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Protecting People and the Environment

Phenomena Identification and Ranking Technique (PIRT) Exercise for Ranking Low-Power Shutdown Plant Operating States and Outage Types

Appendices A - K

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is performing a full-scope, site Level 3 probabilistic risk assessment (PRA), using a four-loop pressurized water reactor (PWR), as the reference plant. During the development of the Level 3 PRA, specifically the low-power shutdown (LPSD) analysis, the need to prioritize the plant operating states, hazards, and outage types to include in the full-scope site Level 3 PRA was identified by the Level 3 PRA project team. This need was further magnified by the fact that realistic LPSD modeling of plant outages involves consideration of the range of types of outages, from planned refueling and maintenance outages to unscheduled maintenance outages, and the significant variation in the types of activities that are performed during these outages. In short, the scope to develop a PRA for each of these LPSD plant configurations is a resource-intensive undertaking.

Because of the significant resources required to develop a full-scope LPSD PRA model, the Level 3 PRA project team decided to use the Phenomena Identification and Ranking Technique (PIRT) process to identify and prioritize the plant operating states, hazards, outage types, and other influences to include in the full-scope site Level 3 PRA. Pacific Northwest National Laboratory (PNNL) was contracted by NRC to coordinate and facilitate this PIRT process.

This report describes the PIRT process developed and used to meet the project objectives and associated results of implementing the process. The objective was to identify the plant operating states (POSs) and plant outage types (POTs), rank them according to their importance to LPSD risk in the context of different hazards, and consider important influences/phenomena (e.g., systems/components out of service, fission product inventory, thermal-hydraulics, status of reactor coolant system pressure and containment boundaries, operator/maintenance activities) associated with an LPSD model in the ranking process. The PIRT process focused on activities that potentially result in damage to fuel during LPSD operations and while the fuel is in the reactor pressure vessel. POSs and POTs specific to LPSD operation at the reference plant were evaluated by the PIRT panel, whose purpose was to identify and rank plant operating states, hazards, and outage types according to their importance to LPSD risk, and to consider important influences/phenomena associated with LPSD in the ranking process. Plant-specific hazards evaluated by the PIRT panel were internal event hazards, internal flooding hazards, internal fire hazards, and seismic hazards. Many of the plant-specific information sources are based on revisions from 2012 and earlier. The information does not necessarily represent the reference plant as currently operated today. However, the information and insights gained are deemed to be generally applicable to the low power and shutdown operation of a four-loop PWR.

FOREWORD

This report provides the results from a formal expert elicitation performed in support of a comprehensive probabilistic risk assessment (PRA) study for a four-loop pressurized water reactor (PWR). The U.S. Nuclear Regulatory Commission (NRC) is performing this work in response to Commission Staff Requirements Memorandum (SRM) SECY 11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities.” This PRA study (commonly referred to as the Level 3 PRA project) covers an ambitious scope and includes all major reactor fuel radiological sources (i.e., reactors, spent fuel pools, and dry cask spent fuel storage), all reactor modes of operation (full power, low power, and shutdown), and all hazard categories (internal and external). A significant challenge in performing PRA assessments for low power and shutdown (LPSD) modes of operation is the large number of potential plant operating states and hazard combinations that must be considered. Each plant operating state represents a unique combination of plant operating conditions (e.g., pressure, temperature, power level, and decay heat generation) and equipment configurations (e.g., reactor coolant system status and potential maintenance configurations). For this PRA study, twenty unique LPSD plant operating states were identified, each of which may require distinct modeling in order to realistically capture the plant response to the variety of hazards considered in the study. Because of the impracticality of analyzing the large number of potential LPSD plant operating state and hazard combinations, it is necessary to prioritize analytical work on the most risk significant areas. Therefore, the objective of the work documented in this report was to prioritize plant operating state and hazard category combinations using a systematic and formalized expert elicitation process. In addition to supporting the Level 3 PRA project, this work also supports efforts to address Commission direction in SRM SECY 11-0172, “Response To Staff Requirements Memorandum COMGEA-11-0001, "Utilization of Expert Judgment in Regulatory Decision Making.” In particular, experience gained form this expert elicitation process has helped to inform guidance being developed for expert elicitation processes and identified further areas for improvement in this important area.

A unique aspect of this work was the application of the Analytic Hierarchy Process (AHP), developed by Thomas L. Saaty, to a Nuclear Regulatory Commission research project. The AHP is a structured, transparent, and reproducible approach that uses the judgment of experts to decompose a decision problem and identify and rank the important or dominant parameters. With the AHP, expert panel members make pair-wise comparisons and develop explicit evaluation criteria to consistently rank the importance of one factor in relation to other factors that are important to the top-level goals, which, for this application, were core damage frequency and fission product release from a damaged core to outside of containment. AHP has been used for several decades in a number of application areas such as engineering, decision support, and resource allocation. The application of this technique to this study has provided insights for the usefulness and practicality of the AHP approach for future NRC research efforts.

This report provides a comprehensive summary of the methods used to plan and execute the expert elicitation, as well as a summary of insights, conclusions, and challenges for LPSD PRA modeling. It should be noted that the results from this study are specifically applicable only to the conditions for the reference plant and are not necessarily generalizable to other facilities. However, the generic aspects of the expert elicitation process, as well as the modeling approach for LPSD PRA, may be useful for other applications.

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The authors extend their thanks to the members of the phenomena ranking and identification technique (PIRT) panel whose insights formed the basis for the results and conclusions reached in this report. The PIRT panel members also provided valuable editorial and technical comments that have been incorporated into this report. The panel members included:

- Mr. Jeffrey Julius — Jensen Hughes, Inc.
- Mr. Ken Kiper — Westinghouse Electric Company
- Mr. James Ledgerwood — Westinghouse Electric Company
- Mr. Jeff Mitman — U.S. Nuclear Regulatory Commission
- Ms. Marie Pohida — U.S. Nuclear Regulatory Commission
- Mr. Stephen Prewitt — The E Group
- Mr. Donald Wakefield — ABS Group

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) is performing a full-scope, site Level 3 probabilistic risk assessment (PRA), using a four-loop pressurized water reactor (PWR) as the reference plant. During the development of the Level 3 PRA, specifically the low-power shutdown (LPSD) analysis, the need to prioritize the plant operating states, hazards, and outage types to include in the full-scope site Level 3 PRA was identified by the Level 3 PRA project team. This need was further magnified by the fact that realistic LPSD modeling of plant outages involves consideration of the range of types of outages, from planned refueling and maintenance outages to unscheduled maintenance outages, and the significant variation in the types of activities that are performed during these outages. Because of the significant resources required to develop a full-scope LPSD PRA model, the Level 3 PRA project team decided to use the Phenomena Identification and Ranking Technique (PIRT) process to identify and prioritize the plant operating states, hazards, outage types, and other influences to include in the full-scope site Level 3 PRA.

The PIRT approach applies a structured process for eliciting judgments from technical experts about difficult technical questions in lieu of other means, such as testing or analysis, which may be resource intensive or implausible. At the heart of a PIRT application is the ranking by experts of the factors important to a particular concern. For this application, the figure of merit is the importance to LPSD risk. Importance ranking requires evaluation criteria by which to judge the importance of the factors.

A technique known as the Analytical Hierarchy Process (AHP) was used to implement the PIRT process on this project. The AHP is a structured, transparent, and reproducible approach that uses the judgment of experts to decompose a decision problem and identify and rank the important or dominant parameters. With the AHP, the PIRT panel members make pair-wise comparisons and develop explicit evaluation criteria to consistently rank the importance of one factor in relation to other factors that are important to the top-level goals. The top-level goals for this application are core damage frequency (CDF) and fission product release from a damaged core to outside of containment.

There are other approaches, besides AHP, that could be employed to implement a PIRT process. While there have been criticisms of AHP, the authors feel that AHP is well-suited to this problem due to its structured approach for comparing the many different factors that can contribute to LPSD risk. Another motivating factor for choosing AHP in this study is the lack of documented applications of AHP in NRC-sponsored studies. The strengths and weaknesses of an application of AHP are documented with this study.

Inherent to eliciting technical judgment from experts are issues such as the possibility of a nonrandom sample of experts, experts with different levels of familiarity with the available data, experts with different motivations, dependent experts, and experts who provide outlier judgments. To address these issues, the PIRT process developed for the LPSD PRA application utilized experience with the application of PIRT and expert elicitation processes on previous NRC projects, and associated NRC guidance. For example, the basic principles incorporated into the NRC expert elicitation guidance for performing probabilistic seismic hazard analyses, referred to as the Senior Seismic Hazard Analysis Committee (SSHAC) process, were embedded in the expert elicitations performed for the PIRT process used in this study.

The PIRT process developed and applied for the LPSD PRA application utilized unique features, including the first application of the AHP technique in an NRC PIRT application and heavy reliance on Web conferencing to conduct the expert elicitations. A summary of the seven step PIRT process follows:

1) Prepare a detailed problem description. The purpose of the detailed problem description is to develop a common understanding from the entire project team, including members of the expert panel, on the scope of the PIRT process. Included in this step was development of the PIRT process, and communication of the process with the project team and with the experts. Also included in this step was providing reference materials pertinent to the LPSD PRA PIRT process to all of the members of the expert panel.

2) Create a PIRT evaluation team. The PIRT evaluation team consisted of a PIRT coordinator and PIRT facilitator, whose responsibilities were to organize the problem being addressed by the PIRT process and facilitate interactions with the experts; the panel of experts whose judgments were to be elicited during the course of the PIRT process; and the participatory peer reviewers (PPRs). The expert panel consisted of seven experts from the nuclear power plant industry and the NRC having experience with LPSD PRA development, the reference plant operation during low power evolutions, and nuclear power plant outage management and operations. Two PPRs from the NRC monitored the expert elicitation process for the purpose of avoiding systemic biases in the elicitation process and enhancing the breadth of the knowledge on which the judgements by the panel experts were based.

3) Hold a series of PIRT process familiarization meetings. Web conferencing was used for four separate PIRT process familiarization meetings using the GoToMeeting® collaboration software in combination with standard audio conference calling technology. The entire project team, including the experts and PPRs, participated in each of these meetings. During these meetings, 1) the problem statement and overall PIRT process was presented and discussed, 2) the general AHP process was described and how it was being employed as the central feature of the PIRT process, 3) an overview of the preliminary set of LPSD PRA PIRT parameters for use in the PIRT exercise was provided to the expert panel for feedback and comment, and 4) the PIRT elicitation forms were presented for feedback and comment by the expert panel.

4) Solicit input from the expert panel about parameters to use in the PIRT. The objective of this step was to ensure that all potentially important POSs, hazards, and POTs were included in the PIRT exercise, that appropriate evaluation criteria were identified, and that the evaluation criteria was sufficient to judge the importance of the POSs for different hazard events relative to CDF and radiological releases due to core damage.

5) Hold individual PIRT elicitation sessions. Elicitation sessions were held individually with each member of the expert panel over a one week period. The same PIRT elicitation forms were used in each elicitation session. Web conferencing (i.e., GoToMeeting®) was used to conduct the elicitation sessions. Trial sessions were held to test the forms and associated process prior to the individual PIRT elicitation sessions. Also, to support the experts and to facilitate consistency, a set of written PIRT Elicitation Instructions were sent to each expert panel member prior to the individual elicitation sessions. To promote consistency in how the individual sessions were conducted, a PIRT facilitator checklist was developed and used by the facilitator during the individual PIRT elicitation sessions.

6) Hold a group PIRT elicitation meeting. This elicitation session was held in a central location with all expert panel members and PPRs present. The results from the individual PIRT elicitation sessions were reviewed. The review focused on looking at the most important POSs and the differences in the importance weights and final priorities between experts. Following the review of these results, for each of the PIRT elicitation forms, each expert, in turn, was given an opportunity to present their responses to the form (based on how they filled it out for the individual sessions) and to describe the rationale and basis for those responses. Discussion among the expert panel members during this process was encouraged but monitored to keep the discussion limited to subjects relevant to the information elicited by the forms and to ensure that there was enough time to discuss all forms during the meeting.

7) Analyze and report the results. Analysis of the results from the group elicitation session was performed during the weeks following the group meeting after the final results were received from the experts. The analysis included compilation of the results, assessment of the results, development of technical insights about the importance of LPSD POSs, POTs, and hazards, and development of insights about the LPSD PIRT PRA process. All members of the project team, including the expert panel members and the PPRs, were provided the opportunity to review and comment on the final results and draft report.

The objective of the PIRT process was to identify the POSs and POTs, rank them according to their importance to LPSD risk in the context of different hazards, and consider important influences/phenomena (e.g., systems/components out of service, fission product inventory, thermal-hydraulics, status of reactor coolant system pressure and containment boundaries, operator/maintenance activities) associated with an LPSD model in the ranking process. The PIRT process focused on activities that potentially result in damage to fuel during LPSD operations and while the fuel is in the reactor pressure vessel. POSs and POTs specific to LPSD operation at the reference plant were evaluated by the PIRT panel. The specific POSs are identified in Table ES-1. The four POTs evaluated were refueling outage, hot standby outage, and maintenance outage with and without draining of the reactor coolant system (RCS).

The AHP evaluation criteria developed for the LPSD PRA PIRT process are organized into a hierarchy as follows: top-level goal, then top-level criteria, then sub-criteria. Each of the criteria and sub-criteria are evaluated by each member of the expert panel in terms of importance to the top level goal using a pair-wise comparison process. The ranking of importance of each sub-criteria for each POS is then evaluated by each of the experts. The top-level evaluation criteria and sub-criteria, along with corresponding evaluation questions that apply the criteria to each POS, are used to determine the relative importance of each POS to the top-level goals. The top-level goals were defined during development of the project scope and problem definition to be the basis for determining the priorities associated with POSs, POTs, and hazards relative to the two risk metrics: core damage frequency and radiological releases from a damaged core.

The AHP hierarchies were developed for each of four hazard groups (internal events, internal fire events, internal flooding events, and seismic events) for each of the two risk metrics. Accordingly, there are a total of eight top-level goals and corresponding hierarchies. Three types of top-level criteria were defined. The first is based on critical safety functions for preventing core damage, which are the same for each of the hazard groups, and consist of RCS inventory control, heat removal from the reactor core, and RCS integrity. The second type of top-level criteria considers the hazard challenges (which is hazard specific). The third type of top-level criteria considers containment integrity (which is specific to the goals for release from a damaged core). Accordingly, there are a total of eight top-level criteria, not all of which apply to each top-level goal. Finally, two sub-level criteria were defined for each top-level criteria, for a total of 16 sub-criteria.

The results of the expert elicitations are primarily presented as plots addressing the following: (1) the aggregated and individual POS priorities for each of the eight top-level goals, (2) the aggregated and individual POS sub-criteria importance weights for each top-level goal, (3) the top-level evaluation criteria importance weights for each expert for each top-level goal, (4) the sub-criteria importance weights for each expert for each top-level criterion, and (5) a summary of the majority POS sub-criteria ranking categories assigned to each POS. These results are presented and discussed extensively in Sections 4.0 and 5.0 of this report.

The primary insights and focus of this study concern the importance ranking of the POSs for different events considered in LPSD PRA. However, lessons were also learned about the process of performing the LPSD PRA PIRT elicitation and suggestions were offered by the experts in a supplementary brainstorming session on LPSD PRA modeling gaps about how to resolve identified uncertainty issues. Each of these is summarized here.

Insights from POS Priority Results

The primary insights derived from this PIRT assessment involved the following:

- Identification and ranking of POSs that are important contributors to risk and that should be considered for inclusion in a detailed LPSD PRA.
- Identification and discussion of factors that are important contributors to the ranking/prioritization of POSs and which may merit further investigation.
- Insights related to the importance of the different POTs, including maintenance outages.

The aggregated POS priorities for each of the eight top-level goals for all POSs are presented in Figure ES-1. A short description of each POS is presented in Table ES-1. Referring to Figure ES-1, the more “red” the POS the more risk-significant the experts considered the POS relative to the other POSs. Conversely, POSs colored “white” were evaluated by the experts to be of low risk-significance relative to the other POSs. The aggregate POS rankings/priorities determined by the LPSD PIRT elicitation for each of the top-level goals supports a shared perception among the experts that POSs with reduced coolant inventory (i.e., mid-loop operations and draining the RCS down to the reactor pressure vessel flange, or POSs 5B, 6, 6-P3, 9, and 10) are the most significant POS contributors to LPSD risk. However, other POSs cannot be entirely dismissed due to the limited state of the knowledge of LPSD risk and the variability of plant conditions and practices. For example, POS 5A in which the pressurizer is water solid for hydrogen degassing is nearly as risk-significant as the mid-loop operation POSs. Generally, POSs most important to core damage are also most important to release from a damaged core.

The PIRT results show that the important contributors to the POS priorities are RCS integrity, human-initiated events, and vulnerability to internal fire, internal flooding, and seismic hazards. RCS isolation is also identified as an important contributor to all top-level goals for some POSs. Across experts, the consistency comparisons of ranking category assignments show that there is, in general, a very high level of agreement between experts on the ranking categories assigned to the RCS isolation. However, because this factor is a significant contributor to risk, and given that the plant configurations and operating conditions during mid-loop operations and RCS draindown operations are not comparable to plant conditions at full power, a more complete understanding of the thermal-hydraulic responses of plants to the loss of RCS isolation events would be helpful. Regarding factors important to release, containment isolation consistently ranked higher than radionuclide suppression.

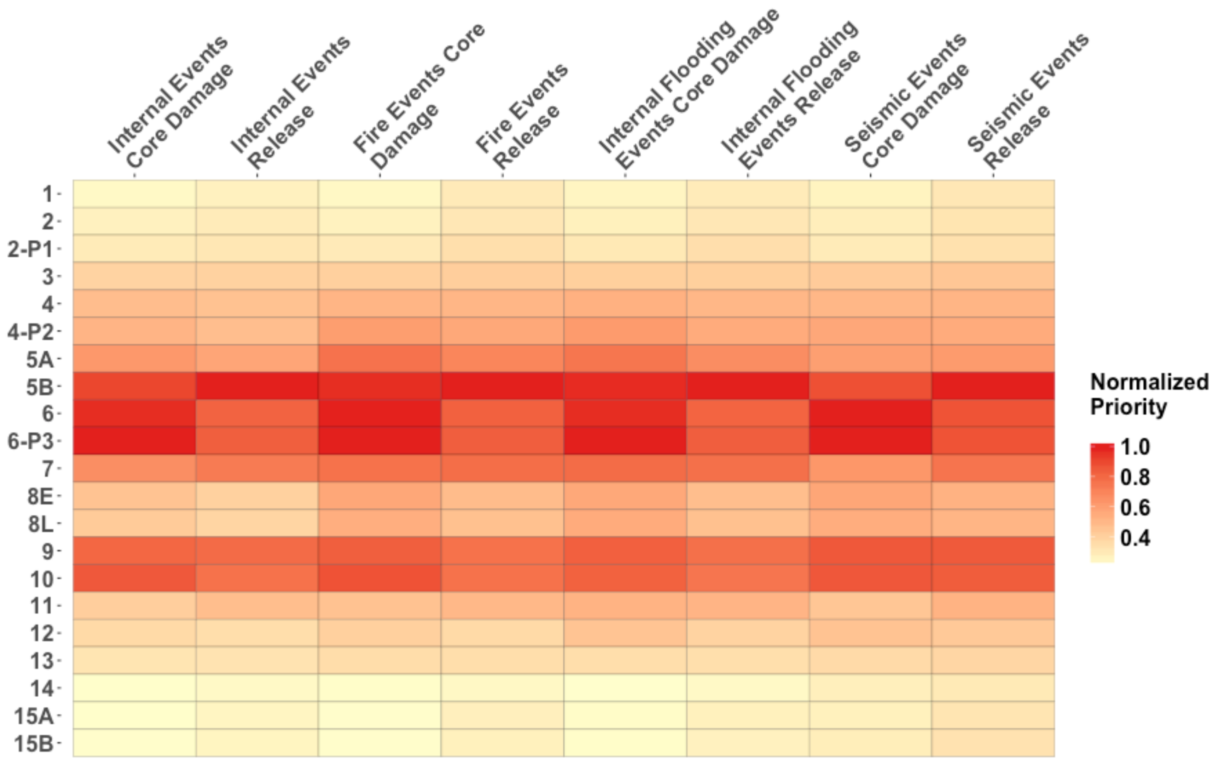

Figure ES-1 Total Aggregated POS Priorities for Each Top-Level Goal

Table ES-1 Short Description of Each POS

POS No.	Short Description	POS No.	Short Description
1	Low power coast down	8E	Refueling (old core)
2	Cooldown Mode 3	8L	Refueling (new core)
2-P1	Hot standby outage	9	Draining after refueling
3	Cooldown Mode 4	10	Mid-loop after refueling
4	Cooldown Mode 5	11	Refill RCS
4-P2	Cooldown Mode 5 maintenance outage	12	Heatup Mode 5
5A	Pressurizer water solid	13	Heatup Mode 4
5B	Reduced water inventory	14	Heatup Mode 3
6	Mid-loop before refueling	15A	Startup <5% power
6-P3	Mid-loop drained maintenance	15B	Startup >5% power
7	Filling refueling cavity		

The PIRT results show that human-initiated events are a significant contributor to POS priority for core damage due to internal events for the mid-loop operation and RCS draindown POSs. The consistency comparison across experts of ranking category assignments shows there is disagreement about the ranking categories assigned to human-initiated events. During low power shutdown operations there is a general increase in operator, maintenance, and other activities and a decrease in the availability of instrumentation and control compared to activity at full-power operation, which can contribute to errors. It would be helpful to better understand the frequency and consequence of human-initiated events at different POSs associated with low power shutdown.

The PIRT results show that vulnerabilities to internal fire, internal flooding, and seismic events are significant contributors to POS priority for core damage and release from a damaged core for most POSs. Across experts, the consistency comparison of ranking category assignments shows there is a relatively significant level of disagreement between the experts about these ranking category assignments for the sub-criteria. The disagreement in these category assignments correlates to the lack of published studies about the vulnerability of PWRs during low power shutdown to internal fires, internal flooding, and seismic events. It would be helpful to understand how differences in plant configurations and activities during low power shutdown operations compared to full-power operations (e.g., temporary removal of fire and flood barriers and pipe snubbers and hangers) contribute to risk.

Lessons Learned Exercising the PIRT Process

The following are the key lessons learned from exercising the PIRT process for this application:

- There may be value in consulting with the expert panel members more explicitly in the early stages of the PIRT process about setup of the PIRT process.
- Significant benefit is derived from conducting trial PIRT elicitation sessions.
- There is a trade-off between whether or not to show the computations associated with determining the importance weights from the pair-wise comparisons on the forms.

In the group meeting, an expert pointed out that LP operating modes are more like full-power operation than like SD modes and if LP and SD POSs were addressed in separate PIRTs, the identified evaluation criteria and corresponding results might have come out differently. In retrospect, it is difficult to know how much difference this could have made in the PIRT elicitation because most experts appeared to make accommodations in their mental models to apply the same evaluation criteria to LP and SD. This observation did not surface during formal solicitation of feedback about the PIRT parameters. It might have proven helpful to consult with the expert panel members more explicitly in the early stages of the PIRT development process.

As a way to gauge the time needed to hold an expert elicitation session and to identify any issues with the PIRT elicitation forms, trial elicitation sessions were performed using the PPRs as substitute experts. These sessions tested how much time would be required to go through the forms with each expert and helped identify and correct mistakes on the forms. It also helped generate insights about how to best improve the elicitation forms and process. Based on comments from the PPRs, a number of improvements were made in the design of the LPSD PRA PIRT elicitation forms to make them easier and more intuitive to fill out. Also, specific instructions and warnings were added to the PIRT Elicitation Instructions provided to the experts prior to sessions based on the reviewers' comments. The information acquired during the trial sessions was invaluable because it not only changed the strategy about what to try to accomplish in online PIRT elicitation sessions but was also the basis for making helpful improvements in the elicitation forms and process.

In designing PIRT elicitation forms, there seems to be a trade-off between whether or not to show the computations associated with determining the importance weights from the pair-wise comparisons on the forms. Many of the forms elicited pair-wise comparisons from the expert to determine the relative importance of one criterion relative to the other criterion and relative to the top-level goal (i.e., importance to core damage or release from a damaged core from internal events, internal fire, internal flooding and seismic events). Prior to the trial PIRT elicitation session, a version of the forms was created that explicitly showed the importance calculations based on the elicited responses. This computational information was deliberately removed from the form to make it less confusing because without a lot of familiarity with the calculations they can be misleading. On the other hand, the experts stated that during the group elicitation meeting they wanted to understand how the calculations were performed (which was provided earlier during the second familiarization but not presented on the forms), so that they could judge whether the resulting importance weights matched their intuition. There seems to be some value in letting the experts perform a sanity check on their results, but there is also some danger in that the results could then be manufactured to match preconceived ideas.

Insights from Brainstorming of LPSD PRA Modeling Challenges

As an activity separate from the PIRT elicitation, the experts brainstormed generic LPSD PRA modeling issues. Identification of these modeling issues is considered an important part of this study because the issues represent an important source of uncertainty for the PIRT elicitation. Of the challenges identified, the following were selected as being representative of the most important kinds of issues identified:

- lack of sufficient information about the frequency of internal, internal fire, and internal flooding events at LPSD and the vulnerability of LPSD POSs to these events
- lack of information about when and which internal fire and internal flooding prevention and mitigation features are bypassed during LPSD
- lack of thermal-hydraulics calculations for LPSD to sufficiently characterize the conditions associated with SD accident sequences
- lack of an enhanced human reliability analysis (HRA) methodology to address the complexity and number of human actions associated with LPSD, particularly those actions taken outside the main control room (MCR).

Some of the experts believed that the frequencies of initiating events for internal events are higher per hour during LPSD compared to full-power operation based on industry information. Furthermore, they presumed that internal fire events and internal flooding events are similarly more frequent at LPSD. This higher frequency may translate to a commensurate increase in risk and therefore challenges the contention that full-power PRA risk bounds low-power risk. However, this higher event frequency may be offset by the fact that initiating events are most likely to occur due to restart from refueling when the decay heat is relatively low. It would be helpful to assess LPSD internal events initiating events frequencies to determine why they are higher per hour at LPSD and to evaluate whether the plant configurations and operations at LPSD are more vulnerable to these events. Likewise, it would be helpful to know whether internal fire and internal flooding events are more frequent during LPSD and whether the plant configurations and operations during LPSD are more vulnerable to internal fire and internal flooding events. A survey of several plants is needed to compile details about LPSD events and their frequency.

Information about standard practices during outages related to internal fire and internal flooding prevention and mitigation features (i.e., bypassing fire and flood barriers, fire detection and suppression systems, and flooding detection systems) would contribute to understanding the risk associated with LPSD POSs. A survey of the administrative controls and operating experience of plants regarding bypassing internal fire and internal flooding prevention and mitigation features during shutdown POSs is needed. This survey should include information about compensatory actions taken when these features or systems are bypassed and information about how quickly these features or systems can be put back into service.

Many of the thermal-hydraulics conditions associated with shutdown accident sequence should be evaluated. These conditions include (1) rapid expulsion of coolant due to loss of residual heat removal (RHR) capability with an opening in cold leg and the lack of a large RCS vent path such as a hot leg steam generator plenum manway; (2) nozzle dams without an adequate RCS vent path; (3) overpressure due to loss of RHR capability; (4) surge line flooding (e.g., pressurizer manway is open, RCS inventory is entrained in the pressurizer, the pressurizer level is increasing while water in the core region is decreasing, and the time to core uncover is reduced); (5) inadequate pressure relief during POSs with RHR pressurized, station blackout (SBO), high-pressure steam through RHR, and an opening size too small to relieve pressure (which may be primarily steam); and (6) vortexing in which air entrainment causes erroneous RCS level indication and possible degradation in performance of the RHR pumps. Particular consideration should be given to scenarios that can lead to increased probability of bypassing containment. A review of industry data for evidence that these kinds of conditions can occur and a thermal-hydraulic analyses of a pilot plant should be performed. An expert elicitation with thermal-hydraulic and LPSD PRA experts should be conducted to develop risk significant scenarios that require analysis using thermal-hydraulic codes that can model risk significant phenomena such as surge line flooding.

HRA of actions taken during shutdown is more complex than for full power because of (1) the reliance of shutdown operations and post-initiating response on manual actions, (2) the potential for dependencies between human-caused initiating events and post-initiating human failure events (HFEs), (3) the need for more decisions associated with which procedure to follow, (4) the reduction in some instances of available instrumentation, (5) multiple work activities, (6) the possibility that an improper action in an early POS could affect a later POS, and (7) the wide variation in time available for operator actions, including Level 2 actions, which range from minutes to days. A study is needed to collect relevant data and analyze the error-forcing context.

ABBREVIATIONS AND ACRONYMS

AC	alternating current
AFW	auxiliary feedwater
AHP	analytical hierarchy process
AOP	Abnormal Operating Procedure
ATWS	anticipated transient without scram
BWR	boiling water reactor
CDF	core damage frequency
ECCS	emergency core cooling system
FMEA	failure mode and effects analysis
HFE	human failure event
HRA	human reliability analysis
IAEA	International Atomic Energy Agency
ISLOCA	interfacing system loss of coolant accident
LOCA	loss of coolant accident
LOOP	loss of offsite power
LP	low power
LPSD	low power shutdown
MCR	Main Control Room
MOV	motor-operated valve
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
PIRT	phenomena ranking and identification technique
PNNL	Pacific Northwest National Laboratory
PORV	power-operated relief valve
POS	plant operating state
POT	plant outage type
PPR	participatory peer reviewer
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPV	reactor pressure vessel
RWST	refueling water storage tank
SBO	station blackout
SD	shutdown

SSHAC	Senior Seismic Hazard Analysis Committee
TLC	top-level criteria
TS	technical specification

APPENDIX A LPSD PRA PIRT PROBLEM STATEMENT

Draft Problem Statement

Background from PNNL Statement of Work (SOW):

During the development of a Level 3 PRA, specifically analysis of low power shutdown (LPSD), the Level 3 PRA Project Team identified the need to prioritize the plant operating states, hazards, and outage types to include in the full-scope site Level 3 PRA given the large number of possible plant operating states in combination with different hazards and outage types. This need is further exacerbated by the fact that LPSD models typically model actual plant outage plans and schedules, which vary greatly from planned outage to planned outage. Since the resources required to perform comprehensive modeling would be impractical, the Level 3 PRA Project Team decided to use the Phenomena Identification and Ranking Table (PIRT) process to identify the plant operating states, hazards, and outage types to include in the full-scope site Level 3 PRA.

Scope of Work from PNNL SOW

PNNL will coordinate and facilitate an expert elicitation process using a PIRT process in accordance with guidance provided by the NRC staff. The objective of this work is twofold: (1) to prepare draft guidance and perform a pilot study based on the draft guidance in order to further enhance the proposed expert elicitation (PIRT) process and (2) resolve specific technical issues that have arisen on the Level 3 PRA project. The staff has identified an initial technical issue to apply the PIRT process to, which is to identify and rank the important plant operating states, hazards, outage types and other influences associated with a LPSD model for inclusion in the full-scope site Level 3 PRA.

The expert elicitation shall be performed in accordance with NRC provided guidance.

Overview of Current L3PRA LPSD Model

The L3PRA model is a full-scope site Level 3 PRA for a four-loop pressurized water reactor (PWR) plant. The L3PRA model is an integrated model that includes internal event scenarios, low power and shutdown event scenarios, and scenarios for other hazards. The current LPSD L3PRA considers LPSD risk during all modes of plant operation, and is structured around plant operating states (POSs) and plant outage types (POTs). The plant operating modes for a PWR[1] are provided in Table A-1. The POTs were identified based on a review of existing studies and are described in Table A-2. The POSs were developed based on a review of existing studies and plant-specific information, and are provided in Table A-3.

[1] Taken from the Westinghouse Standard Technical Specifications.

Table A-1 Plant Operating Modes

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMA L POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (^{0}F)
1	Power Operation	>= 0.99	>5	NA
2	Startup	>=0.99	<5	NA
3	Hot Standby	<0.99	NA	>= 350
4	Hot Shutdown[b]	<0.99	NA	350 > T_{avr} > 200
5	Cold Shutdown[b]	<0.99	NA	<= 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.
(b) All reactor vessel head closure bolts fully tensioned.
(c) One or more reactor vessel head closure bolts less than fully tensioned.

Table A-2 Plant Outage Types

POT	DESCRIPTION
1 Non-Drained Maintenance without the Use of RHR	Corresponds to forced outage maintenance in hot standby; i.e., above 350° F. The reactor is made subcritical (control rods fully inserted), the boron concentration is above the required concentration for hot shutdown, the reactor coolant system (RCS) is maintained hot, the reactor pressure vessel head remains closed, and decay heat is removed by use of the feedwater and steam generators dumping steam to the condensers via turbine bypass, or through the atmospheric dump valves. This plant state will be reached if short outage times are expected, such as due to inadvertent reactor trips, or if there are maintenance activities to be performed that do not require RCS cooldown.
2 Non-Drained Maintenance with the Use of RHR	Corresponds to maintenance while in hot shutdown or cold shutdown without RCS draining. The reactor is made subcritical (control rods fully inserted), the boron concentration is above the required concentration for cold shutdown, the RCS is cooled to below 350°F, the reactor pressure vessel head remains closed, and decay heat is removed by use of the residual heat removal system (RHR) aligned in the shutdown cooling mode. This plant state will be reached if longer outage times are expected, if there is a loss of the main heat sink, or if there are maintenance activities to be performed that require cooldown.
3 Drained Maintenance with the Use of RHR	Corresponds to reduced inventory operation with RCS level just below the flange and with fuel in the reactor pressure vessel (RPV). This state is entered if maintenance requires a low level of the reactor coolant system without unloading the core. The reactor coolant system is drained to below the flange level, but not all the way to midloop, with the reactor vessel head in place. Decay heat is removed by the residual heat removal system aligned in the shutdown cooling mode.
4 Refueling	Corresponds to a refueling outage. The first activities are similar to those for POT 3 up to the point of draining to below the flange. Starting with RCS level below the RPV flange, the reactor basin/fuel transfer canal is filled for transfer of the fuel assemblies to the spent fuel pool. During refueling outages many maintenance activities and surveillance tests are performed. Decay heat is removed by the RHR system and/or spent fuel pool cooling system depending on the location of the spent fuel assemblies.

Table A-3 Plant Operating States

POS No.	POS Description	Plant Operating Mode	RCS Power	RCS Tavg (°F)	RCS Boundary (Vent Status)	POS applicable when transitioning to outage type: Refueling (POT 4)	Maintenance w/o Drain, w/o RHR (POT 1)	Maintenance w/o Drain (POT 2)	Maintenance w/Drain (POT
0	Full power operation	1	100%	Normal Operations Temp (NOT)	RPV Head Intact	X	X	X	X
1	Low power and reactor shutdown	1,2	50 to 20 %	T_{avg}<NOT	RPV Head Intact	X	X	X	X
2	Cooldown with steam generators to 350°F	3	20 to 0%	350<T_{avg}<NOT	RPV Head Intact	X	X	X	X
3	Cooldown with residual heat removal system to 200°F	4	0%	200<T_{avg}<350	RPV Head Intact	X		X	X
4	Cooldown to ambient temperature with residual heat removal system only	5	0%	175<T_{avg}<200	RPV Head Intact	X		X	X
5A	Pressurizer water solid for hydrogen degassing	5	0%	T_{avg}=175	RPV Head Intact	X			X
5B	Draining the reactor coolant system to midloop	5,6	0%	~110<T_{avg}<175	RCS Vented[2], RPV Head Intact	X			X
6	Midloop operation[3]	5,6	0%	71<T_{avg}< 130	RCS Vented[2], RPV Head Intact , PORV may be open	X			X
7	Filling refueling cavity for refueling operation	6	0%	71<T_{avg}< 130	RCS Vented[2], RPV Head de-tensioned	X			
8E	Refueling operation (OLD CORE)	6	0%	71<T_{avg}< 130	RCS Vented[2], RPV Head removed	X			
DF	DEFUELED	n/a	0%	71<T_{avg}< 130	RCS Vented[2], RPV Head removed	X			
8L	Refueling operation (NEW CORE)	6	0%	71<T_{avg}< 130	RCS Vented[2], RPV Head removed	X			
9	Draining the reactor coolant system to midloop after refueling operation	6	0%	71<T_{avg}< 130	RCS Vented[2], RPV Head removed	X			
10	Midloop operation after refueling[3]	5,6	0%	71<T_{avg}< 130	RCS Vented[2], RPV Head tensioned	X			
11	Refill reactor coolant system	5,6	0%	T_{avg} ~125	RPV Head Intact	X			X
12	Reactor coolant system heatup/draw bubble in pressurizer	5	0%	~125<T_{avg}<175	RPV Head Intact	X			X
13	Reactor coolant system heatup to 350°F	4	0%	175<T_{avg}<350	RPV Head Intact	X		X	X
14	Startup with steam generators (AFW) to Hot Standby	3	0%	350<T_{avg}<557	RPV Head Intact	X		X	X
15A	Reactor startup and low power operation	1,2	0=Power<5 %	< NOT	RPV Head Intact	X	X	X	X
15B	Reactor startup and low power operation	1,2	5<Power<50%	< NOT	RPV Head Intact	X	X	X	X

2 RCS may be vented via power-operated relief valves (PORVs), pressurizer manways, or safety valves. Purge valves are not considered vent paths.
3 POS 6 and 10 contains the duration independent initiating event for overdraining the RCS.

Issues to be Addressed by the LPSD PRA PIRT Project

The NRC L3PRA Technical Advisory Group (TAG) provided recommendations for the development of the LPSD element of the L3PRA. These recommendations are to be performed in three phases: Phase 1 – addresses the near-term focus of the LPSD analysis needed to support the overall objectives of the L3PRA Project, Phase 2 – to be conducted in parallel with Phase 1, implements a PIRT process to identify, define, and rank important risk aspects for the LPSD PRA, and Phase 3 – to be conducted after completion of the PIRT process and as resources are available, addresses important technical issues identified by the PIRT panel. With regard to Phase 2, the TAG provided the following recommendation:

> *This phase identifies, defines, and ranks important risk aspects for LPSD PRA. The purposes of the panel are, in order of suggested priority, to:*
>
> a. *Identify activities (if any) that need to be performed as part of the L3PRA LPSD analysis;*
>
> b. *Support the planning of potential future (post-L3PRA Project) activities; and*
>
> c. *Provide a context for the L3PRA LPSD analysis results.*[4]
>
> *The TAG expects that the expert panel will, consistent with a PIRT process, be tasked with identifying, defining, and ranking important risk aspects (e.g., systems, components, processes, and phenomena*[5]*) for LPSD PRA to support these objectives.*

Topics and associated issues recommended for discussion by the PIRT panel are listed in Table A-4 (which are not presented in any order of priority).

Purpose of the LPSD PRA PIRT Panel

The purpose of the PIRT panel is to identify the plant operating states, hazards, and outage types and to rank these according to their importance to LPSD risk, and to consider important influences/phenomena (e.g., systems/components out-of-service, fission product inventory, thermal-hydraulics, status of RCS pressure and containment boundaries, operator/maintenance activities) associated with a LPSD model in the ranking process. However, because of resource (funding) limitations, not all LPSD issues and potential phenomena, including all those identified by the TAG, can be fully considered. Hence, the scope of the LPSD PRA PIRT project, which was informed by an NRC/PNNL staff conference call held on July 7, 2016, is prioritized as described below.

