• Events occurring in the reactor side (e.g., cavity seal failure) do not affect SFP safety, because both sides are separated in this refueling state.
OCP 4	POS 7 ~ POS 9	• Plant in refueling mode (Tech Spec Mode 6) • Consists of 3 different states 1) Off-Load Fuel Movement (POS 7): A full core off-loaded into the SFP in hydraulically connected condition between the reactor side and the SFP side through fuel transfer tube 2) Defueled State: A full core loaded in the SFP with the SFP isolated from the reactor side in POS 8 3) On-Load Fuel Movement: Two thirds of the full core offloaded placed back into the RPV in POS 9 • One DG is on standby, but the other in maintenance in OCP 4	• Some events occurring in the reactor side (e.g., cavity seal failure) while hydraulically connected (i.e., POSs 7 and 9) affect SFP safety because of consequential loss of coolant in the spent fuel pool.
OCP 5	POS 10	• Last period of refueling (Tech Spec Mode 6) • RCS draindown after refueling • RCS isolated from the spent fuel pool • One DG is on standby, but the other in maintenance in OCP 5	• Events occurring in the reactor side (e.g., cavity seal failure) do not affect SFP safety, because both sides are separated in this last stage of refueling.
OCP 6	POS 11 ~ POS 15	• From cold shutdown to power operation (Tech Spec Mode 5 to 1) • Both DGs are on standby in OCP 6 • The main transformer is running and the standby auxiliary transformer (SAT) is on standby	• Since maintenance was performed on the DGs during the refueling cycle, both DGs are on standby in this OCP.

Table 2. An Example of POSs Grouping in the LPSD PRA for APR1400

POS	POS Description
0	Power operation
1	Reduced power to 20% and subsequent trip
2	Cooldown with SG to 350℉
3A	Cooldown with shutdown cooling system to 210 °F
3B	Cooldown with shutdown cooling system to below 210°F
4A	Reactor Coolant System Drain-down (Pressurizer manway closed)
4B	Reactor Coolant System Drain-down (Pressurizer manway Open)
5	Reduced Inventory Operation and Nozzle Dam installation
6	Fill for refueling
7	Off-load
8	Defueled
9	On-load
10	RCS Drain-down after refueling
11	Reduced Inventory Operation with SG manway closure
12A	Refill Reactor Coolant System (Pressurizer manway open)
12B	Refill RCS (Manway closed)
13	RCS heat-up with shutdown cooling system isolation
14	RCS heat up with SG available
15	Reactor startup

III.B. Task 2: Initiating Event Analysis

The objective of the initiating event analysis is to identify and quantify those initiating events that could lead to damage of the fuel stored in the SFP. In case of internal events SFP PRA, the initiating event analysis is carried out only for internal events, i.e., the events caused by failure of equipment within the SFP or SFP-related systems boundary or by human error. The initiating event analysis for external events such as heavy load drop or earthquake shall be performed as part of the external events SFP PRA.

III.B.1. Identification of Initiating Events

Consistent with the identification of initiating evens for the reactor PRA, the initiating events for SFP PRA internal events can be identified from Master Logic Diagram, Failure Mode and Effect Analysis (FMEA), Generic initiating event list and/or operating experiences.

<u>Master Logic Diagram</u>

Master Logic Diagram (MLD) provides a systematic and logical way of analyzing the top event in a deductive manner. For instance, in the case of SFP PRA the top event of ’Offsite Release from SFP’ may be decomposed into ‘Fuel Damage’ and ‘SFP Confinement Failure’. ‘Fuel Damage’ may be further decomposed by asking a question like ‘How can the fuel be damaged?’ Then, these two intermediate events might be obtained: ‘Loss of SFP cooling’ and ‘Loss of SFP coolant inventory’. This kind of deductive process may continue to be followed until all the initiating events are identified.

Failure Mode and Effect Analysis

Failure Mode and Effect Analysis herein primarily aims at identifying failure modes of structures, systems, and components (SSCs) which, if occurring, might challenge SFP safety. In this approach, the SSCs associated with SFP are systematically reviewed to determine whether any failure mode of these SSCs (e.g. failure to operate, spurious operation, breach, disruption, collapse) could lead directly or indirectly to fuel damage.

