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FEASIBILITY STUDY OF RISK INFORMING EMERGENCY ACTION LEVELS OF FISSION PRODUCT BARRIERS USING LEVEL 2 PRA

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ABSTRACT

The existing radiological Emergency Classification (EC) levels, in which Emergency Action Levels (EALs) are assigned to, are established by the NRC according to (1) their relative radiological seriousness, and (2) the time-sensitive onsite and offsite radiological protective actions necessary to respond to such events. In ascending order of severity, these ECs are:

- Notification of Unusual Event (NOUE): Events, which indicate a potential degradation of the level of safety of the plant, or indicate a security threat to facility protection, are in progress or have occurred. No release of radioactive material requiring offsite response or monitoring is expected unless further degradation of safety systems occurs.
- Alert: Events, which involve an actual or potential substantial degradation of the level of safety of the plant, or a security event that involves probable life threatening risk to site personnel, or damage to the site equipment because of intentional malicious dedicated efforts of a hostile act, are in progress or have occurred. Any releases are expected to be limited to a small fraction of the exposure level limits set forth by the EPA's Protective Action Guidelines.
- Site Area Emergency (SAE): Events that are in progress, or have occurred which involve:
    i. Actual or likely major failures of plant functions needed to protect the public
    ii. Security events that result in intentional damage or malicious acts:
        a. toward site personnel or equipment that could lead to the likely failure, or
        b. prevent effective access to equipment needed to protect the public.

Apart from the site boundary, none of the releases are expected to exceed the exposure level limits set forth by the EPA's Protective Action Guidelines.

- General Emergency (GE): Events, which involve actual or imminent substantial core degradation, or melting with a potential for loss of containment integrity, or security events that result in an actual loss of physical control of the facility, are in progress or have occurred. Releases can be reasonably expected to exceed the exposure level limits set forth by the EPA's Protective Action Guidelines for more than the immediate site area.

The objectives of this study were to demonstrate the feasibility, and document insights and major observations of using Level 2 PRA to evaluate ECs for EAL associated with fission product barriers (FPBs). PRA models were applied in a systematic manner to evaluate the consistency of the EC for the FPB related EALs. This feasibility study piloted an approach by using a Level 2 model of a Mark 1 BWR which was created for NRC trial use. The scope of this study is limited to the analyses of the Passive Barrier EAL matrix. The insights and observations made in this feasibility study were consistent with and further supported by the engineering reasoning and deterministic evaluation.

Key Words: Risk-informed evaluation, Level 2 PRA, Emergency Action Levels, Emergency Preparedness, Risk-informed decision making, Emergency Classifications

1 INTRODUCTION

Application of Level 1 PRA to evaluate the likelihood of core damage in different EALs is comprehensively documented [Ref. 1]. Level 2 probabilistic risk assessment (PRA), however, is needed to evaluate the likelihood of various radiological releases. The use of Level 2 PRA can explicitly support the risk-informed evaluation of EALs that are related to fission product barrier (FPB) failures and radiological effluent.

In the past, the NRC has used Level 2 PRA models to provide qualitative insights. However, quantitative results were not used extensively. As risk-informed regulation became more prevalent, both licensees and the NRC have increased their efforts to develop Level 2 PRA models and test their uses. The NRC has increased the use of Level 2 PRA models, both qualitatively and quantitatively, in reviewing new reactor license applications. The increased use of Level 2 PRA models is expected to expedite the development of the Level 2 PRA standards, and to improve the quality of future Level 2 PRA models.

The two primary objectives intended for this study were to demonstrate the technical feasibility of using Level 2 PRA models to risk-inform FPB EALs, and to identify areas within the existing EALs associated with FPBs that could potentially benefit from further examination. Consistent with the objectives noted above, the scope of this study is limited to the analysis of the FPB EAL matrix. However, similar approaches can be applied to analyze other EALs. Furthermore, the emphasis of this study was mainly on the feasibility of technical approaches rather than performing a complete analysis of all FPB related EALs. It consisted of a total of eleven case evaluations for selected representative EAL scenarios for a boiling water reactor (BWR).