In-scope. Topics/issues specific to activities that potentially result in damage to fuel during LPSD operations and while the fuel is in the Containment Building (i.e., while the fuel is in the reactor pressure vessel or during its transfer within the Containment Building), and include:

- Plant operating modes, POTs, and POSs identified in Tables A-1, A-2, and A-3. Additional POTs/POSs identified by the experts will also be included. Identification of

[4] Purpose (c) can help in communicating the results of the analysis (e.g., to provide a qualitative or semi-quantitative indication of the risk importance of the analysis actually done).

[5] Wilson, G. E., and B. E. Boyack, "The Role of the PIRT Process in Experiments, Code Development, and Code Applications Associated with Reactor Safety Analysis," Nuclear Engineering and Design, 186, 2-37 (1998) [cited in NUREG/CR-6844 as one of the general references for the PIRT process].

new/variant POTs/POSs will be based on plant-specific evolutions or changes in plant configuration that result in a potentially significant change in the plant risk posture. Measures of risk posture to be used by the PIRT panel include contribution to core damage frequency (CDF), frequency/consequence of off-site release, and/or other metric(s) defined by the PIRT panel.

- Internal and external event hazards. Previous LPSD analyses (e.g., NUREG/CR-6144) have shown that internal fire events and internal flood events, in addition to internal initiating events, can be major contributors to CDF during LPSD operations. These hazards will be a primary focus of the PIRT panel. However, it is recognized that the focus of these previous analyses was on mid-loop operations, therefore seismic events and other hazards may be considered if determined to be important enough to evaluate by the PIRT panel members, though perhaps not at the same level of detail.

- Standard and plant-specific initiating events identified in at-power PRAs, in previous LPSD analyses, and in the experience of the experts. Because of the recognized importance of operator actions that can cause initiating events (i.e., at-initiator operator actions) during LPSD operations, the potential contribution of these plant-specific actions to risk will be a consideration that the PIRT panel will be explicitly asked to consider. However, a systematic process (e.g., HAZOP analysis, FMEA) to identify all possible initiating events in all POSs in all POTs is beyond the scope of this PIRT study.

- Standard and plant-specific equipment/operator failure events that contribute to risk during LPSD operations. This includes important pre-initiator and post-accident human failure events reflective of available plant procedures/operations.

Not In Scope. Activities that potentially damage fuel during fuel movement and interim storage external to the Containment Building. Examples include: fuel transfers to the spent fuel pool after it leaves the Containment Building, spent fuel pool storage and associated operations, and dry fuel storage. Other constraints on the scope are as discussed above. Also, “other influences,” as described above, will be considered to the extent plant-specific information is available.

<u>Expert Elicitation Process to be Used in this Project</u>

The NRC staff have developed an early draft guidance document for using formal expert judgement in regulatory decision-making, as directed by the Commission in SRM-SECY-11-0172 (ADAMS accession number ML121600282). However, while this guidance is intended to be generally applicable to different types of expert judgements and tailored to specific applications as necessary, it was determined that guidance specific to implementing PIRT was desirable. Based on this it was decided that PNNL would develop an expert elicitation (PIRT) process to use on this project. As noted in the “Scope of Work from PNNL SOW,” one of the objectives of this project is “to prepare draft guidance and perform a pilot of the draft guidance in order to further enhance the proposed expert elicitation (PIRT) process.” PNNL staff are currently developing this process and have submitted the draft to the NRC for comment. Key aspects of the process being developed are as follows:

- PNNL staff will use the draft NRC expert elicitation guidance to the extent practical, including employing the use of web-based meetings to minimize cost.

- PNNL staff will rely heavily on previous NRC experience and lessons learned with implementing the PIRT process.

- The Analytical Hierarchy Process (AHP) developed by Thomas Saaty, and highly recommended by Wilson and Boyack[6] for use in implementing the PIRT process, will be the basic process employed on this project. The AHP process has been extensively used and studied since it was first developed.

- The main objective of the PIRT process is to develop a qualitative ranking of plant operating states, hazards, and outage types according to their importance to LPSD risk, and to explicitly account for important influences/phenomena in this ranking process. Quantitative data values for use in the L3PRA will not be developed.

[6] Wilson, G. E., and B. E. Boyack, "The Role of the PIRT Process in Experiments, Code Development, and Code Applications Associated with Reactor Safety Analysis," *Nuclear Engineering and Design*, **186**, 2-37 (1998) [cited in NUREG/CR-6844 as one of the general references for the PIRT process].

Table A-4 LPSD Topics and Associated Issues Recommended by the TAG

LPSD Topic	Issues
1. Increase in initiating event frequencies at low power	a. Is this increase due to the type of outage evolution? b. Does it occur while shutting down or starting up? Note: This issue is tied to the analysis scope: it depends on what evolutions and conditions are evaluated. For example, if startup is addressed qualitatively, then the level of effort on this issue is greatly (if not entirely) reduced.
2. External hazards at shutdown	a. What aspects of the plant's LPSD operations and conditions might require extensions of the at-power external hazards analyses? For example: i. What are the hazards that can trigger pre-emptive shutdown? ii. What are the hazards that would challenge important outage configurations (e.g., hatch off)? iii. Do POSs screened on the basis of time to core damage require further analysis for extreme external hazards causing extended losses of offsite power and/or ultimate heat sink (and presumably major disruptions of offsite infrastructure)? iv. Will any aspects of LPSD operations challenge screening assumptions made in the at-power external hazards analyses? (Example: screening based on the assumed amount of hazardous materials involved in a transportation accident and the assumption of automatic control room isolation.) v. Will any aspects of LPSD operations and conditions challenge modeling assumptions made in the detailed at-power external hazards analyses? (Examples: taking seismic snubbers out of service, hanging lead shielding from pipes, potentially increased reliance on manual field actions leading to increased exposure to external hazards.) vi. Can LPSD operations lead to offsite conditions that might affect the Level 3 analysis for external hazards? (Example: increased local population increases traffic network load, potentially increasing vulnerability to network disruptions caused by the external hazard.) b. Which are the most important external hazards to address with a detailed analysis? c. Seismic events and other external hazards were not major contributors to LPSD CDF according to NUREG/CR-6144. However, new information (e.g., updated CEUS seismic hazards, changed operational practices during outages) might lead to a different risk understanding. If time permits, consider identifying the operational and analytical changes since NUREG/CR-6144 that could change the external hazard risk insights from that study. (This will likely support the development of the final project report.)
3. Internal hazards at shutdown	a. What aspects of the plant's LPSD operations and conditions might require extensions of the at-power internal hazards analyses? For example: i. Internal floods – increased number of human-induced flooding events; increased presence of workers who can detect water movement; flood barriers that would be expected to be open and not easily closed (e.g., doors with hoses or cables running through) based on historical practice; changes in flood sources (e.g., fire protection water valved open to containment). ii. Internal fires – increased number of human-induced fire events; increased presence of workers who can detect fire; fire barriers that would be

Table A-4 LPSD Topics and Associated Issues Recommended by the TAG (continued)

LPSD Topic	Issues
	expected to be open and not easily closed (e.g., doors with hoses or cables running through) based on historical practice. b. Are there any internal hazards not modeled in detail in the at-power analysis that appear to warrant a detailed analysis for LPSD?[7] (Example: heavy load drops.) i. What processes are in place to control these hazards? ii. What is the operating experience for these hazards? (Major events, causes and impacts.) c. In NUREG/CR-6144, fire was the dominant risk contributor (using pre-NPFA 805 models) for shutdown CDF and internal floods were as important as internal events. Recent events have shown that heavy load drops can have unanticipated effects. Recent RES experiments have shown that HEAF events can have a greater impact than previously believed. New reactor reviews have shown that fires during reduced inventory operation with the RCS vented (open pressurizer manway, etc.) and the SGs unavailable for sustained decay heat removal are a significant contributor to LPSD risk. Flooding is not a major contributor if flooding barriers remain intact during the outage. If time permits, consider identifying the operational and analytical changes since NUREG/CR-6144 that could change the internal hazard risk insights from that study. (This will likely support the development of the final project report.)
4. Containment isolation	a. What are the outage configurations when the equipment hatch is typically removed? b. What time, equipment, and other resources (including offsite power) are required to re-install the hatch? c. What other penetrations are open during the modeled plant outage? (Examples: containment purge valves, penetrations for sludge lancing equipment.)
5. Evolutions	a. Do Mode 1 down-power evolutions (5 to 20% decrease for several hours, then back to full power) have any risk impacts? b. What are the potential impacts of different types of standard refueling outages (midloop early and late vs. midloop late vs. non-midloop outages; fuel offload vs fuel shuffle)? c. What are the potential impacts of other types of plant shutdowns? Examples: i. auto/manual reactor trips ii. forced manual shutdowns iii. accident-initiated outages (recovery after successful initial response to the accident) 1. Does restoration require plant configurations that introduce additional challenges? (Example: SG tube leak requiring midloop operation) 2. Is the increased vulnerability (due to SSCs lost during the accident) to subsequent events (e.g., external hazards) risk important?
6. Treatment of equipment maintenance during outages	What are the plant's practices for equipment maintenance during: a. Planned refueling outages – what is the standard maintenance plan for major safety equipment (e.g., DG, safety bus, service water)?

[7] Note that the L3PRA at-power analysis for "other external hazards" addresses onsite storage of hazardous materials and turbine-generated missiles.

Table A-4 LPSD Topics and Associated Issues Recommended by the TAG (continued)

LPSD Topic	Issues
	b. Forced plant shutdown due to Tech Spec requirement given equipment failed at power? c. Does the plant do opportunistic maintenance (maintenance on SSCs that are not the cause of the forced shutdown)?
7. HRA	a. Which HRA-related high-level requirements and supporting requirements in the LPSD PRA Standard are aspirational and how should they be addressed? b. How should HFEs for long duration LPSD scenarios (with ample time available for equipment recovery, e.g., when the refueling cavity is flooded) be addressed? i. Is scenario screening warranted? ii. If not, under what conditions would screening-level HEPs less than 1.0 be justifiable? c. Pre-initiator HFEs i. What does LPSD operating experience tell us (e.g., regarding types, causes, contextual factors, impacts) about potentially significant latent errors? ii. How can such errors be systematically identified? iii. What screening approaches can be used to focus the analysis? iv. How do the type of latent errors that can occur at shutdown and their treatment (e.g., identification, screening) compare/contrast with at-power PRA? d. HFEs causing initiating events i. What does LPSD operating experience tell us (e.g., regarding types, causes, contextual factors, impacts) about these errors? ii. Can we distinguish maintenance personnel errors from operator errors from the event reports? iii. Are there strong dependencies between these HFEs and post-initiator HFEs? e. Post-IE HFEs i. What does LPSD operating experience tell us (e.g., regarding types, causes, contextual factors, impacts) about these errors? ii. How should the HRA address mode-specific challenges? 1. Modes 3 and 4: challenge of applying EOPs 2. Modes 4-6: cognitive challenge of diagnosing the event sufficiently to choose the correct AOP 3. Mode 4-6: shutdown AOPs (e.g., do they have instructions for utilizing SG cooling via reflux cooling). iii. How do accident mitigation procedures, crew structure, and other factors impacting PSFs at shutdown compare/contrast with at-power PRA? What are the challenges in applying HRA methods commonly used for at-power PRAs for HFEs at shutdown? Note: for all of the above, the PIRT should focus on LPSD-unique aspects, discussing issues that are common to the at-power analysis (Level 2 as well as Level 1) only to the extent that such discussion is critical for the LPSD analysis.

Table A-4 LPSD Topics and Associated Issues Recommended by the TAG (continued)

LPSD Topic	Issues
8. Special activities during outages with potential for challenging plant configurations	What are the plant's practices with respect to: a. Nitrogen fill to accelerate the draining of SG tubes when draining to midloop, b. Water-solid operation for chemical cleanup, c. Vacuum fill of the RCS? d. Nozzel Dam installation and removal with fuel in the vessel, and e. Cold leg maintenance with fuel in the vessel.
9. Level 2 / Level 3 PRA	What aspects of containment response and offsite consequences are significantly different for accidents at shutdown, in comparison with accidents at power? For example, are containment phenomena and containment releases impacted by the presence of an open containment (i.e., with a large equipment hatch off)? Does LERF go to ~0 some days after shutdown? If so, what is a more appropriate risk metric to account for offsite consequences?

APPENDIX B DESCRIPTION OF THE ANALYTICAL HIERARCHY PROCESS APPLIED TO PIRT ELICITATION

Early PIRT developers, in writing about the role of the PIRT process, recommended use of the Analytical Hierarchy Process (AHP) – developed by Thomas Saaty, professor of statistics and operations research – as a way to formalize subjective decision-making into a product that is defensible, transparent, and complete. PNNL notes that while the AHP does not appear to have been used in PIRTs performed and documented for the NRC up to this point, it has been studied and used extensively since its development in the early 1980s. Moreover, PNNL finds AHP's application of pair-wise comparisons, a central feature of the AHP, and use of explicit evaluation criteria as a practical way to consistently rank the importance of one factor (or alternative) in relation to other factors that are also important to an overarching goal. This appendix provides a simple discussion of how the AHP is employed and how it was used as the central feature of the PIRT process to determine LPSD PRA priorities associated with adequately assessing risk. As such this discussion does not provide description of the PIRT parameters and AHP models used in this exercise as those details are provided in Section 3.0 of the report.

The AHP is based on a theory of measurement through pairwise comparisons and relies on the judgement of experts to derive priority scales and assign comparison and ranking categories (Saaty2008). Cognitive psychologists assert that comparative judgement versus absolute judgment is easier because of the way that information must be held and compared. The steps of AHP are to:

1. Define a decision problem and determine the kind of knowledge sought

2. Structure a decision hierarchy in which the top level of the hierarchy is the goal of the decision, the intermediate levels are factors (criteria) important to the goal and on which lower level factors depend, and the lowest level is a set of alternatives to be prioritized

3. Construct sets of pairwise comparisons to determine the relative importance of the criteria to the level immediately above it

4. Use the comparison to weight the priorities for each level until the final priorities of the alternatives can be determined.

A simple example diagram (i.e., hierarchy) of how the AHP process works is shown in Figure B-1 consisting of a top level goal, four evaluation criteria, and four alternatives to be ranked. The top level goal used in this example is one of top level goals from the Low Power Shutdown (LPSD) Probabilistic Risk Assessment (PRA) Phenomena Identification Ranking Table (PIRT) process. However, the actual AHP hierarchies developed for the LPSD PRA PIRT process are more complex as they consist of more criteria, an additional level of criteria (i.e., sub-criteria) and a significant number of alternatives (i.e., Plant Operating States) to be prioritized.

Example pair-wise comparisons of criteria to obtain a weight for each criterion is illustrated in Table B-1. The importance of one criterion over another is assigned according to the definitions presented in Table B-2 (adapted from Saaty 2008 for this illustration). The importance of each criterion used in judging the risk significance of an alternative listed the Column A of Table B-1 is compared to the importance of the criterion listed in Row B row of that same table by selecting the appropriate Comparison Category from Table B-2. If this involves comparison of a criterion to itself then the obvious result is that they have the same importance so the Comparison Category would Category E with an Importance Ratio of 1.0. If two criteria have already been compared elsewhere in the table, then the result should be the reciprocal ratio of the earlier comparison. In Table B-2, the total priority for each criterion is based on the sum of the assigned ratios for that row. From that, as discussed by Saaty (2008), the normalized totals (i.e., “Normalized Priorities”) are determined as shown in the last column of Table B-1.

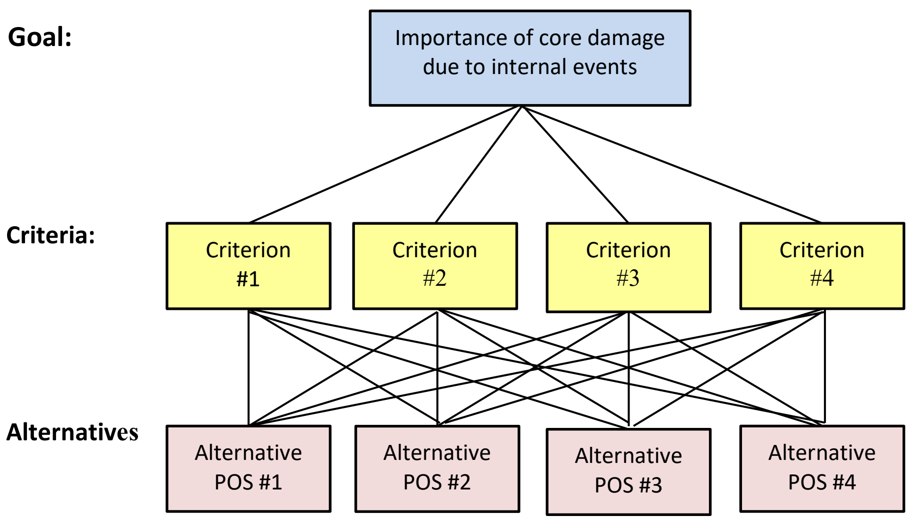

Figure B-1 Application of AHP to the PIRT Process

Table B-1 Pair-wise Comparison of Evaluation Criteria

Column A						
Row B	**Criterion #1**	**Criterion #2**	**Criterion #3**	**Criterion #4**	**Totals**	**Normalized Priorities**
Criterion #1	1	.14	.25	.11	.04	.04
Criterion #2	7	1	4	.50	.35	.35
Criterion #3	4	.25	1	.25	5.5	.15
Criterion #4	9	2	4	1	16	.45
Total					35.5	1.00

Table B-2 Importance Category and Ratios Based on Pairwise Comparisons

Importance of Criterion listed in Column A versus Row B

Comparison Category	Definition	Importance Ratio
A	Exceptionally more important	9
B	Strongly more important	7
C	Moderately more important	4
D	Slightly more important	2
E	Equally important	1
F	Slightly less important	1/2
G	Moderately less important	1/4
H	Strongly less important	1/7
I	Exceptionally less important	1/9

The ranking for each alternative is recorded in a manner similar to the form illustrated in Table B-3. In this example, the top level goal is the importance of POSs to core damage from a fire event. In this case, the four alternatives being ranked are four POSs. For each POS in Table B-3), a High (H), Moderate (M) or Low (L) was assigned corresponding to the level that the evaluation question was judged to be met. For example, if the evaluation criterion question was: “What level of importance does heat load during the POS contribute to fire event accident sequences that lead to core damage?”, then the response would be to assign H, M, or L to that POS and hazard combination. For the LPSD PRA PIRT elicitation, the experts were asked to describe the ranking categories. For, example the High ranking category for the evaluation criterion question above might be defined as “The decay heat in the reactor the first few hours after shutdown.”

Table B-3 Example Importance Ranking of Alternatives

Relative Importance of Plant Operating States for Fire Events that Result in Core Damage

	Evaluation Criterion # 1 Question	Evaluation Criterion #3 Question	Evaluation Criterion #3 Question	Evaluation Criterion #4 Question
POS #1	H	L	M	L
POS #2	M	H	L	L
POS #3	M	H	H	H

POS #4	L	H	L	M

The weights associated with ranking categories (i.e., H, M, and L) are determined by the experts for each evaluation criterion question using pairwise comparison. The comparison is performed in the same manner as shown in Table B-2 to determine evaluation criterion weights. Table B-4 shows weights determined for the ranking categories be determined by pairwise comparison in which ranking category listed in the first column of Table B-4 is compared to the ranking categories listed in the first row of the table by selecting the appropriate Comparison Category from Table B-3 and applying the corresponding Importance Ratio. The total for weight for each ranking categories is determined by first summing the assigned ratios, determining the normalized weight, and then calculating the "idealized weight". The idealized weights are then determined by dividing by the normalized weights by the largest normalized weight as shown in the last column of Table B-4. The idealized weights, per Saaty (2008) are used to weight the ranking categories assigned to different alternatives.

Table B-4 Pair-wise Comparison of Ranking Weights for Question #1

Level at Which Criterion is Met	High (H)	Moderate (M)	Low (L)	Totals	Normalized Weights	Idealized Weights
High (H)	1	7	9	17	0.72	1.00
Moderate (M)	.14	1	4	5.14	0.22	0.30
Low (L)	.11	.25	1	1.36	0.06	0.08
Total				15.73	1.00	

Then each alternative is assigned a ranking category for each evaluation criterion question so that the overall importance of each alternative to the top level goal can be determined. Table B-5 shows the ranking category assignments for the four alternatives from Table B-4 (i.e., POS #1, POS #2, POS #3, and POS #4) against the top level goal of the importance to core damage from a fire event, and shows how the final POS priorities are calculated. The normalized priorities for each evaluation criterion question shown in Table B-2 and ranking category weights determined like shown in Table B-4 are shown Table B-5 under the ranking category assignments (i.e., H. M. or L) to illustrate the calculation. Idealized category ranking weights for Evaluation Criterion Questions 2, 3, and 4, in addition to Questions #1, are invented in Table B-5 to illustrate how final priorities are calculated.

For the three kinds of elicited values and assignments (i.e., the evaluation criteria weights, the ranking category weights, and the rankings themselves) forms are be provided to the experts to capture the elicitations. The forms provide space in each entry for the experts to provide, if they choose, the bases or reasons for the values elicited.

For typical PIRT performed for NRC top level goals were defined but underlying evaluation criteria were not explicitly defined (e.g., in NUREG/CR-6742, NUREG/CR-6743, and NUREG/CR-6744). Also, the weighting for ranking phenomena importance was were priori set to High equals 1.0, Medium equals to 0.5, and Low equal to 0 times the number of experts who assigned that phenomena importance ranking to the phenomena being assessed. Integrating AHP in to the PIRT process. PNNL finds AHP's application of pair-wise comparisons, a central feature of the AHP, and use of explicit evaluation criteria as a practical way to consistently rank the importance of one factor (or alternative) in relation to other factors that are also important to an overarching goal.

Table B-5 Example Importance Ranking of Alternatives

Relative Importance of Plant Operating States for Fire Events that Result in Core Damage

	Question #1 (0.04)	**Question #2** 0.35	**Question #3** 0.15	**Question #4** 0.45	**Importance Total**	**Normalized Importance**
POS #1	H (0.04*1.00)	L (0.35*0.25)	M (0.15*0.44)	L (0.45*0.09)	0.27	0.13
POS #2	M (0.04*0.30)	H (0.35*1.00)	L (0.15*0.12)	L (0.45*0.09)	0.41	0.20
POS #3	M (0.04*0.30)	H (0.35*1.00)	H (0.15*1.00)	H (0.45*1.00)	0.95	0.46
POS #4	L (0.04*0.08)	H (0.35*1.00)	L (0.15*0.12)	M (0.45*0.18)	0.43	0.21
Total					2.06	1.00

APPENDIX C LPSD PRA PIRT PROCESS DESCRIPTION

C.1 LPSD PRA PIRT Process Description

The Phenomena Identification and Ranking Table (PIRT[1]) process is a systemic way to gather information from experts about identification of important nuclear power plant systems, components, processes and phenomena and ranking them in importance to the objectives of decisions that need to be made (Diamond 2006) (Wilson and Boyack 1998). The PIRT process was first developed and applied in the late 1980s (Shaw 1988) (Boyack 1989) and later progressed into a generalized process (Wilson and Boyack 1998). The PIRT process has been successfully applied to several Nuclear Regulatory Commission (NRC) applications, such as NUREG/CR-6742 (Boyack 2001a), NUREG/CR-6743 (Boyack 2001b), NUREG/CR-6744 (Boyack 2001c), NUREG/CR-6764 (Bidinger 2002), and NUREG/CR-7150, Volume 1 (Salley M.H., and Wachowiak 2012). Though the PIRT process typically identifies phenomena that are relevant to a particular figure-of-merit, it can also be applied to reactor, system, or component conditions; physical or engineering approximations; reactor process parameters; or other factors that influence the figure-of-merit that is of interest (Diamond 2006). For this application, the PIRT process is used to identify plant operating states[2] (POSs), hazards, plant outage types[3] (POTs), and other influences that are important to include in a Low Power Shutdown (LPSD) analysis supporting a full-scope plant Level 3 PRA.

At the heart of a PIRT application is ranking the factors of interest. For this application, the figure of merit is the importance of the factor to LPSD risk. Importance ranking requires identifying criteria[4] to judge the importance of the factors. The decisions about what the criteria are and the weights that should be assigned to different criteria are considered by PNNL to be a key to the PIRT process for this application. Wilson and Boyack in their paper on use of the PIRT process (Wilson and Boyack 1998) state that use of the Analytical Hierarchy Process (AHP) (developed by Thomas Saaty, professor of statistics and operations research) “is highly recommended to formalize subjective decision making into a product that is defensible,

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[1] In some references reviewed by PNNL such as Diamond 2006 and Holbrook 2007, PIRT was defined as Phenomena Identification and Ranking Technique as opposed to Phenomena Identification and Ranking Table.

[2] From the LPSD PRA Standard (ANS/ASE 2015): <u>A standard arrangement of the plant during which the plant conditions are relatively constant, are modeled as constant, and are distinct from other configurations in ways that impact risk. POS is a basic modeling device used for a phased-mission risk assessment that discretizes the plant conditions for specific phases of an LPSD evolution. Examples of such plant conditions include core decay heat level, primary water level, primary temperature, primary vent status, containment status, and decay heat removal mechanisms. Examples of risk impacts that are dependent on POS definition include the selection of initiating events, initiating event frequencies, definition of accident sequences, success criteria, and accident sequence quantification.</u>

[3] From the LPSD PRA Standard (ANS/ASE 2015): <u>Term used to describe the general cause of the plant being subcritical. Different outage types result from maintenance and refueling requirements that necessitate different LPSD evolutions and resulting POSs. For example, a “refueling” outage type leads to cold shutdown with some or all of the fuel elements transferred out of the reactor pressure vessel. In contrast, a “maintenance” outage conducted at cold shutdown to repair steam piping would be a different outage type.</u>

[4] The word “criteria” as it is used in this document refers to “a standard on which a judgement or decision may be based.”

scrutinizible, and complete." PNNL notes that though AHP has not been used in PIRTs performed and documented for NRC up to this point, AHP has in general been extensively used and studied since it was first developed in the early 1980s. Moreover, PNNL sees that AHP's application of pair-wise comparisons explicitly helps resolve one of the challenges of this particular expert elicitation (i.e., consistently ranking the importance of sizable sets of factors that those different are similar in many respects.) Accordingly, AHP as it is described in (Saaty 2008) will be employed as part of the PIRT process used in this application.

This PIRT approach includes a structured process for eliciting judgments from technical experts on Difficult technical questions in lieu of other means, such as testing or analysis, which may be implausible. Inherent to eliciting technical judgment from experts are issues such as the possibility of a nonrandom sample of experts, experts with different levels of familiarity with the available data, experts with different motivations, dependent experts, and experts that provide outlier judgments. The project team notes that NRC has produced guidance on performing expert elicitation for probabilistic seismic hazard analysis in NUREG-2117, Rev. 1 (Kammerer 2012) and NUREG/CR-6372 (Budnitz 1997) that addresses these kinds of concerns. This process is referred to as the Senior Seismic Hazard Analysis Committee (SSHAC) process. In light of this, the project team considered the following basic SSHAC principles as it put together its PIRT process for this application:

- Structure – A structured team and process to facilitate elicitation and minimize biases.
- Breadth of State-of-Knowledge – A team that represents the breadth of expertise required, includes a balance of experts with diverse opinions, and has full access to all available data.
- Independence – Judgments by each team member that are based on the individual's knowledge and expertise, not that of their peers or employers.
- Interaction among the team members during the assessment process to 1) develop a common understanding of the problem and data and 2) ensure that differences among the assessments of individual team members represent genuine epistemic uncertainty[5] and do not result from misunderstandings or from exposure to different sets of data or models.
- Integration (rather than consensus) and aggregation of all team members' interpretations and judgments, including assessment of uncertainties.

In general, these principles will be embedded in the elicitations performed for the PIRT process used in this application. The LPSD PIRT process follows the steps shown in Figure C-1. The PIRT process consists of: 1) online meetings that provide to a panel of experts the problem definition, description of the PIRT process, and an initial list of important POSs, hazards, and evaluation questions, 2) gathering feedback by email from expert panel members on the list of POSs, hazards, and evaluation questions that should be addressed in the PIRT evaluation, 3) remotely eliciting in individual interviews information to determine the importance ranking of

[5] Epistemic uncertainty is uncertainty due to limited knowledge and data (opposed to uncertainty due to randomness or variability).

each POS and hazard combination, and 4) eliciting information in a face-to-face group meeting with the experts to determine the importance ranking of each POS and hazard combination.

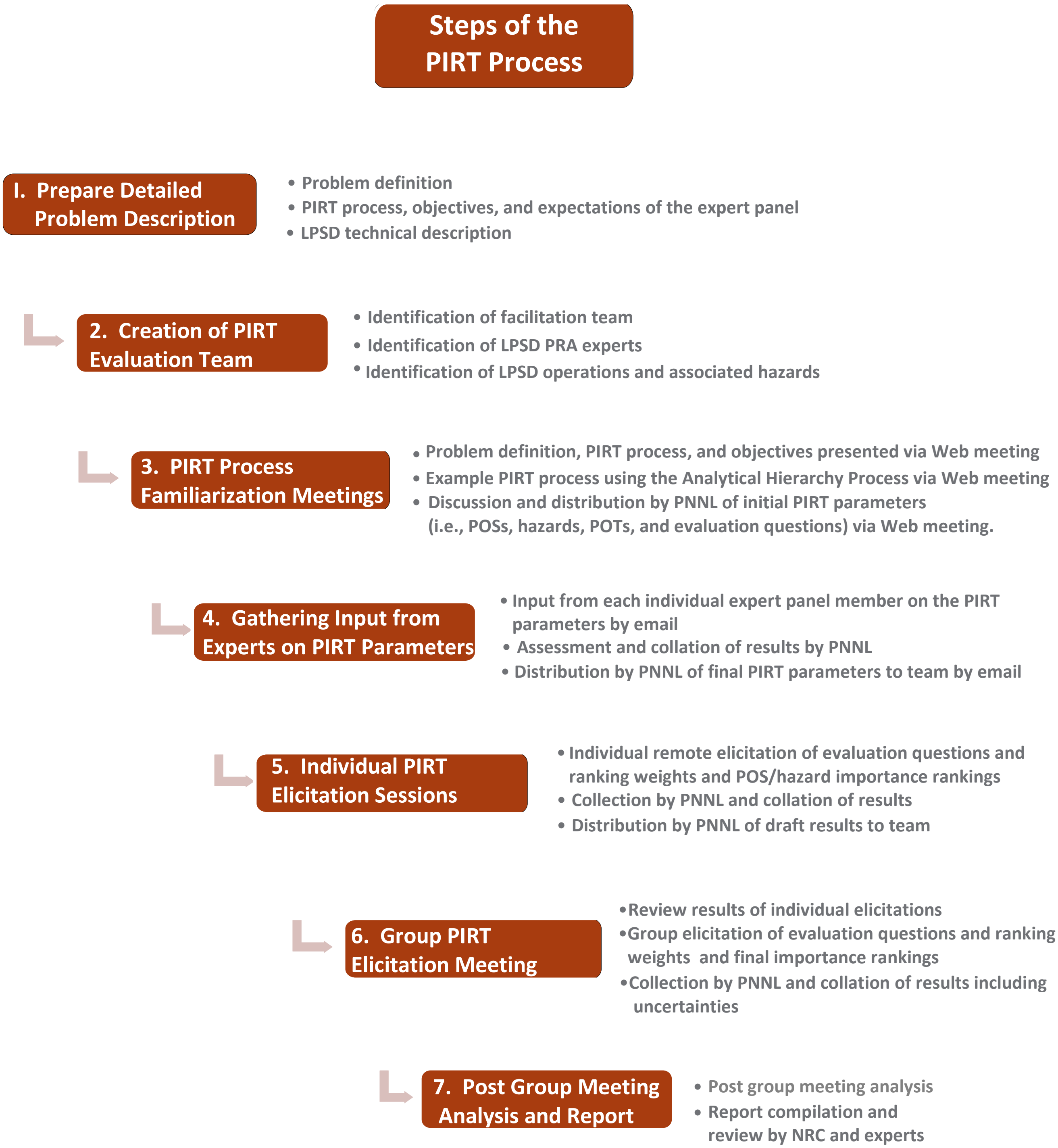

Figure C-1 PIRT Process Description

The outcome of the PIRT process will be the importance ranking of POSs for internal events and other hazards (such as internal fires or seismic events, determined to be important contributors to risk in terms of both Core Damage Frequency (CDF) and radioactive release from core damage sequences. Given that POTs will encompass or be limited to specific POSs, the importance of POTs will be determined based on the importance of the POSs that make up specific POTs. If the same POS is meaningfully different for different POTs, then the POS will be subdivided into separate POSs (e.g., If POS No.1 is different for one POT compared to the same POS for other POTs, then the POS will be divided in POS No. 1A and 1B to account for the two different variations of the POS).

This section describes each step of the PIRT process used in this LPSD application.

C.2 Preparation of Detailed Problem Description

The first step in the elicitation process is the preparation of materials relevant to the LPSD PIRT. These materials include:

1. A detailed statement of the problem being addressed by the LPSD PIRT expert panel. The detailed problem statement will be provided to each member of the panel for review and consideration prior to becoming a member of the panel, and will be documented in final project report.
2. Description of the PIRT process. The PIRT process, which is summarized in Figure C-1, is discussed in detail in the remainder of this PIRT process description. The PIRT process, including the elicitation approach, will be provided to the NRC for review and comment prior to the start of the elicitation sessions. A description of the LPSD PIRT process after comments by NRC staff are incorporated will be provided to each member of the expert panel prior to the first familiarization meeting. This initial version of the LPSD PIRT process description will be included in the final project report as an appendix. Documentation of how the LSPD PIRT process was actually implemented will be provided in the final project report.
3. LPSD operations and outage information. Plant-specific LPSD operations and outage information will be compiled and provided to each expert prior to the first familiarization meeting. This will include the plant shutdown operating procedures, outage reports, and a draft PRA report on POSs. The shutdown operating procedures that will be provided include:
    - Normal Operating Procedure for Heatup to Hot Shutdown
    - Normal Operating Procedure for Cooldown to Cold Shutdown
    - Normal Operating Procedure for Refueling Operations
    - Normal Operating Procedure for Mid-Loop Operations
    - Normal Operating Procedure for RCS Vacuum Refill
    - Normal Operating Procedure for Residual Heat Removal System

- Abnormal Operating Procedure (AOP) for leakage of the RCS
- AOP for loss of RHR
- AOP for loss of AC Class 1E Electrical System
- Procedure for Outage Risk Assessment Monitoring

The Plant-specific outage reports that will be provided, detailing the experiences from ten recent refueling outages. These documents contain proprietary information that cannot be duplicated or disclosed without first obtaining the written permission of the NRC.

4. Other reference material. A set of current and historical reference materials considered to be relevant to LPSD PRA implementation or to the PIRT process will be provided to each member of the panel prior to the first elicitation session. Included in these references is guidance from the International Atomic Energy Agency (IAEA) on performing probabilistic safety assessments of LPSD modes at nuclear power plants (IAEA 2000) which provides a set of defined POSs for PWRs than those identified in the NUREG and EPRI documents cited below.

    - NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States" (NRC 1993)
    - NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, unit 1" (Chu 1995) [6]
    - NUREG/CR-6093, "An Analysis of Operational Experience During Low Power and Shutdown and a Plan for Addressing Human Reliability Assessment Issues" (Barriere 1994)
    - NUREG/CR-7114, "A Framework for low Power/Shutdown Fire PRA" (Nowlen 2013)
    - ANS/ASME-58.22-2014 for Trial use and Pilot Application, "Requirement for Low Power and Shutdown Probabilistic Risk Assessment" (ANS 2014)
    - EPRI 1003465, "Low Power and Shutdown Risk Assessment Benchmarking Study" (Mitman 2002)
    - EPRI 3002005295296, "EPRI Low Power and Shutdown Probabilistic Risk Assessment Standard Pilot: Palo Verde Self-Assessment" (Hance 2015)
    - IAEA-TECDOC-1144, "Probabilistic safety assessments of nuclear power plants for low power and shutdown modes,"
    - Decision Making with the Analytical Hierarchy Process," (Saaty 2008)

[6] NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," Vols. 1–6 October 1995, is not publicly available.

- NRC generic letters on loss of residual heat removal (RHR) while the reactor coolant system (RCS) is partially filled:
  - Generic Letter No. 87-12, "Loss of Residual Heat Removal (RHR) while the Reactor Coolant System (RCS) is Partially Filled" (NRC 1987)
  - Generic Letter No. 88-17, "Loss of Decay Heat Removal – 10 CFR 50.54(f)" (NRC 1988)

C.3 <u>Creation of a PIRT Evaluation Team</u>

The PIRT evaluation team will consist of a PIRT coordinator and facilitator team, whose responsibility is to organize the problem being addressed by the PIRT process and facilitate interactions with the experts, the panel of experts whose judgements will be elicited during the course of the PIRT process, and a participating peer reviewer.

C.3.1 Coordinator and Facilitator Team

The coordinator and facilitator team will setup, organize, and coordinate the problem being addressed by the PIRT process; facilitate interactions with the experts; and assess and put together the elicited information in a way that addresses the problem being addressed in the PIRT process. The team will consist of staff that fullfill the following roles (in some cases a PNNL staff member may fulfill more than one role or trade roles):

- Facilitator. The primary role of the facilitator will be to elicit judgement from the experts in the individual interviews and in the group meetings. An important attribute of the facilitator will be the ability to communicate effectively and clearly and the willingness to challenge participants to fulfill their roles while maintaining a structured and efficient process. In group interaction, the facilitator will encourage the evaluators to challenge one another and facilitate interactions among the experts on the expert panel to ensure that all assessments are challenged and adequately defended and that the experts act at all times as objective and impartial assessors. The facilitator will ensure that the evaluators consider the views of the larger technical community. The role and function of the expert elicitation as described in NUREG-2117, Rev. 1 (Kammerer 2012) was used to define the facilitator role above.
- Technical integrator and elicitation recorder. The technical integrator will have a broad-based knowledge of PRA methods and applications, LPSD modeling as it pertains to this application, and the PIRT process. The role of the technical integrator will be to provide technical leadership toward achieving the objectives of the project. The technical integrator may also serve as the recorder for the elicitation meetings.
- Project manager. The project manager will be responsible for ensuring adherence to scope, schedule and budget. The project manager will develop contracts with all technical personnel and subcontractors, organize the workshops (including issuing invitations to all participants and observers), and keep NRC apprised of progress in

terms of scope, schedule, and budget. The responsibilities of the project manager will include holding each participant to their contractual roles and responsibilities.

The coordination and facilitation team as a group will work together to formulate the PIRT process objectives and scope; coordinate and provide background technical information to the experts; develop the PIRT evaluation format; guide and record the individual and group elicitation sessions; and analyze and summarize the panel's findings.

Several PNNL staff with expertise in expert elicitation, PRA modeling, and statistical analysis will contribute to the project. Contributions are expected to include aggregation of the elicited information and characterization of the uncertainty created by any significant spread or differences in responses from the expert panel members. Lastly, the NRC project manager and NRC staff not on the expert panel but having expertise in LPSD risk or expert elicitation may participate in the group PIRT elicitation meeting as observers.