A review of each SSC associated with SFP is also performed to identify initiating events that may result from multiple failures (e.g., common cause failure) or from system misalignments following preventive or corrective maintenance. Special attention should be given to common cause initiators, such as the failure of support systems (e.g., electric power buses, component cooling water, instrument air, room cooling, etc.).

Generic Initiating Events List

The starting point of identifying initiating events is usually the generic list of initiating events. For instance, the NRC identified nine initiating event categories in the NUREG-1738 study (Ref. 5):

1) Loss of offsite power from plant centered and grid-related events
2) Loss of offsite power from events initiated by severe weather
3) Internal fire
4) Loss of pool cooling
5) Loss of coolant inventory
6) Seismic event
7) Cask drop
8) Aircraft impact
9) Tornado missile

Among these nine initiating events, the loss of offsite power events, loss of pool cooling, and loss of coolant inventory are internal events.

On the other hand, the EPRI report on SFP-Reactor PRA (Ref. 3) presents the following categories for internal events:

1) Loss of SFP Cooling: Loss of SFP cooling system which leads to SFP heatup and fuel damage.
2) Loss of SFP Support Systems: Loss of SFP support systems (i.e., systems supporting SFP cooling and make-up) as it affects operability of the SFP cooling and makeup system.
3) SFP Draindown: SFP draindown causing a rapid loss of coolant inventory. It has previously been identified as the dominant contributor to SFP risk. The loss of SFP inventory scenarios is primarily attributed to external events such as earthquake. However, SFP draindown may be caused by internal events such as the following:
   - Failure of the SFP gates or seals when at-power due to loss of air supply to maintain pressure for the SFP gate seals.
   - A break in a penetration below the normal SFP level. Typically there are no penetrations present below SFP normal water level except the SFP gates.
   - Siphoning of the SFP due to a break in a line that extends down below the surface of the SFP. Typically suction and discharge lines on SFPs do not extend far down into the pool and may be equipped with anti-siphon devices that are usually located just below the normal SFP water level.
4) Re-criticality: PWRs maintain a subcritical margin using borated water, boral (or equivalent) plates in the SFP racks, and procedural constraints on fuel location and water temperature. Breakdown in procedural controls could lead to criticality events under some extreme circumstances (e.g., gradual boron dilution events, rapid boron dilution events, fuel loading errors into uncontrolled fuel cells). In addition to the effectiveness of administrative controls, the chance of a prompt criticality accident strongly depends on the refueling method employed. Based on the physical and procedural measures to maintain subcriticality in the SFP, no formal quantification of the frequency of criticality events in the SFP was conducted in the EPRI study for SFP-Reactor PRA.

Comparison of only internal event categories between the NRC and EPRI studies indicates that the two sets are essentially identical, because inadvertent criticality in the SFP was also considered in the NRC study although not specifically included in the nine categories mentioned above.

Operational Experience

The operational experiences of the SFPs in similar nuclear power plants are reviewed to identify any events that may need to be included in the list of initiating events. If such events are found, efforts should be made to find out the status of the SFP-related SSCs at the time of the event occurrence.

III.B.2. Initiating Events Grouping

Once a complete list of initiating events is developed for each operating cycle phase, the identified initiating events shall be grouped to facilitate an efficient but realistic estimation of fuel damage frequency. A systematic approach like Master Logic Diagrams may be employed as an aid in grouping initiating events.

Each initiating event group shall be composed of events that essentially impose the same success criteria on SFP-related SSCs. In addition, other factors such as the following should also be taken into account in grouping initiating events: a) SFP-related systems response; b) accident progression timing (including time available for mitigating systems and operator actions to be performed); and c) effect of the initiating event on the operability of mitigating systems and on the performance of plant operators.

III.B.3. Initiating Events Frequency Analysis

Initiating event frequencies are calculated accounting for relevant generic data, and plant-specific data if available. Screening criteria similar to those used in the Reactor PRA will be used in the SFP PRA to eliminate initiating events from further evaluation. Where frequency data applicable to the plant being analyzed are not readily available, the frequencies will be estimated using expert judgment or a systematic approach such as fault tree.