The fission product barrier matrix for BWR EALs is shown in Table 1. The EALs for a typical Mark 1 BWR were taken from NUMARC/NESP-007 [Ref. 2]. Three major barriers are defined: Fuel clad, Reactor coolant system, and Primary containment. EAL threshold conditions were defined by different combinations of the loss or the potential loss of each barrier. This is shown in the matrix located on the top of the table. This matrix shows that for single barrier degradation, the loss or a potential loss of primary containment barrier is assigned to a NOUE (FU1), whereas either a loss or a potential loss of fuel cladding or reactor coolant system barriers is assigned to an Alert (FA1). The loss or a potential loss of primary containment barrier is assigned to an NOUE because it can occur even when no accident is in progress. However, the conditions for the loss or potential loss of containment barrier as defined in Table 1 could also occur during accidents. Therefore, while doing risk evaluations, it is necessary to consider containment leakages during both accident and non-accident conditions.

Concurrent losses or potential losses of two barriers as shown in the barrier matrix constitute a SAE (FS1). The EALs do not differentiate between the potential loss and the loss of the barriers. A concurrent degradation of all three barriers with either “three losses”, or “two losses plus one potential loss”, would constitute a GE (FG1).

The goal of using PRA methods to risk inform EAL is to confirm that the conditional risk increases as the EC becomes more severe. The use of Level 2 PRA can significantly broaden the scope of risk evaluation by providing information about reliability of active containment systems, failure probability of passive barriers, magnitude and timing of radiological releases. As an example, the sequential transition and the transition time from a lower EC (i.e. NOUE or Alert) to a more severe EC (i.e. SAE or GE) is important for emergency planning. A longer transition time would allow more time to conduct an effective evacuation if it becomes necessary. Should the activation of a GE occur, it is desirable that it follows the activation of a SAE, an Alert, or an NOUE.

While the quantification of Level 1 PRA only provides CCDP (Conditional Core Damage Probability) as a risk metric, a full Level 2 PRA provides information about the probability, the timing, and the magnitude of releases during an accident. The magnitude of releases can be divided into a limited number of categories: large, medium, small, and negligible; the timing of releases can be divided into early releases and late releases. Depending on the timing and magnitude of release, the importance of GE activation and timely evacuation becomes apparent.

Table 1. Fission Product Barrier EAL

FISSION PRODUCT BARRIER MATRIX (Applicability: Modes 1 2 3)

FISSION PRODUCT BARRIER STATUS	FG1: GENERAL EMERGENCY				FS1: SITE AREA EMERGENCY												FA1: ALERT				FU1: UNUSUAL EVENT	
Fuel Clad – LOSS	X	X		X	X	X	X	X									X					
Fuel Clad – POTENTIAL LOSS			X						X	X	X	X						X				
Reactor Coolant System – LOSS	X	X	X		X				X				X	X					X			
Reactor Coolant System – POTENTIAL LOSS				X		X				X					X	X				X		
Primary Containment – LOSS	X		X	X			X				X		X		X						X	
Primary Containment – POTENTIAL LOSS		X						X				X		X		X						X