C.3.2 Expert Panel Members

Two types of experts will be selected as members of the LPSD PIRT expert panel. The first type are LPSD PRA experts who have experience performing, teaching, and/or reviewing LPSD PRA; understand the risk significant contributors and modeling issues associated with LPSD PRA; and are familiar with the results of LPSD PRAs that have been performed in the past, including any external events LPSD PRAs that have been performed. The second type are experts in LPSD operations and how the reference plant responds to hazards, such as internal fires or flooding, that can occur during LPSD operations. In some cases, an expert may legitimately qualify as both types of experts. Important to the elicitation process is the selection of appropriate experts needed to credibly perform the PIRT exercise.

Members of the panel having the necessary expertise and technical credibility in the above subject areas will be selected based on consideration of 1) recommendations by PNNL PRA staff, 2) recommendations by NRC LPSD PRA staff, 3) recommendations by the operating company of the participating reference plant, and 4) recommendations by other members of the expert panel.

The members that are planned to be selected for the expert panel are listed in Table C-1. The panel members represent the experience necessary to elicit the information identified in the final project report, and include 1) five members who are experts in LPSD PRA, with emphasis, in some cases, on specific aspects of LPSD PRA, such as HRA and thermal hydraulic modeling; 2) one member who is an expert in nuclear power plant (NPP) outage management and operations; and 3) one member who is an expert in the reference NPP procedures and operation during LPSD evolutions.

During the PIRT process training, the experts will be cautioned that their assessments should represent their own individual knowledge, experience, and judgment, and not the opinions or positions of their organizations.

Table C-1 LPSD Expert Panel

Panel Member	Organization	Expertise
Ken Kiper	Westinghouse Electric Company	LPSD PRA
Jeff Julius	Jensen Hughes	LPSD Human Reliability Analysis (HRA) and PRA
Don Wakefield	ABS Group	LPSD HRA and PRA
Jeff Mitman	NRC	LPSD PRA including thermal hydraulic (T-H) success criteria
Marie Pohida	NRC	LPSD PRA including T-H success criteria
Jim Ledgerwood	Westinghouse Electric Company	NPP outage management and operations
Steve Prewitt	Retired Senior Reactor Operator	NPP procedures and operation during LPSD evolutions

C.3.3 Participatory Peer Reviewer

The SSHAC process recommends conducting a participatory peer review to monitor the expert elicitation process for the purpose of avoiding significant systematic biases in the elicitation and enhancing the breath of the knowledge on which the judgments are based. The guidance states that participatory peer reviewers should be independent from the process though they are present during elicitation sessions and can participate in the process.

Participatory peer reviewers interact with the project team and the experts at all stages throughout the project. Their review includes determining whether the project is consistent with the basic principles of expert elicitation, whether it follows a formal elicitation process, and whether the technical assessment has been adequately defended and documented. The benefit of involving participatory peer reviewers is the opportunity to identify problems early on so they can be corrected before the project reaches an end state. However, peer reviewers must have a well-defined role and preserve their independent status throughout the project, particularly because frequent interactions with the project can lead to a loss of objectivity.

For the LPSD PRA PIRT panel project, a staff member (or staff members) from the NRC who has expertise in expert elicitation will fulfill the participatory peer reviewer (PPR) role. The specific purpose of this peer reviewer is to review the PIRT process, which means ensuring that the project conforms to the basic principles and formal process for eliciting expert judgment. Because the purpose of this project is to provide judgement on priorities for further LPSD PRA model development, and not to provide input to a regulatory decision or to a process that will be used in regulatory decision-making, a PPR to review the technical aspects of the elicitation is judged to not be necessary by PNNL.

C.4 PIRT Process Familiarization Meeting

Web conferencing will be utilized for three separate PIRT process familiarization meetings using the GoToMeeting® collaboration software in combination with standard audio conference calling technology. In the first meeting, PNNL will define and familiarize the expert panel about the problem that will be addressed by the PIRT exercises and will describe the PIRT process. In the second meeting PNNL will provide an example problem of how the PIRT evaluation will be performed using the Analytical Hierarchy Process (AHP). In the third meeting PNNL will provide an overview of the initial parameters identified by PNNL for use in the PIRT exercise.

Online Meeting #1 will be held on January 4, 2017, and last two hours. The purpose of this familiarization meeting will be to:

1. Define for the panel of experts the problem that will be addressed by the PIRT exercise,
2. Review the objectives of the PIRT exercise,
3. Provide an overview of the PIRT process and schedule logistics,
4. Review what is expected from the PIRT panel members,
5. Train expert panel members on avoiding elicitation bias

The Powerpoint® slides for this meeting will be distributed to the expert panel members before the meeting and will be documented in an appendix of the final project report.

Online Meeting #2 will be held on January 10, 2017, and last two hours. This meeting will primarily consist of working through an example problem using the AHP approach identified by PNNL as applicable to the problem being addressed. PNNL will:

1. Describe the AHP and how it applies to the problem being addressed,
2. Provide an easy-to-understand example of AHP in which judgements are elicited from the experts as a way of illustrating the AHP and how it can be used in a PIRT exercise,
3. Lead a discussion with the expert panel about applying AHP to the defined problem.

The Powerpoint® slides for this meeting will be distributed to the expert panel members before the meeting and will be documented in an appendix of the final project report.

Online Meeting #3 will be held on January 12, 2017, and last two hours. In this meeting, PNNL will present to the expert panel the initial parameters identified by PNNL to use in the PIRT exercise. This meeting will consist of:

1. POS definitions to use in the PIRT exercise,
2. POT definitions to use in the PIRT exercise,
3. Hazards to be addressed in the PIRT exercise, and

4. Evaluation questions to be used in the PIRT exercise, which will be weighted according to the relative importance of different POS and hazard combinations to Level 3 PRA (i.e., CDF and radioactive release from core damage) using AHP.

The Powerpoint® slides for this meeting will be distributed to the expert panel members before the meeting and will be documented in an appendix of the final project report.

C.5 Gathering Input from Experts on PIRT Parameters

After the January 12, 2017 meeting, input from individual expert panel members will be solicited about augmentation or refinements needed to the list of POSs, POTs, hazards, and PIRT evaluation questions presented by PNNL in the January 12, 2011, PIRT process familiarization meeting. This information will be collected from the expert panel members about a week later, on January 20, 2017. The PIRT parameters will then be finalized by PNNL and distributed back to the expert panel members a week later on January 27, 2017.

The first objective of this step is to ensure that all potentially important POSs, hazards, and POTs are included in the PIRT exercise. For the most part, POSs have been defined for the reference plant, but there may be reasons to subdivide a POS to get better resolution, join POSs to gain assessment efficiency, or add POSs that are important to considered and have not been previously identified. The POSs represent the fundamental assessment element of the PIRT process. POSs may need to be added to reflect specific plant configurations and conditions not encompassed by POSs associated with standard POTs if they are judged by the expert panel to merit inclusion.

The hazards important to LPSD risk will also be identified. Though there is a lack of information about external event LPSD risk across a full set of POSs, PNNL notes that NUREG-6144[7] presents risk results associated with LPSD internal events, internal fires, internal flooding events, and seismic events for a limited set of POSs. These results indicate that the internal fires and floods are as important (or more important) to LPSD risk as internal events. However, the expert panel members will need to address whether these results are generalizable to the reference plant. NUREG-1855 (Drouin 2009) provides guidance in Section 6 on the kinds of external hazards that should be considered in a "risk assessment." This list and other information will be used by the experts to decide what hazards should be addressed in the PIRT exercise. It very possible that the frequency of certain kinds of hazard events increase during LPSD compared to full-power.

Lastly, POTs important to LPSD will be identified. It is anticipated that the importance of POTs will be determined based on the importance of the POSs that make up a POT. Therefore, the primary focus for this parameter will be on accurately defining the POTs. Evolutions that involve reduction in power but do not result in an outage will be addressed to the extent the panel

[7] NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," Vols. 1–6 October 1995, is not publicly available.

determines is necessary within the bounds of an LPSD model[8]. Plant configurations for these situation are bound by Technical Specifications for Mode 1. The number of possible non-standard plant configurations could be quite large, and so consideration of all plant configuration variations will considered outside the scope of this project because it challenge the schedule and budget constraints of the project.

The second objective is to identify the evaluation questions to be used in the PIRT exercise to judge the importance of the POS and hazard combinations. Development of these questions will be based on factors that contribute to CDF or radioactive release from a core damage sequence (e.g., open containment penetrations). It is expected that the evaluation questions will be consistent across different POSs but be somewhat different for different hazards because the accident sequence phenomena may be different between different hazards. Risk factors associated with CDF identified by PNNL that seem relevant to the PIRT evaluation include:

- Heat load during the POS,
- RCS inventory level during the POS,
- Duration of the POS,
- Availability of RCS venting during the POS,
- Opportunity for operator-induced initiating events during the POS, and
- Unavailability of shutdown cooling trains and SSCs during the POS.

The evaluation questions will ask about the extent to which the risk factors important to CDF and radioactive release contribute to accident sequences associated with particular POS and hazard combinations.

- What level of importance does heat load during the POS contribute to accident sequences that lead to core damage?
- What level of importance does low RCS inventory during the POS contribute to accident sequences that lead to core damage?
- What level of importance does POS duration contribute to accident sequences that lead to core damage?
- What level of importance does the opportunity for operator-induced initiating events during the POS contribute to accident sequences that lead to core damage?
- What level of importance does the unavailability of shutdown cooling trains and systems during the POS contribute to accident sequences that lead to core damage?

Additional risk factors associated with radioactive release from core damage sequences will be identified and a separate evaluation performed for radioactive release. For example, risk factors associated with release identified by PNNL that seem relevant to the PIRT evaluation include:

[8] The ANS/ASME LPSD PRA standard (ANS/ASME 2015) defines a LPSD evolution to include “reducing power to 30% in order to conduct maintenance or an operational activity.”

- Unavailability of containment systems during the POS
- Status of containment isolation during the POS

Because release of radioactive material from core damage accidents associated with LPSD are not as likely to be as large or early as at full-power, consideration of release factors will not be limited to large early release frequency (LERF). An additional evaluation question or modification of the set of evaluation questions determined for CDF may be warranted.

POTs will not be explicitly assessed, given that the POSs from all POTs will evaluated. The importance of POTs will be determined based on the importance of the POSs that make up specific POTs. If the same POS is meaningfully different for different POTs, then the POS will be subdivided into separate POSs (e.g., If POS No.1 is different for a specific POT compared to the other POTs, then the POS will be divided in POS No. 1A and 1B for the two different variations of the POS).

To facilitate this activity, PNNL will transmit descriptions of the POS and POT attributes and hazard information relevant to LPSD that it has compiled, along with the solicitation request to each panel member.

Based on suggestions by the expert panel to augment or refine the list of POSs, POTs, hazards, and PIRT evaluation questions, PNNL will finalize these PIRT evaluation parameters. PNNL will distribute the final PIRT evaluation parameters back to the expert panel members by about January 27, 2017, ahead of elicitation from individual panel members on determining the relative weights to be associated with the evaluation questions and the ranking for each POS and hazard combination. These descriptions of the POS and POT attributes, hazards, importance factors, and evaluation questions relevant to LPSD will be provided in an appendix of the final project report.

C.6 Individual PIRT Elicitation Sessions

Elicitation sessions are expected to be held individually with each member of the expert panel during the week of January 30 through February 3, 2017. After the completion of each individual elicitation session, PNNL will provide each expert panel member the results of the session as scribed by PNNL staff for that panel member. Each expert panel member will review the scribed results of the individual sessions and return them to PNNL with any corrections and additions needed to finalize the assessment by February 10, 2017. The individual PIRT sessions will be performed using AHP. In these sessions, three kinds of information will be elicited: (1) the relative weights associated with the evaluation questions (certain factors are more significant than others to risk), (2) the ranking categories that define the level at which the evaluation question are met (e.g., High, Moderate or Low) for each POS and hazard combination, and 3) the ranking weights associated with the ranking categories. After the elicitations are completed, the priorities determined by the elicitation will be reviewed for reasonableness. As with the PIRT process familiarization meetings, web conferencing (i.e., GoToMeeting®) will be used to conduct these elicitation sessions, which will generally last between 1 and 2 hours.

Prior to the start of each of these elicitation sessions, forms will be distributed to each member of the expert panel on which the weights and rankings for each POS and hazard combination should be recorded. These forms will incorporate the POSs, hazards, and evaluation questions determined to be important to the PIRT evaluation by the expert panel and PNNL in the previous step of the LSPD PIRT process. This will allow the time for each expert to consider the specific information being elicited. The forms will be developed in Excel® and designed to elicit the information described in this section of this PIRT process description. The results from the individual elicitations and will be documented in an appendix of the final project report.

The weights associated with the evaluation questions will be determined by ratios based on pair-wise comparison of the criteria (This is a central element of AHP). As discussed previously, the evaluation questions will address importance factors such as heat load or unavailability of systems during the POS. A diagram of how the AHP process works is shown in Figure C-2 using an example involving four evaluation questions and four alternatives (e.g., the POSs.) to be ranked.

Example pair-wise comparisons of criteria to obtain a weight for each criterion is illustrated in Table C-2. The importance of one criterion over another is assigned according to the definitions and associated ratios presented in Table C-3 (adapted from Saaty 2008). The importance of each criterion in judging the risk significance of a POS and hazard combination listed the first column of Table C-2 is compared to the importance of the criterion listed in the first row of that same table by selecting the appropriate ratio from Table C-3. If this involves comparison of a criterion to itself then the obvious result is that they have the same importance. If two criteria have already been compared elsewhere in the table, then the result should be the reciprocal of that earlier comparison. Then the total for each question is based on the sum of assigned ratios are for that row. From that, as discussed by Saaty (2008), the normalized totals (referred to here as “Normalized Weights”) are determined as shown in the last column of Table C-2.

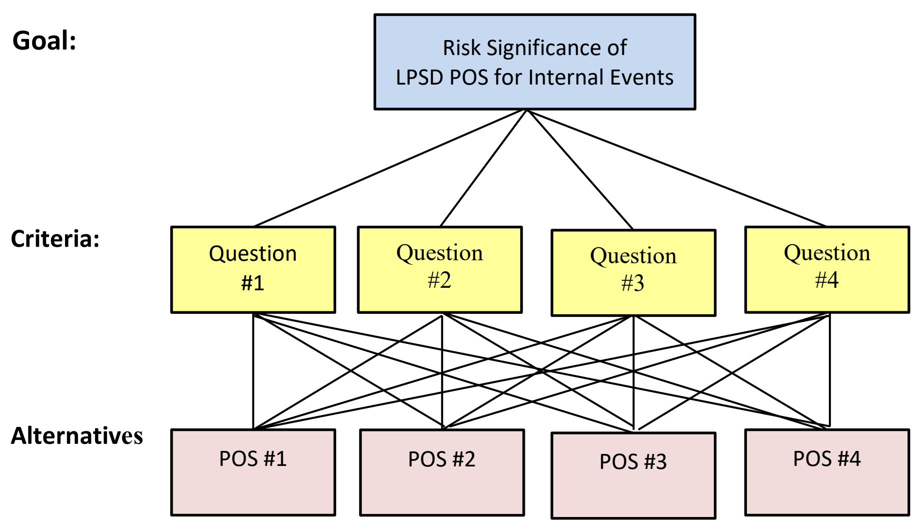

Figure C-2 Application of AHP to the PIRT Process

Table C-2 Pair-wise Comparison of Evaluation Questions

Column A						
Row B	**Question #1**	**Question #2**	**Question #3**	**Question #4**	**Totals**	**Normalized Weights**
Question #1	1	.14	.25	.11	.04	.04
Question #2	7	1	4	.50	.35	.35
Question #3	4	.25	1	.25	5.5	.15
Question #4	9	2	4	1	16	.45
Total					35.5	1.00

Table C-3 Importance Ratios Based on a Pair-wise Comparison

Ratio	**Definition**
	Importance of Criterion listed in Column A versus Row B
9	Exceptionally more important
7	Strongly more important
4	Moderately more important
2	Slightly more important
1	Equally important
.50	Slightly less important
.25	Moderately less important
.14	Strongly less important
.11	Exceptionally less important

The ranking for each POS and hazard combination using the evaluation questions will be recorded on a form similar to the form illustrated in Table C-4 which shows a ranking of POSs to CDF. In this example, for each POS and hazard combination (e.g., see entry for POS #1 through #4 for a fire event in Table C-4), a High (H), Moderate (M) or Low (L) was assigned corresponding to the level that the evaluation question was judged to be met. For example, if the evaluation question was: “What level of importance does heat load during the POS contribute to accident sequences that lead to core damage?”, then the response would be to assign H, M, or L to that POS and hazard combination. A separate importance ranking of POSs and hazard combinations to radioactive release will be performed with a somewhat modified set of evaluation questions. Additional sets of tables (for CDF and release) will be filled out for internal event POSs and other hazards (e.g., fire).

Table C-4 Example Importance Ranking of POSs to CDF

Relative Importance of Plant Operating States for Fire Events				
	Question #1	**Question #2**	**Question #3**	**Question #4**
POS #1	H	L	M	L
POS #2	M	H	L	L
POS #3	M	H	H	H
POS #4	L	H	L	M

The quantitative values used in a number of PIRTs (e.g., NUREG/CR-6742, NUREG/CR-6743, and NUREG/CR-6744) for ranking phenomena importance was H equal to 1, M equal to 0.5, and L equal to 0. These values will not be used in this PIRT process, instead the weight associated with ranking categories will determined by the expert panel for each evaluation question using pairwise comparison. This comparison will be performed in the same manner as shown in Table C-2 for the evaluation weights. As shown in Table C-5, the weights assigned to the ranking categories will be determined by comparing the levels listed in the first column of Table C-5 to the level listed in the first row of the table and selecting the appropriate ratio from Table C-3 that applies.

The total for weight for ranking categories is determined by the sum of assigned ratios. The normalized weights are determined from the totals and the idealized weights, per Saaty (2008) the idealized weights are determined by dividing by the normalized weights by the largest normalized weight as shown in the last column of Table C-5.

Table C-5 Pair-wise Comparison of Ranking Weights for Question #1

Level at Which Criterion is Met	High (H)	Moderate (M)	Low (L)	**Totals**	**Normalized Weights**	**Idealized Weights**
High (H)	1	7	9	17	0.72	1.00
Moderate (M)	.14	1	4	5.14	0.22	0.30
Low (L)	.11	.25	1	1.36	0.06	0.08
Total				15.73	1.00	

When each POS and hazard combination is assigned a ranking category for each evaluation question then the overall importance of each POS to CDF or release in relation to other POSs can be determined as shown in Table C-6. This is by done by using the relative weights determined for each evaluation questions (a shown in in Table C-2) and the ranking weights associated with the rankings. In Table C-6, idealized ranking weights are provided for Questions 2, 3, and 4 (in addition to Questions #1) to illustrate how the table is filled out. It should be noted that though the same ranking categories were used in this example case, different categories could be used for each evaluation question. PNNL finds that assigning ranking values to each evaluation question for each POS and hazard combination, produces a more refined evaluation than just applying an importance ranking directly to the POS hazard combinations.

Table C-6 Example Quantified Importance Ranking of Plant Operating States to CDF

Relative Importance of Plant Operating States for Fire Events						
	Question #1 (0.04)	**Question #2** 0.35	**Question #3** 0.15	**Question #4** 0.45	**Importance Total**	**Normalized Importance**
POS #1	H (0.04*1.00)	L (0.35*0.25)	M (0.15*0.44)	L (0.45*0.09)	0.27	0.13
POS #2	M (0.04*0.30)	H (0.35*1.00)	L (0.15*0.12)	L (0.45*0.09)	0.41	0.20
POS #3	M (0.04*0.30)	H (0.35*1.00)	H (0.15*1.00)	H (0.45*1.00)	0.95	0.46
POS #4	L (0.04*0.08)	H (0.35*1.00)	L (0.15*0.12)	M (0.45*0.18)	0.43	0.21
Total					2.06	1.00

As explained above, separate importance ranking of POSs and hazard combinations to radioactive release will be performed with a somewhat modified (perhaps not significantly) set of evaluation questions.

For the three kinds of elicited values and assignments (i.e., the evaluation question weights, the ranking weights associated with the ranking categories for each POS and hazard combinations, and the rankings themselves) forms will be provided to capture the elicitations. The forms will provide space in each entry for the experts to provide, if they choose, the bases or reasons for the values elicited. Though not the specific focus of the PIRT elicitation process, the forms will document comments and explanations that provide bases for elicited values. This information could include identification of relevant modelling challenges that inject uncertainty into an elicited value. The experts will be expected to provide this information to the extent it is needed to maintain internal consistency and explain results that are not obvious. At the end of the process this qualitative information will be reviewed for insights.

After the elicitation is completed and the quantitative importance values (i.e., priorities) associated with the POSs are determined, the elicitation will be reviewed for reasonableness. Of particular interest, is whether the resulting priorities match the overall views of expert panelists. For cases in which the elicited priorities do not appear to match the panelist's overall view, PNNL will review and adjust the AHP process if needed to produce results that are more internally consistent.

Prior to the start of each individual elicitation, there will be a brief Powerpoint® presentation that provides general instructions about how the elicitation will be conducted and what specific information will be elicited on each form. The Powerpoint® presentation and forms for the elicitation will be provided in an appendix of the final project report.

During each of the individual elicitation sessions, the PNNL Facilitator will facilitate the discussion while the PNNL Integrator will record on the forms the information elicited from the expert. The process described above will be followed for all of the experts during the individual elicitation sessions.

Following the completion of each of the individual elicitation sessions, the compiled forms will be emailed to of the experts for review of and comment on the accuracy and completeness of the recorded information. Each expert will be asked to provide an updated form with corrected information, if the expert determines that PNNL has not accurately recorded their input. Generally, it is expected that the returned forms will include comments by the experts that describes the basis for the input as described above. Space will be provided in the forms for each elicited value to capture this information. Since these forms will be subsequently updated by the experts following the group elicitation session, these preliminary results will not be included in the final project document.

C.7 Group PIRT Elicitation Meeting

A final combined PIRT elicitation session will be held with all members of the expert panel during a 2½-day period from February 21 through 23, 2017. NRC staff are also expected to attend the meeting, including the PPR. A Powerpoint® presentation will be used to facilitate the meetings and elicitation process.

In the first phase of the meeting there will be an overview of the agenda for the meetings, a brief review of the problem description, a review of the project status, introduction of panel members and observers (and directions to the observers on their role in the elicitation), a discussion of comments on the PIRT process to-date, and a review of changes made to the completed forms from the individual elicitation sessions. At this point, each panel member will be provided with a hardcopy of the completed forms from their individual elicitation sessions.

In the second phase of the meeting, the results from the individual elicitation sessions will be reviewed by each expert panelist with the group. During this review, PNNL will point out instances in which there was divergence of opinion in either the weights assigned to the evaluation questions or in the ranking of POS and hazard combinations. PNNL will also present the average results using the geometric mean of the aggregate results from the individual PIRT sessions[9]. Also during this phase of the meeting the results of an uncertainty analysis performed by PNNL across the individual PIRT results will be presented that characterizes the differences in the elicited inputs between experts and provides assessment of how the differences should be considered in the final importance ranking of the POSs (for particular hazard). Saaty (2008) recommends calculating each expert's alternative rankings separately

[9] Saaty (2008) points out that in order to maintain the reciprocity principle of criteria comparisons, you should use the geometric mean rather than the arithmetic mean to combine multiple responses.

(i.e. with their individual criteria comparisons and individual alternative/criteria weightings) then aggregating the computed scores across participants. Assigned weights and rankings will be normalized for each panel participant so that the results are comparable between experts. The uncertainty analysis will allow identification of POSs/hazards/POTs where a minority of experts believe the risk to be higher. Areas of greatest disagreement with be identified from the variabilities of the elicited values.

The third phase of the meeting will be the group PIRT elicitation process. Similar to the individual PIRT sessions, the group session will elicit three the kinds of information. Each expert panel member, led by the PNNL facilitator, will perform the following steps for each assignment or value being elicited:

1) Each expert will review their input from the individual elicitation meetings for the parameter being elicited. This will include: 1) pair-wise comparison of evaluation questions like shown in Table C-2 to determine evaluation question weights, 2) pair-wise comparison of the ranking categories (e.g., H, M, and L) to determine ranking weights like shown in Table C-5, and 3) determination of importance ranking of each POS (by hazard) by assigning a ranking category to each evaluation question shown in Table C-6.
2) Each expert will be given an opportunity to present the basis and rationale for their individual responses. Discussion amongst the expert panel members will be encouraged. Space on the forms will be provided to document comments by the experts and could include identification of modelling challenges that inject some uncertainty into an elicited value. The experts will be expected to provide this information to the extent it is needed to maintain internal consistency and explain results that are not obvious. At the end of the process this qualitative information will be reviewed for insights.
3) Following completion of discussions, each expert will be given an opportunity to modify their original elicitation results based on the additional information presented by the various experts. These changes will be made by the experts either electronically or by hand on a hardcopy of the form.

Each expert will update their forms during a two-week period following the conclusion of the elicitation meetings and provide them to PNNL by about March 10, 2017. This time period will provide each expert with the opportunity to review their changes, add comments to the forms in a less pressured setting, and reconsider their input in light of the additional information discussed during the meeting or made available subsequent to the meeting.

The final forms filled out with input elicited from the experts and updated will be provided in an appendix of the final project report.

C.8 Post Group Meeting Analysis

Analysis of the results will be performed by PNNL in the weeks following the group meeting (i.e., February 27 through March 10). The post group meeting activities will include: 1) compilation of the results, 2) assessment of the quantitative results, and 3) summarization of qualitative output, and 4) formulation of insights.

The idealized ranking values obtained from the PIRT elicitation process (e.g., see Table C-6) reflect the importance of the assessed POS and hazard combinations to CDF and release. These elicitation results will be reviewed for insights. Insights from the quantitative results that are generalizable to future LPSD PRAs or PIRTs will be summarized. An assessment of the differences in opinions between expert panel members that reflect areas of uncertainty be identified and characterized, and insights formulated. Also though not the specific focus of the PIRT elicitation process, comments, explanations, bases, and assumptions provided by the experts associated with specific elicited values or category assignments will be summarized and assessed to some extent to gain further insights that are generalizable to future LPSD PRAs or PIRTs.

The final project report will document the PIRT elicitation process as it was implemented, the elicitation results provided by each expert based on the group PIRT elicitation meeting performed February 21 through 23 and updated two weeks later, and the potential sources of bias and lessons learned from the PIRT elicitation process. The draft report will be provided to all of the expert panel members, and to the NRC, for review and comment.

C.9 References

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APPENDIX D   PANEL MEMBER QUALIFICATIONS

This appendix provides a summary of the qualifications of each member of the expert panel and participatory peer reviewers.

Jeffrey A. Julius
Director of Safety and Risk

Summary
- Technical Manager coordinating Scientech engineers conducting probabilistic risk analyses (PRA) and PRA applications for U.S. nuclear power utilities - 21 years.
- Applied probabilistic models in Risk Management/Decision Analyses - 25 years.
- Extensive experience in the conduct and management of PRAs, probabilistic safety analyses (PSA), and reliability analyses in the commercial nuclear field - 30 years.
- Researched and developed new Risk Assessment methods and PRA techniques in the areas of Shutdown PRA and Human Reliability Analyses - 25 years.
- Experienced in the operation, maintenance and analysis of nuclear reactors - 36 years.
- Extensive experience in personnel management, training, and schedule optimization - 36 years.

Education
Bachelor of Science in Engineering (B.S.E.), Nuclear Engineering, University of Washington, 1980
U.S. Navy Nuclear Power Training School, Orlando, FL, 1981
U.S. Navy Nuclear Prototype Training, Idaho Falls, ID, 1982

Licenses and Certifications
Certified Chief Engineer Officer of the S5W Naval Nuclear Propulsion Plant

Security Clearance
Inactive DoD Top Secret Clearance

Qualifications
Probabilistic Risk Assessment/Probabilistic Safety Assessment/Reliability Assessments
Over twenty-nine years of experience in the performance, management and review of probabilistic safety, risk, and reliability assessments of nuclear reactor plants and non-nuclear systems. Principal Investigator in the NUREG-1150/4550 research program of the Surry plant and the NUREG-1921 Fire HRA Guidelines. Experienced in all task disciplines and all types of PSA; including internal events, Human Reliability, Fire, and Shutdown PRA.

Project Manager. Project Manager of PRA analytical and software development projects for over twenty-five years. Currently managing the Internal Fire and Internal Flood PRA projects for the Barakah Nuclear Power Plant. Recently completed three NFPA-805 Fire PRA projects at DC Cook, Diablo Canyon, and Callaway; with two of the plants having successfully completed the transition of their fire protection program to NFPA 805. Additionally, acted as senior Technical Advisor for the Hatch Fire PRA (BWR). The most comprehensive PSA project consisted of two, complete (Level 1 through Level 3) probabilistic safety assessments (PSAs) for Power and Non-Power operating states, including a Fire and external events PRA. One PSA served as a baseline analysis, and one evaluated the impact of proposed modifications affecting all systems, including revising the fire suppression systems, at the Borssele (Dutch) PWR. This project was initially conducted in 1992, maintained as a living PSA as part of the plant's licensing basis, and then re-evaluated in 2002 for potential future modifications to ensure plant safety for the next 20 years. The 2007 update developed the as-built PSA reflecting credit of the implemented plant modifications. The PSA analyses have been included as part of an environmental impact statement and is now part of the plant's licensing basis, modeling internal events, external events, and human errors of commission during power and non-power operations as part of a complete (Level 1, 2, and 3) PRA. As Project Manager was responsible for the technical quality, administration, and results of assigned projects. Primary tasks included the performance, direction, and review of all technical work; and administrative functions such as work scope definition, budget, and schedules. For Borssele this included defending the PRA during three international peer reviews, addressing findings and observations. Principal investigator for the accident sequence delineation, human reliability analysis (including errors of commission), results interpretation, and non-power plant response tasks; also performed the errors of commission during non-power and task integration, including the Level 1/Level 2/Level 3 interfaces. Also, Project Manager/Senior Technical Advisor of the EPRI HRA Users Group for sixteen years providing HRA modeling guidance and tools to all U.S. plants plus vendors and international users.

Human Factors/Human Reliability. Currently President of the HRA professional society. Project Manager or Principal Analyst in the review and update of the Human Reliability Analysis (HRA) of the Borssele, Comanche Peak, DC Cook, Diablo Canyon, Farley, Hatch, Indian Point 2, Salem, VC Summer, Vogtle and Wolf Creek probabilistic risk assessments. Employed THERP modeling for execution actions; and employed multiple cognitive methods such as Cause-Based Decision Tree and Human Cognitive Reliability modeling emphasizing the proper timing, spatial, and functional dependencies under a wide range of accident scenarios. Addressed PRA Certification Review findings and observations on the HRA. Instructed HRA techniques at the Callaway, DC Cook, VC Summer, Diablo Canyon, Prairie Island, Fort Calhoun and Millstone nuclear power plants and provided HRA review to Byron, Braidwood, Fort Calhoun, Indian Point 3, Kewaunee, Phillipsburg, Pilgrim, Quad Cities, Surry and Zion. Assisted several plants with human reliability as part of the significance determination process, including Turkey Point, Point Beach, VC Summer, Wolf Creek, and Callaway. Developed and implemented new techniques in Human Errors of Commission in an analysis for Power and Shutdown PRAs. Developed techniques for dynamic HRA modeling in Risk Monitors for on-line maintenance or Shutdown conditions with varying plant configuration and thermal-hydraulic time windows. Additionally, he advised the development of human error probabilities to be used in a reactor trip-monitoring (Trip Meter) project and the Fire PRA HRA modeling in NUREG/CR-6850 via the Diablo Canyon pilot plant. Project Manager for the EPRI HRA Calculator™ and EPRI HRA Users Group (representing 17 US utilities with over 60 plants; plus 9 vendors and utilities outside of the USA) since 2001. Currently leading the EPRI HRA Users Group participation in joint projects with the USNRC; one on Fire HRA methods development and the other comparing HRA methods to simulator empirical data.

Shutdown PRA Task Leader, Principal Investigator, or Senior Advisor for eighteen nuclear power plants. Developed integrated models for all plant operational states (combining the Power and Non-Power models), producing plant response (event tree) models, system models, data, and results for boiling and core damage in steady-state and transient shutdown conditions, including the Spent Fuel Pool. Provided Shutdown PRA Training to Wolf Creek, Callaway, Comanche Peak, Borssele, Perry, Nine Mile Point, Hope Creek, and the Korean Electric Power Research Institute. Assisted in developing an IAEA technical document describing Low Power and Shutdown PRA methods. Member of the ANS writing committee to develop a Low Power and Shutdown PRA Standard. Project Manager for an EPRI project benchmarking Low Power and Shutdown qualitative analyses with quantitative results. Participated in the review of a shutdown PRA for a German boiling water reactor.

Reliability Analyses. Developed and presented utility workshops on the applications of the GO methodology in improving plant reliability, availability, and maintainability. Principal Investigator for systems reliability and availability analyses of nuclear plant systems at FitzPatrick and Indian Point 2, and non-nuclear analyses of uninterruptible power supplies and a hazardous waste incinerator using GO.

Probabilistic Risk Assessment Program Development

Risk-Informed PRA Applications Technical Manager and Office Manager for Seattle, Washington, overseeing and directing nuclear probabilistic risk assessments (PRA) and chemical process safety management (PSM) projects conducted by a group of approximately 10 engineers. Responsible for both the technical quality and administrative aspects (scope, level of detail, quality assurance, budget, and schedule) in successfully completing PRA projects. Coordinates PRA business development activities. Provides guidance and insight into model development and application. Senior Consulting Engineer with over twenty-nine years of experience in risk assessment and management of probabilistic models for Power and Shutdown states, reliability and availability modeling, and integrated analysis and operations of complex engineering systems. Participated in various analytical capacities in major risk assessments for numerous foreign and domestic nuclear power plants evaluating electrical generation and distribution, mechanical, hydraulic, fluid, and pneumatic systems. Project manager directing engineering, software, training and review projects for the last twenty-five years.

Risk Management/Decision Analysis Using PRA/PSA
Applied probabilistic models in Risk Management/Decision Analyses through two types of projects, one from the perspective of plant hardware and procedural change evaluation and the other in developing a tool for the day-to-day management of plant configuration.

Hardware and Procedural Change Evaluation. International review team, one of three experts selected by the International Atomic Energy Agency to review the application of the PSA and Markov models to changing Technical Specification Allowed Outage Times (AOT) at a Hungarian VVER reactor. Project Manager of a risk-informed AOT extension of 6.9kV AC components for Comanche Peak using the Full Power and Shutdown PRA.

Project Manager for a series of technical support projects using a Living PSA model as a decision-making tool for evaluating proposed hardware and procedural changes. Vendor and regulator proposed modifications were evaluated qualitatively and quantitatively for two 10-year periodic safety evaluations, and potential areas of improvement were identified for a third periodic safety evaluation. Designs were modified and some modifications eliminated as a result of the first program (the second has just started), with the savings more than large enough to pay for this program. Provided technology transfer in PSA procedures and techniques, demonstrating the applications of PSA in the evaluation of proposed modifications at the Borssele nuclear power plant. These projects included extensions to technical specification allowed outage times.

Shutdown Risk Management. Shutdown PRA Task Leader or Senior Adviser for an EPRI project to develop Shutdown PRA models into Safety Monitor™ (risk meter) models for the Callaway, Wolf Creek, Comanche Peak, Pt Beach, North Anna, Pt. Beach, D.C. Cook, Kewaunee, Borssele, Clinton, and Surry nuclear power plants in the US. Expanded the Individual Plant Examination PRA models, developing a single, integrated model for all plant operational states (Power and Non-Power). The resulting Living Risk Monitor model evaluates the impact of changing plant configuration and maintenance schedules on plant safety. Provided senior advisor support (developed task plans and conducted reviews) to project teams performing a detailed shutdown PRA for the Perry and Hope Creek (BWR) and Diablo Canyon (PWR) stations and a focused study of the Nine Mile Point (BWR) plant. The Borssele model for all modes included spatial and external events. Conducted a Low Power and Shutdown Benchmarking study for EPRI to compare quantitative risk levels to qualitative outage management controls for a PWR and BWR.

Risk Assessment Methods Development
Significant methods development programs include Shutdown PSA and Human Reliability Analyses. Participated as a member, and then team leader, in two International Atomic Energy Agency Technical Committee meetings developing international shutdown PRA standards. Developed a procedure for constructing Shutdown PRA models that are focused and integrated. The models consist of one set of system fault trees for both Power and Non-Power, which can be employed in typical PRA sequence quantification or in a Risk Meter type of application. Developed and implemented Human Errors of Commission techniques two years prior to the USNRC ATHEANA program. Identified and quantified realistic instances of human errors of commission following accidents during Power and Non-Power operations at a European PWR, using a probabilistic/systems perspective to justify screening. Project Manager for an EPRI project to assess methods and propose research of the impact of organizational factors on PRA.

Nuclear Power Operations and Maintenance
Four Commanding Officer tours of Navy Reserve detachments, and was Executive Officer for the Trident Refit Facility (TRF) Reserve Detachment, performing maintenance and repairs of nuclear submarines. First Reservist to qualify and stand Repair Duty Officer at the Bangor Refit Facility. Shift Engineer at the S1W Navy submarine prototype reactor plant. Supervised and conducted normal startups, shutdowns, power transients, casualty drills for training, scrams, and scram recoveries at the plant. Oversaw all operations, maintenance, and training activities of staff and students. Instructed students and staff on reactor operations and theory. Main Propulsion Assistant and Water Chemistry/Radiological Controls Officer during overhaul and post-overhaul testing.

Computer Skills
Hardware - IBM-compatible (MS DOS or Windows)

Software - Microsoft Office Suite (Word, Excel, PowerPoint, Access), MS Project, Word Perfect, Internet Access Software, PRA Codes (NUPRA, CAFTA, SETS, HRA Calculator), Reliability/Availability Codes (GO, MicroGO).

Employment

Scientech, a Curtiss-Wright Flow Control company, Director of Safety and Risk,
2016 - Present
Division Manager responsible for quality, staff development as well as commercial and technical goals. Continued work in the areas of.Program Development, Risk Management/Decision Analysis, Probabilistic Risk Assessment/Probabilistic Safety Assessment, Human Factors, Project Management, and Technical Review

Scientech, a Curtiss-Wright Flow Control company, Technical Manager of Risk and Reliability Programs,
1996 - 2015
Program Development, Risk Management/Decision Analysis, Probabilistic Risk Assessment/Probabilistic Safety Assessment, Human Factors, Project Management, Technical Review

NUS Corporation, Deputy General Manager, 1995-1996
Program Development, Risk Management/Decision Analysis, Probabilistic Risk Assessment/Probabilistic Safety Assessment, Human Factors, Project Management, Technical Review

NUS Corporation, Project Manager, 1989-1995
Risk Management/Decision Analysis, Probabilistic Risk Assessment/Probabilistic Safety Assessment, Human Factors, Project Management, Reliability Assessment, Task Management

Energy Incorporated, Consulting Engineer, 1987-1989
Probabilistic Risk Assessment/Probabilistic Safety Assessment, Human Factors, Reliability Assessment, Task Management, Accident Analysis, Software Development

U.S. Naval Reserve, Officer, 1987 - 2010
Project Management, Administration Management, Nuclear Operations & Maintenance, Task Management, Quality Assurance, Total Quality Management (TQM), Commanded four Naval Reserve units.