III.C. Task 3: Accident Sequence Analysis

The objective of accident sequence analysis is to ensure that the response of the SFP systems and operators to an initiating event is reflected in the assessment of Fuel Damage Frequency (FDF) and Large Release Frequency (LRF) in such a way that:

- Significant operator actions, mitigation systems, and phenomena that can alter sequences are appropriately included in the accident sequence model, event tree structure, and sequence definition.
- Plant-specific dependencies are reflected in the accident sequence structure.
- Success criteria are available to support the individual function successes, mission times, and time windows for operator actions for each critical safety function modeled in the accident sequences.
- End states are clearly defined to be fuel damage or successful mitigation with capability to support the Level 1 to Level 2 interface.

III.C.1. Internal Events SFP Level 1 Event Trees

Internal events Level 1 event trees shall be developed for the SFP by capturing system performance, key uncertainties in structural boundary conditions, and key operator actions up to an end state that will be defined in the SFP PRA. Based on the information on various SFP end states from the state-of-the-art review (e.g., zirconium fire end state in NUREG-1738), the end state to be used in the SFP Level 1 PRA should be first determined. In NUREG-1738, a simplified end state of an SFP water level 3 feet above the top of the fuel was used as a surrogate for onset of the zirconium fire. It was because recovery below this level, given failure to recover before reaching this level, was judged to be unlikely given the significant radiation field in and around the SFP at lowered water levels. The simplified end state was judged to provide a slightly conservative, but adequate measure to determine time frames important to human error and recovery estimates. It also greatly simplifies the analysis by eliminating the need to accurately model the complex heat transfer mechanisms and chemical reactions that are occurring in the fuel assemblies as they are being slowly uncovered. Hence, the simplified end state of an SFP water level 3 feet above the top of the fuel may be used as the end state in the SFP Level 1 PRA.

In developing the SFP Level 1 event trees, the information from the Level 1 event trees for nuclear reactor (e.g., loss of offsite power event tree) could be used to the extent as applicable to the SFP. Human actions affecting the event sequences will be explicitly identified and described to facilitate performance of the human reliability analysis (HRA) task.

III.C.2. Internal Events SFP Level 2 Event Trees

SFP Level 2 event trees cover system responses and operator actions including recovery actions taken after the surrogate condition of fuel damage or the end state defined in Level 1 SFP PRA has been reached. Severe accident phenomena that the SFP may encounter during accident conditions, such as exothermic zirconium reaction, hydrogen deflagration, and zircaloy interactions, will be taken into account in the SFP Level 2 event trees along with the fuel degradation and melting process in the presence of air as well as burnup effects.

These SFP Level 2 event trees can be developed in a similar manner as in the Reactor Level 2 PRA. Split fractions for the SFP event tree model will be quantified in consideration of relevant data for similar situations or by use of the results of thermal hydraulic analyses. Various dependencies and mitigation actions shall be taken into account in the event tree modeling. Furthermore, it will be necessary to adopt, as necessary, the binning logic developed in the Reactor Level 2 PRA in order to bin the SFP Level 2 event tree end-states into a small number of release categories. Thermal hydraulic computer code analysis such as MAAP code can be utilized to provide estimates of the release magnitudes and timings (source terms) during the spent fuel damage accidents.

III.D. Task 4: Success Criteria Analysis

The objective of the success criteria element is to define the plant-specific measures of success and failure that support the other technical elements of the SFP PRA in such a way that:

- Overall success criteria (i.e., fuel damage and large release) are defined.
- Success criteria are defined for critical safety functions, supporting systems, structures, components, and operator actions necessary to support accident sequence development.
- The methods and approaches have a firm technical basis.
- The resulting success criteria are referenced to the specific deterministic calculation.

III.D.1. Definition of Fuel Damage Criteria

The definition of 'fuel damage' will be used in estimating the 'fuel damage frequency (FDF)' as a risk metric in the SFP PRA. It will also be used in evaluating the consequences of selected accident scenarios by thermal hydraulic computer code.