	1. FUEL CLAD BARRIER		2. REACTOR COOLANT SYSTEM BARRIER		3. PRIMARY CONTAINMENT BARRIER	
	LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
a. Reactor Pressure Vessel (RPV) Water Level	1. RPV level **< -195 inches** **without** at least one Core Spray loop flow **> 6250 gpm**. **OR** 2. RPV level **< -226 inches**.	3. RPV level **< -172 inches**. **OR** 4. RPV level **cannot** be determined.	1. RPV level **< -172 inches**. **OR** 2. RPV level **cannot** be determined.			1. Plant conditions indicate that Primary containment flooding is required.
b. Drywell (DW) Pressure / Hydrogen Concentration			1. Drywell pressure **> 2.0 psig** **AND** 1. Drywell pressure rise due to RCS leakage.		1. Rapid unexplained drop in Drywell pressure following initial pressure rise. **OR** 2. Drywell pressure response **not** consistent with LOCA conditions.	3. Drywell pressure **> 56 psig** and rising. **OR** 4. a. Drywell or Torus Hydrogen concentration **> 6%**. **AND** b. Drywell or Torus Oxygen concentration **> 5%**.
c. Drywell (DW) High Range Rad Monitor	1. Drywell radiation **> Fuel Cladding Loss** Threshold Table F1.		1. Drywell radiation **> 100 R/hr**. **AND** 2. Indications of RCS leakage into the Drywell.			1. Drywell radiation **> Primary Containment Potential Loss** Threshold, Table F2.
d. Breached / Bypassed	1. Coolant activity **> 300 µCi/gm** Dose Equivalent I-131.		1. UNISOLABLE Main Steam Line (MSL) Break. **AND** 2. a. High MSL Flow and High Steam Tunnel Temperature alarms. **OR** b. Direct report of steam release.	3. RCS Leakage **> 50 gpm** inside the Drywell. **OR** 4. UNISOLABLE primary system leakage outside drywell resulting in Secondary Containment area temperatures or area radiation levels **> T-103 Alarm Setpoint**.	1. a. Failure of all isolation valves in any one line to close. **AND** b. Downstream pathway to the environment exists. **OR** 2. Intentional venting/purging of Primary Containment per TRIPs or SAMPs due to accident conditions. **OR** 3. UNISOLABLE primary system leakage outside drywell resulting in Secondary Containment area temperatures or area radiation levels **> T-103 Action Level**.	
e. Emergency Director Judgment	1. Any condition in the opinion of the Emergency Director that indicates a Loss of the Fuel Clad Barrier.	2. Any condition in the opinion of the Emergency Director that indicates a Potential Loss of the Fuel Clad Barrier.	1. Any condition in the opinion of the Emergency Director that indicates a Loss of the Reactor Coolant System Barrier.	2. Any condition in the opinion of the Emergency Director that indicates a Potential Loss of the Reactor Coolant System Barrier.	1. Any condition in the opinion of the Emergency Director that indicates a Loss of the Primary Containment Barrier.	2. Any condition in the opinion of the Emergency Director that indicates a Potential Loss of the Primary Containment Barrier.

Table F1 – Fuel Clad Loss

Time After Shutdown (hrs)	Drywell Radiation (R/hr)
≤ 2	9.55E+02
>2 – 4	8.40E+02
>4 – 8	7.25E+02
>8 – 16	6.05E+02
>16 - 23	5.45E+02
> 23	5.40E+02

Table F2 – PC Potential Loss

Time After Shutdown (hrs)	Drywell Radiation (R/hr)
≤ 2	2.20E+03
>2 – 4	1.95E+03
>4 – 8	1.70E+03
>8 – 16	1.40E+03
>16 - 23	1.25E+03
> 23	1.25E+03

1.1 Technical Approach

The technical approach for risk-informing the EAL associated with FPBs involve two major tasks: (1) scenario identification and (2) quantitative analysis.

Scenario identification deals with selecting the PRA accident conditions, which can produce the same symptoms as FPB EALs. It consists of three steps, and it establishes the bridge between FPB EALs and the PRA models by defining the accident conditions.

Quantitative analysis follows a methodology similar to that used in Accident Sequence Precursor (ASP) for analysis of events and conditions for estimating the change in CCDPs and extended it for estimating the changes in conditional probabilities in various release categories. This was done by identifying the accident conditions capable of producing the same symptoms as those described in Table-1 for loss or potential loss of barriers and mapping them to the PRA model for reevaluating the frequencies of various release categories. In most cases, the accident condition was mapped to the PRA model by modifying the appropriate basic events. In rare occasions, however, changes had to be made to the actual model or the associated PRA rules in order to simulate the accident condition of interest. Two risk metrics were used to bin and categorize the quantitative results: CCDP and CCRP. CCRP stands for Conditional Consequential Release Probability. It includes the early releases (both small and large) and large late release.

1.1.1 Technical Approach – Scenario Identification

Possible accident conditions that could produce the same symptoms of each of the FPB EAL were determined by examining the plant specific accident response evaluation. This involved reviewing accident analysis documents and/or performing simplified calculations to determine the necessary conditions that can produce the symptoms identified for each FPB-EAL in Table 1.