U.S. Navy, Division Officer, 1980-1987
Nuclear Systems, Nuclear Operations & Maintenance, Project Management, Marine Operations, Quality Assurance, Training

Affiliations/Honors

American Nuclear Society, 1980
American Nuclear Society Low Power and Shutdown PRA Standard writing committee

Publications

Julius, J.A., et al, 1993 "Application of the KCB PSA in a "Living" Program" (coauthor), presented at the PSA '93 Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida, January 26-29, 1993.

Julius, J.A., Bertucio, R.C., *Analysis of Core Damage Frequency from Internal Events: Surry Unit 1*, Sandia National Laboratory and EI International, NUREG/CR-4550, Revision 1, Volume 3, USNRC, June 1989.

Julius, J.A., et al, 1988 "Performance of a Detailed Analysis of the Balance of Plant Systems for the James A. FitzPatrick Nuclear Power Plant" (coauthor), presented at the 15th Inter-RAM International Reliability Conference, Portland, Oregon, June 14-17, 1988.

Julius, J.A., et al, 1992 *Probabilistic Safety Assessment for the Borssele Nuclear Power Plant - For Power Conditions, Preliminary Report*, PSAB, HALLIBURTON NUS and Siemens-KWU, March 1992.

Julius, J.A., Jones, D.M., 1992 *Evaluation of Proposed Modifications - First Report - Coverage of Proposed Modifications*, MODdocnr.032-002R1, August 7, 1992, HALLIBURTON NUS Environmental Corporation.

Julius, J.A., Jones, D.M., 1992 *Evaluation of Proposed Modifications - Second Report - Initial Review Report*, MODdocnr.032-003R1, August 28, 1992, HALLIBURTON NUS Environmental Corporation.

Julius, J.A., Jones, D.M., 1992 *Evaluation of Proposed Modifications - Third Report - Detailed Evaluation Report*, MODdocnr.032-004R1, September 29, 1992, HALLIBURTON NUS Environmental Corporation.

Julius, J.A., Jones, D.M., 1992 *Evaluation of Proposed Modifications: Miscellaneous PSA Issues Examined as Part of the Modifications Project*, MODdocnr.032-005R3, October 18, 1992, HALLIBURTON NUS Environmental Corporation.

Julius, J.A., et al, 1993 *Probabilistic Safety Assessment as Part of an Environmental Impact Statement for NPP Borssele, Volumes 1, 2, and 3*, (PSA-MER) 059-003 rev. 0, 059-004 rev. 0, 059-005 rev. 0, HALLIBURTON NUS, Siemens-KWU, KEMA, and EPZ, September 1993.

Julius, J.A., et al, 1993 *Low Power and Shutdown PSA for NPP Borssele, Phase A*, (PSAS-A) PSAS-C-SR-01-R1, HALLIBURTON NUS and Siemens-KWU, December 1993.

Julius, J.A., et al, 1995 *Integrated Probabilistic Safety Assessment for the NPP Borssele, Volumes 1-5* (PSA-3), PSA3-94-1, Rev. 0, Halliburton NUS, June 1995.

Julius, J.A., et al, 1995 *Integrated Probabilistic Safety Assessment for the NPP Borssele, Post Modifications, Volumes 1 and 2* (PSA-3MOD), PSA3-MOD-1, Rev. 0, Halliburton NUS, June 1995.

Julius, J.A., Parry, G.W., Jorgenson, E.J., Mosleh, A, *An Analysis of the Potential for Significant Errors of Commission during the Response Phase to Full Power Transients and Accidents at the Borssele Nuclear Power Plant*, PSAS-N-HI-01-R1, HALLIBURTON NUS, University of Maryland, and Siemens-KWU, December 1993.

Julius, J.A., Parry, G.W., Jorgenson, E.J., Mosleh, A, 1994 "A Procedure for the Analysis of Errors of Commission in a PSA" (coauthor), presented at PSAM-II, An International Conference on the Advancement of System-Based Methods for the Design and Operation of Technological Systems and Processes, San Diego, California, March 20-25, 1994.

Julius, J.A., Jorgenson, E.J., Parry, G.W. and Mosleh, A.M., 1995 "A Procedure for the Analysis of Errors of Commission in a Probabilistic Safety Assessment of a Nuclear Power Plant at Full Power", Reliability Engineering and System Safety, Vol. 50, (1995), pages 189-201.

Julius, J.A., Jorgenson, E.J., Parry, G.W. and Mosleh, A.M., 1996 "A Procedure for the Analysis of Errors of Commission in a Probabilistic Safety Assessment of a Nuclear Power Plant during Non-Power", presented at PSAM-III, An International Conference on the Advancement of System-Based Methods for the Design and Operation of Technological Systems and Processes, Crete, June 17-21, 1996.

J.A.Julius, et al 1996 "Safety Monitor Implementation Project at Callaway, Wolf Creek, and Comanche Peak Stations", presented at PSA-96, An International Conference on the Probabilistic Safety Analyses, Park City, Utah, October 7-10, 1996.

J.A.Julius, E.J.Jorgenson, G.W.Parry, and A.M.Mosleh, 1996 "A Procedure for the Analysis of Errors of Commission in a Probabilistic Safety Assessment of a Nuclear Power Plant During Non-Power", Reliability Engineering and System Safety, Vol. 53, (1996) , pages 139-154.

Julius, J.A., 1999 "Dynamic Human Interaction Models for Risk Monitoring", presented at the PSA '99 Probabilistic Safety Assessment International Topical Meeting, Washington, D.C., August 23-26, 1999.

Julius, J.A., D. M. Jones, 1999 "Insights from Developing Shutdown Risk Monitor Models", presented at the PSA '99 Probabilistic Safety Assessment International Topical Meeting, Washington, D.C., August 23-26, 1999.

Julius, J.A., D. M. Jones, Mike Phillips, 2000 "Integrated Maintenance Rule Assessments Using the Safety Monitor ", presented at the 8th International Conference on Nuclear Engineering, Baltimore, MD, April 2-6, 2000.

Julius, J. A., et al, Probabilistic Safety Assessments of Nuclear Power Plants for Low Power and Shutdown Modes, IAEA-TECDOC-1144, International Atomic Energy Agency, March 2000.

Julius, J. A., et al, "Overview of Risk-Informed Decision Making in Recent US Applications", IAEA Technical Committee Meeting, International Atomic Energy Agency, November 2001.

Julius, J.A., et al, "EPRI Human Reliability Analysis Initiatives in the United States", OECD Nuclear Energy Agency Workshop: Building the new HRA, Strengthening the Link between Experience and HRA, Munich, Germany, January 28-30, 2002.

Julius, J.A., et al, "EPRI Human Reliability Analysis Calculator", presented at PSAM-6, An International Association of Probabilistic Safety Assessment and Management Conference, San Juan, Puerto Rico, June 23-28, 2002.

Julius, J.A., Grobbelaar, J.F. "Second Generation Shutdown Safety Monitor Modeling", presented at PSAM-6, An International Association of Probabilistic Safety Assessment and Management Conference, San Juan, Puerto Rico, June 23-28, 2002.

Julius, J.A., et al, "EPRI Human Reliability Analysis Guidelines", presented at PSA'02, American Nuclear Society sponsored Probabilistic Safety Assessment Conference, Detroit, MI, October 6-10, 2002.

Julius, J.A., et al, "Low Power and Shutdown Risk Assessment Benchmarking", presented at PSA'02, American Nuclear Society sponsored Probabilistic Safety Assessment Conference, Detroit, MI, October 6-10, 2002.

Julius, J. A., et al, Guidance for Incorporating Organizational Factors into Nuclear Power Plant Risk Assessments-Phase 1 Workshop, EPRI-TR-1003322, Electric Power Research Institute, December 2002.

Julius, J. A., et al, Low Power and Shutdown Risk Assessment Benchmarking Study, EPRI-TR-1003465, Electric Power Research Institute, December 2002.

Julius, J.A., et al, "Differences Between Full Power Operation and Low Power / Shutdown Operation, and the Implications on the LPSD Standard", presented at PSAM-7, An International Association of Probabilistic Safety Assessment and Management Conference, Berlin, Germany, June 14-18, 2004.

Julius, J.A., Grobbelaar, J.F., et al, "EPRI HRA Calculator™ - Version 3", presented at PSA'05, American Nuclear Society sponsored Probabilistic Safety Assessment Conference, San Francisco, CA, September 12-15, 2005.

Julius, J.A., Grobbelaar, J.F., et al, "HRA Dependency Analysis Using the EPRI HRA Calculator™", presented at PSA'05, American Nuclear Society sponsored Probabilistic Safety Assessment Conference, San Francisco, CA, September 12-15, 2005.

Julius, J.A., Grobbelaar, J.F., "Integrating Human Reliability Analysis Approaches in the EPRI HRA Calculator®", presented at PSAM-8, An International Association of Probabilistic Safety Assessment and Management Conference, New Orleans, Louisiana, May, 2006.

Julius, J.A., Grobbelaar, J.F., "New Advances in Human Reliability Analysis Using the EPRI HRA Calculator®", presented at the American Nuclear Society 2006 Winter Meeting, Albuquerque, New Mexico, November 12-16, 2006.

Julius, J.A., Grobbelaar, J.F., "Development of Human Reliability Analysis Approach to Fire Probabilistic Risk Assessment", PSAM-9, An International Association of Probabilistic Safety Assessment and Management Conference, Hong Kong, May, 2008.

Julius, J.A., et al, "Benchmarking Human Reliability Analysis (HRA) Methods Against Simulator Data – Method for Comparison", PSAM-9, An International Association of Probabilistic Safety Assessment and Management Conference, Hong Kong, May, 2008.

Julius, J.A., et al, "Results from Comparison of HRA Method Predictions with Empirical Data on Human Performance in Accident Scenarios", PSAM-9, An International Association of Probabilistic Safety Assessment and Management Conference, Hong Kong, May, 2008.

Julius, J.A., et al, "Insights from Comparison of HRA Method Predictions with Empirical Data on Human Performance in Accident Scenarios", PSAM-9, An International Association of Probabilistic Safety Assessment and Management Conference, Hong Kong, May, 2008.

Julius, J.A., Grobbelaar, J.F., Kohlhepp, K.D., et al "EPRI/NRC Fire Human Reliability Analysis Guidelines", presented at PSA'08, American Nuclear Society sponsored Probabilistic Safety Assessment Conference, Knoxville, TN, September 7-11, 2008.

Julius, J.A., Grobbelaar, et al "Automated Human Reliability Dependency Analysis Using the EPRI HRA Calculator®", presented at PSA'08, American Nuclear Society sponsored Probabilistic Safety Assessment Conference, Knoxville, TN, September 7-11, 2008.

Julius, J.A., et al, "The International Empirical HRA Study Using Simulator Human Performance Data", presented at PSA'08, American Nuclear Society sponsored Probabilistic Safety Assessment Conference, Knoxville, TN, September 7-11, 2008.

Julius, J.A., et al, International HRA Empirical Study – Description of Overall Approach and First Pilot Results from Comparing HRA Methods to Simulator Data, NUREG/IA-0215, U.S. Nuclear Regulatory Commission, Washington DC, USA, in publication.

Julius, J. A., et al, Support System Initiating Events – Identification and Quantification Guideline, EPRI 1016741, Electric Power Research Institute, Palo Alto, CA, in publication.

Julius, J.A., et al, EPRI/NRC-RES Fire Human Reliability Analysis Guidelines, NUREG-1921, EPRI 1016741 Electric Power Research Institute, Palo Alto, CA and U.S. Nuclear Regulatory Commission, Washington DC, USA, December 2009.

Julius, J.A., et al, EA Preliminary Approach to Human Reliability Analysis for External Events with a Focus on Seismic, EPRI TR-1025294, Electric Power Research Institute, Palo Alto, CA, USA, December 2012.

KENNETH KIPER

Technical Manager
Risk Applications and Methods,
Engineering Center of Excellence,
Westinghouse Electric Company

EDUCATION

M.S. Nuclear Engineering - Ohio State University, Columbus, Ohio, 1980

M.S. Physics - Ohio State University, Columbus, Ohio, 1978

B.S. Physics and Mathematics - Olivet Nazarene University, Bourbonnais, Illinois, 1976

OVERVIEW

Mr. Kiper is a multi-disciplined Nuclear Engineer, specializing in risk assessment, risk management, and risk applications for nuclear power plants. Most of his 32-year career has been spent as a risk management engineer at the Seabrook Station nuclear power plant. Because of the extensive use of risk assessment during Seabrook's licensing and operation, he has been engaged in every aspect of modern probabilistic risk assessment (PRA), including leading efforts in risk assessment of non-power modes of plant operation. He also participates in a number of industry activities, including leading Standards efforts, owners groups, and peer reviews.

EMPLOYMENT HISTORY

Westinghouse Electric Company (2014 to present)

- Technical Manager, Risk Applications & Methods I Department

  Following his retirement from NextEra Energy Company, Mr. Kiper joined Westinghouse Electric Company's PRA group. There he is responsible for a number of risk-application projects, including leading the PRA peer reviews for several utilities.

NextEra Energy Company, Seabrook Station, Seabrook, NH (1982 to 2014)

- Consultant Engineer, Nuclear Risk Management Department

  Mr. Kiper has more than thirty years nuclear experience at Seabrook Station in every technical area of probabilistic risk assessment, including directing major risk analyses, performing applications, and developing and maintaining an all-modes living PRA. He developed his technical background by participating in the original Seabrook PRA, where he was mentored by a number of experts in risk assessment. He was lead analyst for the IPE and IPEEE reports for Seabrook. He was also the lead analyst for the Shutdown PRA and Spent Fuel Pool PRA that were developed for Seabrook. He utilized the Shutdown PRA as a tool to assess the risk for each refueling outage. He was responsible for periodic major updates of the Seabrook PRA, including recent work to comply with the ASME/ANS PRA Standard. He regularly performs and reviews analyses and applications in the areas of systems analysis, plant sequence modeling, human action analysis, data analysis, external events analysis, containment analysis, and site/consequence analysis.

US Nuclear Regulatory Commission, Washington, DC (1980 to 1982)

- Project Manager, Licensing Division, NRR

Mr. Kiper served two years at the US Nuclear Regulatory Commission as a Project Manager in the Division of Licensing. He was involved in initial licensing of commercial nuclear power plants, including Byron and Braidwood. At NRC, Mr. Kiper received introductory training on probabilistic risk assessment techniques.

PROFESSIONAL EXPERIENCE

ASME / ANS Joint Committee on Nuclear Risk Management (2008 to Present)

Mr. Kiper serves on the Joint Committee on Nuclear Risk Management (JCNRM), which is responsible for development and maintenance of technical standards for risk assessment and risk management for the nuclear industry. He also serves on the executive committee, which provides direction to the standards effort.

Subcommittee on Standards Applications, JCNRM (2012 to Present)

Mr. Kiper helped to organize the new Subcommittee on Standards Applications of the JCNRM and serves as its first chair. This subcommittee is responsible for interfacing with other standards developing organizations to strengthen and standardize the uses of risk assessment techniques.

Subcommittee on Standards Maintenance, JCNRM (2008 to 2012)

Mr. Kiper served as Chair of the Subcommittee on Standards Maintenance (SCSM) of the JCNRM during the major revision to the PRA Standard that resulted in Addendum B. In this role, he lead the SCSM and its associated writing groups responsible for all technical changes to supporting requirements in the combined PRA Standard.

Low Power & Shutdown PRA Standards Writing Group (2003 to Present)

Mr. Kiper is a member of the Low Power & Shutdown (LPSD) PRA Standards Writing Group, as well as past writing group chair. In this role, he is a principle contributor to the first comprehensive LPSD PRA Standard.

Independent Technical Reviewer (2004 to Present)

Mr. Kiper has participated as an independent reviewer on a number of PRA peer reviews and technical projects. This includes international reviews of PRAs in Switzerland (Goesgen, 2004) and Mexico (Laguna Verde, 2007); reviews of Internal Flood PRA (St Lucie, 2010), high winds PRA (Pt. Beach, 2012) and seismic PRA (Diablo Canyon 2013); and technical reviews of HRA projects (ATHEANA users guide, 2006 and Internal Flood HRA Guidelines, 2008).

Risk Management Subcommittee, PWROG (1996 to 2014)

While at Seabrook Station, Mr. Kiper was a member of the Risk Management Subcommittee (RMSC) of the PWROG, where he serves as chair of the External Events Working Group. He is past chair and vice-chair of the working group that preceded the RMSC. Among other projects, he provided technical oversight and review of the WOG seal leakage model (WCAP-16141).

TECHNICAL SPECIALTIES

Low Power & Shutdown Risk Assessment

Mr. Kiper is one of the leading authorities on the development of probabilistic risk models for low power and shutdown states at nuclear power stations. He was one of the lead authors of the original Shutdown PRA for Seabrook Station in 1988 and developed that original Shutdown PRA into an integrated all modes PRA that is used for both on-line and outage risk management. He is an active member and past chair of the ASME/ANS LPSD PRA Standard writing group.

Seismic Risk Assessment

Mr. Kiper has been responsible for the development and maintenance of plant response model for the Seabrook SPRA. He was project manager for IPEEE work in 1992, which included expansion of the original SPRA. He was the lead PRA engineer responsible for the SPRA update in 2004 and integration of the SPRA into the Seabrook PRA model and documentation. He led the seismic walk down effort at Seabrook in 2012 in response to the NRC near term task force request. He has lead several peer reviews of seismic PRAs for the US industry.

Human Reliability Assessment

Through the efforts to upgrade and expand the Seabrook PRA, Mr. Kiper has extensive experience with a number of HRA methodologies, including success (failure) likelihood index methods, cause-based and time-based methods, as well as THERP. He participated in development of the ATHEANA methodology, including trial use as Seabrook in 1997 and was a technical review of the ATHEANA users' guide. Mr. Kiper was a technical reviewer for Internal Flood PRA HRA methods developed by EPRI & NRC and has been a peer reviewer focusing on HRA at several reviews, including at a Swiss power plant (Goesgen) in 2004.

RISKMAN Software

Mr. Kiper is proficient in the RISKMAN risk analysis software and has exploited the capabilities of RISKMAN to create event sequence models using descriptive logic structures. This allows creation of complex sequence models that are also self-documenting so that a reviewer who understands the plant could read the logic rules directly.

James D. Ledgerwood
Outage Manager, Westinghouse Electric Company

SUMMARY OF EXPERIENCE

Mr. Ledgerwood has extensive experience as an accomplished project manager, program manager, and maintenance manager of organizations up to 300 employees. He has successfully demonstrated the ability to engineer, plan, schedule and budget for diverse organizations, including extensive experience with Nuclear Regulatory Commission interface, labor unions, negotiating and administering large contract services, maintenance, outage, quality assurance, design and construction contracts, and operating physical plants for large industrial facilities.

EXPERTISE

Large Industrial Projects

- Main Turbine EHC Digital Upgrade
- Fukushima Daiichi Response Plant Modifications.
- Independent Nuclear Spent Fuel Storage Installation Development and Implementation.
- Large Air Compressor Installation Program Development.
- Underwater Installation in place repair projects.
- Extensive underground piping repair projects with equipment (ESF D/G's) required to remain in service.
- Ground water pollution mitigation projects including geological and ultrasonic mapping (BFNP tritium elimination).
- Numerous Instrumentation and Control projects including Bentley Nevada vibration monitoring systems Westinghouse control systems, Siemens breakers, General Electric Controls, Square D and others.
- Projects including all aspects of major motor/generator repairs including exciter and motor control centers. Also familiar with variable frequency and DC controls.

Maintenance Management

- Participated as a Maintenance Manager in over 16 major industrial outages, and several EAOT ESF D/G outages, all scheduled in accordance with P-6 waterfall critical path scheduling processes.
- Participated as a Maintenance Support Manager responsible for procedures, preventive maintenance, predictive maintenance including vibration, oil, and thermographic analysis in a reliability centered maintenance program.

PERSONAL DATA

Bachelor of Science, University of New York 1986

PROFESSIONAL CERTIFICATIONS (see attached)

United States Nuclear Regulatory Commission Nuclear Plant Senior Reactor Operator License
License Number SOP-43530/ Docket 55-40541
Institute of Nuclear Power Operations Certified Training Instructor
Crane Nuclear Certified Advanced Signature Analyst (Viper) for Motor Operated and Air Operated Valves
Honorable Discharge United States Navy 1986

RELATED SOFTWARE EXPERIENCE

Microsoft Office Suite including EXCEL
Primavera P-6 Scheduling Tools

PROFESSIONAL MEMBERSHIPS/RELATED COURSES

Member of the Project Management Institute
Membership ID 829350
Completed Project Management Institute Courses 2100 (PMP Exam Prep) and 4100 (Tricks of the Trade)
Completed Facilitative Leadership Training (Maximizing Meeting Effectiveness).
Also have extensive training in Labor Relations and Management skills

EMPLOYMENT HISTORY

Westinghouse Electric Company	June 2016 – Present
Dominion North Anna Power Station	May 2015 – May 2016
South Texas Project Nuclear Operating Co.	Nov. 2013 – Apr. 2015
Crane Nuclear Services, Inc.	Feb. 2012 – Nov. 2013
Real Estate Investment and Development	Feb. 2007 – Feb. 2012
Tennessee Valley Authority	July 1997 – Feb. 2007
Houston Industries (Now Reliant Energy)	Jan. 1986 – July 1997
United States Navy	Jan. 1980 – Jan. 1986

DETAILED PROFESSIONAL EXPERIENCE

Westinghouse Electric Company

Outage Manager June 2016 – Present

Responsible for managing all Westinghouse services at various clients, both with the United States and abroad.

Dominion North Anna Power Station May 2015 – June 2016

Project Manager

Responsible for several projects including: Digital upgrade of Main Turbine EHC, Upgrade/replacement of over 700 safety related motor control center buckets/breakers, and three balance of plant upgrade projects. Also responsible for training North Anna PMs and Schedulers on basic resource loading/Level III schedule development.

South Texas Project Nuclear Operating Company Nov. 2013 – Apr. 2015

Sr. Project Manager

Solely responsible for overall implementation of physical/regulatory plant changes and integrated start-up testing associated with complying with NRC EA-12-049/51 (Response to the Fukushima Event). Installed two 1 MW Diesel generators, associated missile proof buildings, 480V distribution system, multiple pumps and plant electrical/mechanical tie ins. Total budget $40M.

Crane Nuclear Services, Inc. Feb. 2012 – Nov. 2013

Project Manager MOV/AOV/Valve Outage Services

Responsible for total project management including upgrade, repair, restoration, set up and testing of safety related motor operated and air operated valves at commercial nuclear plants. Detailed responsibilities include scope determination, schedule development, associated man power and cost plan development, contract development, and project oversight through completion.

Pool Protector L.L.C.

Owner - Real Estate Investment and Development Feb. 2007 - Feb. 2012

Following early retirement in February, 2007 established a company responsible for the procurement, restoration and resale for profit of residential properties.

EMPLOYMENT HISTORY

DETAILED PROFESSIONAL EXPERIENCE

Tennessee Valley Authority — July 1997 – Feb. 2007
Browns Ferry Nuclear Plant

Senior Manager D/Rotational

Project Manager — June 2004 – Feb. 2007

Responsible for development, preparation and implementation of numerous large projects. Each project involved research requiring scoping, initial estimating and budgeting, project presentation to an approving board followed by detailed schedule development and implementation utilizing critical path scheduling processes. Employees utilized were obtained from all local union halls via a sub-contractor under an existing president's agreement. Note that most projects were associated with extensive Nuclear Regulatory oversight. Most notably, the Independent Spent Fuel Storage Installation project ($ 52M) included a five man team review for two weeks by members of the Nuclear Regulatory Commission. Prior to entering private business, I had just completed negotiation with HOLTEC International and TRIVIS Inc. for the next phase of project implementation ($10M).

Maintenance Support Manager — July 1997 – June 2004

Responsible for the Maintenance Support Department. The department consisted of procedure writers, preventive and predictive maintenance engineers for electrical, mechanical and instrumentation and controls disciplines (all members of the Engineering Association Union). The group was also responsible for the corrective action and reliability centered maintenance programs for the Maintenance and Modifications organization (730 personnel) and dealt with all quality assurance and regulatory related interface for the Department. During this period, I participated in 5 major outages (including a generator rewind and exciter replacement with major turbine overhauls) as a Maintenance Manager and implemented numerous major upgrades.

Maintenance Superintendent

Responsible for the oversight of the Maintenance organization consisting of over 300 employees. Responsibilities included all three maintenance divisions (Electrical, Mechanical, Instrumentation and Controls) and the rapid response shift team (fix it now or FIN group). Employee make-up varied form approximately 245 trades and labor personnel and 50 supervisors. During this period, I participated in numerous scheduled outages which, due to rigid critical path schedule monitoring and implementation resulted in Browns Ferry being recognized as a world leader in outage scheduling processes.

Houston Industries (Now Reliant) — Jan. 1986 – July 1997
South Texas Project Electric Generating Station

Instrumentation and Controls (I&C) Maintenance Manager — Sept. 1993 - July 1997

Responsible for all I&C production activities including; corrective maintenance, preventive maintenance, training program implementation, and procedure writing for a 90 employee union organization (International Brotherhood of Electrical Workers). During this period we drastically reduced the corrective and preventive maintenance backlogs, totally redesigned procedures and associated technical programs and achieved world record outage performance during two major outages. As a result, the group was individually recognized by both the Institute of Nuclear Power (INPO) organization and the Nuclear Regulatory Commission. Major projects included development and implementation of control and monitoring system upgrades, plant computer upgrades and complete upgrade of the plants chemical laboratory facilities.

EMPLOYMENT HISTORY

DETAILED PROFESSIONAL EXPERIENCE

Consulting Engineering Specialist, Corrective Action Group Apr. 1992 – Sept. 1993
General Supervisor
Served as the lead station event investigator. During this period, conducted numerous root cause investigations involving serious regulatory interest events and re-engineered the station corrective action program.

General Maintenance Supervisor Sept. 1991 – Apr. 1992
General Supervisor

Acted as the Maintenance Department Manager for all on-shift maintenance activities. Provided oversight, coordination and prioritization for on-shift personnel. Acted as the primary interface between the Plant Operations Department, Engineering Department and the Outage Group.

Lead Reactor Operations Specialist June 1989 – Sept. 1991
Senior Specialist (Level 12)

Served as a licensed Senior Reactor Operator assigned to the Unit 1 Control Room. During this period, I experience one major refueling outage, several forced outages and numerous power manipulations.

Lead Quality Assurance Specialist Apr. 1988 – June 1989
Senior Specialist

Lead evaluator and supervisor for an eight member Nuclear Assurance Surveillance Group.

EMPLOYMENT HISTORY

Senior Quality Assurance /Quality Control Specialist Feb. 1986 – Apr. 1988

Specialist

Performed surveillance audits, QC inspections including electrical, mechanical, I&C and NDE.

United States Navy Jan. 1980 – Jan. 1986

EM-1 SS (E-6)

Served aboard USS Dallas (SSN-700)/Instructor Windsor, Ct. Acted as the refit coordinator aboard ship. Responsible for all shipboard repairs in port. Assigned as Leading Electrical Division First Class P.O. responsible for supervision of a twelve member crew.

Industry/Regulatory Certifications/Plant Committee Memberships

USNRC Senior Reactor Operator License (43530)
Senior Manager Plant Operations Review Committee (PORC) Member
Senior Manager Corrective Action Review Group (CRG) Member
INPO National Academy of Nuclear Training Certified Instructor
10 CFR 50.59 Preparer, Evaluator, Approver and Instructor Certification
10 CFR 72.48 Preparer, Evaluator, Approver and Instructor Certification
Root Cause Analysis Evaluator and Approver Certification
10 CFR 50.49/NUREG 588 (Environmental Qualification) QA Evaluator Certification
ANSI 45.2.6 Level II Electrical Inspector Certification
ANSI 45.2.6 Level II Instrumentation and Controls Inspector Certification
ANSI 45.2.6 Level II Mechanical Inspector Certification
SNTC-1A Level 1 Dye Penetrant NDE Technician Certification
SNTC-1A Level I Magnetic Particle NDE Technician Certification

Leadership/Management/Project Management Credentials

Zenger-Miller Leadership Certification
MARC Labor Relations Management Certification
Interaction Associates Facilitative Leadership Management Certification
Senn-Delaney Leadership Management Certification
Technical Contact Manager Certifications (various)

Technical Certifications/Classes

Motor Operated Valves
Limitorque Actuator Technician
Motor Operated Valve Data Acquisition Technician
Motor Operated Valve Advanced Signature Analysis Technician
Air Operated Valves
Actuator/ Instrument Maintenance and Repair Technician
Data Acquisition and Analysis Technician
Advanced Signature Analysis Technician
CRANE Instructor Certification

JEFFREY T. MITMAN

Rockville, MD

Project Management / PRA Position in the Nuclear Industry

QUALIFICATIONS

Senior Reliability and Risk Analyst with more than 35 years experience in the Nuclear Industry. Responsible for managing risk analysis projects and teams. Solid record of bringing projects in on schedule and budget.

MAJOR ACCOMPLISHMENTS

- Transitioned NRC to detailed PRA models for low power and shutdown significance determinations process evaluations.
- Guided development of and managed industry's first configuration risk management software tool.
- Obtained regulatory approval of EPRI's RI-ISI methodology.
- Managed first PRA of bolted spent fuel storage cask.

EXPERIENCE

US NUCLEAR REGULATORY COMMISSION (Rockville, MD) 2005 - Present

Senior Reliability and Risk Analyst (NRC Office of Nuclear Reactor Regulation)

- Conducted Significance Determination Process (SDP) evaluations of reactor events including development and/or modification of required models.
- Lead analyst for low power and shutdown event issues and concerns.
- Guided development of shutdown Standardized Plant Analysis Risk (SPAR) models.
- Conducted Human Reliability Analysis (HRA).
- Evaluated external event risk from dam failures.
- Participated in post NRC's Fukushima NTTF flooding guidance development.
- Developed NRC's guidance on crediting FLEX in risk-informed regulatory applications.
- Advised NRC NFPA-805 team on issues related to shutdown fire risk.
- Performed evaluations of risk informed license applications.

Reliability and Risk Analyst (NRC Office of Nuclear Regulatory Research)

- Project Manager for the development of shutdown SPAR models

ERIN ENGINEERING AND RESEARCH, INC. (Walnut Creek, CA) 2004 - 2005

Lead Senior Engineer

- Configuration risk management evaluation of at-power fire risk.
- Configuration risk management evaluation of loss of offsite power.

ABE STAFFING SERVICES (Palo Alto, CA) 2003 - 2005

Consultant to EPRI

- Brought project to closure involving Dry Cask Storage PRA project and team, involving Transnuclear bolted cask containing PWR fuel.

EPRI (Palo Alto, CA) 1998 - 2003

Project Manager

- **Outage Risk Assessment and Management** (ORAM-Sentinel): Grew first of a kind software application for performing configuration risk management in nuclear power plants.
  - Conducted research in low power and shutdown risk; shutdown initiating event and event frequency derivation.
  - Delivered multiple versions (including alpha, beta & production), testing and full documentation.
  - Administered utility user group, marketing, contract preparation, technology transfer, technical report publication and training.
  - Actively managed both development and application contracts with multiple suppliers and customers. Managed annual $1M budget.

- **Dry Cask Storage PRA**: Initiated innovative analysis of Transnuclear cask containing PWR fuel.
  - Managed unique team with diverse experience in both cask design and PRA backgrounds.
- **Risk Informed In-service Inspections** Project (RI-ISI): Lead team in obtaining regulatory approval of methodology to safely reduce piping weld inspection requirements using combination of probabilistic and degradation analysis.
  - Responsible for methodology finalization and acceptance by industry and U.S. NRC.
  - Conducted marketing, sales, contract preparation, technology transfer, training and technical report publication.
  - Actively managed both development and application contracts with both suppliers and customers. Managed annual $1M budget.
- **Human Reliability Analysis Project**: Managed project to bring consistency to on industry use of HRA methods.
  - Responsible for EPRI HRA area, including development of HRA Calculator software and establishment of associated users group.

ERIN ENGINEERING AND RESEARCH, INC. (Palo Alto, CA) 1992 - 1998
Lead Senior Engineer
Collaborated with EPRI ORAM-SENTINEL Project Manager in project development and administration, user group administration, contract preparation, technology transfer workshops, technical report generation and editing. Performed ORAM analysis of the Diablo Canyon plant. Performed ORAM Probabilistic Analysis of Perry spent fuel pool. Drafted and edited ORAM V2.0 User's Manual. Assisted in ORAM-SENTINEL software design, performed software debugging. Principle researcher and author of BWR outage contingency report. Prepared marketing and training, materials.

ABB IMPELL CORPORATION (King of Prussia, PA) 1990 - 1992
Lead Senior Engineer

- **Design Basis Documentation**: directed team of three engineers to review PECO Feedwater System Design. Wrote Design Basis Documentation reports for Limerick and Peach Bottom power plants, identifying licensing and design concerns by reviewing the system design as documented in drawings, calculations, vendor manuals, Technical Specifications, UFSAR, SER, SRP, 10CFR50.59 safety evaluations etc. and by interfacing with utility engineering personnel. Prepared Engineering Change Requests as necessary.
- **Shift Outages**: during Limerick Nuclear Power Plant refueling / maintenance outage. Coordinated all shift maintenance work and testing. Collaborated with all groups in power plant, allocating resources as needed to maintain schedule and reporting to senior plant outage management. Performed system reviews prior to placing them back in service. Conducted shift outage meetings. Tracked work group performance against schedule. Advised utility management on techniques for schedule and outage organizational improvements.

GENERAL ELECTRIC COMPANY (San Jose, CA) Experience Prior to 1990
Startup-Test Engineer

- **Shift Startup Engineer:** During power ascension phase coordinated all system testing on shift and startup interface with operations. During preoperational phase, acted as operations shift supervisor responsible for coordinating all system testing and flushing on shift from main control room. Updated senior utility management daily on testing status.
- **Additional positions:** Shift Technical Advisor, Test Engineer, Lead QC / Welding Inspector

EDUCATION / PROFESSIONAL DEVELOPMENT

- BSE, Nuclear Engineering, University of Michigan, Ann Arbor, MI
- Introductory VBA class, University of California, Berkeley, CA
- Misc. business courses at various colleges and universities
- Senior Reactor Operator Certified
- GE Station Nuclear Engineering
- Effective Utilization of PSA, ERIN Engineering & Research, Walnut Creek, CA.

PROFESSIONAL ASSOCIATIONS

- American Nuclear Society (ANS) member since 1978
- ANS Risk Informed Standards Committee (RISC)
- ANS Risk Informed Standards Writing Group on Shutdown PRA Standard
- American Society of Mechanical Engineers (ASME)
- ASME Section XI, Working Group on Implementation of Risk Based Examination
- MIT Professional Summer Programs Guest Lecturer at Risk-Informed Operational Decision Management Course

MARIE POHIDA

Monrovia, MD

SUMMARY

Senior Reliability and Risk Analyst with more than 30 years' experience performing PRA analyses focusing on nuclear reactor shutdown risk and external event risk.

MAJOR ACCOMPLISHMENTS

- Wrote the NRC final safety evaluation report sections pertaining to shutdown risk for internal and external events for the ABWR, AP600, AP1000, and the ESBWR Design Certification.
- Wrote the NRC final safety evaluation report sections pertaining to shutdown risk, high winds, and external flooding risk for the South Texas Combined Licensee Application.
- Developed the Significance Determination Process for low-power and shutdown operation of light-water reactors for the Office of Nuclear Reactor Regulation.
- Developed detailed PRA models for low power and shutdown significance determinations process evaluations.
- Developed and implemented temporary inspection guidance to evaluate licensees' shutdown mitigation capability in response to the Staff Requirements Memorandum (SRM) to SECY 97-168 "Issuance For Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation".
- Participated in inspections of light water reactors in the area of shutdown risk.
- Developed the risk evaluation for SECY 97-168.

EXPERIENCE

U.S. NUCLEAR REGULATORY COMMISSION (Rockville, MD) 1990-Present

Senior Reliability and Risk Analyst (NRC Office of New Reactors 2007-present)

- Lead analyst for low power and shutdown risk, high wind, and external flooding risk.
- Recently completed the Phase 2 safety evaluation report sections pertaining to low power and shutdown internal events risk Level 1 and Level 2 for the APR1400 Design Certification.
- Wrote the NRC final safety evaluation report sections pertaining to shutdown risk, high winds, and external flooding risk for the South Texas Combined Licensee Application
- Reviewed the AREVA EPR design and the Mitsubishi Heavy Industries US-APWR design in the area of low power and shutdown risk for internal events, fires, floods, and seismic.
- Wrote the NRC final safety evaluation report sections pertaining to shutdown Level 1 and Level 2 risk for internal events, fires, floods, and high winds for the ESBWR Design Certification.
- Reviewed AP1000 licensee amendment requests pertaining to shutdown risk and shutdown operations.
- Participated in post NRC's Fukushima NTTF flooding guidance development.
- Guided development of shutdown Standardized Plant Analysis Risk (SPAR) models for AP1000.

Senior Reliability and Risk Analyst (NRC Office of Nuclear Reactor Regulation 1990-2007)

- Lead analyst in NRR for low power and shutdown risk Level 1 and Level 2.
- Developed the Significance Determination Process (SDP) for low-power and shutdown operation of light water reactors.
- Developed detailed PRA models for low power and shutdown SDP phase 3 event evaluations.
- Instructed regional senior reactor analysts how to use the low power and shutdown Phase 1 and Phase 2 SDP tools.
- Conducted human reliability analysis (HRA).
- Reviewed licensee amendment requests in the area of full power, low power, and shutdown risk.
- Developed and implemented temporary inspection guidance to evaluate licensee's shutdown mitigation capability in response to the SRM to SECY 97-168.
- Participated in inspections of light water reactors in the area of shutdown risk.
- Evaluated the risk significance of shutdown operating events prior to NRC implementation of the Reactor Oversight Process.
- Developed the risk evaluation for SECY 97-168 for the proposed shutdown rule.
- Participated in the Senior Consultant Group to guide development of the Grand Gulf and Surry Shutdown

PRAs for the Office of Research (NUREG/CR 6143 and NUREG/CR 6144).

- Wrote the risk sections of the safety evaluation report on the Oconee Emergency AC Power System which required a detailed review of the Keowee Hydroelectric Units PRA.
- Reviewed daily 10CFR 50.72 reports for risk significance to support NRR operating experience briefings.
- Wrote the risk sections in NUREG 1449, "Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States".

BALTIMORE GAS AND ELECTRIC- CALVERT CLIFFS NUCLEAR POWER PLANT 1986-1989
Engineer

- Performed design basis containment response analyses using the CONTEMPT computer code.
- Developed fault tree systems models for the Calvert Cliffs Individual Plant Examination (IPE) in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities".
- Co-author of the Calvert Cliffs Unit 1 Industry Degraded Core Rulemaking (IDCOR) individual plant examination.
- Wrote justifications for continued operation (JCO) using risk insights.
- Used CAFTA, IRRAS, and GO reliability software.
- Reviewed the NRC's risk-based inspection program for Calvert Cliffs

PUBLICATIONS

- "Technical Challenges Associated with Shutdown Risk when Licensing Advanced Light Water Reactors," Marie Pohida, Jeffrey Mitman, Probabilistic Safety Assessment and Management Conference June 2014.