In the PRAs for nuclear power reactors, 'core damage frequency (CDF)' is typically used as a risk metric. As part of the success criteria, the ASME/ANS RA-Sa-2009 Standard (Ref. 1) states the following requirement (i.e., SC-A2) with respect to 'measures for core damage' suitable for Capability Categories II and III.

"SPECIFY the plant parameters (e.g., highest node temperature, core collapsed liquid level) and associated acceptance criteria (e.g., temperature limit) to be used in determining core damage. SELECT these parameters such that the determination of core damage is as realistic as practical, in a manner consistent with current best practice. DEFINE computer code-predicted acceptance criteria with sufficient margin on the code-calculated values to allow for limitations of the code, sophistication of the models, and uncertainties in the results, in a manner consistent with the requirements specified under HLR-SC-B.

Examples of measures for core damage suitable for Capability Category II/III, that have been used in PRAs, include (a) collapsed liquid level less than core height or code predicted peak core temperature >2,500°F (BWR); (b) collapsed liquid level below top of active fuel for a prolonged period, or code-predicted core peak node temperature >2,200°F using a code with detailed core modeling; or code-predicted core peak node temperature >1,800°F using a code with simplified (e.g., single-node core model, lumped parameter) core modeling; or code-predicted core exit temperature >1,200°F for 30 min using a code with simplified core modeling (PWR)."

It is also notable that the NRC recently made some statements related to definition of fuel damage in SFP. In the draft NRC document of "Technical Analysis Approach Plan for Level 3 PRA Project" (Ref. 2), an example definition of 'fuel damage' for SFP is mentioned as: water level at 1/2 fuel height. In addition, the following are given as example figures-of-merit in connection with the SFP PRA (Ref. 13):

1)	SFP water temperature = 212°F	≈	Onset of SFP boiling
2)	SFP water level < 4 feet above top of fuel	≈	Radiation levels on the refuel floor begin to noticeably increase
3)	SFP water level reaches top of fuel	≈	Radiation levels on the refuel floor are reaching high levels
4)	Fuel Handling Building (FHB) bulk air temperature = 140°F	≈	Personnel safety due to a hot, wet environment is a concern
5)	FHB bulk air temperature = 200°F	≈	Significant steam burns to personnel in the vicinity of the pool are an intermediate concern
6)	SFP water level reaches 2/3 active fuel height	≈	Adequate steam cooling is being lost and fuel heatup is commencing
7)	Water-loss-rate when level reaches 2/3 active fuel height	≈	Provides a sense of whether mitigation deployed at that point would result in level recovery

On the other hand, the EPRI study on PWR SFP risk assessment (Ref. 3) indicates that at approximately peak nodal temperature of 1800°F the clad would begin an exothermic reaction with the steam, and the clad temperature would rapidly rise. Furthermore, when the used nuclear fuel is exposed to a continuous supply of air (e.g., oxygen), the zircaloy clad can undergo a rapid exothermic reaction (i.e., zirconium fire) if the fuel clad is in the 600°C to 1100°C (1100°F to 2000°F) temperature range. In light of all these considerations, the EPRI definition of spent fuel damage, i.e., water level below one-third core-height and falling plus calculated peak core temperatures greater than 1800°F (980°C) for more than 10 minutes, may be adopted in performing SFP PRAs.

III.D.2. Special Considerations in the SFP PRA Success Criteria Development

Thermal/hydraulic (T/H), structural, or other supporting engineering analyses will be performed as necessary to provide success criteria and event timing sufficient for quantification of FDF and LRF, determination of the relative impact of success criteria on SSCs and human actions, and the impact of uncertainty on this determination. The computer codes (e.g., MAAP code) and analytic models shall have sufficient capability to model the conditions of interest in determining success criteria for FDF and evaluating specific event scenarios, and should provide results representative of the plant.