As an example, the conditions necessary for drywell radiation to be greater than the values noted in Table F1 (embedded in Table 1), was found to necessitate more than 3% of fuel failures and the existence of a flow path or a rupture from the vessel to drywell. There could be many different accident scenarios that can meet the conditions noted above. Short of examining all possible scenarios, the study limited the evaluation to those that had higher likelihood of occurrence and were contained in dominant accident sequences in the PRA. The following example sequences were selected:

1. Medium or Large LOCA with failure of early or late injection.
2. An extended SBO beyond battery depletion and failure to recover Alternative AC power from other sources.

This process was followed for each of the symptoms for potential loss, or the loss of a passive barrier. A representative set of scenarios was developed by examining each symptom individually.

It was found that the accident scenarios developed for loss or potential loss of one FPB could also cause the loss or potential loss of another FPB. It was, therefore, necessary to assemble all the accident scenarios and find their integrated impact on loss or potential loss of all FPBs. Appropriate ECs then were assigned to each accident scenarios, and any unnecessary duplication were removed. An example of the result of this process is shown in Table 2.

Table 2. Example list of a sequences and its mapping against EALs

Accident Condition AC # and [Case #]	Accident Condition Description	Potential Loss FB (Fuel Barrier)	Loss FB	Potential Loss RCB (Reactor Coolant Barrier)	Loss RCB	Potential Loss PCB (Primary Containment Barrier)	Loss PCB	EC
5 [Case 7]	MLOCA with failure of HPCI and CS systems		X		X	X		GE
10 [Case 8]	Long SBO (>10 hrs.) resulting in battery depletion and failure of all containment heat removal including venting.		X		X	X		GE
1 [Case 5]	LOMFW transients with failure of HPCI, RCIC, and CRD	X			X			SAE
2 [Case 6]	LOOP with loss of HPCI, RCIC, and CRD	X			X			SAE
11 [Case 4]	SLOCA				X			ALERT
20 [Case 3]	SLOCA with isolation of steam lines and main feedwater system				X			ALERT
36 [Case 1a]	Containment leakage during plant normal operation						X	NOUE
34 [Case 2]	LOMFW/Steam with failure of SPC and SDC						X	NOUE

1.1.2 Technical Approach – Quantitative Analysis

A Level 2 PRA model for a Mark 1 BWR that was created for trial use was used along with the version 8 of SAPHIRE code to determine the quantitative risk metrics associated with each accident condition. This PRA model has not been verified by the NRC staff or the licensee to reflect the as-built, as operated conditions. However it is adequate for this proof of concept study. In most cases, the accident condition was mapped to the PRA model by modifying the appropriate basic events. In rare occasions, however, changes were made to the model or the associated PRA rules in order to simulate the accident condition of interest.

The risk metrics are shown in table 3. The definition column includes general interpretations of the risk metrics for the use of this study.

Table 3. Risk Metrics

Acronym	Name	Definition
CCDP	Conditional Core Damage Probability	The probability that the selected accident condition progress to onset of the Core Damage.
CPLER	Conditional Probability of Large-Early Release	The probability of Large release (>5% of I and Cs) to environment within at most12 hours, but generally less than 8 hours after the Core Damage
CPLLR	Conditional Probability of Large-Late Release	The probability of Large release to environment greater than 12 hours after the Core Damage
CPMER	Conditional Probability of Medium-Early Release	The probability of Medium Release (< 5% and >1% of Cs and I) to the environment within at most12 hours but generally less than 8 hours after the Core Damage
CPMLR	Conditional Probability of Medium-Late Release	The probability of Medium release to the environment more than 8 hours after the Core Damage
CPSER	Conditional Probability of Small-Early Release	The probability of small release (<0.1% and >0.01% of Cs and I) to the environment within at most12 hours, but generally less than 8 hours after the Core Damage
CP SLR	Conditional Probability of Small-Late Release	The probability of small Release to the environment more than 12 hours after the Core Damage
CPCLR	Conditional Probability of no or negligible releases (Controlled releases)	The probability of controlled or negligible release (<0.01% of Cs and I) to the environment more than 12 hours after the Core Damage

Two risk metrics were used to bin and categorize the quantitative results: CCDP and CCRP. CCRP is defined as the summation of the CPLER, CPMER, CPSER and CPLLR. The CCRP risk metric is particularly important in terms of EP because it covers all the early and large releases, which could pose significant risk to the public.