EDUCATION / PROFESSIONAL DEVELOPMENT

- BS in Nuclear Engineering, University of Maryland, 1986
- Graduate coursework in reliability engineering, University of Maryland, College Park, Maryland.
- Completed NRC BWR full-series-training

RESUME

Stephen E. Prewitt

Objective

Apply Nuclear Power Plant knowledge and experience in providing support to the Commercial Nuclear Power industry.

Education

July 1990 - Vogtle Units 1 & 2 Senior Reactor Operator License

May 1987 - Vogtle Units 1 & 2 Reactor Operator License

June 1986 - Georgia Institute of Technology Research Reactor Training Course

November 1983 – Nuclear Reactor Fundamentals, Memphis State University

Professional experience

May 2013 to Present - Contract Consultant developing flexible and diverse coping strategies for Vogtle Electric Generating Plant in response to NRC orders as related to the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station. This included development of procedures for FLEX Strategy Guidelines and Strategy Implementation Guidelines addressing the ability to respond to an extended loss of AC power at the station using installed plant equipment and the deployment and operation of temporary equipment. Participated in the INPO and NRC audits to verify plant compliance with the orders.

May 2012 to January, 10th, 2013 – Contract Training instructor Units 1 & 2. SRO Certified and qualified as classroom instructor, simulator instructor, and lesson plan developer. Taught Hot License18 and License Operator Continuing Training simulator and classroom.

April 2011 to May 2012 – Contract consultant as Plant Vogtle Operations Department Corrective Action Program coordinator. Duties include daily review, analysis and resolution of condition reports. Fill the required quorum position of Senor Reactor Operator in all CAPCO meetings. Also attend site minor and major design change meetings to determine impact to the Operations Department processes and procedures.

January 2011 to March 2011 – Westinghouse Site Coordinator for Job Familiarization Project at Plant Vogtle. Scheduled and provided oversight of all site activities for 25 Chinese power plant personnel from the Hiayang Nuclear Plant in mainland China. Duties included scheduling task observations in Operations, Maintenance, Training and Engineering for these disciplines to familiarize personnel with normal daily operation of a US commercial Nuclear Power Plant. This involved peer to peer contact/interviews and meeting formats for daily work processes.

January 2010 to January 2011 – Project Manager for implementation of Central Plant Procedure Group concept at Plant Vogtle.

January 2000 to January 2010 – Vogtle Units 1 & 2 Operations Procedure Group Supervisor. Directly involved with development and implementation of the electronic review, approval, and issue of plant procedures. Initiated work process that effectively reduced the Operations procedure revision backlog from over 900 to less than 100. This was accomplished by creating and executing new work flow processes to eliminate duplication and streamline applications. Covered on Shift assignments as Unit Shift Supervisor when required.

January 1994 to January 2000 – Vogtle Units 1 & 2, on shift as Shift Support Supervisor providing supervision and oversight for System Operators, clearance and tagging, system startup and shutdown, refueling supervisor support during outages and all aspects in the field of day to day plant operation.

September 1989 to January 1994 – Vogtle Training Center as Operations instructor. During this time upgraded RO Licensed to SRO, worked in lesson plan development, SO training, and Licensed Operator Requal program as a classroom and simulator instructor.

May 1985 to September 1989 – Attended Licensed Operator training obtaining Reactor Operator License in May 1987. Worked on shift as Assistant Plant Operator and Plant Operator during initial startup and subsequent Commercial operation of both Vogtle Units 1 & 2.

January 1984 to May 1985 – System Operator (SO) in initial hiring of Operation staff for Vogtle Units 1 & 2. Worked on shift in support of new system turnover from construction and construction acceptance testing.

DONALD J. WAKEFIELD

Senior Consultant, Operational Risk and Performance Consulting

PROFESSIONAL HISTORY

ABSG Consulting Inc., Irvine, California, Senior Consultant, Operational Risk and Performance Consulting, 2000–Present

EQE International, Inc., Irvine, California, Senior Consultant, 1997–2000

PLG, Inc., Irvine, California, Senior Consultant, 1983–1997

Cygna Energy Services, Associate, 1981–1983

General Atomic Company, Engineer, 1974–1981

PROFESSIONAL SUMMARY

Mr. Donald J. Wakefield has more than 40 years of experience in all phases of the risk analysis of nuclear power plants and other complex facilities, including human reliability analysis (HRA). From 2014 to 2015, he performed a Level 2 analysis for both full power and shutdown events of the Kernkraftwerk Gösgen plant in Switzerland and a Level 2 analysis for both full power and shutdown conditions for the Axpo AG plant, also in Switzerland. From 2012 to 2014, Mr. Wakefield served as the lead probabilistic risk assessment (PRA) consultant for the seismic PRAs of four nuclear plants for FirstEnergy Nuclear Operating Company (FENOC), who was among the industry first movers in responding to the Task 2.1 and 2.3 Fukushima Near-Term Task Force requirements.

He has served as principal investigator and project manager for the risk assessment of several nuclear plants in the United States and the Far East. He served as a key risk analyst on assessments of a floating, production, offloading and storage facility (FPSO), an oil tanker, and for the handling of abandoned chemical weapons in China. Mr. Wakefield is also project manager for the development of ABSG Consulting Inc.'s (ABS Consulting) RISKMAN™ software for quantitative risk assessment applications. He is now serving as the Chairman of the Low Power and Shutdown PRA Standard Writing Group (ANS 58.22) and serves on the American Society of Mechanical Engineers' Committee on Nuclear Risk Management and American Nuclear Society's (ANS) Risk Informed Standards Committee.

PROFESSIONAL EXPERIENCE

From March 2012 through December 2014, Mr. Wakefield served as the lead PRA consultant to FENOC for seismic PRAs at Perry Nuclear Power Plant, Davis-Besse Nuclear Power Station, and both Beaver Valley Nuclear Power Station units. Under his direction, new seismic PRA models were developed for each unit, two using the CAFTA suite of codes and the Beaver Valley units using the RISKMAN code.

In late 2006, Mr. Wakefield became the writing group chairman for the ANS PRA Standard for Low Power and Shutdown Events (ANS-58.22). This standard, under his direction, was approved for trial use. Mr. Wakefield has also been active in the modeling of shutdown events. He performed a review of the Seabrook Station, all power modes PRA model. He recently performed a Level 2 analysis for both full power and shutdown events of the Kernkraftwerk Gösgen plant in Switzerland, and a Level 2 analysis for both full power and shutdown conditions for the Axpo AG plant, also in Switzerland. These efforts are in addition to his past Level 1 shutdown studies for the High Flux Australian Reactor in Australia, Takahama 3 and 4, and for other plants in Japan.

Mr. Wakefield served as the principle investigator for a fire risk analysis of the Watts Bar Unit 2 plant to satisfy its FIVE licensing requirement. This study was performed using CAFTA.

Mr. Wakefield has also performed human reliability analysis for nuclear plants. He served as task leader for the human factors analysis of the Three Mile Island Unit 1 probabilistic safety assessment (PSA). Performed the original human factors analysis for the PSA and then, nearly 20 years later, worked with the plant safety staff to update the analysis using the Electric Power Research Institute (EPRI) HRA Calculator®. Mr. Wakefield served as an independent reviewer for the South Texas Project upgrade to the latest EPRI HRA Calculator and performed a similar review effort for Pacific Gas and Electric Company. Mr. Wakefield was co-author of the EPRI report on the SHARP-1 approach to HRA analyses for PSAs.

Mr. Wakefield served as principal investigator for the Beaver Valley Units 1 and 2 PSA performed to satisfy U.S. Nuclear Regulatory Commission (NRC) individual plant examination (IPE) and individual plant examination of external events (IPEEE) requirements. He also provided expertise in developing and analyzing the Sequoyah and Watts Bar PSA plant models to satisfy the IPE.

He served as project manager for the Salem PSA update and as technical consultant for a PSA of the new production (i.e., weapons materials) modular gas-cooled reactor.

He was a key contributor to accident sequence modeling, including human factors analysis, and seismic analysis for the Diablo Canyon PSA.

Mr. Wakefield served as principal investigator in charge of extending a fault tree linking PSA plant model for a pressurized water reactor in the Far East to accommodate the assessment of plant internal fires and seismic events.

He is a consultant specializing in accident sequence modeling and plant systems analysis for probabilistic safety assessments. Recently he served as technical advisor and sequence model architect for a risk assessment model for the excavation and disposal of abandoned chemical weapons in China. The study considered weapon handling errors, plant fires and weapon explosions there from. This assessment looked at all initiating events and the sequence development extended to payouts resulting from worker and population exposures, building and equipment losses and from environmental cleanup costs. Mr. Wakefield served as the technical lead and coordinated inputs from the Knoxville, Tennessee, San Antonio, Texas, and Irvine, California, offices for use by the ABS Consulting Tokyo, Japan, office.

Served as senior analyst for the development of a quantitative risk assessment model for an FPSO facility hypothetically located in the Gulf of Mexico. This model, funded internally by ABS Consulting, looked at risk to the workers from pool fires, jet fires, and environmental damage from potential oil spills. Also, in 1995, Mr. Wakefield performed the risk assessment portion of an explosion analysis for the Agbami FPSO owned by Star Deep Water Petroleum Limited, and one for the GX Platform owned by Exxon Mobil for Mustang Engineering. He also served as advisor for the PSA of a new, double-hulled oil tanker.

Mr. Wakefield developed the CAFTA-based accident sequence model for a seismic margins assessment for the ACR-700 design for Atomic Energy of Canada Limited.

Mr. Wakefield served as instructor for numerous PSA courses and provided extensive utility training sessions both in the U.S. and abroad. He served as course instructor to the NRC for the risk assessment of external events and to describe the large event tree approach to sequence modeling.

Mr. Wakefield provides technical direction and project management for the development of ABS Consulting's RISKMAN PSA software and administers the RISKMAN Technology Group (a utility users' group). This user's group, now in its twenty-seventh year, funds the maintenance and development of RISKMAN upgrades. Mr. Wakefield provides the interface between the user's group members, and the RISKMAN development team.

Mr. Wakefield was a substantial contributor to a 5-year high temperature gas-cooled reactor (HTGR) risk assessment study. He developed numerous improvements to severe accident consequence computer programs for the HTGR. Quantified uncertainties in severe accident source terms and dose assessment for the HTGR, the first such assessment ever accomplished for any reactor type. Developed a procedure for prioritizing HTGR safety research programs using PSA and formulated an initial set of research recommendations. Prepared test specifications to implement research recommendations.

Mr. Wakefield has authored numerous scientific papers on the subject of probabilistic risk assessment methods including such topics as importance measures, comparison between event tree and fault tree linking, and human reliability analysis techniques.

EDUCATION

M.S., Nuclear Engineering, University of California, Berkeley, 1974
B.S., with Highest Honors, Engineering Mathematics, University of California, Berkeley, 1973

LANGUAGE

Fluent: English

MEMBERSHIPS

American Nuclear Society
Phi Beta Kappa, National Scholastic Honor Society
Tau Beta Pi, National Engineering Honor Society

AWARDS

ANS Standards Service Award, 2015
Regents Fellowship, University of California, 1974
Department of Engineering Certificate Award, 1973

PUBLICATIONS

Klügel, J.-U., S. B. Rao, et al., "SPSA 2016 – Shutdown Probabilistic Safety Assessment," ABSG Consulting Inc., prepared for Kernkraftwerk Gösgen-Däniken AG, R-2129227-1853, July 31, 2016.

Klügel, J.-U., S. B. Rao, et al., "GPSA 2015 – Gösgen Probabilistic Safety Assessment," ABSG Consulting Inc., prepared for Kernkraftwerk Gösgen-Däniken AG, R-2129227-1853, October 31, 2015.

Wakefield, D. J., F. R. Beigi, and K. R. Fine, "An Approach to Seismic Probabilistic Risk Assessment Systems, Structures, and Components Screening," presented at the 2015 International Topical Meeting on Probabilistic Safety Assessment and Analysis (PSA 15), Sun Valley, Idaho, April 26–30, 2015.

Wakefield, D. J., and J. C. Lin, "A Unified Approach to PSA Accident Sequence Model Quantification," presented at the 12th International Probabilistic Safety Assessment and Management Conference (PSAM 12), Honolulu, Hawaii, June 2014.

Wakefield, D. J., "2012 Status of 'Low Power and Shutdown PRA Methodology Standard', ANSI/ANS-58.22," presented at 2012 20th International Conference on Nuclear Engineering (ICONE 20), Anaheim, California, July 20–August 3, 2012.

Wakefield, D. J., et al, "RISKMAN®, Celebrating 20+ Years of Excellence!," presented at 10th International Conference on Probabilistic Safety Assessment and Management (PSAM 10), Seattle, Washington, June 7–11, 2010.

Wakefield, D. J., "Quantification of Fault Tree Models for Initiating Events," Proceedings of 2010 18th International Conference of Nuclear Engineering (ICONE 18), Xi'an, China, May 17–21, 2010.

Wakefield, D. J., and Y. Xiong, "Importance Measures Computed in RISKMAN® for Windows," 5th International Conference on Probabilistic Safety Assessment and Management (PSAM 5), Osaka, Japan, November 27–December 1, 2000.

Johnson, D. H., D. J. Wakefield, and R. Cameron, "Use of PSA in Risk Management at a Research Reactor," presented at the 1999 International Topical Meeting on Probabilistic Safety Assessment (PSA 99), Washington, D.C., August 22–25, 1999.

Quilici, M., W. T. Loh, and D. J. Wakefield, "IPEEE Reports Survey," prepared for Computer Software Development Co., Ltd., PLG-1194, Tokyo, Japan, March 1998.

Wakefield, D. J., "PSA and RISKMAN® Software Training Course," presented to Tennessee Valley Authority, Newport Beach, California, PLG-1195, February 2–6, 1998.

Wakefield, D. J., and D. H. Johnson, "A Level 1+ Probabilistic Safety Assessment of the High Flux Australian Reactor," prepared for Department of Industry, Science and Tourism, Canberra, Australia, PLG-1200, January 1998.

Wakefield, D. J., and D. H. Johnson, "Summary Report – A Level 1+ Probabilistic Safety Assessment of the High Flux Australian Reactor," prepared for Department of Industry, Science and Tourism, Canberra, Australia, PLG-1201, January 1998.

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EDUCATION

Ph.D in Neuroscience, University of Pennsylvania, 1993
M.S in Computer Vision and Biophysics, China Academy of Science, 1986
B.S in Electrical Engineering, Shandong University, China, 1983

PROFESSIONAL EXPERIENCE

2008 - Present, Sr. **Human Performance Engineer,** Human Factors and Reliability Branch, Division of Risk Analysis, Office of Nuclear Regulatory Research, US Nuclear Regulatory Commission, Rockville, MD

Responsibilities: Served as a project manager and technical expert for conducting and directing human factors and reliability research for nuclear safety; Led the development of the NRC's human reliability analysis method, an Integrated Human Event Analysis System (IDHEAS); Developed the NRC's guidance for conducting expert elicitation and authored the White Paper Guidance for expert elicitation.

2002 –2008, Engineering Research Psychologist, Human Factors Research Division, Federal Aviation Administration (FAA), Oklahoma City, OK.

Responsibilities: Served as a principal investigator and project manager in human factors and cognitive engineering. Areas of research included human performance analysis, performance measurements, human-system interface design and evaluation, personnel selection and training, and safety intervention.

2000 – 2002 Senior Research Scientist, San Jose State University Foundation at Human Factors Division, NASA Ames Research Center, Moffett Field, CA

Responsibilities: Conducted research for space shuttle cockpit design, focusing on perceptual learning, attention, and image quality assessment. Conducted human factors engineering and practices in aerospace systems.

1998 - 1999 NIH Research Fellow, Department of Psychology, Stanford University, Stanford, CA

Responsibilities: Conducted research in cognitive psychology, visual attention, object representation, and human performance measurement. Research projects involved fMRI measures, psychometric experimental design, statistic data analysis, and human performance modeling.

1993 - 1997 Research Associate, Brain and Cognitive Science, MIT, Cambridge, MA, and Computational Neural Systems, Caltech, Pasadena, CA.

Responsibilities: Conducted research in brain learning, attention and memory, neural network, and planning and cognitive control. Developed computational models of sensory-motor integration and cognitive intention.

1988 - 1993 Biomedical Fellowship Research Assistant, Departments of Physiology and Neuroscience, University of Pennsylvania, Philadelphia, PA

Graduate study focused in cognitive neuroscience, behavioral psychology, and modeling/simulation of complex systems. Thesis addressed brain learning, memory, and visual information processing. Performed a series of neural network simulation of brain learning mechanisms and developed models of training and learning.

Jeffery Wood, PhD
Reliability and Risk Analyst
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Work Experience

NRC Reliability and Risk Analyst, GG-14 9/2010 – present

- Program manager for development of SAPHIRE risk assessment software and promotion of use by risk analysts throughout the agency
- Lead analyst for internal flooding and shutdown events supporting NRC's comprehensive Level 3 PRA study
- Nominated to serve as vice chairperson of international common cause failure data exchange project
- Briefing management team and Advisory Committee for Reactor Safety members on status of projects
- Represent NRC at technical conferences and standards committees

NRC Reliability and Risk Analyst, GG-13 9/2008 – 9/2010

- Program manager and quality assurance auditor for risk assessment software development
- Interface with other government agencies for sharing risk assessment software and methods
- Assisted in extending NRC's SPAR risk assessment models to include additional external hazards (e.g. fire, flooding and seismic)
- NRC representative to international data exchange project
- Interface with university grantees on risk assessment research

NRC Nuclear Safety Professional Development Program 8/2006 – 9/2008

- On-the-job and classroom training on nuclear power plant design, operation, and accident analysis
- Develop familiarity with statistical analysis and probabilistic risk assessment techniques
- Assembled plant information database for use by emergency response center staff

Brookhaven National Laboratory, Research Consultant 7/2005 – 7/2006

- Maintained radiation detection system in support of large particle collider experiment
- Supervised data production and distribution of results to end users

Education

Doctor of Philosophy degree in Physics 9/2005

University of California Los Angeles
Dissertation: *Polarimetry at the Brookhaven AGS Using Proton-Carbon Coulomb-Nuclear Interference*

Education (continued)

Master of Science degree in Physics 3/2001
University of California Los Angeles

Bachelor of Arts degree double major in Physics and Mathematics 5/1999
Indiana University
Graduated with High Distinction and Phi Beta Kappa Honor Society

Awards & Accomplishments

- Special act award for outstanding response to Inspector General audit of software development program
- Special act award for establishing contacts with international counterparts and organizing technical exchange meetings
- Group award for documenting and responding to comments in support of external expert peer review of a comprehensive risk assessment project
- Special act award for organizing academic and government participants for expert panel on uncertainty analysis in risk-informed decision-making
- Recipient of multiple performance awards
- Completed graduate-level Nuclear Engineering courses in Thermal Hydraulics and Environmental Risk Assessment (University of Maryland)
- Nominated to serve on American Society of Mechanical Engineers / American Nuclear Society Joint Committee on Nuclear Risk – Working group for probabilistic risk assessment standards

Selected Papers & Publications

- J. Wood, et al, “A Compendium of Risk Assessment Studies by US Nuclear Regulatory Commission Office of Nuclear Regulatory Research,” Proceedings of ESREL 2017, Portoroz, Slovenia, June 2017.
- J. Wood, et al, “ICDE Project Collection and Analysis of Common-cause Failures of Emergency Diesel Generators,” Proceedings of 13th International Conference on Probabilistic Safety Assessment & Management (PSAM13), Seoul, October 2016.
- J. Wood, et al, “Estimating Conditional Failure Probabilities of Observed Piping Degradations,” Proceedings of 11th International Probabilistic Safety Assessment and Management Conference and the Annual European Safety and Reliability Conference, Helsinki, June 2012.
- S. Khericha, et al, “Development of Probabilistic Risk Assessment Model for BWR Shutdown Modes 4 and 5 Integrated in SPAR Model,” Proceedings of 10th International Conference on Probabilistic Safety Assessment & Management (PSAM10), Seattle, June 2010.
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APPENDIX E LPSD PRA PIRT PARTICIPATORY PEER REVIEW REPORT

Two NRC staff members, Jing Xing and Jeff Wood, formed the participatory peer review team for this project. The two reviewers participated in the entire elicitation process. Jing Xing is the author for the NRC White Paper of Guidance for Expert Elicitation (Xing and Morrow, 2016[1]), referred to as White Paper Guidance in this peer review report; her review was primarily focused on the elicitation process. Jeff Wood focused on both technical and process aspects. The review panel provided oral or verbal comments to the PIRT coordinator and facilitator throughout the elicitation process. The PIRT team essentially addressed all the comments and improved the process accordingly. This peer review report summarizes the participatory peer review panel's key observations.

E.1 Compliance with the Basic Principles of Use of Expert Judgment in Decision Making

The ultimate objective of conducting an expert elicitation is to appropriately represent the center, body, and range of the technical community's views about a technical problem. The NRC White Paper Guidance delineates seven basic principles for performing expert elicitation. The LPSD-PIRT complies with these principles:

Representation of technical community – The expert panel consists seven experts from the US nuclear industry and the NRC. Collectively, the panel possesses sufficient knowledge in areas of PRA, low power/shutdown operations, and risk management. The experts in the panel are recognized technical leaders in one or several areas.

Independent intellectual ownership – The project team ensured that all the inputs to the elicitation were shared with every experts. The expert panel understood that they were not representing their employer or organization on the panel, but were serving in their own right as a recognized leader in their respective field. The project team made this clear to the panel. The reviewers observed that expert judgment was based on the experts' knowledge and expertise, not the positions of the project sponsors or organizations the experts were associated with. However, as discussed later, some experts occasionally discussed the positions of the organizations they represented during the workshop, although those conversations were mostly off-topic discussion with negligible impact on the elicitation results.

Avoidance of conflicts of interest – The nature of the technical issues in this project does not post any concern for conflicts of interest. The project team assured that the panel members have no duties or responsibilities that would create the appearance of a conflict of interest. Yet, the experts were not asked to make explicit disclosure on conflicts of interest.

Breadth of state of knowledge – This refers to the range of knowledge and interpretations about the technical issue. The breadth of state of knowledge was primarily addressed through the selection of the expert panel members. The panel members bring diverse experience with different aspects of LPSD PRA, including areas such as HRA and thermal hydraulic modeling. The panel also included experts with experience with nuclear power plant outage management and operations, and one expert that was experienced with the specific outage practices and

[1] Xing, Jing and Morrow, Stephanie. *White Paper: Practical Insights and Lessons Learned on Implementing Expert Elicitation*, ADAMS accession number ML16287A734, U.S. Nuclear Regulatory Commission, Washington, DC, October 2016.

procedures used at the reference plant. The expert panel members bring experience in developing LPSD PRA for a variety of nuclear plants and different plant design types. However, due to the plant specific nature of the problem statement, the experts were asked to consider their past experience and apply their judgment as applicable to the specific design and operation of the reference plant. It is noted that some models and data that the experts are familiar with may not be applicable to the reference plant. The breadth of knowledge brought to bear on this problem must be limited to that which is applicable to the reference plant low power and shutdown risk.

Interaction and integration – The elicitation process was performed through interaction and integration. The experts began interactions in the preparation stage of the process, e.g., identifying and refining technical issues to be elicited, compiling available data and models. At the face-to-face meeting, the experts presented and defended their interpretations of the technical issues, and they challenged other's interpretations. The final results represent the integrated belief of the expert panel.

Structured process – The LPSD PRA PIRT employed a well-structured process to facilitate interaction and integration. The process is further discussed in Section E.4.

Transparency - The process and the information generated in LPSD PRA PIRT are documented in a transparent way. The documentation includes the input data and models that were considered, the process employed, the results obtained, and the caveats and limitations of the inputs, process, and results. Such transparency helps to demonstrate the stability and integrity of the results.

E.2 Technical Challenges

The LPSD PIRT process presented a number of technical challenges to the experts participating in the panel. Four key types of technical challenges are identified and presented below. Dealing with these challenges appears to be unavoidable due to the nature of the PIRT subject matter. However, the PIRT team has taken effort to address each challenge. The steps that the PIRT team took to address the challenges are discussed further throughout this peer review report. The strategies that were taken to address the technical challenges included: selecting qualified panel members, disseminating thorough background information relevant to the problem statement (discussed further in Section E.3,) executing the elicitation process (discussed further in Section E.4,) and adhering to good practices such as piloting and training on the elicitation process (discussed further in Section E.5.) The PIRT team's effectiveness in managing these challenges allowed the panel members to provide their expert judgment to process and successfully complete the PIRT.

Challenge 1. The large number of parameters required for the PIRT

The purpose of the PIRT is to identify the plant operating states, hazards, and outage types and rank these according to their importance to LPSD risk. The ranking must also taking into consideration other important influences, including variations in plant configuration, out-of-service equipment, operator/maintenance activities, and thermal-hydraulic analyses. Covering all of these aspects of LPSD risk creates for a large scope problem. A large number of parameters had to be defined and evaluated by the experts.

The final ranking considers 21 plant operating states, and the risk associated with each state is evaluated for 16 different evaluation criteria, which consider factors that are important to safety four hazard categories (internal events, internal flooding, internal fire, and seismic events) and for both core damage and radiological release. This yields 336 individual plant operating state ranking evaluations. Underlying these rankings are many more parameters that define the top-level criteria and sub-criteria that were used in the ranking process. In support of 336 the plant operating state rankings, the experts also performed 64 top-level criteria comparisons, 8 sub-criteria comparisons, 48 sub-criteria ranking category comparisons, and 48 sub-criteria ranking category definitions. In addition, experts were requested to include justification comments for all of their comparisons.

In discussions with the participatory peer reviewers, the PIRT team acknowledged the challenge of having so many comparisons that were needed for this process. The PIRT team took several steps to limit the burden on the experts. First, the PIRT team went through significant effort to develop the PIRT elicitation forms, including revisions to the forms after piloting them with the participatory peer reviewers. The forms structured the comparisons in a logic way and included embedded information and guidance to assist with the process. The PIRT team also developed a useful summary table of PIRT parameters, evaluation questions, ranking categories and considerations. The summary table was included in the elicitation instructions along with other key reference information that the experts needed when formulating their comparisons. This gave experts an easy-to-use reference when working through their evaluations.

Challenge 2. The problem statement is plant-specific

The problem that is addressed by this PIRT process involves ranking of plant operating states, hazards, and outage types according to their importance to LPSD risk. The management of outages can vary significantly from plant to plant. In order to address the problem statement, the expert panel must be familiar with the way that outages are managed at the reference plant. While the panel members have expert knowledge of outage risk, they do not necessarily have the familiarity with the reference plant outage management.

This challenge was addressed by selecting panel members that were able to help address the plant specific nature of the PIRT. Two panel members were selected based on their experience with managing outages at the reference plant and similar plants. One member has experience and an outage manager for Westinghouse Corporation, supporting outages at PWR plants similar to the reference plant. Another panel member worked as a Senior Reactor Operator and later a training instructor at the reference plant. These members bring significant experience with the reference plant and how outages are managed at the plant and similar plants. Having these panel members greatly assisted the entire panel in developing a common understanding of the reference plant operating details.

Challenge 3. The large amount of reference information provided to the panel

The scope of this PIRT and the plant specific nature of the problem required that a large amount of plant specific information be provided to the expert panel. Information was provided to the panel members to support their common understanding of the reference plant approach to outage operations and management. The provided information included a proposed set of POS definitions, the proposed PIRT ranking criteria definitions, plant procedures, reference plant outage reports documenting past refueling outages, historical LPSD PRA references, PIRT process references, and a summary of plant specific calculations used for estimating heat up, accident sequence timing, and success criteria.

The experts were not expected to review all of the reference materials. A complete and thorough review of all the material could not practically be completed during the timeframe of the PIRT process. The experts were expected to bring their own knowledge and experience base to the

PIRT process, and they could refer to the reference material as needed. The extent of review that each expert performed was not tracked.

Challenge 4. The unclear relationship between criteria ranking and overall results

The expert panel members were briefed on the PIRT process during the familiarization meetings. The stated purpose to identify the plant operating states, hazards, and outage types and rank these according to their importance to LPSD risk appeared to be clearly understood by the experts. However, the AHP ranking process was less clear to the experts. The relationship between ultimate results and the top level criteria and sub-criteria comparisons was not apparent. This led to many questions from the experts about the elicitation process. Some voiced concern that the process may misconstrue their intended rankings.

The PIRT team took several actions to alleviate the experts' concerns. First, the PIRT team walked through examples and explained the AHP approach. The PIRT team also demonstrated how to complete the PIRT forms during the individual elicitation sessions. The experts were also given ample time to complete the forms. The experts were allotted up to two weeks to complete the forms on their own. Finally, the PIRT team facilitated the group meeting and reviewed the individual elicitation results. During the group meeting discussions, the experts were able to clarify their understanding of the process and hear the perspectives of the other experts. The experts were permitted to revise their forms, if they thought it was necessary.

E.3 Assemble and Disseminate Background Information

This is to provide the expert panel the most complete and up-to-date information that adequately represents available data regarding the technical issue. The scope of this PIRT and the plant specific nature of the problem required that a large amount of plant specific information be provided to the expert panel. Information was provided to the panel members to support their common understanding of the reference plant approach to outage operations and management.

The project team initially identified the information to be used as background information. As the expert panel was formed, the experts recommended additional sources of data. The project team and expert panel identified more data sources needed during the training and technical issue familiarization sessions. The provided information included:

- a proposed set of plant operating state definitions;
- a proposed set of plant outage type definitions;
- the proposed PIRT ranking criteria definitions;
- several plant procedures;
- reference plant outage reports documenting past refueling outages;
- historical references on LPSD risk and PRA studies;
- references on past PIRT processes; and
- a summary of plant specific calculations used for estimating heat up, accident sequence timing, and success criteria.

The PIRT team discussed the key reference information during the PIRT familiarization meetings that were held prior to the individual elicitation. These discussions helped the experts focus on the most important reference documents and helped to establish a common understanding among the panel members. The panel members were also asked to provide their feedback on the PIRT parameters. This set of information included the plant operating state and outage type definitions, hazard categories, and ranking criteria. All of the panel members provided input either during the discussions or by written comments to the PIRT team. This allowed the experts to redefine aspects of the PIRT parameters and gave them some feeling of ownership of the process. The comments provided by the experts were incorporated into the final PIRT instructions, which were distributed prior to the individual elicitations.

The review panel assessed that the background information met the following criteria:

- Representativeness – Information covers the most important aspects of LPSD risk and is relevant to the problem statement.

The information provided to the experts was representative of the important aspects of LPSD risk. The proposed PIRT parameters defined the scope of the ranking that was to be performed. The experts provided comments on the PIRT parameters. The final parameters are representative of the most important aspects of LPSD risk. Also, the information was relevant to the plant specific evaluation of outage risk at the reference plant. The provided documents summarized the reference plant experience with recent refueling outages. Several plant procedures relevant to LPSD operations were also provided to the expert panel.

- Balanced – Information balances the needs from the experts in different technical areas involved in the study.

A large amount of background information was provided to the panel. The experts were not expected to thoroughly review all the documents. Experts may not have needed some reference materials. The experts did request additional information during the PIRT familiarization meetings. In particular, experts requested plant specific calculations of the time to reach boiling temperatures. A summary of these calculations were provided, but experts were reminded that they should consider this information along with their past experiences with LPSD PRAs. The PIRT team reminded the experts that a critical review of these calculations or other documents was not their goal. The experts were to use the background information as needed to supplement their own knowledge bases. The PIRT team provided a balanced set of information. They were responsive to the experts' requests, but also cautioned them to avoid bias and not be over-burdened by providing critical review comments.

- Usability – Information is readily accessible and searchable by the experts.

The background information was made accessible to the experts. Much of the information was supplied by the reference plant and was proprietary information. The PIRT team collected a non-disclosure agreement from each panel member to ensure the members understood that the information was only to be used in support of this PIRT process. All of the background information files were provided to the experts. The experts were able to review and refer to the information as needed throughout the process. The PIRT team highlighted the key documents and important information during familiarization meetings. This helped the experts focus on the most relevant information. The PIRT team also summarized the PIRT parameters and reference material in the final elicitation instructions to the experts. These instructions were formatted in a succinct and useable form.

E.4 Elicitation and Integration of Expert Judgments

The expert panel interacts to evaluate the information available, make interpretations, and form judgments. Inheriting from the formal SSHAC process (NUREG/CR-6372[2], the White Paper Guidance recommends three elicitation workshops, while recognizing the flexibility of having the workshops:

Workshop 1. The first workshop is focused on evaluating the data and models relevant to the technical issue. This workshop should also seek to elicit experts' experience, knowledge, and interpretations of the technical issue, while emphasizing the uncertainties, limitations, and caveats in the data and models.

Workshop 2. The second workshop is for proponent experts to make judgments of the technical issues based on structured interaction at this workshop and the outputs of Workshop 1.

Workshop 3. The third workshop is to evaluate the preliminarily integrated results and use the feedback to finalize the judgments.

The project utilized web-based process familiarization meetings and individual elicitation to fulfill Workshop 1. The LPSD PIRT had one face-to-face group meeting as to fulfill the functions of Workshop 2. The function of Workshop 3 was partially fulfilled with a discussion session at the end of the face-to-face meeting and the following-up email communications between the technical integrators and the expert panel.

E.4.1 PIRT process familiarization meetings

A series of web-based meetings were held to familiarize the panel members with the PIRT process. The panel members participated by telephone, and the PIRT team gave a presentation that was viewed via the web-meeting. The first two familiarization meetings focused on training on the elicitation process. These meetings included training on avoiding biases and simple examples of Analytical Hierarchy Process (AHP. The third meeting presented the PIRT parameters to the panel and requested feedback from the panel members. The experts discussed the technical issues and dataset in an interactive manner. Of the discussion topics were the effect of human actions in LPSD and the diversity of fire hazard. Two experts shared past experience and knowledge in these areas. Due to the limited time of the meetings, the dataset was not explored in depth for variabilities, uncertainties, and caveats in the relevant data and models. However, additional written comments were collected within two weeks after the web-meeting. The experts' comments were incorporated into the final PIRT parameters that were used for the elicitation. A final familiarization meeting was then held to describe the elicitation forms and explain how the PIRT parameters were to be evaluated during the individual elicitations.

E.4.2 Individual elicitation sessions

The individual elicitation consisted two parts: a facilitated web elicitation session for a portion of the technical issues followed by the expert's homework to complete the judgments of all the technical issues. The elicitation achieved the following Workshop 1 functions defined in the White Paper Guidance:

[2] Budnitz, R.J., et al. *Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts*, U.S. Nuclear Regulatory Commission NUREG/CR-6372, Livermore, CA, April 1997.

Understanding and interpreting the technical issues. The one-to-one interaction between the elicitation facilitator and the expert allowed the expert to think through the technical issues and better understand their context. However, due to the time limitation, the elicitation was only performed with a subset of the technical issues.

Exercise of elicitation. The facilitator walked the expert through each type of the worksheets. This provided practice for the expert to formally articulate his/her judgments as well as explicitly identify the associated assumptions, rationale, and factors contributing to uncertainties.

E.4.3 Group meeting

The face-to-face group meeting has three sections. The first morning of the meeting was devoted to experts' sharing their understanding of the technical issues and assessment of the available data, model, and evidence. The majority of the meeting was for the experts presenting and defending their evaluation of the technical issues. The last section of the meeting was for the panel to identify uncertainties in the technical issues and considerations in LPSD PRA modeling.

Our review of the meeting focuses on the following areas, referencing the White Paper Guidance about Workshop 2: achieving the objectives, structured process, interaction and facilitation, biases and mitigating strategies, good practices, and lessons learned.

E.4.3.1 Achieving the objectives

The meeting achieved its objectives with the expected outputs:

The experts made judgments about the technical issues as documented on the elicitation worksheets, along with experts' justification, reasoning, and uncertainty considerations.

The group meeting supported a systematic and integrated evaluation of the technical issues, incorporating:

- judgment;
- evidence, examples, and anchors supporting the judgment;
- boundary conditions within which the judgment is valid;
- exceptions and what-if consideration where the judgment does not apply; and
- uncertainties as well as assumptions made for the uncertainties.

Group discussions helped to improve the common understanding and interpretation of PIRT parameters.

The group meeting outcome represents the state of knowledge about the technical issues represented by the technical community.

The experts acknowledged several areas where consensus data and models to support LPSD PRA are lacking. This led to the experts creating a "parking lot" of technical issues that need more research in LPSD PRA modeling. Given the limited experience with developing LPSD PRAs in the U.S. nuclear industry, the need for additional research is a logical outcome of this meeting. This highlights the value of the LPSD PIRT process, as the ranking results may help guide which technical areas are most important for investing in additional research.

E.4.3.2 Structured process

The workshop began with the technical integrator and facilitator's briefing about the goals of the workshop, an explanation for the process and ground rules that would be followed, the roles of all participants. The following key roles of workshop participants were implemented throughout the meeting:

The technical integrator presented the workshop objectives and technical issues, facilitated discussion of the technical issues, fostered experts' challenging of others' judgments, and resolved technical disagreement.

The meeting facilitator ensured that the workshop procedures were followed, the experts used the worksheets as intended, and the ground rules of interaction were consistently enforced.

The experts presented and defended their evaluation of available data and initial judgments of the technical issues, and considered revising their judgments based on discussions. They also challenged other experts' judgments.

The peer reviewers observed the process and provided their comments to the technical integrator and facilitator. The comments were incorporated into the meeting process. The reviewers did not observe any significant process issues, thus the comments were minor.

E.4.3.3 Interaction and facilitation

The meeting was well facilitated. The expert panel evaluated the technical issues in an interactive and integrative manner through highly engaged and active discussion.

All the experts presented and were queried in a uniform manner; they were asked to provide specific answers to questions about the issues and the reasoning behind their responses.

The meeting was focused on elicitation of experts' judgments. Every expert had the opportunity to discuss his/her views without the pressure of reaching consensus with other experts' judgments.

The experts were very motivated for challenging others' judgment, sharing his/her expertise and knowledge, and defending his/her own points of view as well as others. The balanced composition of the expert panel fostered in-depth discussion of the technical issues from different perspectives (e.g., regulation, design, PRA, plant operation).

The experts were open to information or considerations brought up by other experts, and they considered and defended others' points of view.

Although, the experts considered others' opinions, there was no evidence that the group was pressured to converge on consensus results. There were only a few instances of significant outliers resulting from the individual elicitation ranking results. It is possible that those experts proposing outlier positions could feel pressure to change results to be more aligned with the other experts. However, after the group discussion of the results, those experts with significant outliers tended to uphold their original assessments.

E.4.3.4 Biases and mitigating strategies

The facilitator gave a brief biasing training at the beginning of the meeting, with emphasis on group biasing, anchoring bias, and potential social bias. Throughout the meeting, the reviewers did not observe systematic biases in the elicitation process. Individual experts may have had other types of cognitive biases (e.g., representational bias, confirmatory bias) that were not readily apparent to the reviewers.

E.4.3.5 Good practices

The reviewers observed some good practices contributing to a successful meeting:

- Preparation – The project team did a good preparation work for the meeting.