The success criteria for the SFP accident prevention and mitigation measures are functions of the following:

- Decay heat as determined primarily by time since last fuel off-load from the reactor, and the quantity of recently discharged fuel (e.g., 1/3 core or full core)
- Size and location of LOCA or drain-down (if applicable)
- Available make-up systems
- Available Fuel Pool Cooling equipment
- Connectivity of the SFP and the refueling cavity (i.e., fuel transfer tube opened/closed)

Hence, these aspects will be taken into account in developing success criteria for spent fuel pool. The results of T/H computer code calculations will be utilized in defining success criteria when deemed necessary.

III.E. Task 5: System Analysis

The objective of the systems analysis element is to identify and quantify the causes of failure for each system represented in the initiating event analysis and accident sequence analysis in such a way that:

- System-level success criteria, mission times, time windows for operator actions, and assumptions provide the basis for the system logic models as reflected in the model. A reasonably complete set of system failure and unavailability modes for each system is represented.

- Human errors and operator actions that could influence the system unavailability or the system's contribution to accident sequences are identified for development as part of the HRA element.
- Different initial system alignments (e.g., fuel transfer canal open in certain OCPs) are evaluated to the extent needed for FDF and LRF determination.
- Intersystem dependencies and intra-system dependencies including functional, human, phenomenological, and common-cause failures that could influence system unavailability or the system's contribution to accident sequence frequencies are identified and accounted for.

III.E.1. Development of Systems Model

Plant information sources relevant to SFP risk shall be reviewed to define or establish:

- system components and boundaries
- dependencies on other systems
- instrumentation and control requirements
- testing and maintenance requirements and practices
- operating limitations such as those imposed by Technical Specifications
- component operability and design limits
- procedures for the operation of the system during normal and accident conditions
- system configuration during normal and accident conditions

Detailed systems models will be developed for systems that influence SFP risk, unless: a) sufficient system-level data are available to quantify the system failure probability; or b) system failure is dominated by operator actions, and omitting the model does not mask contributions to the results of support systems or other dependent-failure modes. Those system models developed for the Reactor PRA will be used in the SFP PRA to the extent possible for the sake of consistency and efficiency in the PRA work.

The boundaries of the components required for system operation will be established such that they match the definitions used to establish the component failure data. For example, a control circuit for a pump will not be included as a separate basic event (or events) in the system model if the pump failure data used in quantifying the system model include control circuit failures. Intersystem dependencies and intra-system dependencies including functional, human, phenomenological, and common-cause failures that could influence system unavailability or the system's contribution to accident sequence frequencies shall be identified and accounted for. Human errors and operator actions affecting system unavailability or accident sequence frequencies shall be explicitly identified and described to facilitate performance of the HRA task.

III.E.2. Special Considerations in SFP System Modeling

It is expected that systems analysis need to be carried out for the spent fuel pool cooling and makeup systems, and also support systems (e.g., component cooling water system or electrical system) including the ultimate heat sink, as well as the SFP instrumentation and control (I&C) systems and SFP HVAC system to the extent affecting the SFP system unavailability or SFP-related accident sequence frequencies. Modeling consideration will also be given to SFP boundary seal and fuel transfer canal if they are found to affect system performance during any operating cycle phase (OCP).

In addition, fire pumps and any beyond-design-basis mitigation systems that could be used for SFP cooling during SFP emergency conditions (e.g., 10 CFR 50.54(h)(2) equipment) will be modeled if they are available for spent fuel accident mitigation. System-level success criteria, mission times, time windows for operator actions, operation and maintenance practices assumed for the SFP, and so on, shall be considered in performing the systems analysis.

III.F. Task 6: Human Reliability Analysis

The objective of the human reliability element of the SFP PRA is to ensure that the impacts of plant personnel actions are reflected in the assessment of SFP risk in such a way that:

- Both pre-initiating event and post-initiating event activities, including those modeled in support system initiating event fault trees, are addressed.

- Logic model elements are defined to represent the effect of such personnel actions on SFP-related systems availability and on accident sequence development.
- Plant-specific and scenario-specific factors are accounted for in the evaluation of human performance.
- Human performance issues are addressed in an integral way so that issues of dependency are captured.