The timing associated with EAL activation compared to the onset of the core damage and the release time cannot be explicitly obtained from the Level 2 PRA models, however they are implicitly embedded in the definition of Plant Damage States (PDS) and the release categories. An attempt was also made to estimate additional timing information by examining accident analysis reports such as NUREG-1953.

This study performed a limited number of case studies to demonstrate the feasibility of the approach and document the steps involved. These limited quantitative results mostly indicated consistency between the CCDP and CCRP for the cases analyzed.

2 INSIGHTS AND OBSERVATIONS

Although this study was intended as a feasibility study, it identified a number of insights that could help with the future risk-informing process of FPB-EALs and other EALs using a Level 2 PRA.

The following insights were obtained from mapping the FPB EALs to PRA accident conditions. These insights are, therefore, specific to FPB EALs. Further PRA analyses should be performed, in order to determine whether they are applicable to other types of EALs.

1. Most accidents that cause the symptoms of loss or potential loss of one barrier could also cause the loss or potential loss of at least one more barrier. In some representative accidents three barriers are affected. It would be difficult to postulate an accident that affects only one barrier. As a result, most representative accidents are classified as a SAE or a GE, and not as an NOUE or an Alert. This study found that NOUEs or Alerts will not be initially triggered often as a result of FPB EALs. They will be triggered by EALs that are triggered earlier in the accident scenario, many of which are discussed in NUREG-1754 [Ref. 1].

2. BWRs with Mark I containments are inerted during operation. A loss or potential loss of containment barrier (FU1) during normal operation is expected to result in plant shutdown in a short period. Duration of operation before shutdown was assumed to be 72 hours consistent with plant technical specification requirements. This condition would currently trigger an NOUE.

3. Treating the impact of the loss and the potential loss of FPBs in the EAL matrix the same (e.g. any combinations of two losses or potential losses are classified as an SAE) is expected to result in accident conditions with wide ranges of CCDPs and CCRPs, i.e. it causes different levels of risk significance to be grouped under same EC. As an example, an accident condition (AC2) involved LOOP with the loss of HPCI, RCIC and CRD injection systems, which triggers loss of RCB and potential loss of FB. Another accident condition (AC6) involved LOOP with the loss of HPCI, RCIC, CRD, CS and LPCI systems, which triggers the loss of both RCB and the loss of FB. Both of these accident

conditions are classified as SAE, although the CCDP/CCRP for AC6 is much higher than CCDP/CCRP for AC2.

4. Plant parameters associated with the Heat Capacity Temperature Limit (HCTL), such as RPV pressure, suppression pool level, and suppression pool temperature, play an important role in defining the loss or potential loss of containment barrier. These parameters which could indicate the potential loss of suppression pool are not explicitly used in the symptom-based definitions for EAL threshold conditions. Suppression pool heat-up, which is closely related to HCTL, is a risk significant event in Level 2 PRA model.

The following insights were gained based on the use of Level 2 PRA models for risk-informing FPB-EALs.

1. Risk-informing requires a full understanding of the models and assumptions in the Level 2 PRA which can allow the analyst to trace back the quantitative results to their major contributors. The use and understanding of Level 2 PRA models can be significantly improved if explicit modeling practices are implemented.
2. More direct coupling of Level 1 and Level 2 PRA models can help the user to understand and better characterize the resulting accident sequences from the initiating events to the final release categories.
3. The timing associated with the major events during an accident progression should be obtained from the accident analysis codes such as MELCOR. This information would be important for emergency planning and they have to be noted as a part of the end states of the event trees and propagated throughout accident progression.
4. The post core damage activities such as SAMG (Severe Accident Management Guidelines) using the EDMG (Extended Damage Mitigation Guidelines) and systems for post Fukushima enhancements could affect or delay symptoms that trigger various FPB EALs. PRA models should reflect these activities and the performance of the associated components including their survivability in post core damage environment.

3 REFERENCES

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