All the materials were distributed to the experts before the meeting. The materials were well organized for easy use. The meeting topics were careful thought about. For example, in addition to cover the technical issues, the meeting had a discussion session for experts to identify and discuss uncertainties and challenges in LPSD modeling. Potential pitfalls were identified and mitigating strategies were planned. The meeting logistics were well arranged.

- Facilitation – The meeting facilitator and technical integrator facilitated the meeting properly.

They fostered discussion asking probing questions as needed without interrupting the experts' thinking process. The experts were comfortable expressing their points of view without feeling pressed.

- Visualization – The project team used visualization aids to help the experts organize information, maintain situational awareness, and reduce their cognitive loads.

The large amount of the technical issues and the criteria in hierarchal quantitative approach are difficult for anyone to be cognizant of them all while working on an individual topic. The project team prepared large printed charts of the key definitions of the issues, criteria, and the relationships of criteria and sub-criteria. The charts were hung on the wall so that experts can easily view them to maintain the situational awareness.

It was a good practice that the facilitator and technical integrator used the parking lot to set aside questions that need further research. The parking lot also helped to set aside disputes on issue needing refinement and the scope of the issues.

E.4.3.6 Lessons learned

The reviewers observed some caveats of the meeting. These can be useful lessons to inform future practices of the PIRT method.

It was unclear to most experts how their judgments would be integrated. Although at the beginning of the meeting it was declared as a ground rule that the workshop was to develop community distribution instead of consensus, some experts were not clear how that worked. Toward the late part of the meeting, one expert was still unclear about the purpose of the meeting and questioned whether the meeting was to reduce the tail of the distribution. Some experts also expressed uncomfortableness for being an outliner. A good practice recommended in the White Paper

Guidance was that the technical integrator should demonstrate how the results will be integrated. The technical integrator should review all the key definitions and assumptions about the technical issues. This was not done systematically. The technical integrators assumed that those were already made clear in individual elicitation. In fact, the experts still had confusion about the definitions of some technical issues. The technical integrators led the panel discussion and redefined some issues.

Experts complained that sometimes they got lost with the hierarchal quantitative approach. They did not see how everything would get together at the end to address the technical question of the project. In additional to the training provided to the experts on the method, an upfront demonstration of how the method works with intuitive examples would have helped the experts to maintain a clear and consistent understanding of the method.

The White Paper Guidance states the “Independent intellectual Ownership” principle: “The expert panel must clearly understand that they are not representing their employer or organization on the panel, but are serving in their own right as a recognized leader in their respective field. Each expert should also maintain independence from the other experts in the team in order to avoid (or mitigate an organizational or groupthink bias risk.” This principle should have been better emphasized at the meeting. From time to time, there were some “small-talks” in the experts discussing topics related to the technical issues. Occasionally experts debated off-tracking topics from the perspectives of the organizations they represented. Although the reviewers did not observe systematic groupthink bias, such small-talks did not conform to the basic principle.

Time management during the meeting was a challenge. Although the project team carefully thought and planned time allocation to meeting topics, it proved that time management was still challenging. As the result, toward the later portion of the meeting the discussion for the technical issues were moderately compressed.

E.5 General Discussion and Summary

A general discussion of the overall impression of the LPSD PIRT process is provided below. The peer reviewers’ findings are presented in five key areas: the analytical hierarchy process, individual elicitation sessions, familiarization of the technical issues, training and piloting, and the effects of panel size. A final summary and conclusions from the participatory peer reviewers are also presented.

E.5.1 The analytical hierarchy process

The project team employed the Analytical Hierarchy Process (AHP for the expert panel to assess the technical issues. The AHP is a PIRT method intended to make the PIRT process consistent and transparent. Instead of having the experts directly ranking the technical issues, AHP has the experts pair-wisely compare the relative importance of a set of pre-defined criteria that contribute to assessment of the technical issues. Moreover, the experts were asked to define their own sub-criteria and compared the importance of sub-criteria that consist a criterion.

The AHP requires the experts to express their belief of relevant importance of the criteria/sub-criteria in given scales. The use of these comparison scales prompts some questions. How can one make sure that the experts are all on the same scale? Does “exceptionally more important” mean the same to all the experts? The experts discussed their own understanding of the scales. Through the discussion, the experts were more aggregable on the meanings of the scales. For example, if one choose “Factor X is exceptionally more important than Factor Y,” it means that the contribution of Factor Y to the technical issue is nearly negligible compared to that of Factor X. In other words, when one chooses “E” for a criterion, it means that one would evaluate the POSs highly based on that criterion while giving weak consideration on other criteria.

The experts felt that they needed to see the weights calculated from the assigned scales to assure themselves. Future training on the AHP could be better performed with more intuitive examples to demonstrate the meanings of the scales, i.e., how the various choices of comparison scales affect the relative weights of the criteria.

The experts were asked to define their own sub-criteria ranking categories. They defined the sub-criteria ranking categories from different perspectives. Collectively those perspectives made a better understanding of the technical issue compared to using a single-dimensional predefined sub-criteria or not probing sub-criteria at all.

The experts felt that they sometimes got lost in the criteria, sub-criteria, and comparative scales – it was not apparent how they were contributing to the technical issue being evaluated. In fact, it was not clear to the reviewers how the experts carried their AHP outcomes into their overall judgment of the technical issue. It was also unclear whether all the experts used their AHP outputs in a consistent manner.

Overall, it was a useful piloting of the AHP in a PIRT process. The reviewers observed some advantages of the approach. Yet, more studies are needed to demonstrate the pros and cons of the approach.

E.5.2 Individual elicitation sessions

Having individual elicitation sessions in lieu of a face-to-face Workshop 1 was mainly to compromise the resource demands of conducting workshops. Nevertheless, the reviewers observed two advantages of using the individual elicitation sessions. The individual elicitation sessions allowed experts to evaluate and interpret the available data/models, and the sessions assisted in effective time management during the group meeting.

Evaluation and interpretation of the available data/models. The experts could allocated adequate time on his/her own to work on all the technical issues. This allowed the expert to fully evaluate and interpret the available data/models and make judgments with thoughtful rationales.

Time management. The amount and complexity of the technical issues made it impossible for the experts to begin from scratch developing their judgments while still perform all the functions of Workshop 2 at the 3-day face-to-face group meeting. The initial assessment allowed the experts focus the face-to-face meeting on interacting, challenging each other, and obtaining the community understanding of the technical issues.

One potential shortcoming of having individual elicitation is that experts made their initial judgment without interactively evaluating or interpreting the available data, models, and evidence. For example, the project team realized that the understanding of fire hazard and its impacts on plant safety varies greatly among the experts; it would be better to have the fire hazard experts to share their knowledge and evaluation with the rest of the panel before starting individual elicitation. Theoretically, this shortcoming can be made up through the face-to-face meeting, where the experts present their judgments and challenge each other. Indeed, several experts indicated that they would modify their initial judgments about fire hazard based on the meeting discussion.

However, the reviewers are unable to assess to what extent that the face-to-face meeting compensated the lack of interaction in the individual elicitation.

E.5.3 Familiarization of the technical issues

Familiarization of the technical issues is to ensure that all the experts have a clear, precise, and thorough understanding of the technical issues. The project team familiarized the expert panel on the technical issues through two web meetings. The experts asked questions and made recommendations on refining the technical issues. In addition, the project team further interacted with the experts on understanding the technical issues during individual elicitation sessions. The review panel considers that the experts achieved consistent understanding of the issues through the familiarization process.

The familiarization web-meeting was conducted in a tutorial way, i.e., the facilitators explained the technical issues to the experts. Due to the limited time available for the meeting, strategies such as probing the experts' mental models of the technical issues were not used. It appeared that although the experts understood the issues, the boundary conditions or assumptions of some issues were still confusing to the experts. The group meeting clarified the confusing issues and refined the definition of several POSs.

E.5.4 Training and piloting

Training and piloting are essential before the elicitation. The project team conducted four web-based training sessions on the following areas:

1) Familiarizing the subject matter (including the necessary background information on why the elicitation was being performed and how the results will be used) and the technical problems being asked;
2) Familiarizing the basic principles of elicitation and the elicitation process, including the analytical hierarchy approach;
3) Educating on possible biases that could be present and influence the judgments; and
4) Familiarizing the worksheets of the individual elicitations.

The review panel considers that the training achieved its goals of item 1), 3), and 4) above. Yet, the training on the analytical hierarchy approach was inadequate. It was not apparent to the experts the meanings of different scales of criterion importance comparison, and how the scales impact their final ranking of the technical issues. This shortcoming was overcome through interaction and discussion during the group meeting. Some experts re-worked on their forms after the group meeting because of their misunderstanding about the approach. One reason contributing to this shortcoming was that the training was primarily tutorial rather than interactive. The experts asked questions during the training sessions but they were not provided with opportunities of exercises or practices. This was in part due to the large number of comparisons required in the forms. Much of the time during the individual elicitation sessions was used explaining and orienting the experts to the forms, leaving little time available for exercise or practice.

The project team also performed a piloting of the individual elicitation. The piloting was to evaluate feasibility, time, cost, and potentially adverse events so that the project team could improve the design of the face-to-face meeting prior to performance of a full-scale project. The piloting led the project team to refine several technical issues, the elicitation instructions, and mostly, the elicitation forms. The piloting proved to be very helpful. Yet, due to the schedule and resource limitations, the piloting was conducted on one “mock” expert. Whenever it is possible, the piloting should be performed with a small subset of the experts from the formed expert panel.

E.5.5 The effects of the panel size

The panel has seven experts, well balanced and complimentary in their areas of expertise. On the other hand, this relatively larger size of experts lead to mental fatigue. The expert took turns to present and defend their evaluation. Reviewers observed that some experts started to lose attention after four or five presentations, especially for complex issues.

E.5.6 Final summary and conclusions

In summary, the review panel considers that the LPSD PIRT successfully achieves the project goals. Technically, the PIRT achieved the goal of ranking plant operating states according to their importance to LPSD risk. The ranking was performed with consideration for different plant outage types, two different risk metrics (i.e., core damage and radiological release), and four different hazard categories (i.e., internal events, fire events, internal flooding events, and seismic events.) The expert panel contributed to identifying and defining the parameters that were considered in the ranking process. The LPSD PIRT employed a formal expert elicitation process; the process complies the principles and guidelines in the NRC’s White Paper Guidance. The use of a formal expert solicitation provides good regulatory assurance of the results. Valuable good practices and lessons learned from the process can inform future work.

APPENDIX F FORMS USED IN INDIVIDUAL LPSD PRA PIRT ELICITATION MEETINGS

The PIRT facilitators presented to the expert panel the LPSD PRA PIRT elicitation forms that would be used in the individual remote elicitation meetings during an online meeting held on January 31, 2017. The purpose of the presentation was to accomplish the following:

- Familiarize the expert panel members with the LPSD PRA PIRT elicitation forms.
- Explain to experts the purpose of each form and the information that was intended to be elicited.
- Answer any questions from the experts about the purpose or intent of the forms.

The LPSD PRA PIRT elicitation forms are presented in this appendix.

Note: The PIRT expert panel members were presented with additional plant specific information used for defining the POS details. The version of the Plant Operating State (POS) Importance Ranking Elicitation Form (PR1) table, as presented in this report, omits some of the POS details that were used in the actual expert elicitation forms.

TLC1

Internal Events Core Damage Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Justification for Evaluation
RCS Inventory Control is more important than **Heat Removal**		
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**		
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Internal Events Hazard**		
Internal Events Hazard is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Internal Events Hazard**		
Internal Events Hazard is more important than **Heat Removal**		
RCS Integrity is more important than **Internal Events Hazard**		
Internal Events Hazard is more important than **RCS Integrity**		

TLC2

Internal Events Release Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**		
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**		
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Internal Events Hazard**		
Internal Events Hazard is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Containment Performance**		
Containment Performance is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Internal Events Hazard**		
Internal Events Hazard is more important than **Heat Removal**		
Heat Removal is more important than **Containment Performance**		
Containment Performance is more important than **Heat Removal**		
RCS Integrity is more important than **Internal Events Hazard**		
Internal Events Hazard is more important than **RCS Integrity**		
RCS Integrity is more important than **Containment Performance**		
Containment Performance is more important than **RCS Integrity**		
Internal Events Hazard is more important than **Containment Performance**		
Containment Performance is more important than **Internal Events Hazard**		

TLC3

Fire Events Core Damage Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**		
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**		
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Fire Hazard**		
Fire Hazard is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Fire Hazard**		
Fire Hazard is more important than **Heat Removal**		
RCS Integrity is more important than **Fire Hazard**		
Fire Hazard is more important than **RCS Integrity**		

TLC4

Fire Events Release Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**		
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**		
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Fire Hazard**		
Fire Hazard is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Containment Performance**		
Containment Performance is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Fire Hazard**		
Fire Hazard is more important than **Heat Removal**		
Heat Removal is more important than **Containment Performance**		
Containment Performance is more important than **Heat Removal**		
RCS Integrity is more important than **Fire Hazard**		
Fire Hazard is more important than **RCS Integrity**		
RCS Integrity is more important than **Containment Performance**		
Containment Performance is more important than **RCS Integrity**		
Fire Hazard is more important than **Containment Performance**		
Containment Performance is more important than **Fire Hazard**		

TLC5

Internal Flooding Events Core Damage Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**		
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**		
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Internal Flooding Hazard**		
Internal Flooding Hazard is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Internal Flooding Hazard**		
Internal Flooding Hazard is more important than **Heat Removal**		
RCS Integrity is more important than **Internal Flooding Hazard**		
Internal Flooding Hazard is more important than **RCS Integrity**		

TLC6

Internal Flooding Events Release Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**		
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**		
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Internal Flooding Hazard**		
Internal Flooding Hazard is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Containment Performance**		
Containment Performance is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Internal Flooding Hazard**		
Internal Flooding Hazard is more important than **Heat Removal**		
Heat Removal is more important than **Containment Performance**		
Containment Performance is more important than **Heat Removal**		
RCS Integrity is more important than **Internal Flooding Hazard**		
Internal Flooding Hazard is more important than **RCS Integrity**		
RCS Integrity is more important than **Containment Performance**		
Containment Performance is more important than **RCS Integrity**		
Internal Flooding Hazard is more important than **Containment Performance**		
Containment Performance is more important than **Internal Flooding Hazard**		

TLC7

Seismic Events Core Damage Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**		
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**		
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Seismic Hazard**		
Seismic Hazard is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Seismic Hazard**		
Seismic Hazard is more important than **Heat Removal**		
RCS Integrity is more important than **Seismic Hazard**		
Seismic Hazard is more important than **RCS Integrity**		

TLC8

Seismic Events Release Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**		
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**		
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Seismic Hazard**		
Seismic Hazard is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Containment Performance**		
Containment Performance is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Seismic Hazard**		
Seismic Hazard is more important than **Heat Removal**		
Heat Removal is more important than **Containment Performance**		
Containment Performance is more important than **Heat Removal**		
RCS Integrity is more important than **Seismic Hazard**		
Seismic Hazard is more important than **RCS Integrity**		
RCS Integrity is more important than **Containment Performance**		
Containment Performance is more important than **RCS Integrity**		
Seismic Hazard is more important than **Containment Performance**		
Containment Performance is more important than **Seismic Hazard**		

SC1

Sub-Criteria Elicitation Form

Top-Level Criteria	Pair-wise Comparison of Two Sub-Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise sub-criteria comparison]	Expert Comments
RCS Inventory Control	**RCS Water Level** is more important than **Availability of Systems to Make-up Inventory**		
	Availability of Systems to Make-up Inventory is more important than **RCS Water Level**		
Heat Removal	**Heat Load** is more important than **Availability of Reactor Cooling Systems**		
	Availability of Reactor Cooling Systems is more important than **Heat Load**		
RCS Integrity	**RCS Isolation** is more important than **RCS Pressure Relief Capability**		
	RCS Pressure Relief Capability is more important than **RCS Isolation**		
Internal Events Hazard	**Human Initiated Errors** is more important than **Important Equipment Failures**		
	Important Equipment Failures is more important than **Human Initiated Errors**		
Containment Performance	**Containment Isolation Capability** is more important than **Availability of Radionuclide Suppression Systems**		
	Availability of Radionuclide Suppression Systems is more important than **Containment Isolation Capability**		
Fire Hazard	**Fire Frequency** is more important than **Vulnerability to Fire Damage**		
	Vulnerability to Fire Damage is more important than **Fire Frequency**		
Internal Flooding Hazard	**Internal Flooding Frequency** is more important than **Vulnerability to Internal Flooding Damage**		
	Vulnerability to Internal Flooding Damage is more important than **Internal Flooding Frequency**		
Seismic Hazard	**Seismic Frequency** is more important than **Vulnerability to Seismic Damage**		
	Vulnerability to Seismic Damage is more important than **Seismic Frequency**		

SCR1

RCS Water Level Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the RCS water inventory level?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Low Water Level is more important than **Medium Water Level**		
Low Water Level is more important than **High Water Level**		
Medium Water Level is more important than **High Water Level**		

Ranking Category	Category Definition by Expert
Low Water Level	
Medium Water Level	
High Water Level	

SCR2

Availability of Systems to Make-up Inventory Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the availability of systems to keep the core covered?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Not Very Available is more important than **Available**		
Not Very Available is more important than **Very Available**		
Available is more important than **Very Available**		

Ranking Category	Category Definition by Expert
Not Very Available	
Available	
Very Available	

SCR3

Heat Load Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the heat load from power, decay heat, and RCS temperature?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Load is more important than **Medium Load**		
High Load is more important than **Low Load**		
Medium Load is more important than **Low Load**		

Ranking Category	Category Definition by Expert
High Load	
Medium Load	
Low Load	

SCR4

Availability of Reactor Cooling Systems Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the availability of reactor cooling systems?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Not Very Available is more important than **Available**		
Not Very Available is more important than **Very Available**		
Available is more important than **Very Available**		

Ranking Category	Category Definition by Expert
Not Very Available	
Available	
Very Available	

SCR5

RCS Isolation Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the level of vulnerability to maintaining RCS loop isolation?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**		
Very Vulnerable is more important than **Normal Vulnerability**		
Vulnerable is more important than **Normal Vulnerability**		

Ranking Category	Category Definition by Expert
Very Vulnerable	
Vulnerable	
Normal Vulnerability	

SCR6

RCS Pressure Relief Capability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the level of vulnerability to over-pressurization of the RCS?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**		
Very Vulnerable is more important than **Normal Vulnerability**		
Vulnerable is more important than **Normal Vulnerability**		

Ranking Category	Category Definition by Expert
Very Vulnerable	
Vulnerable	
Normal Vulnerability	

SCR7

Human Initiated Errors Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the level of opportunity for accident sequences initiated by human errors?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Opportunity for Errors is more important than **Medium Opportunity for Errors**		
High Opportunity for Errors is more important than **Low Opportunity for Errors**		
Medium Opportunity for Errors is more important than **Low Opportunity for Errors**		

Ranking Category	Category Definition by Expert
High Opportunity for Errors	
Moderate Opportunity for Errors	
Low Opportunity for Errors	

SCR8

Important Equipment Failures Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the level of opportunity for accident sequences initiated by equipment failures?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Opportunity for Failures is more important than **Medium Opportunity for Failures**		
High Opportunity for Failures is more important than **Low Opportunity for Failures**		
Medium Opportunity for Failures is more important than **Low Opportunity for Failures**		

Ranking Category	Category Definition by Expert
High Opportunity for Failures	
Moderate Opportunity for Failures	
Low Opportunity for Failures	

SCR9

Containment Isolation Capability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the time required to close containment versus time available?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Short Time is more important than **Moderate Time**		
Short Time is more important than **Long Time**		
Moderate Time is more important than **Long Time**		

Ranking Category	Category Definition by Expert
Short Time	
Moderate Time	
Long Time	

SCR10

Availability of Radionuclide Suppression Systems Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the availability of radionuclide suppression systems?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Not Very Available is more important than **Available**		
Not Very Available is more important than **Very Available**		
Available is more important than **Very Available**		

Ranking Category	Category Definition by Expert
Not Very Available	
Available	
Very Available	

SCR11

Fire Frequency Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the fire frequency for this plant configurations and set of activities?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Frequency is more important than **Medium Frequency**		
High Frequency is more important than **Low Frequency**		
Medium Frequency is more important than **Low Frequency**		

Ranking Category	Category Definition by Expert
High Frequency	
Moderate Frequency	
Low Frequency	

SCR12

Fire Damage Vulnerability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the chance that fire damage initiates an accident sequence?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**		
Very Vulnerable is more important than **Normal Vulnerability**		
Vulnerable is more important than **Normal Vulnerability**		

Ranking Category	Category Definition by Expert
Very Vulnerable	
Vulnerable	
Normal Vulnerability	

SCR13

Internal Flooding Frequency Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the internal flooding event frequency for this plant configurations and set of activities?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Frequency is more important than **Medium Frequency**		
High Frequency is more important than **Low Frequency**		
Medium Frequency is more important than **Low Frequency**		

Ranking Category	Category Definition by Expert
High Frequency	
Moderate Frequency	
Low Frequency	

SCR14

Internal Flooding Damage Vulnerability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the chance that internal flooding damage initiates an accident sequence?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**		
Very Vulnerable is more important than **Normal Vulnerability**		
Vulnerable is more important than **Normal Vulnerability**		

Ranking Category	Category Definition by Expert
Very Vulnerable	
Vulnerable	
Normal Vulnerability	

SCR15

Seismic Frequency Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the seismic event frequency for this plant?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Frequency is more important than **Medium Frequency**		
High Frequency is more important than **Low Frequency**		
Medium Frequency is more important than **Low Frequency**		

Ranking Category	Category Definition by Expert
High Frequency	
Moderate Frequency	
Low Frequency	

SCR16

Seismic Damage Vulnerability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the chance that seismic damage initiates an accident sequence?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**		
Very Vulnerable is more important than **Normal Vulnerability**		
Vulnerable is more important than **Normal Vulnerability**		

Ranking Category	Category Definition by Expert
Very Vulnerable	
Vulnerable	
Normal Vulnerability	

Plant Operating State (POS) Importance Ranking Elicitation Form (PR1)

POS		Technical Specification (TS)	POS Duration (hours)	Evaluation Criteria Rank															
No.	Description	TS Mode	Range over POTs	*Heat Load*	*Availability of Systems to Make-up RCS Inventory*	*RCS Water Level*	*Availability of Reactor Cooling Systems*	*RCS Isolation*	*RCS Pressure Relief Capability*	*Human Initiated Errors*	*Important Equipment Failures*	*Containment Isolation Capability*	*Availability of Radionuclide Suppression Systems*	*Fire Frequency*	*Vulnerability to Fire Damage*	*Internal Flooding Frequency*	*Vulnerability to Internal Flooding Damage*	*Seismic Frequency*	*Vulnerability to Seismic Damage*
				SCR3	*SCR2*	*SCR1*	*SCR4*	*SCR5*	*SCR6*	*SCR7*	*SCR8*	*SCR9*	*SCR10*	*SCR11*	*SCR12*	*SCR13*	*SCR14*	*SCR15*	*SCR16*
				High Load (H) Medium Load (M) Low Load (L)	*Not Very Available (H) Available (M) Very Available (L)*	*Low Water Level (H) Medium Water Level (M) High Water Level (L)*	*Not Very Available (H) Available (M) Very Available (L)*	*Not Very Available (H) Available (M) Very Available (L)*	*Very Vulnerable (H) Vulnerable (M) Normal (L)*	*High Opportunity (H) Moderate Opportunity (M) Low Opportunity (L)*	*High Opportunity (H) Moderate Opportunity (M) Low Opportunity (L)*	*Short Time (H) Moderate Time (M) Long Time (L)*	*Not Very Available (H) Available (M) Very Available (L)*	*High Frequency (H) Moderate Frequency (M) Low Frequency (L)*	*Very Vulnerable (H) Vulnerable (M) Normal (L)*	*High Frequency (H) Moderate Frequency (M) Low Frequency (L)*	*Very Vulnerable (H) Vulnerable (M) Normal (L)*	*High Frequency (H) Moderate Frequency (M) Low Frequency (L)*	*Very Vulnerable (H) Vulnerable (M) Normal (L)*
1	Low power and reactor shutdown	1,2	6																
			Expert Comments:																
2	Cooldown with steam generators to 350°F	3	5																
			Expert Comments:																
2-P1	Cooldown with steam generators to 350°F	3	16																
			Expert Comments:																
3	Cooldown with RHR system to 200°F	4	3																
			Expert Comments:																
4	Cooldown to ambient temperature with RHR only	5	4																
			Expert Comments:																
4-P2	Cooldown to ambient temperature with RHR only	5	118																
			Expert Comments:																
5A	Pressurizer water solid for degassing	5	20-30																
			Expert Comments:																
5B	Draining the RCS to reduce inventory, RCS is vented	5	20-40																
			Expert Comments:																
6	Mid-loop operation prior to refueling	5	14-20																
			Expert Comments:																
6-P3	Mid-loop operations	5	greater than 20																
			Expert Comments:																
7	Filling refueling cavity for refueling operation	6	45																
			Expert Comments:																
8E	Refueling operation (offloading old core)	6	68																
			Expert Comments:																
DF	Defueled	n/a	120																
8L	Refueling operation (loading new core)	6	52																
			Expert Comments:																
9	Draining the RCS refueling operation	6	120																
			Expert Comments:																
10	Mid-loop operation after refueling	5,6	25																
			Expert Comments:																
11	Refill RCS, reactor vents are closed	5,6	20-24																
			Expert Comments:																
12	RCS heatup/draw bubble in pressurizer	5	48-54																
			Expert Comments:																
13	RCS heatup to 350°F	4	20																
			Expert Comments:																
14	Startup with steam generators to Hot Standby	3	40																
			Expert Comments:																
15A	Reactor startup and low power operation (0=Power<5%)	2	20																
			Expert Comments:																
15B	Reactor startup and low power operation (5<Power<50%)	1,2	20-36																
			Expert Comments:																

LOK1

Self Evaluation of Level of Knowledge

Form	Description of Form	Level (H, M, L)	Comments
TCL1	Comparison of TLC for Internal Events Core Damage	H	This risk is generally understood, although there are aspects that have not been fully investigated.
TCL2	Comparison of TLC for Internal Events Core Release	H	This risk is generally understood, although there are aspects that have not been fully investigated. This risk has the benefit of both the decay heat and source term decreasing with increasing time after shutdown, so the back-end of long outages tend to be no be so important to risk.
TCL3	Comparison of TLC for Fire Core Damage	M	Fire risk at LPSD is the intersection of two complicated analyses. I am familiar with fire risk at power and somewhat with fire frequency and fire protection controls at shutdown, so it is possible to make some judgments about fire risk at shutdown.
TCL4	Comparison of TLC for Fire Core Release	M	
TCL5	Comparison of TLC for Internal Flooding Core Damage	M	Flood risk is somewhat less complicated than fire and the at-power flood risk can give a good picture of flood risk at shutdown. However, we don't have a good handle on the increase in frequency that may occur in some POSs and the potential for flood barriers to be removed in a way that would not allow prompt reinstallation (e.g., a door held open).
TCL6	Comparison of TLC for Internal Flooding Core Release	M	
TCL7	Comparison of TLC for Seismic Events Core Damage	M	Seismic risk is expected to be not significant because of (a) the short duration of most POSs and (b) the Seismic-Cat 1 equipment that is performing a function during shutdown. However, there are short-duration events (e.g., reactor head on the polar crane) where the conditional risk may be higher. Also, the change in tank levels may change the tank fragility analysis.
TCL8	Comparison of TLC for Sismic Events Core Release	M	
SCL1	Sub-Criteria Elicitaion Form	H	
SCR1	RCS Water Level Ranking	H	
SCR2	Availability of Systems to Make-up Inventory Ranking	H	
SCR3	Heat Load Ranking	H	
SCR4	Availability of Reactor Cooling Systems Ranking	H	
SCR5	RCS Isolation Ranking	H	
SCR6	Pressure Relief Capability Ranking	H	
SCR7	Containment Isolation Capability	H	
SCL8	Availability of Radionuclide Suppression Systems Ranking	H	
SCR9	Operator Initiated Events	H	
SCR10	Important Equipment Failures	H	
SCR11	Fire Frequency	M	
SCR12	Fire Damage Vulnerability	M	
SCR13	Internal Flooding Event Frequency	M	
SCR14	Internal Flooding Damage Vulnerability	M	
SCR15	Seismic Event Frequency	M	
SCR16	Seismic Damage Vulnerability	M	
PR1	POS Importance Ranking Elicitation Form	H	

APPENDIX G PIRT PARAMETERS AFTER SOLICITATION OF INPUT FROM EXPERT PANEL MEMBERS

Based by solicitation of input from the expert member members received during the week of January 16, 2017, the final Phenomena Identification and Ranking table (PIRT) parameters to be used in the Low Power Shutdown (LPSD) Probabilistic Risk Assessment (PRA) PIRT exercise consist of the following PIRT parameters groups:

1) Plant Outage Types (POTs) along with their definitions
2) Plant Operating States (POSs) along with their definitions
3) Hazardous events to evaluated in the PIRT exercise
4) Evaluation criteria to be used in the PIRT Analytical Hierarchy Process (AHP) and associated Ranking Categories

The PIRT parameters are identified below.

Plant Outage Types

The POTs are the four types identified and described in the preparation of the PIRT detailed problem statement. The four planned outage types presented in Attachment 1. A table that identifies the POS that comprise the POT are presented in Attachment 2.

Plant Operating States

The POSs proposed to be addressed in the PIRT exercise are the 21 POS identified and described in the preparation of the PIRT detailed problem statement. During the development of the POS descriptions, minor revisions were implemented based on amendments by PNNL and the expert panel.

Hazard Types

The hazards proposed to be addressed in the PIRT exercise are:

1) Internal events
2) Internal flooding
3) Internal fire
4) Seismic events

Evaluation Criteria

The Evaluation Criteria (top-level criteria and sub-criteria), are presented in Figures G-1 and G-2 for importance to CDF and release respectively. The Evaluation Questions associated with each criterion and corresponding Ranking Categories are shown in Table G-1. Criteria associated with fire, internal flooding and seismic events are shown in Table 1 but shown only for internal events in Figures G-1 and G-2.

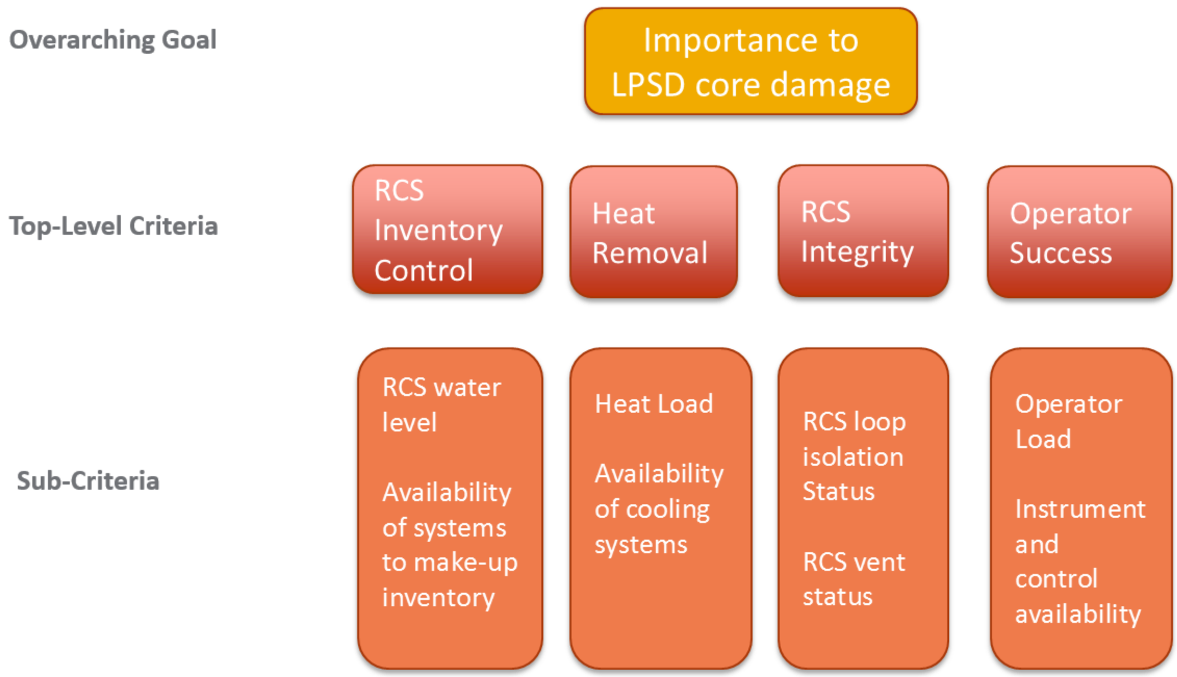


Figure G-1 Proposed set of Top-Level Evaluation Criteria and Sub-criteria for Importance to LPSD Core Damage

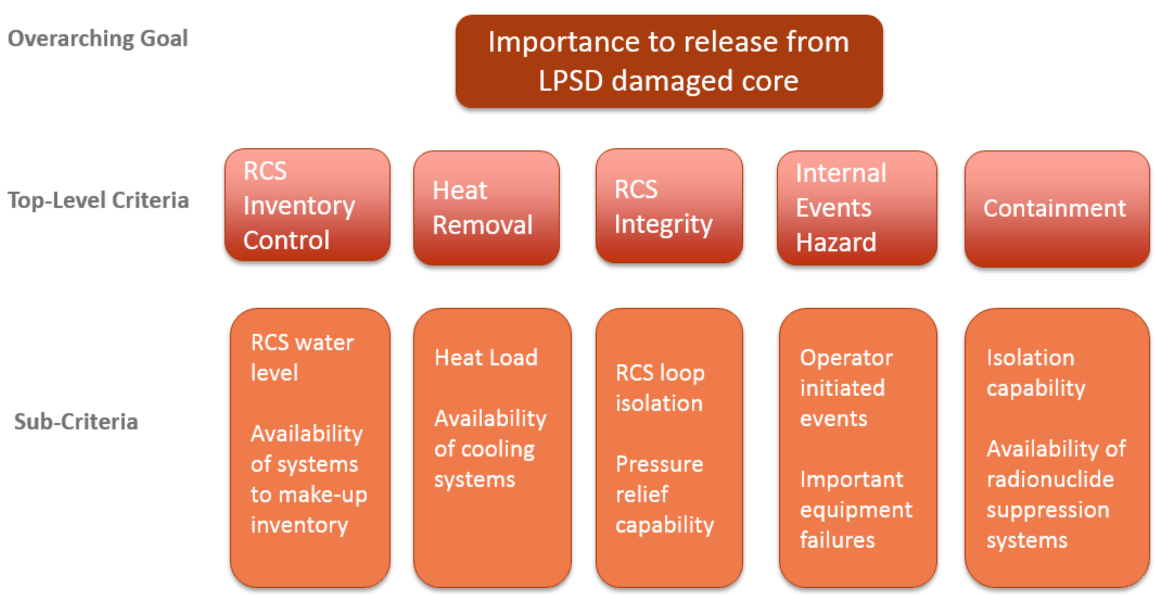


Figure G-2 Proposed set of Top-Level Evaluation Criteria and Sub-criteria for Importance to LPSD Release from the Containment

Table G-1 Evaluation Criteria for Importance of Core Damage and Release from Damaged Core

TOP-LEVEL CRITERIA	SUB-CRITERIA	EVALUATION QUESTIONS	RANKING CATEGORIES	CONSIDERATIONS
RCS Inventory Control				
	RCS water level	What is the RCS water inventory level?	Low Medium High	During mid-loop operations the coolant is drained to its lowest level.
	Availability of systems to make-up inventory	What is the availability of systems to keep the core covered?	Not Very Available Available Very Available	Systems to keep the core covered include charging pumps, ECCS or gravity feed from the RWST.
Heat Removal				
	Heat load	What is the heat load from power, decay heat, and RCS temperature?	High Medium Low	Power is being produced for the low power POSs. Decay heat load is a function of time since reactor shutdown and as such an attribute of each POS.
	Availability of reactor cooling systems	What is the availability of reactor cooling systems?	Not Very Available Available Very Available	Includes the number of RHR trains available and whether SG cooling is functional.
RCS Integrity				
	RCS isolation	What is the level of vulnerability to maintaining RCS isolation?	Very Vulnerable Vulnerable Normal	Challenges to RCS isolation include: isolation the presence of nozzle dams, low pressure seals such as instrument tube seals, RCP shutdown seals, maintenance activities that could drain the primary inventory, and pressurizer manways and SG manways,
	Pressure relief capability	What is the level of vulnerability to over-pressurization of the RCS?	Very Vulnerable Vulnerable Not Vulnerable	Relief capability consideration include PORVs, overpressure events through the RHR relief valves, whether head vents are open, and water addition when the RCS is water solid.
Containment Performance				
	Containment isolation capability	What is the time required to close containment versus time available?	Short Moderate Long	The time-to-boiling is an important consideration if it is sooner than the containment can be closed. Time to boiling is a function of heat load so is not repeated under this criterion.
			Not Very Available	

Table G-1 Evaluation Criteria for Importance of Core Damage and Release from Damaged Core (continued)

TOP-LEVEL CRITERIA	SUB-CRITERIA	EVALUATION QUESTIONS	RANKING CATEGORIES	CONSIDERATIONS
	Availability of radionuclide suppression systems	What is the availability of radionuclide suppression systems?	Available Very Available	Radionuclide suppression includes sprays and filters.
Internal Events Hazard*				
	Operator initiated events	What is the level of opportunity for accident sequences initiated by operator errors?	High Moderate Low	Considerations for this criterion include operator load, stress, and distractions, number of required actions, as well as maintenance and other activities not directly related to transitioning between POSs. Considerations also include duration of the POSs and the availability of valid instrumentation and control.
	Important equipment failures	What is the level of opportunity for accident sequences initiated by equipment failures?	High Moderate Low	This criteria concerns the likelihood of initiating events associated with equipment failure. It does not concern the vulnerability of plant configuration due to unavailable systems. Unavailability of systems is considered under other top-level criterion.
Fire Hazard*				
	Fire frequency	What is the fire frequency for this plant configurations and set of activities?	High Frequency Moderate Frequency Low Frequency	Contributors to fire frequency include the level of operational and maintenance activity and level of combustible material loading during LPSD and the duration of the POS.
	Fire damage vulnerability	What is the chance that fire damage initiates an accident sequence?	Very Vulnerable Vulnerable Normal	This criteria concerns the likelihood that fires initiate accident sequences. It does not concern the vulnerability of plant configuration due to unavailable systems. Unavailability of systems is considered under other top-level criterion.
Internal Flooding Hazard*				
	Internal flooding event frequency	What is the internal flooding event frequency for this plant configurations and set of activities?	High Frequency Moderate Frequency Low Frequency	Contributors to internal flooding event frequency include the large number of actions associated with transitioning from one POS to another and maintenance activities. Considerations also include duration of the POSs.
	Internal flooding damage vulnerability	What is the chance that internal flooding damage	Very Vulnerable Vulnerable	This criteria concerns the likelihood that flooding events initiate accident sequences. It does not

Table G-1 Evaluation Criteria for Importance of Core Damage and Release from Damaged Core (continued)

LEVEL	SUB-CRITERIA	EVALUATION QUESTIONS	RANKING CATEGORIES	CONSIDERATIONS
		initiates an accident sequence?	Normal	concern the vulnerability of plant configuration due to unavailable systems. Unavailability of systems is considered under other top-level criterion. The status of features designed to minimize of impact of flooding could vary from full-power.
Seismic Event Hazard*				
	Seismic event frequency	What is the seismic event frequency for this plant?	High Frequency	The ranking should be the same for all POSs. Considerations also include duration of the POSs.
			Moderate Frequency	
			Low Frequency	
	Seismic damage vulnerability	What is the chance that a seismic damage initiates an accident sequence?	Very Vulnerable	This criteria concerns the vulnerability of the plant configuration to a seismic event (i.e., fragility). It does not concern the vulnerability of plant configuration due to unavailable systems. Unavailability of systems is considered under other top-level criterion.
			Vulnerable	
			Normal	

Note:
*The internal events, fire, internal flooding, and seismic event top criteria will be compared separately with the other top level criteria.