The Human Reliability Analysis for SFP Internal Events Level 1/2 PRA will be conducted to quantify the human failure events associated with SFP or mitigation systems operation in case of abnormal conditions in the SFP. In addition to the post-initiator human errors, pre-initiator human errors such as restoration error shall also be considered if found to be significant contributors to system unavailability or accident sequence frequency. Quantification of the probabilities of these human failure events and recovery actions will be based on accident-specific conditions associated with the SFP, where applicable, including any dependencies among actions and conditions.

III.G. Task 7: Data Analysis

The objective of SFP data analysis is to provide estimates of the parameters used to determine the probabilities of the basic events representing equipment failures and unavailabilities modeled in the SFP PRA in such a way that:

- Parameters, whether estimated on the basis of plant-specific or generic data, appropriately reflect the configuration and operation of the SFP.
- Component or system unavailabilities due to maintenance or repair are accounted for.
- Uncertainties in the data are understood and appropriately accounted for.

In general, three types of data are basically needed in performing a PRA: 1) initiating event frequencies, 2) component unreliability/unavailability data (i.e., due to component failure or maintenance), and 3) human error probabilities. Depending on the specific PRA (e.g., shutdown PRA, SFP PRA), special data applicable to the operating mode or facility under analysis may be additionally needed, and hence, the data parameters to be used should be carefully chosen or developed in consideration of the modeling context.

Data sources that may be usefully applied in performing the SFP PRA are listed below.

<u>Initiating Event Data:</u>

- SFP-Relevant Initiating Event Data (e.g., loss of SFP coolant inventory): Refer to initiating event frequencies based on the SFP operating experience (e.g., NUREG-1275 Volume 12 (Ref. 7), EPRI-3002002691 (Ref. 3), NUREG/CR-4982 (Ref. 8))
- Other Generic Initiating Event Data (e.g., loss of offsite power): The initiating event parameter estimates developed by the Idaho National Laboratory (INL) for the NRC's Level 1 SPAR models (i.e., NUREG/CR-6928 (Ref. 9)) may be used for the SFP PRA

<u>Component Unreliability/Unavailability Data:</u>

- Failure and Unavailability Data for Generic Components (e.g., pumps, valves): The component failure and unavailability data included in NUREG/CR-6928 may be applied to the SFP PRA in consideration of the operational environment and maintenance practices of the SFP.
- Failure Data for SFP-Relevant Equipment (e.g., reactor cavity seal or gate seal, or anti-siphon devices): Refer to NUREG-1275 Volume 12, NUREG-1738 (Ref. 5), EPRI-3002002691, etc.

<u>Human Performance Data:</u>

- SFP-Specific Human Performance Data: Refer to NUREG-1738

Table 3 presents the data sources potentially applicable to the SFP PRA with description of the salient features with respect to SFP data or shutdown-specific data.

Table 3. Data Sources Potentially Applicable to the SFP PRA

Report	Report Title	Salient Features with respect to SFP/Shutdown Data
NUREG/CR-4982 (Ref. 8)	Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82	• SFP accident initiating events and probability estimates (loss of water circulating capability; structural failure of pool due to seismic events, missiles, or heavy load drop; partial draindown of pool due to refueling cavity seal failures) • Evaluation of probability of a clad fire given a pool drainage event estimated
NUREG-1353 (Ref. 10)	Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools	• Provides frequency for SFP accidents: structural failures due to missiles, aircraft crashes, heavy load drop, earthquakes; pneumatic seal failure; inadvertent drainage; loss of cooling/makeup • Provides SFP-specific data such as estimated likelihoods of propagation of zircaloy fire to older spent fuel and self-sustaining zircaloy clad oxidation for various spent fuel rack configurations and decay heat loads, and shows events in which pneumatic inflated seals have failed
NUREG-1275, Volume 12 (Ref. 7)	Operating Experience Feedback Report – Assessment of Spent Fuel Pool Cooling	• Provides generic failure data for loss of SFP coolant inventory and loss of SFP cooling events • Classification of SFP events into; 1) loss of SFP coolant inventory through connected systems, gates and seals, structures or liner; 2) loss of SFP cooling due to loss of power to SFP cooling pumps, etc., or ineffective heat sinks
NUREG-1738 (Ref. 5)	Operating Experience Feedback Report – Assessment of Spent Fuel Pool Cooling	• Provides generic failure data for loss of SFP coolant inventory and loss of SFP cooling events • Classification of SFP events into; 1) loss of SFP coolant inventory through connected systems, gates and seals, structures or liner; 2) loss of SFP cooling due to loss of power to SFP cooling pumps, etc., or ineffective heat sinks
EPRI 1003113 (Ref. 11)	An Analysis of Loss of Decay Heat Removal Trends and Initiation Events Frequencies (1989-2000)	• The most comprehensive database for shutdown initiating events • Includes initiating event recovery probabilities as well as data for loss of pump at midloop, large draindown, small draindown, etc.
EPRI 1021176 (Ref. 12)	An Analysis of Loss of Decay Heat Removal and Loss of Inventory Event Trends (1990-2009)	• Updated the database developed in EPRI 1003113 study by incorporating improved operating experience during plant shutdown • Use of this document along with EPRI 1003113 is recommended by the NRC for shutdown-related initiating events
EPRI 3002002691 (Ref. 6)	PWR Spent Fuel Pool Risk Assessment Integration Framework and Pilot Plant Application	• Provides the most comprehensive list of SFP accident initiating events and their frequencies

The reliability data (e.g. failure probabilities, unavailabilities, CCF parameters) for various basic events relevant to the mitigation systems performance shall be compiled from the Reactor PRA if available, with adjustment associated with differences between SFP and reactor made as necessary. The unique SFP condition reflected in each OCP shall be taken into account in the data analysis.

III.H. Task 8: Fuel Damage Frequency Quantification

The objective of the FDF quantification element is to provide an estimate of fuel damage frequency (and support the quantification of large release frequency) based upon the specific fuel damage scenarios, in such a way that:

- The results reflect the design, operation, and maintenance of the plant.
- Significant contributors to FDF are identified such as initiating events, accident sequences, and basic events (equipment unavailability and human failure events).
- Dependencies are accounted for.
- Uncertainties are understood.

In accordance with the ASME/ANS PRA standard (Ref. 1), the fault trees and Level 1 event trees that were developed for the SFP will be linked as a part of this task. In addition to common cause failures, all other dependencies such as those due to design (i.e., system dependencies), basic event probabilities, human action dependencies, and dependencies caused by specific sequences of accidents will be modeled.

Accident minimal cutsets for the surrogate criteria of fuel damage will be generated for each OCP. The minimal cutsets will be examined to ensure that they accurately reflect the modeling assumptions, and the dominant contributors will be identified for the release categories for each OCP.

Mitigation of accidents involving the SFP will likely involve accident sequences with multiple human failure events (HFEs). In accordance with the ASME/ANS PRA standard element QU-C1, cutsets with multiple HFEs that potentially impact quantification of significant accident sequences/cutsets will be identified. The PRA model will be then requantified with the human event probabilities (HEPs) set to values that are sufficiently high so that the cutsets multiple HFE are not truncated. The dependency between the HFEs in the cutsets will be evaluated, and all the cutsets including those with multiple HFEs will be then requantified to obtain the FDF.

Major sources of uncertainties and assumptions made in each analysis step of the SFP Internal Events Level 1 PRA shall be identified in a similar manner as done in the Reactor PRA. The description for each source of uncertainties or assumptions will be complied along with possible means to address them (e.g., through propagation, sensitivity analysis, or expert judgment). To the extent possible, the various sources of logic model uncertainties will be propagated through the Level 1 event tree models. When the uncertainties cannot be quantitatively addressed, sensitivity analyses will be performed in order to characterize them. Uncertainties in the deterministic models will be identified, and their effects will be addressed either qualitatively through expert judgment or by sensitivity analysis.