ATTACHMENT 1

PLANT OUTAGE TYPES

POT 1: Non-Drained Maintenance without the Use of RHR.

POT 1 corresponds to a maintenance outage in hot standby; i.e., RCS temperature above 350° F. The reactor is made subcritical, and decay heat is removed by use of the feedwater and steam generators. This plant state will be reached if short outage times are expected, such as due to inadvertent reactor trips, or if there are maintenance activities to be performed that do not require RCS cooldown.

POT 2: Non-Drained Maintenance with the Use of RHR.

POT 2 corresponds to a maintenance outage that requires the plant to go to Mode 4 (hot shutdown) or Mode 5 (cold shutdown) without RCS draining. The reactor is made subcritical, and the RCS is cooled to below 350°F. Decay heat is removed by use of the residual heat removal system aligned in the shutdown cooling mode. This plant state will be reached if longer outage times are expected, if heat removal with secondary side is not available, or if there are maintenance activities to be performed that require cooldown.

POT 3: Drained Maintenance with the Use of RHR

POT 3 corresponds to a maintenance outage in Cold Shutdown with RCS level drained to reduced inventory operation. Decay heat is removed by the residual heat removal system aligned in the shutdown cooling mode. The reactor vessel head is in place with fuel in the reactor pressure vessel. This state is entered if maintenance requires a low level of the reactor coolant system, e.g., reactor coolant pump seal maintenance or steam generator tube leakage.

POT 4: Refueling

POT 4 corresponds to a refueling outage with fuel offloaded to the spent fuel pool. During refueling outages many maintenance activities and surveillance tests are performed. While fuel assemblies remain in the reactor vessel, the decay heat is removed by the residual hear removal system aligned in the shutdown cooling mode.

ATTACHMENT 2

PLANT OUTAGE TYPE AND PLANT OPERATING STATE SUMMARY TABLE

POS		TS	POS Applicable when Transitioning to Outage Type			
No.	Description	TS Mode	Refueling (POT-4)	Hot Standby (POT-1)	Maintenance w/o Drain (POT-2)	Maintenance w/Drain (POT-3)
1	Low power and reactor shutdown	1,2	X	N/A	X	X
2	Cooldown with steam generators to 350 °F	3	X		X	X
2-P1	Cooldown with steam generators to 350 °F	3	X	X		
3	Cooldown with residual heat removal system to 200 °F	4	X		X	X
4	Cooldown to ambient temperature with residual heat removal system only	5	X			X
4-P2	Cooldown to ambient temperature with residual heat removal system only	5			X	
5A	Pressurizer water solid for degassing	5	X			X
5B	Draining the reactor coolant system to reduced inventory, RCS is vented	5,6	X			X
6	Mid-loop operation prior to refueling	5,6	X			X
7	Filling refueling cavity for refueling operation	6	X			
8E	Refueling operation (offloading old core)	6	X			
DF[1]	Defueled	n/a	X			
8L	Refueling operation (loading new core)	6	X			
9	Draining the reactor coolant system after refueling operation	6	X			
10	Mid-loop operation after refueling	5,6	X			
11	Refill reactor coolant system, reactor vents are closed	5,6	X			X

PLANT OUTAGE TYPE AND PLANT OPERATING STATE SUMMARY TABLE (continued)

POS		TS	POS Applicable when Transitioning to Outage Type			
No.	**Description**	**TS Mode**	**Refueling (POT-4)**	**Hot Standby (POT-1)**	**Maintenance w/o Drain (POT-2)**	**Maintenance w/Drain (POT-3)**
12	Reactor coolant system heatup/draw bubble in pressurizer	5	X			X
13	Reactor coolant system heatup to 350 °F	4	X		X	X
14	Startup with steam generators to Hot Standby	3	X		X	X
15A	Reactor startup and low power operation (0=Power<5%)	2	X	X	X	X
15B	Reactor startup and low power operation (5<Power<50%)	1,2	X	X	X	X

[1]POS DF is provided in this table for completeness but is not evaluated by the PIRT panel since the reactor is defueled.

Note: The PIRT expert panel members were presented with additional plant specific information used for defining the POS details. The table, as presented in this report, omits some of the POS details that were used in the expert elicitation forms.

APPENDIX H INSTRUCTIONS FOR FILLING OUT LPSD PRA PIRT ELICITATION FORMS

PIRT Elicitation Instructions

The PIRT elicitation instructions are organized into general instruction that apply to all the elicitation forms and a set of specific instructions for each elicitation form.

General Instructions

The following are general instructions and for filling out the PIRT elicitation forms and reminders about pitfalls to avoid such as those that could lead to bias.

1. Avoid bias that might be caused by a conflict of interest by providing your best objective technical judgment. Remember that you are experts in Low Power Shutdown (LPSD) PRA and the challenges and methods associated with performing LPSD PRA and as such you represent a the community of experts.
2. Avoid misinterpretation by asking for clarification if there is confusion about what is being elicited during the online interview or as you are filling out the forms later.
3. Avoid other motivational and cognitive biases discussed in the first familiarization meeting such as being overly influenced by recent events or social pressure to respond in a particular way.
4. In forms in which you are providing pairwise comparisons of one attribute to another attribute (e.g., How does A compare to B), the comparisons only needs to be performed once, because the reciprocal relationship is assumed for the reciprocal comparison (i.e., How does B compare to A?).
5. When using the Comparison Scale to compare one attribute to another ask yourself why the Comparison Category just above or below the one you have chosen might not be more appropriate.
6. To the extent possible, provide consistent responses and avoid inconsistent responses. For example, if A is judged to be greater than B, and B is judged to be greater than C, then C should not be judged to be greater than A.
7. To the extent practical provide justification of your elicited judgment in the comment field provided in the elicitation form as a way to document your thinking and to facilitate internal consistency.
8. If there are LPSD PRA modelling challenges that make an elicitation response (or set of responses) uncertain or for which assumptions must be made, then identify those uncertainties and assumptions in the comment field provided in the elicitation form.

Form-Specific Instructions

The following are instructions, explanations, and identification of specific relevant information associated with understanding and filling out each elicitation form. These instructions primarily consist of explaining what judgements are being elicited and why and a list of key sources of information relevant to filling out the forms. In many cases, this key information, such as definitions of the PIRT parameters and terms used in the PIRT elicitation forms are included in these instructions as separate attachments. In other cases the information resides in reports that we have previously transmitted to you.

<u>Form TLC1 – Internal Events Core Damage Top-Level Criteria Elicitation Form</u>

This form elicits comparisons of the top-level evaluation criteria to each other using the Comparison Scale and includes internal event hazards as one of the top-level criteria. The expert must choose a Comparison Category from the Comparison Scale by assigning an “A”, “B”, “C”, “D”, or “E” to the appropriate fields in the form. These comparisons are used to quantitatively determine the relative importance of the top-level evaluation criteria to each against the overarching goal of importance to core damage. The top-level evaluation criteria were defined in a way that the importance of the criteria is more-or-less proportionally consistent with respect to the other top-level criteria across the different POSs. This means that each criterion is meant to be thought of as an independent importance contributor to the internal events core damage. This relationship is needed so that each top-level criteria can be weighted in importance against the other criteria.

Internal events for the sake of this elicitation is defined as plant equipment failures (including equipment failures caused by random loss of off-site power), vessel and line breaks, and operator errors that initiate an accident sequences during LPSD which could lead to core damage.

Information needed to fill out this form:

1. The Comparison Scale is provided in Attachment A.
2. The Summary Table of PIRT parameters, evaluation questions, and ranking categories and associated considerations are provided in Attachment B.
3. A Hierarchy Diagram of the evaluation criteria associated with internal events importance to core damage is provided in Attachment C.

Other information relevant to filling out the form:

- NUREG-1449, “Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States”

- NUREG/CR-6144, “Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, unit 1” .[1]
- NUREG/CR-6093, “An Analysis of Operational Experience During Low Power and Shutdown and a Plan for Addressing Human Reliability Assessment Issues”
- EPRI 1003465, “Low Power and Shutdown Risk Assessment Benchmarking Study”
- EPRI 3002005295296, “EPRI Low Power and Shutdown Probabilistic Risk Assessment Standard Pilot: Palo Verde Self-Assessment”
- IAEA-TECDOC-1144, “Probabilistic safety assessments of nuclear power plants for low power and shutdown modes”

Form TLC2 – Internal Events Release Top-Level Criteria Elicitation Form

This form elicits comparisons of the top-level evaluation criteria to each other using the Comparison Scale and includes internal event hazards as one of the top-level criteria. The expert must choose a Comparison Category from the Comparison Scale by assigning an “A”, “B”, “C”, “D”, or “E” to the appropriate fields in the form. These comparisons are used to quantitatively determine the relative importance of the top-level evaluation criteria to each against the overarching goal of importance to release from a damaged core. The top-level evaluation criteria were defined in a way that the importance of the criteria is more-or-less proportionally consistent with respect to the other top-level criteria across the different POSs. This means that each criterion is meant to be thought of as an independent importance contributor to the internal events core damage release. This relationship is needed so that each top-level criteria can be weighted in importance against the other criteria.

The top-level criteria for this table is the same as the top-level criteria for the previous table against the overarching goal of importance to core damage, with the exception that a criterion for containment functionality was added. It should not necessarily be assumed that the relative importance between the criteria in this table is similar to the relative importance between the criteria in the previous table.

Information needed and other information relevant to filling out this form are the same as for Form TCL1.

Form TLC3 – Fire Event Core Damage Top-Level Criteria Elicitation Form

This form elicits comparisons of the top-level evaluation criteria to each other using the Comparison Scale and includes internal flooding hazards as one of the top-level criteria. As in the case of other top-level evaluation criteria, these comparisons are used to quantitatively determine the relative importance of the top-level evaluation criteria to each against the

[1] NUREG/CR-6144, “Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1,” Vols. 1–6 October 1995, is not publicly available.

overarching goal of importance to core damage. The top-level evaluation criteria were defined in a way that the importance of the criteria is more-or-less proportionally consistent with respect to the other top-level criteria across the different POSs. This means that each criterion is meant to be thought of as an independent importance contributor to the internal events core damage. This relationship is needed so that each top-level criteria can be weighted in importance against the other criteria.

The top-level criteria for this table (i.e., Form TCL3) is similar to the top-level criteria for Form TCL1 associated with Internal Events importance to core damage, with the exception that the criterion “Internal Events Hazard” is replaced by “Fire Hazard’. In spite of this similarity, it should not necessarily be assumed that the relative importance between the criteria for the two forms are similar.

The fire hazard for the sake of this elicitation is defined as fires in the plant during LPSD associated with electrical cabinets, electoral cables, transient combustibles, and equipment that causes fire damage that can lead to the failure of equipment and components, or spurious equipment actuations that initiate an accident sequence at LPSD which could lead to core damage.

Information needed to fill out this form:

1. The Comparison Scale is provided in Attachment A.
2. The Summary Table of PIRT parameters, evaluation questions, and ranking categories and associated considerations are provided in Attachment B.
3. A Hierarchy Diagram of the evaluation criteria associated with internal events importance to core damage is provided in Attachment D.

Other information relevant to filling out the form:

- NUREG/CR-7114, “A Framework for low Power/Shutdown Fire PRA” (Nowlen 2013)
- Plant-specific proprietary report on Fire PRA quantification

<u>Form TLC4 – Fire Event Release Top-Level Criteria Elicitation Form</u>

This form elicits comparisons of the top-level evaluation criteria to each other using the Comparison Scale and includes internal flooding hazards as one of the top-level criteria. As in the case of other top-level evaluation criteria, these comparisons are used to quantitatively determine the relative importance of the top-level evaluation criteria to each against the overarching goal of importance to release from a damage core. The top-level evaluation criteria were defined in a way that the importance of the criteria is more-or-less proportionally consistent with respect to the other top-level criteria across the different POSs. This means that each criterion is meant to be thought of as an independent importance contributor to the internal events core damage release. This relationship is needed so that each top-level criteria can be weighted in importance against the other criteria.

The top-level criteria for this table (i.e., Form TCL4) is similar to the top-level criteria for Form TCL2 associated with Internal Events importance to release from a damaged core, with the exception that the criterion “Internal Events Hazard” is replaced by “Fire Hazard’. In spite of this similarity, it should not necessarily be assumed that the relative importance between the criteria for the two forms are similar.

Information needed and other information relevant to filling out this form are the same as for Form TCL3 above.

<u>Form TLC5 – Internal Flooding Event Core Damage Top-Level Criteria Elicitation Form</u>

This form elicits comparisons of the top-level evaluation criteria to each other using the Comparison Scale and includes internal flooding hazards as one of the top-level criteria. As in the case of other top-level evaluation criteria, these comparisons are used to quantitatively determine the relative importance of the top-level evaluation criteria to each against the overarching goal of importance to core damage.

Internal flooding events for the sake of this elicitation is defined as internal flooding events caused by a line or vessel breaches or operator error that lead to loss of plant systems as a result of water impact that initiate an accident sequence at LPSD which could lead to core damage.

Information needed to fill out this form:

1. The Comparison Scale is provided in Attachment A.
2. The Summary Table of PIRT parameters, evaluation questions, and ranking categories and associated considerations are provided in Attachment B.
3. A Hierarchy Diagram of the evaluation criteria associated with internal flooding importance to core damage is provided in Attachment E.

Other information relevant to filling out the form is the plant-specific proprietary report on Internal Flooding PRA.

<u>Form TLC6 – Internal Flooding Event Release Top-Level Criteria Elicitation Form</u>

This form elicits comparisons of the top-level evaluation criteria to each other using the Comparison Scale and includes internal flooding hazards as one of the top-level criteria. These comparisons will be used to weight the relative importance of the top-level evaluation criteria used to evaluate each Plant Operating State (POS) against the overarching goal of “Importance to release from damaged core.”

Information needed and other information relevant to filling out this form are the same as for Form TCL5 above.

Form TLC7 – Seismic Event Core Damage Top-Level Criteria Elicitation Form

This form elicits comparisons of the top-level evaluation criteria to each other using the Comparison Scale and includes the seismic hazard as one of the top-level criteria. The purpose of this form is like the other top-level evaluation criteria forms.

Seismic events for the sake of this elicitation are defined as seismic events of differing magnitudes and severity that cause enough structural and/or equipment damage to initiate an accident sequence during LPSD which could lead to core damage.

Information needed to fill out this form:

1. The Comparison Scale is provided in Attachment A.
2. The Summary Table of PIRT parameters, evaluation questions, and ranking categories and associated considerations are provided in Attachment B.
3. A Hierarchy Diagram of the evaluation criteria associated with seismic events is similar to the ones for fire and internal flooding and is not presented elicitation instructions. However, the sub-criteria for the seismic hazard can be found in the Summary Table of PIRT parameters provided in Attachment F.

Other information relevant to filling out this form is the plant-specific proprietary report on Seismic PRA.

Form TLC8 – Seismic Event Release Top-Level Criteria Elicitation Form

This form elicits comparisons of the top-level evaluation criteria to each other using the Comparison Scale and includes the seismic hazard as one of the top-level criteria. The purpose of this form is like the other top-level evaluation criteria forms.

Information needed and other information relevant to filling out this form are the same as for Form TCL7 above.

Form SC1 – Sub-criteria Elicitation Form

This form elicits comparisons of the sub-criteria using the Comparison Scale associated with each top-level evaluation criterion to each other using the Comparison Scale. In each case (i.e., for each top-level criterion), there are just two sub-criterion. These comparisons will be used to weight the relative importance of sub-criteria to the top-level evaluation criteria. Accordingly, the relative weight of the importance of the sub-criteria to the top-level criterion multiplied by relative weight of the importance of that top-level criterion produces the relative weight of that sub-criterion to the overarching goal of importance to core damage or release from a damage core. The evaluation sub-criteria were defined in a way that the importance of the criteria is more-or-less proportionally consistent with respect to the other sub-criteria. This means that each sub-criterion is meant to be thought of as an independent importance contributor to the fire event core damage. This relationship is needed so that the evaluation sub-criteria can be weighted in importance against the other evaluation sub-criteria.

Information needed to fill out this form:

1. The Comparison Scale is provided in Attachment A.
2. The Summary Table of PIRT parameters, evaluation questions, and ranking categories and associated considerations are provided in Attachment B.

Other information relevant to filling out this form is the same as for the TCL tables above.

<u>Form SCR1 –RCS Water Level Sub-Criterion Ranking Elicitation Form</u>

This forms elicits two kinds of information pertaining to ranking categories. First, though the ranking categories have labels (e.g., High, Medium, and Low) that in general correspond to the level of importance of the sub-criteria to overarching goals of core damage and release from a damaged core, they must be further defined by the expert to make them useful for prioritizing the importance of POSs. This can be accomplished using qualitative or quantitative descriptors, or both. To facilitate a clear understanding of what judgements are being elicited from the experts, the sub-criteria in the elicitation form is presented as an evaluation question (i.e., “What is the RCS water inventory level?”) The purpose of providing more definitive definitions of the ranking categories (in this case the ranking categories are High water Level, Medium Water Level, and Low Water Level) is to provide anchors that will help the expert provide judgement that are internally consistent. In this case, the expert might choose to define the ranking categories according to feet of water in the reactor vessel. This sub-criteria supports the top-level evaluation criterion “RCS Inventory Control.”

The second kind of information that will be elicited is comparisons of the ranking categories to each other using the Comparison Scale. These comparisons will be used to weight the relative importance of the ranking categories amongst themselves to the sub-criteria and ultimately to the overarching importance goals. The relative weight between ranking categories does not need to be equally distributed. It is possible, for example, that only one of the categories (e.g., Low Water Level) might be significant to importance.

During mid-loop operations the coolant is drained to its lowest level. Systems to keep the core covered include charging pumps, ECCS or gravity feed from the RWST.

Information needed to fill out this form:

1. The Summary Table of PIRT parameters, evaluation questions, and ranking categories and associated considerations are provided in Attachment B.
2. The plant-specific proprietary report on Low Power and Shutdown PRA

Other information relevant to filling out this form:

- Plant-specific proprietary report on Shutdown PRA
- Plant-specific proprietary schematic showing elevations of equipment and water levels in the RCS
- Plant-specific proprietary related procedures

Form SCR2 – Availability of Systems to Make-up Inventory Sub-Criterion Ranking Elicitation Form

As for all ranking category forms, this forms elicits two kinds of information. First, though the ranking categories have labels (e.g., High, Medium, and Low) that in general correspond to the level of importance of the sub-criteria to overarching goals of core damage and release from a damaged core, they must be further defined by the expert to make them useful for prioritizing the importance of POSs. This can be accomplished using qualitative or quantitative descriptors, or both. To facilitate a clear understanding of what judgements are being elicited from the experts, the sub-criteria in the elicitation form is presented as an evaluation question (i.e., “What is the availability of systems to keep the core covered?”). The purpose of providing more definitive definitions of the ranking categories (in this case the ranking categories are Not Very available, Available, Very Available) is to provide anchors that will help the expert provide judgement that are internally consistent. In this case, the expert might choose to define the ranking categories in terms of the number of available systems and trains that could provide enough water to keep the core covered. This sub-criteria supports the top-level evaluation criterion “RCS Inventory Control.”

The second kind of information that will be elicited is comparisons of the ranking categories to each other using the Comparison Scale. These comparisons will be used to weight the relative importance of the ranking categories amongst themselves to the sub-criteria and ultimately to the overarching importance goals. The relative weight between ranking categories does not need to be equally distributed. It is possible, for example, that only one of the categories (e.g., Not very Available) might be significant to importance.

Systems to keep the core covered include charging pumps, ECCS or gravity feed from the RWST.

Information needed to fill out this form:

1. The Summary Table of PIRT parameters, evaluation questions, and ranking categories and associated considerations are provided in Attachment B.
2. The Plant-specific proprietary report on Low Power and Shutdown PRA

Other information relevant to filling out this form:

- Plant-specific proprietary report on Shutdown PRA
- Plant-specific proprietary related procedures

Forms SCR3 and SCR4 – Heat Load and Availability of Reactor Cooling Systems Sub-Criterion Ranking Elicitation Form

This form elicits the same two kinds of information as all the sub-criteria ranking elicitation forms. The elicitation questions and corresponding ranking category labels presented for the sub-criteria in these forms are:

- "What is the heat load from power, decay heat, and RCS temperature?" - High Load, Medium Load, and Low Load
- "What is the availability of reactor cooling systems?" – Not Very Available, Available, and Very Available

These sub-criteria support the top-level evaluation criterion "RCS Inventory Control."

Some POSs are low power operating states rather than shutdown operating states, thus power is being produced. Decay heat load is a function of time since reactor shutdown and as such is attribute of each POS. Considerations for RCS cooling include the number of RHR trains available and whether steam generator (SG) cooling is functional.

Information needed and other information relevant to filling out this form is similar to the other sub-criteria ranking elicitation forms.

Forms SCR5 and SCR6 – RCS Loop Isolation and Pressure Relief Sub-Criterion Ranking Elicitation Forms

These form elicits the same two kinds of information as all the sub-criteria ranking elicitation forms. The elicitation questions and corresponding ranking categories presented for the sub-criteria in these forms are:

- "What is the level of vulnerability to loss of RCS loop isolation?" – Very Vulnerable, Vulnerable, and Not vulnerable
- "What is the level of vulnerability to over-pressurization of the RCS?" - Very Vulnerable, Vulnerable, and Not vulnerable

These sub-criteria support the top-level evaluation criterion "RCS Integrity."

Challenges to loop isolation include the presence of nozzle dams, low pressure seals such as instrument tube seals, water addition when the RCS is water solid, over-pressure events through the RHR relief valves, RCS shutdown seals, and maintenance activities that could drain the primary inventory. Relief capability consideration include the status of the PORVs, pressurizer manway, SG manways, or head vents are open.

Information needed and other information relevant to filling out this form is similar to the other sub-criteria ranking elicitation forms.

Forms SCR7 and SCR8 – Operator Initiated Events and Important Equipment Failures Sub-Criterion Ranking Elicitation Forms

These forms elicit the same two kinds of information as all the sub-criteria ranking elicitation forms. The elicitation questions and corresponding ranking categories presented for the sub-criteria in these forms are:

- "What is the level of opportunity for accident sequences initiated by operator error?" – High, Moderate, Low
- "What is the level of opportunity for accident sequences initiated by equipment failures?" - High, Moderate, Low

These sub-criteria support the top-level evaluation criterion "Internal Events Hazard."

Considerations for this criterion include operator load, stress, and distractions, number of required actions, as well as availability of valid instrumentation and control.

This criterion concerns the likelihood of initiating events associated with equipment failure. It does not concern the vulnerability of plant configuration due to unavailable systems. Unavailability of systems is considered under other top-level criterion.

Information needed and other information relevant to filling out this form is similar to the other sub-criteria ranking elicitation forms.

Forms SCR9 and SCR10 – Containment Isolation Capability and Availability of Radionuclide Suppression Systems Sub-Criterion Ranking Elicitation Forms

These forms elicit the same two kinds of information as all the sub-criteria ranking elicitation forms. The elicitation questions and corresponding ranking categories presented for the sub-criteria in these forms are:

- "What is the time required to close containment versus time available?" – Short, Moderate, Long
- "What is the availability of the radionuclide suppression systems?" - Not Very Available, Available, and Very Available

These sub-criteria support the top-level evaluation criterion "Containment Performance."

The time-to-boiling is an important consideration if it is sooner than the containment can be closed. Time to boiling is a function of heat load so is not repeated under this criterion. Radionuclide suppression includes sprays and filters.

Information needed and other information relevant to filling out this form is similar to the other sub-criteria ranking elicitation forms except that following:

1. Plant-specific proprietary information on LPSD containment integrity and isolation
2. Plant-specific proprietary information on Time-to-Boil

Forms SCR11, SCR12, SCR13, SCR14, SCR15, and SCR16 – Fire Frequency, Fire Damage Vulnerability, Internal Flooding Frequency, Internal Flooding Damage Vulnerability, Seismic Frequency, and Seismic Damage Vulnerability Sub-Criterion Ranking Elicitation Forms

These form elicits the same two kinds of information as all the sub-criteria ranking elicitation forms. The elicitation questions and corresponding ranking categories presented for the sub-criteria in these forms are related to hazard events and conform to the following format:

- "What is the of the hazard event frequency? – High Frequency, Moderate Frequency, Low Frequency
- "What is the chance that damage from the hazard event initiates an accident sequence?" – Very Vulnerable, Vulnerable, Normal

Information needed and other information relevant to filling out this form is similar to the other sub-criteria ranking elicitation forms except that following may be useful:

- Plant-specific proprietary report on Fire PRA quantification
- Plant-specific proprietary report on Internal Flooding PRA
- Plant-specific proprietary report on Seismic PRA

Form PR1 - Plant Operating State (POS) Importance Ranking Elicitation Form

This form elicits from the expert for each POS assignment the ranking category judged to be most appropriate responses to the evaluation equations associated with each criterion (i.e., the 16 sub-criteria). There is not enough room on this form to present the evaluation questions associated with each criterion, so only the criteria are presented. Therefore, it will be useful to refer to Attachment B which provides the evaluation question for each criterion along with the options available (i.e., the ranking categories). In this form, the ranking categories options are listed just under each criterion. The expert will enter an "H", "M", or "L" in the parenthesis just right of the ranking category selected (This approach, rather than writing out the actual ranking category label facilitates quantification of priorities).

The ranking categories assigned in this form have weights that has been previously determined by pair-wise comparison of the sets of ranking categories defined for each sub-criterion. The sub-criteria have weights that have been determined by pair-wise comparison of the sets sub-criteria associated with top-level criteria. The top-level criteria have weights that have been determined by pair-wise comparison of the top-level criteria amongst themselves. By assigning a ranking category to each POS for each criterion an importance priority can be calculated for each POS for each of the 10 following top-level cases:

- POS priorities for internal events associated with importance to core damage
- POS priorities for internal events associated with importance to release from a damaged core
- POS priorities for fire associated with importance to core damage

- POS priorities for fire associated with importance to release from a damaged core
- POS priorities for internal flooding associated with importance to core damage
- POS priorities for internal flooding associated with importance to release from a damaged core
- POS priorities for seismic events associated with importance to core damage
- POS priorities for seismic events associated with importance to release from a damaged core

Information needed to fill out this form:

1. The Summary Table of PIRT parameters, evaluation questions, and ranking categories and associated considerations are provided in Attachment B.
2. The Plant Outage Type and Plant Operating State Summary Table provided in Attachment I
3. Plant-specific Outage Frequencies and Durations
4. The plant-specific proprietary report on Low Power and Shutdown PRA

Other information relevant to filling out this form:

- Plant-specific proprietary report on Shutdown PRA
- Plant-specific proprietary related procedures and outage reports
- All previously identified information sources earlier in these instructions

Form LOK1 – Self-Evaluation of Level of Knowledge

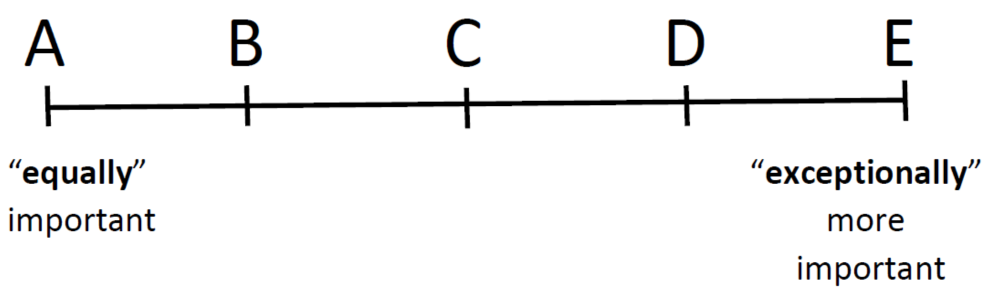
This form elicits from the experts a self-evaluation of the level-of-knowledge they possess about the information that is elicited in the forms. This information will not be used to calculate POS, POT, or hazard priorities, rather it will be used along with other uncertainty information to help characterize uncertainties in the elicitation results and to identify related insights. That said, given that each of you are experts in LPSD PRA and/or outage operations, it expected that the fields for most forms for most experts will be assigned a “High Level of Knowledge.” The expert enter an “H”, “M”, or “L” for each PIRT evaluation table used in the PIRT evaluation process. If a particular portion of a particular form merits a different category assignment then the rest of the form than those details can be explained in the comments column.

ATTACHMENT A

COMPARISON SCALE

Table 1 Comparison Scale

Scale	Scale Definition
A	Criteria X and Criteria Y **"equally"** important
B	Criteria X **"slightly"** more important than Criteria Y
C	Criteria X **"moderately"** more important than Criteria Y
D	Criteria X **"strongly"** more important than Criteria Y
E	Criteria X **"exceptionally"** more important than Criteria Y



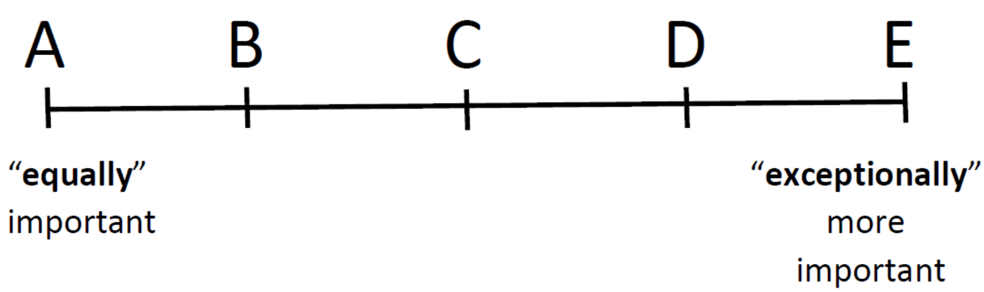


Figure 1 Visual Representation of Comparison Scale

ATTACHMENT B

SUMMARY TABLE OF PIRT PARAMETERS, EVALUATION QUESTIONS, RANKING CATEGORIES AND CONSIDERATIONS

Table 2 Summary Table of Evaluation Criteria, Ranking Categories and Associated Considerations

TOP-LEVEL CRITERIA	SUB-CRITERIA	EVALUATION QUESTIONS	RANKING CATEGORIES	CONSIDERATIONS
RCS Inventory Control				
	RCS water level	What is the RCS water inventory level?	Low Medium High	During mid-loop operations the coolant is drained to its lowest level.
	Availability of systems to make-up inventory	What is the availability of systems to keep the core covered?	Not Very Available Available Very Available	Systems to keep the core covered include charging pumps, ECCS or gravity feed from the RWST.
Heat Removal				
	Heat load	What is the heat load from power, decay heat, and RCS temperature?	High Medium Low	Power is being produced for the low power POSs. Decay heat load is a function of time since reactor shutdown and as such an attribute of each POS.
	Availability of reactor cooling systems	What is the availability of reactor cooling systems?	Not Very Available Available Very Available	Includes the number of RHR trains available and whether SG cooling is functional.
RCS Integrity				
	RCS isolation	What is the level of vulnerability to maintaining RCS isolation?	Very Vulnerable Vulnerable Normal	Challenges to RCS isolation include: isolation the presence of nozzle dams, low pressure seals such as instrument tube seals, RCP shutdown seals, maintenance activities that could drain the primary inventory, and pressurizer manways and SG manways,
	Pressure relief capability	What is the level of vulnerability to over-pressurization of the RCS?	Very Vulnerable Vulnerable Not Vulnerable	Relief capability consideration include PORVs, overpressure events through the RHR relief valves, whether head vents are open, and water addition when the RCS is water solid.
Containment Performance				
	Containment isolation capability	What is the time required to close containment versus time available?	Short Moderate Long	The time-to-boiling is an important consideration if it is sooner than the containment can be closed. Time to boiling is a function of heat load so is not repeated under this criterion.
	Availability of radionuclide suppression systems	What is the availability of radionuclide suppression systems?	Not Very Available Available Very Available	Radionuclide suppression includes sprays and filters.

Table 2 Summary Table of Evaluation Criteria, Ranking Categories and Associated Considerations (continued)

TOP-LEVEL CRITERIA	SUB-CRITERIA	EVALUATION QUESTIONS	RANKING CATEGORIES	CONSIDERATIONS
Internal Events Hazard*				
	Operator initiated events	What is the level of opportunity for accident sequences initiated by operator errors?	High Moderate Low	Considerations for this criterion include operator load, stress, and distractions, number of required actions, as well as maintenance and other activities not directly related to transitioning between POSs. Considerations also include duration of the POSs and the availability of valid instrumentation and control.
	Important equipment failures	What is the level of opportunity for accident sequences initiated by equipment failures?	High Moderate Low	This criteria concerns the likelihood of initiating events associated with equipment failure. It does not concern the vulnerability of plant configuration due to unavailable systems. Unavailability of systems is considered under other top-level criterion.
Fire Hazard*				
	Fire frequency	What is the fire frequency for this plant configurations and set of activities?	High Frequency Moderate Frequency Low Frequency	Contributors to fire frequency include the level of operational and maintenance activity and level of combustible material loading during LPSD and the duration of the POS.
	Fire damage vulnerability	What is the chance that fire damage initiates an accident sequence?	Very Vulnerable Vulnerable Normal	This criteria concerns the likelihood that fires initiate accident sequences. It does not concern the vulnerability of plant configuration due to unavailable systems. Unavailability of systems is considered under other top-level criterion.
Internal Flooding Hazard*				
	Internal flooding event frequency	What is the internal flooding event frequency for this plant configurations and set of activities?	High Frequency Moderate Frequency Low Frequency	Contributors to internal flooding event frequency include the large number of actions associated with transitioning from one POS to another and maintenance activities. Considerations also include duration of the POSs.
	Internal flooding damage vulnerability	What is the chance that internal flooding damage initiates an accident sequence?	Very Vulnerable Vulnerable Normal	This criteria concerns the likelihood that flooding events initiate accident sequences. It does not concern the vulnerability of plant configuration due to

Table 2 Summary Table of Evaluation Criteria, Ranking Categories and Associated Considerations (continued)

TOP-LEVEL CRITERIA	SUB-CRITERIA	EVALUATION QUESTIONS	RANKING CATEGORIES	CONSIDERATIONS
				unavailable systems. Unavailability of systems is considered under other top-level criterion. The status of features designed to minimize of impact of flooding could vary from full-power.
Seismic Event Hazard*				
	Seismic event frequency	What is the seismic event frequency for this plant?	High Frequency	The ranking should be the same for all POSs. Considerations also include duration of the POSs.
			Moderate Frequency	
			Low Frequency	
	Seismic damage vulnerability	What is the chance that a seismic damage initiates an accident sequence?	Very Vulnerable	This criteria concerns the vulnerability of the plant configuration to a seismic event (i.e., fragility). It does not concern the vulnerability of plant configuration due to unavailable systems. Unavailability of systems is considered under other top-level criterion.
			Vulnerable	
			Normal	

Note:
*The internal events, fire, internal flooding, and seismic event top criteria will be compared separately with the other top level criteria.