III.I. Task 9: Spent Fuel Pool Degradation Analysis

The objective of spent fuel degradation analysis is to characterize degradation of the SFP for each fuel damage scenario identified in the SFP Level 1 PRA by assuming fuel storage configuration, fuel burnup, and spent fuel pool rack configuration for the plant being analyzed, and then performing thermal-hydraulic calculations and evaluating physical phenomena for some representative scenarios in such a way that:

- Parameters for defining spent fuel damage states are appropriately identified so that an interface between Level 1 and Level 2 PRAs can be established to facilitate Level 2 PRA.
- Fuel damage scenarios are binned together in terms of the spent fuel damage states for evaluation in Level 2 accident progression event trees.
- Potential for zirconium fire is appropriately identified in consideration of decay heat rate, fuel burnup, fuel storage configuration, fuel assembly geometry, building ventilation rates and air flow paths, and fuel cladding oxidation rates.
- The time available before fuel uncovery and the time available for plant operators to take actions to prevent a zirconium fire are appropriately determined for the HRA in Level 2 PRA.

Spent fuel degradation analysis will be carried out by use of a thermal-hydraulic code (e.g., MAAP code) which provides a best estimate representation for severe accident analyses. The capabilities and limitations of the spent fuel pool model as represented by the code need to be understood in order to properly interpret the code computational results. The key physical phenomena and associated timing that may be evaluated with the thermal-hydraulic code include the following:

- Prolonged loss of spent fuel pool cooling leading to long term boil-off of pool inventory
- Slow or sudden loss of spent fuel pool inventory due to various reasons (e.g., SFP cooling system pipe break, leak of the fuel transfer canal)
- Extent of zircaloy-steam exothermic heat and hydrogen production
- Fuel cladding failure timing and noble gas release timing
- Radionuclide release magnitudes and timings

III.J. Task 10: Large Release Frequency Analysis

The objective of the LRF analysis element is to identify and quantify the contributors to large early releases, based upon the specific spent fuel damage scenarios, in such a way that:

- The methodology is clear and consistent with the Level 1 evaluation, and creates an adequate transition from Level 1.
- Operator actions, mitigation systems, and phenomena that can alter sequences are appropriately included in the Level 2 event tree structure and sequence definition.
- Dependencies are reflected in the accident sequence model structure, if necessary.
- Success criteria are available to support the individual function successes, mission times, and time windows for operator actions and equipment recovery for each critical safety function modeled in the accident sequences
- End states are clearly defined to be LRF or non-LRF.

Based upon the timing determined to be available to mitigate a loss of spent fuel pool cooling or a loss of water inventory, the potential for recovery and mitigation actions will be evaluated as part of the Level 2 event tree analysis. The recovery analysis will credit human recovery actions subject to the following limitations:

- Close-in recovery actions (e.g. add water, create a path to remove heat) must be completed before the pool water level reaches a point where ambient air temperatures or, the gamma and neutron doses preclude human habitation.
- The recovery actions must be addressed in a written emergency procedure and personnel must have been trained in carrying out the required actions, preferably via a drill or emergency exercise.

Also considered as part of Level 2 event tree analysis are the success and failure of venting of the fuel building, its structural integrity, and potential for scrubbing of the radiological releases. In order to properly characterize different release categories, the magnitude, timing, and energy of releases are taken into account. Additional considerations in evaluating the frequency of various Level 2 scenarios and end states are:

- Success and failure probabilities of the events of concern in Level 2 analysis
- The radiation level, existence of harsh environment for equipment survivability, and time available
- The system responses and their impact on radionuclide scrubbing or the effectiveness in controlling the source terms

IV. CONCLUSION

This paper describes the process to develop the SFP Level 1 and Level 2 PRA for internal events. Ten major technical tasks were defined to systematically perform the SFP internal events Level 1 and Level 2 PRA for APR1400 nuclear power plants. These tasks can also be applied for other SFP internal events Level 1 and Level 2 PRAs. Hence, it is expected that the methodological approach presented in this paper will facilitate to quantify plant-specific SFP risks, i.e., fuel damage frequency (FDF) and large release frequency (LRF), and identify significant risk contributors for SFP internal events including loss of SFP coolant inventory and loss of SFP cooling.

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