ATTACHMENT C

HIERARCHY DIAGRAMS OF EVALUATION CRITERIA FOR INTERNAL EVENTS IMPORTANCE

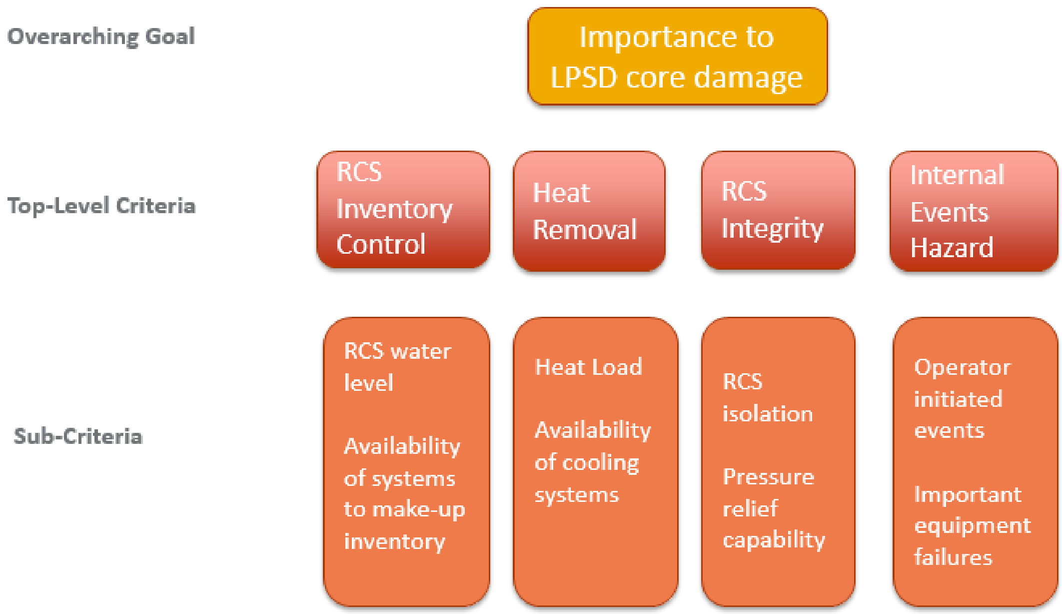


Figure 2 Hierarchy Diagram of Evaluation Criteria for Internal events Importance to Core Damage

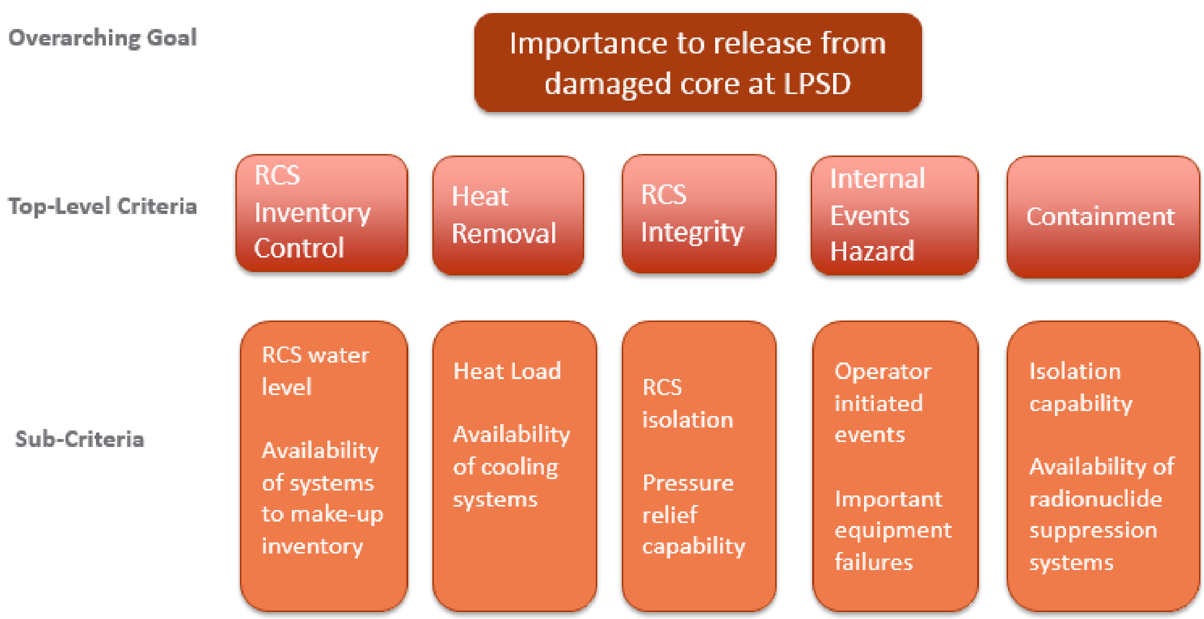


Figure 3 Hierarchy Diagram of Evaluation Criteria for Internal Events Importance to Release from Damaged Core

ATTACHMENT D

HIERARCHY DIAGRAM OF EVALUATION CRITERIA FOR FIRE IMPORTANCE

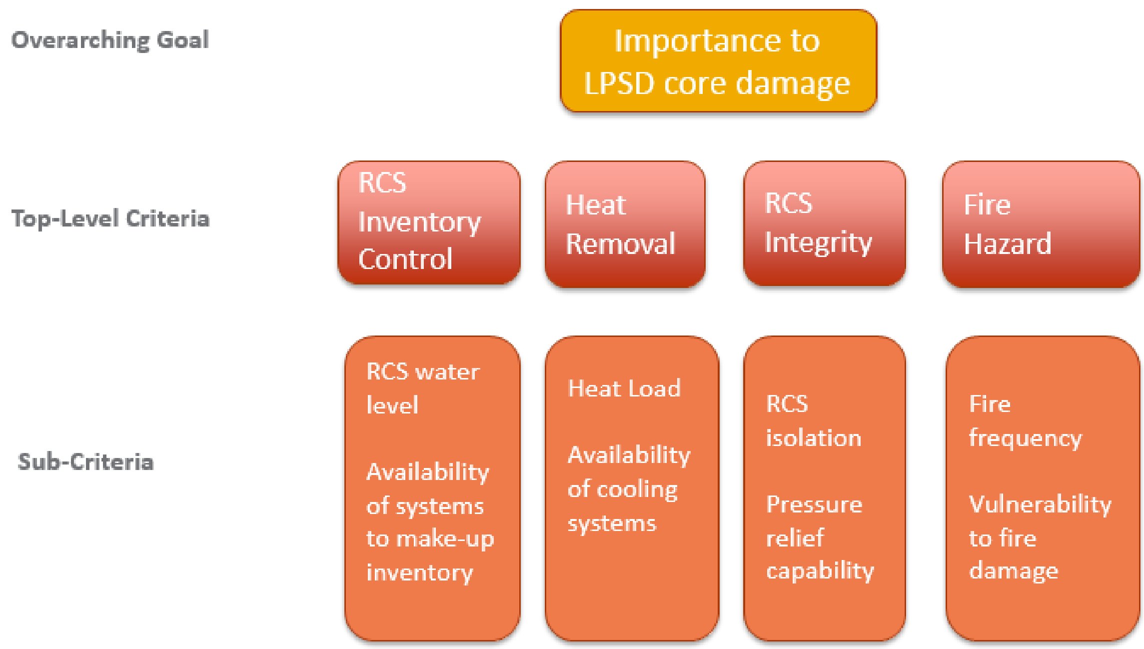


Figure 4 Hierarchy Diagram of Evaluation Criteria for Fire Importance to Core Damage

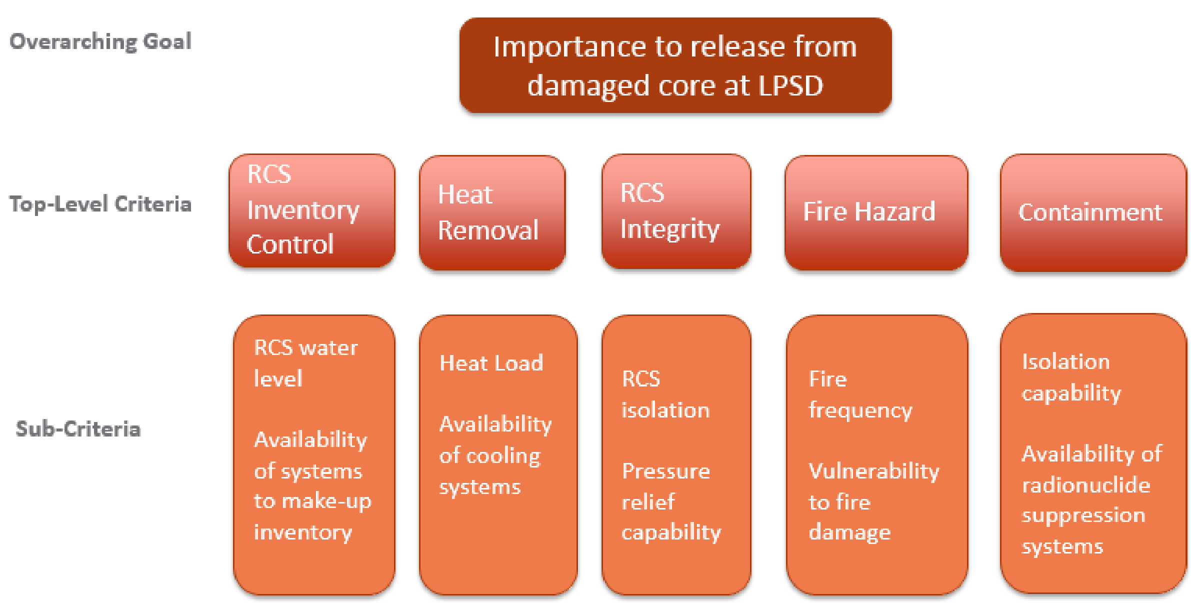


Figure 5 Hierarchy Diagram of Evaluation Criteria for Fire Importance to Release from Damaged Core

ATTACHMENT E

HIERARCHY DIAGRAM OF EVALUATION CRITERIA FOR INTERNAL FLOODING IMPORTANCE

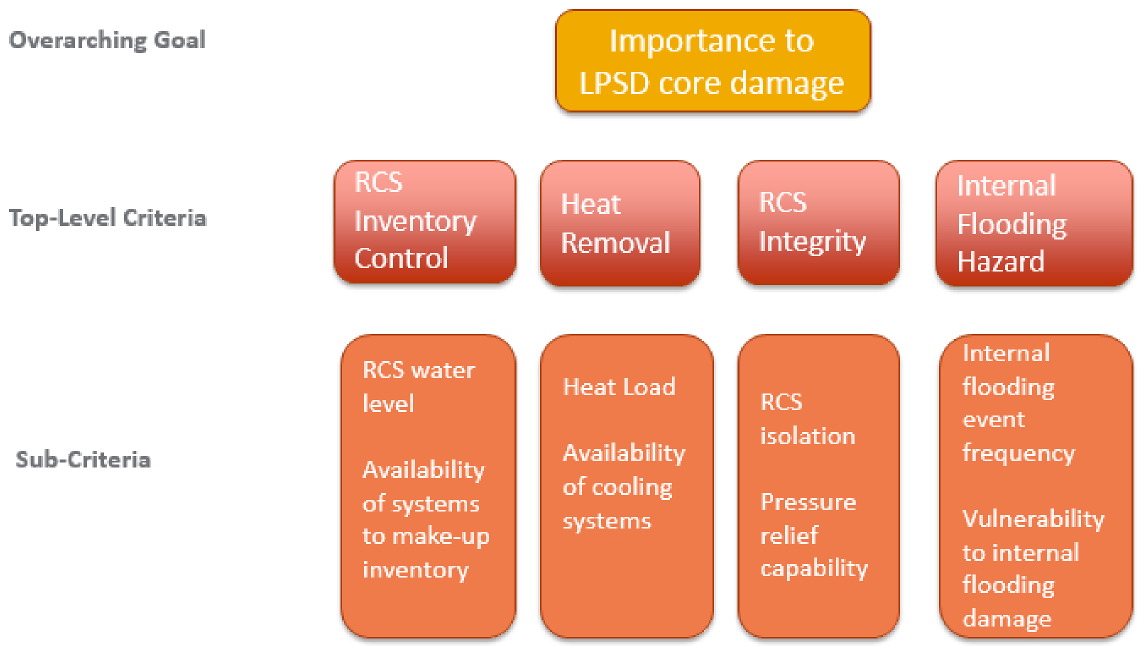


Figure 6 Hierarchy Diagram of Evaluation Criteria for Internal Flooding Importance to Core Damage

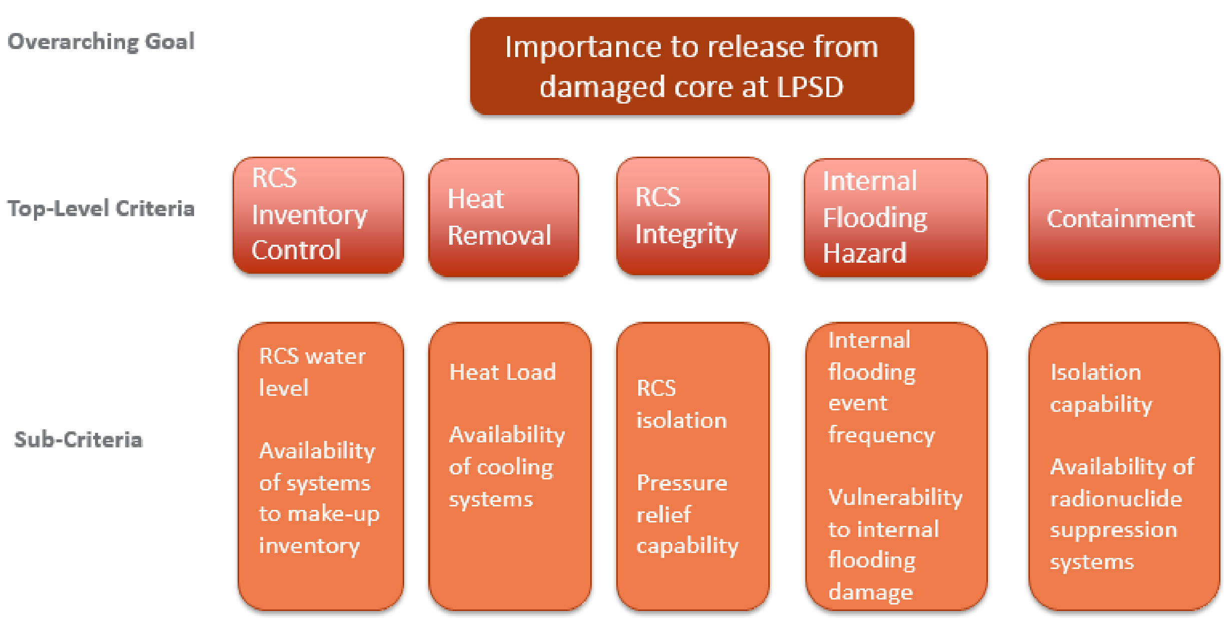


Figure 7 Hierarchy Diagram of Evaluation Criteria for Internal Flooding Importance to Release from Damaged Core

ATTACHMENT F

HIERARCHY DIAGRAM OF EVALUATION CRITERIA FOR SIESMIC EVENT IMPORTANCE

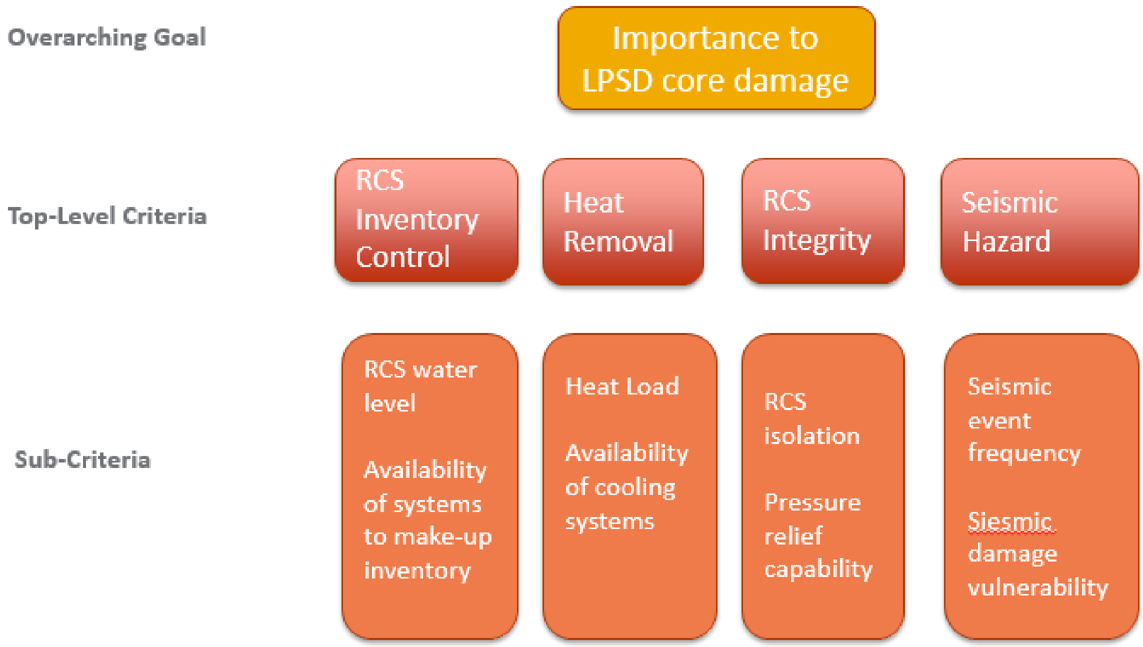


Figure 8 Hierarchy Diagram of Evaluation Criteria for Seismic Event Importance to Core Damage

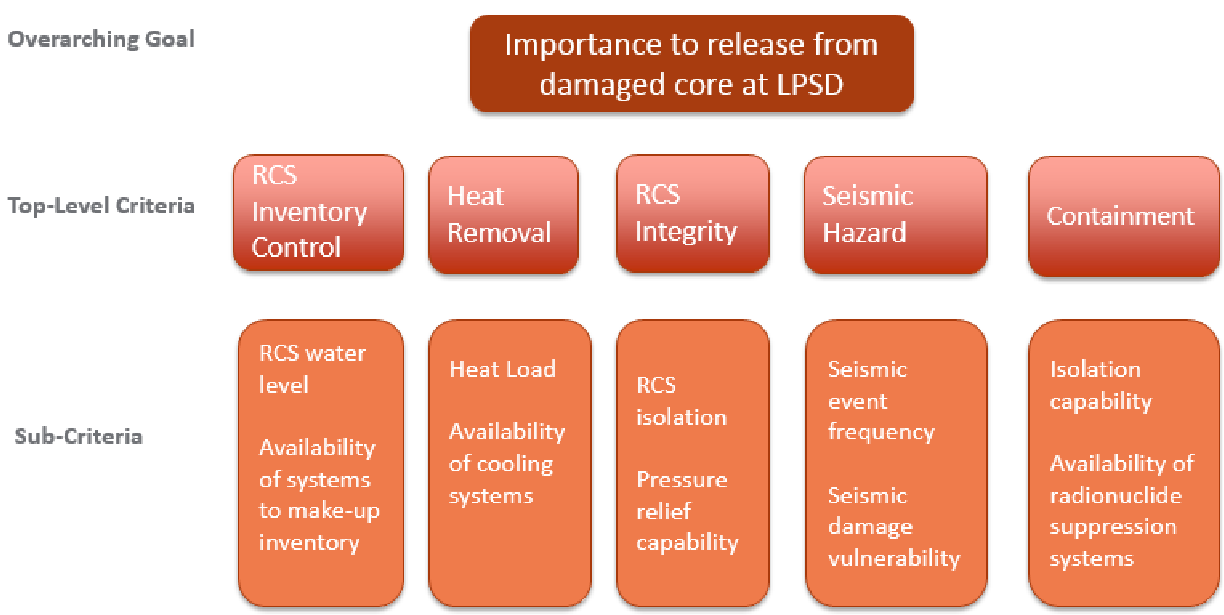


Figure 9 Hierarchy Diagram of Evaluation Criteria for Seismic Event Importance to Release from Damaged Core

ATTACHMENT G

PLANT OUTAGE TYPE AND PLANT OPERATING STATE SUMMARY TABLE

Table 3 Plant Outage Type and Plant Operating State Summary

POS		TS	POS Applicable when Transitioning to Outage Type			
No.	Description	TS Mode	Refueling (POT-4)	Hot Standby (POT-1)	Maintenance w/o Drain (POT-2)	Maintenance w/Drain (POT-3)
1	Low power and reactor shutdown	1,2	X	N/A	X	X
2	Cooldown with steam generators to 350 °F	3	X		X	X
2-P1	Cooldown with steam generators to 350 °F	3		X		
3	Cooldown with residual heat removal system to 200 °F	4	X		X	X
4	Cooldown to ambient temperature with residual heat removal system only	5	X			X
4-P2	Cooldown to ambient temperature with residual heat removal system only	5			X	
5A	Pressurizer water solid for degassing	5	X			X
5B	Draining the reactor coolant system to reduced inventory, RCS is vented	5,6	X			X
6	Mid-loop operation prior to refueling	5,6	X			X
7	Filling refueling cavity for refueling operation	6	X			
8E	Refueling operation (offloading old core)	6	X			
DF[1]	Defueled	n/a	X			
8L	Refueling operation (loading new core)	6	X			
9	Draining the reactor coolant system after refueling operation	6	X			
10	Mid-loop operation after refueling	5,6	X			
11	Refill reactor coolant system, reactor vents are closed	5,6	X			X
12	Reactor coolant system heatup/draw bubble in pressurizer	5	X			X
13	Reactor coolant system heatup to 350 °F	4	X		X	X

Table 3 Plant Outage Type and Plant Operating State Summary (continued)

POS		TS	POS Applicable when Transitioning to Outage Type			
No.	**Description**	**TS Mode**	**Refueling (POT-4)**	**Hot Standby (POT-1)**	**Maintenance w/o Drain (POT-2)**	**Maintenance w/Drain (POT-3)**
14	Startup with steam generators to Hot Standby	3	X		X	X
15A	Reactor startup and low power operation (0=Power<5%)	2	X	X	X	X
15B	Reactor startup and low power operation (5<Power<50%)	1,2	X	X	X	X

[1]POS DF is provided in this table for completeness but is not evaluated by the PIRT panel since the reactor is defueled.
Note: The PIRT expert panel members were presented with additional plant specific information used for defining the POS details. The table, as presented in this report, omits some of the POS details that were used in the expert elicitation forms.

APPENDIX I LPSD PRA PIRT FACILITATOR CHECKLIST

PIRT Elicitation Facilitator Checklist

1. Check on whether the expert received the PIRT elicitation instructions and evaluation forms and had time to look at them.
2. Check on whether the expert had time to exercise the forms and to what extent.
3. Explain how the online meeting will work (We will ask for example responses from each kind of evaluation form to ensure that expert understands what information is being elicited. We give the expert three days to return the completed table to PNNL.).
4. Read the general instructions. (For #7, providing a basis for selecting "Strong or Exceptionally" is particularly important).
5. Open Form TLC1. Explain what information is being elicited and how the form works.
6. Ask the expert to assign a Comparison Category to one of the TLC pairs and a basis for the category assignment.
7. Ask why one Comparison Category up or down wouldn't be more appropriate. Point out the resources available in the Attachment 2 Summary Table.
8. Move to Form TLC2 and ask the expert to assign a Comparison Category to one of the TLC pairs involving Containment and to provide a category assignment and basis.
9. Move to one of the other TLC comparison tables and ask the expert to assign a Comparison Category to one of the TLC pairs and that involve an Initiating Event (e.g., Fire).
10. Open Form SC1 and explain what information is being elicited and how the form works.
11. Ask the expert to assign a Comparison Category to one of the SC pairs and a basis for the category assignment.
12. Ask why one Comparison Category up or down wouldn't be more appropriate.
13. Open Form SCR1 and explain the two kinds of information being elicited and how the form works.
14. Suggest the kinds of information (quantitative or descriptive) could be used to define RCS Water Level.
15. Mention that in general the Ranking Category labels are intuitively arranged from high to low risk (importance to overarching goal), with the exception of the Ranking Categories for RCS Water Level which are :Low, Water Level, Medium water Level, and High Water Level.

16. Ask to expert to pick a SCR table and define the three categories.
17. For the same table ask the expert to assign Comparison Categories to each of the Ranking Categories.
18. Emphasize that the quantitative weights between the categories do not have to be evenly distributed. Use RCS Water Level as an example of why not.
19. Move to Form SCR9 and show them the error in Ranking Category labels that should Short Time, Moderate Time, and Long Time.
20. Open Form PR1 and explain what information is being elicited and how the form works.
21. Show the expert that RCS Water Level was deliberately placed out-of-order, but that for the rest of the form the sub-criteria are presented in the same order as the previous set of forms.
22. Emphasize that we want to expert enter a H, M, or L.
23. Ask the expert to pick a few fields and to assign Ranking Categories and verbally share the basis for those rankling.
24. Emphasize that we want bases for the assigned categories but that the bases could be provided with a whole column (i.e., a criterion).
25. Move to POS # 4-P2 and ask how POS #4-P2 is evaluated different from POS #4.
26. (After refueling there is a difference in the same POSs for different POTs. For refueling outage the POSs after the refueling will have significantly reduced decay heat due to the time it takes to perform refueling.)
27. Open Form LOK1 and explain the kinds of information being elicited and how the form works.

APPENDIX J COMPLETED ELICITATION FORMS FROM GROUP MEETING

This appendix provides an example set of LPSD PRA elicitation forms with the filled in responses. The seven expert members, each provided a set of LPSD PRA elicitation forms following the group elicitation meeting held from February 21-23, 2017. The example set of LPSD PRA elicitation forms is a compilation of forms selected from the seven expert members. The specific sets of LPSD PRA elicitation forms filled in by each expert member are not included for brevity.

TLC1

Internal Events Core Damage Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Justification for Evaluation
RCS Inventory Control is more important than **Heat Removal**	C	While heat load is important, water level is more important. Both criteria consider availability of systems, there are numerous systems to add water, but limited systems to remove decay hea (RHR and SG cooling) but the SG when available are passive and very reliable.
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**	A	RCS integrity is only important because of the inclusion of loss of inventory in the definition. The following are not important: If the PORVs fail open they create a hole in the top of the system. The failure to isolate probability is low and pipe break probability is very low. Inventory control includes level and risk is highly correlated to level.
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Internal Events Hazard**	A	As stated above risk is highly correlated with RCS level. However, risk is directly correlated with hazard (i.e., frequency) and IEF at shutdown are frequent on the order of 1 per year to 1 per ten years.
Internal Events Hazard is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		As described above, RCS integrity is important because it includes the LOI initator. While heat removal is moderately important.
RCS Integrity is more important than **Heat Removal**	D	
Heat Removal is more important than **Internal Events Hazard**		Risk is directly correlated with internal events hazard (frequency) and the frequencies are high. Heat removal is important but is a second order effect.
Internal Events Hazard is more important than **Heat Removal**	C	
RCS Integrity is more important than **Internal Events Hazard**		As described above, integrity is important only because it includes the LOI initator. The frequency of initiating event is large and directly correlated with risk.
Internal Events Hazard is more important than **RCS Integrity**	A	

TLC2

Internal Events Release Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**	E	(Really an F) The ability to add water (keep the core covered and in excess of what is needed for boiloff -prevent boiling) allows the operator more time to: (1) recover RHR and (2) close containment
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**	D	The ability to add water (keep the core covered and in excess of what is needed for boiloff -prevent steaming inside containment) given any size of RCS penetration allows the operator more time to: (1) recover RHR and (2) close containment
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Internal Events Hazard**	D	The ability to add water (keep the core covered and in excess of what is needed for boiloff -prevent boiling) given any internal events hazard allows the operator more time to: (1) recover RHR and (2) close containment
Internal Events Hazard is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Containment Performance**	C	The two biggest LPSD risk reduction measurre are: (1) Having inventory to keep the core covere and to delay steamining inside containment and (2) closing the containment
Containment Performance is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		If there is a hole in the RCS that communicates with containment atmosphere, then containment closure must take place before RCS boiling and steamiing inside containmnent.
RCS Integrity is more important than **Heat Removal**	D	
Heat Removal is more important than **Internal Events Hazard**		
Internal Events Hazard is more important than **Heat Removal**	B	
Heat Removal is more important than **Containment Performance**		Containment Closure is most important for reducing large release
Containment Performance is more important than **Heat Removal**	D	
RCS Integrity is more important than **Internal Events Hazard**	C	The status of the RCS such as whether nozzle dams are open, or the pressurize manway is open, or if a cold leg penetration exists without a large hot leg vent path determines the time to core uncovery and the need for RCS inventory control
Internal Events Hazard is more important than **RCS Integrity**		
RCS Integrity is more important than **Containment Performance**		If the containment is closed then you can handle any RCS Integrity challenge that is encountered.
Containment Performance is more important than **RCS Integrity**	C	
Internal Events Hazard is more important than **Containment Performance**		Containment Closure is most important for reducing large release
Containment Performance is more important than **Internal Events Hazard**	D	

TLC3

Fire Events Core Damage Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**		Heat removal is judged strongly more important than RCS inventory control since fires taking out AC power are likely prominent and it's loss alone causes core damage
Heat Removal is more important than **RCS Inventory Control**	D	
RCS Inventory Control is more important than **RCS Integrity**		Both loss of RCS Integrity and loss of RCS inventory control are needed to result in core damage
RCS Integrity is more important than **RCS Inventory Control**	A	
RCS Inventory Control is more important than **Fire Hazard**		Fire hazard is judged strongly more important than RCS inventory control since fires taking out AC power are likely prominent and it's loss alone causes core damage
Fire Hazard is more important than **RCS Inventory Control**	D	
Heat Removal is more important than **RCS Integrity**	D	Heat removal is judged strongly more important than RCS integrity since fires taking out AC power are likely prominent and it's loss alone causes core damage
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Fire Hazard**		Both fire hazard and loss of heat removal are needed to result in core damage
Fire Hazard is more important than **Heat Removal**	A	
RCS Integrity is more important than **Fire Hazard**		Fire hazard is judged strongly more important than RCS integrity since fires taking out AC power are likely prominent and it's loss alone causes core damage
Fire Hazard is more important than **RCS Integrity**	D	

TLC4		
Fire Events Release Top-Level Criteria Elicitation Form		
Pair-wise Comparison of Two Criteria (X and Y)	**Comparison Result** [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	**Expert Comments**
RCS Inventory Control is more important than **Heat Removal**		RCS Inventory Control has an even stronger impact on Release than on Core Damage.
Heat Removal is more important than **RCS Inventory Control**	A	
RCS Inventory Control is more important than **RCS Integrity**	C	RCS Inventory Control has an even stronger impact on Release than on Core Damage.
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Internal Events Hazard**		RCS Inventory Control has an even stronger impact on Release than on Core Damage.
Internal Events Hazard is more important than **RCS Inventory Control**	A	
RCS Inventory Control is more important than **Containment Performance**		Containment isolation is the ultimate barrier for release, RCS inventory control is also important as it can prevent damage, as well as delay the release allowing for radionuclide decay.
Containment Performance is more important than **RCS Inventory Control**	B	
Heat Removal is more important than **RCS Integrity**	C	Same as TLC1
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Internal Events Hazard**	A	Same as TLC3
Internal Events Hazard is more important than **Heat Removal**		
Heat Removal is more important than **Containment Performance**		Containment isolation is the ultimate barrier for release.
Containment Performance is more important than **Heat Removal**	C	
RCS Integrity is more important than **Internal Events Hazard**		Same as TLC1
Internal Events Hazard is more important than **RCS Integrity**	C	
RCS Integrity is more important than **Containment Performance**		Containment isolation is the ultimate barrier for release.
Containment Performance is more important than **RCS Integrity**	C	
Internal Events Hazard is more important than **Containment Performance**		Containment isolation is the ultimate barrier for release.
Containment Performance is more important than **Internal Events Hazard**	C	

TLC5

Internal Flooding Events Core Damage Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**	C	As long as inventory can be maintained then the core can be cooled by boil off though the high point vent path.
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**	B	This ranking assumes that no event can be postulated where a flooding directly causes a loss of RCS integrity (piping). Internal flooding events are much more likely in the early shutdown of the plant vice during most shutdown POS and outage states.
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Internal Flooding Hazard**	C	Internal flooding events are much more likely in the early shutdown of the plant vice during most shutdown POS and outage states. Most internal flooding contributors are secured when RHR and RCS inventory control are in use during shutdown.
Internal Flooding Hazard is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**	B	This ranking assumes that no event can be postulated where a flooding directly causes a loss of RCS integrity (piping). Internal flooding events are much more likely in the early shutdown of the plant vice during most shutdown POS and outage states.
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Internal Flooding Hazard**	D	Internal flooding events are much more likely in the early shutdown of the plant vice during most shutdown POS and outage states. Most internal flooding contributors are secured when RHR and RCS inventory control are in use during shutdown.
Internal Flooding Hazard is more important than **Heat Removal**		
RCS Integrity is more important than **Internal Flooding Hazard**	B	This ranking assumes that no event can be postulated where a flooding directly causes a loss of RCS integrity (piping). Internal flooding events are much more likely in the early shutdown of the plant vice during most shutdown POS and outage states. A loss of RCS integrity in any case can lead to core dame and release.
Internal Flooding Hazard is more important than **RCS Integrity**		

TLC6

Internal Flooding Events Release Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**		Three means available to remove heat. Even if you have a loss of inventory, heat can be removed via feed and bleed from RWST.
Heat Removal is more important than **RCS Inventory Control**	B	
RCS Inventory Control is more important than **RCS Integrity**		Maintenance of RCS integrtiy will reduce possiblilty of core damage due to loss of inventory resulting in a release.
RCS Integrity is more important than **RCS Inventory Control**	B	
RCS Inventory Control is more important than **Internal Flooding Hazard**		Internal Flooding will possibly impact the systems and component required for inventory control.
Internal Flooding Hazard is more important than **RCS Inventory Control**	B	
RCS Inventory Control is more important than **Containment Performance**	C	If inventory is maintained, likelihood of core damage and resulting release is reduces. Without damage and release containment integrity is not a concern.
Containment Performance is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		Loss if intergrity due to internal flooding can impact ability to remove heat due to loss of inventory.
RCS Integrity is more important than **Heat Removal**	C	
Heat Removal is more important than **Internal Flooding Hazard**		Internal flooding is a direct threat to heat removal systems and components.
Internal Flooding Hazard is more important than **Heat Removal**	C	
Heat Removal is more important than **Containment Performance**		As long as heat removal systems and components are available, containment should not be threatened.
Containment Performance is more important than **Heat Removal**	C	
RCS Integrity is more important than **Internal Flooding Hazard**	C	Loss of RCS integrity could be the direct source of internal flooding.
Internal Flooding Hazard is more important than **RCS Integrity**		
RCS Integrity is more important than **Containment Performance**		Loss of integrity is just one way to to get to core damage. You need contaiment failure to get release.
Containment Performance is more important than **RCS Integrity**	D	
Internal Flooding Hazard is more important than **Containment Performance**	C	As long as internal Flooding issues are addressed and the sources of flooding evaluated and mitigation strategies employed, containment performance should not be challenged.
Containment Performance is more important than **Internal Flooding Hazard**		

TLC7

Seismic Events Core Damage Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**	C	Both RCS Inventory Control and Heat Removal equipment are equally susceptible to earthquakes. Therefore, no change from base case internal events case.
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**		I doubt that the steam generator nozzle dams are seismically qualified and therefore are susceptible to earthquakes. Therefore, I've increased the RCS integrity criteria from the base internal events case.
RCS Integrity is more important than **RCS Inventory Control**	B	
RCS Inventory Control is more important than **Seismic Hazard**	C	I expect the Seismic Hazard to be a couple of orders of magnitude lower than the internal events hazard. Therefore, I've increase the importance of RCS Inventory Control from the base internal events case.
Seismic Hazard is more important than **RCS Inventory Control**		
Heat Removal is more important than **RCS Integrity**		I doubt that the steam generator nozzle dams are seismically qualified and therefore are susceptible to earthquakes. Therefore, I've increased the RCS integrity criteria from the base internal events case.
RCS Integrity is more important than **Heat Removal**	E	
Heat Removal is more important than **Seismic Hazard**		I expect the Seismic Hazard to be a couple of orders of magnitude lower than the internal events hazard. Therefore, I've increase the importance of Heat Removal from the base internal events case.
Seismic Hazard is more important than **Heat Removal**	A	
RCS Integrity is more important than **Seismic Hazard**	B	I doubt that the steam generator nozzle dams are seismically qualified and therefore are susceptible to earthquakes. Plus, the Seismic Hazard is probably lower than the Internal Events Hazard. Therefore, I've increased the RCS integrity criteria from the base internal events case.
Seismic Hazard is more important than **RCS Integrity**		

TLC8

Seismic Events Release Top-Level Criteria Elicitation Form

Pair-wise Comparison of Two Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise criteria comparison]	Expert Comments
RCS Inventory Control is more important than **Heat Removal**	A	These criteria do not have a clear distinction in importance to seismic-Release when their contribution is integrated across all POSs. Differences in contribution to CDF are captured by TLC7
Heat Removal is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **RCS Integrity**	A	No clear distinction - same as "A" above
RCS Integrity is more important than **RCS Inventory Control**		
RCS Inventory Control is more important than **Seismic Hazard**		It is not clear how important Seismic Hazard would be to release, but in may be more important than other criteria (except for Containment Performance).
Seismic Hazard is more important than **RCS Inventory Control**	B	
RCS Inventory Control is more important than **Containment Performance**		Containment performance and time after shutdown (size of the source term) are the primary contributors to the risk of release, give a core melt. This comparison is "C" (and not "D") based on the need for a core melt.
Containment Performance is more important than **RCS Inventory Control**	C	
Heat Removal is more important than **RCS Integrity**	A	No clear distinction - same as "A" above
RCS Integrity is more important than **Heat Removal**		
Heat Removal is more important than **Seismic Hazard**		Seismic Hazard slight importance - same as "B" above
Seismic Hazard is more important than **Heat Removal**	B	
Heat Removal is more important than **Containment Performance**		Containment performance is important to release. Same as "C" above
Containment Performance is more important than **Heat Removal**	C	
RCS Integrity is more important than **Seismic Hazard**		Seismic Hazard slight importance - same as "B" above
Seismic Hazard is more important than **RCS Integrity**	B	
RCS Integrity is more important than **Containment Performance**		Containment performance is important to release. Same as "C" above
Containment Performance is more important than **RCS Integrity**	C	
Seismic Hazard is more important than **Containment Performance**		Both are expected to be important, but a clear distinction is not possible because of the limited experience base.
Containment Performance is more important than **Seismic Hazard**	B	

SC1

Sub-Criteria Elicitation Form

Top-Level Criteria	Pair-wise Comparison of Two Sub-Criteria (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise sub-criteria comparison]	Expert Comments
RCS Inventory Control	**RCS Water Level** is more important than **Availability of Systems to Make-up Inventory**	E	RCS water level determines how quickly make-up inventory must be initiated (midloop versus 23 feet above the refueling flange)
	Availability of Systems to Make-up Inventory is more important than **RCS Water Level**		
Heat Removal	**Heat Load** is more important than **Availability of Reactor Cooling Systems**		In a PWR, each train of RHR is designed for High Heat load. SG heat removal with the SGs in wet layup and a steam removal path (atmosphereic dump valves being available) can handle high heat load.
	Availability of Reactor Cooling Systems is more important than **Heat Load**	D	
RCS Integrity	**RCS Isolation** is more important than **RCS Pressure Relief Capability**		It is important to ensure RCS does not re-pressurize beyond the capability of temporary RCS penetrations (eg nozzle dams) or more than 3 lbs in reactor vessel head with cold leg penetrations(most important to prevent a rapid loss of RCS inventory).
	RCS Pressure Relief Capability is more important than **RCS Isolation**	E	
Internal Events Hazard	**Human Initiated Errors** is more important than **Important Equipment Failures**	E	Operator Initiated Errors will impact the operator's ability to mitigate the error which is risk significant since most recovery actions during LPSD are not automated like full power.
	Important Equipment Failures is more important than **Human Initiated Errors**		
Containment Performance	**Containment Isolation Capability** is more important than **Availability of Radionuclide Suppression Systems**	E	Getting the containment closed prior to RCS boiling and steaming inside containment is the greater challenge
	Availability of Radionuclide Suppression Systems is more important than **Containment Isolation Capability**		
Fire Hazard	**Fire Frequency** is more important than **Vulnerability to Fire Damage**		If a postulated hazard creates a vulnerability resulting in fire damage to core heat removal and the ECCS pumps, then this fire vulnerability has greater risk significance
	Vulnerability to Fire Damage is more important than **Fire Frequency**	E	
Internal Flooding Hazard	**Internal Flooding Frequency** is more important than **Vulnerability to Internal Flooding Damage**		If a postulated hazard creates a vulnerability resulting in flood damage to core heat removal and the ECCS pumps, then this fire vulnerability has greater risk significance
	Vulnerability to Internal Flooding Damage is more important than **Internal Flooding Frequency**	E	
Seismic Hazard	**Seismic Frequency** is more important than **Vulnerability to Seismic Damage**	C	What is the frequency of the lowest seismic acceleration that can cause a loss of the switchyard and failure of ECCS and DHR due to snubber removal and scaffolding? What is the frequeqncy of the lowest seismic acceleration that can cause failure to re-close an open containment and failure of SG nozzle dams?
	Vulnerability to Seismic Damage is more important than **Seismic Frequency**		

SCR1

RCS Water Level Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the RCS water inventory level?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Low Water Level is more important than Medium Water Level	D	Low water level significantly reduces time to respond to an accident, and in some cases affects success criteria.
Low Water Level is more important than High Water Level	E	Low water level significantly reduces time to respond to an accident, and in some cases affects success criteria.
Medium Water Level is more important than High Water Level	D	High water level provides significantly more time for operator response, and effectively reflects successful injection prior to the initiating event.

Ranking Category	Category Definition by Expert
Low Water Level	midloop to six inches below RPV head flange
Medium Water Level	water level in the pressurizer or water solid
High Water Level	refueling cavity flooded

SCR2

Availability of Systems to Make-up Inventory Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the availability of systems to keep the core covered?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Not Very Available is more important than **Available**	B	The fewer number of SI/CCP trains makes it slightly more important but both require manual start
Not Very Available is more important than **Very Available**	D	Requirement for manual start makes it strongly more important than auto start condition.
Available is more important than **Very Available**	D	Requirement for manual start makes it strongly more important than auto start condition.

Ranking Category	Category Definition by Expert
Not Very Available	only 2 of SI and CCP trains available and must be manually started
Available	3 or 4 trains of SI and CCP available and must be manually started
Very Available	Greater than or equal to 2 trains of SI/CCP and automatically started, or cavity full

SCR3

Heat Load Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the heat load from power, decay heat, and RCS temperature?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Load is more important than **Medium Load**	B	Heat load just doesn't cross the boundaries of the heat removal capabilities of any systems so I don't see much of an impact
High Load is more important than **Low Load**	C	
Medium Load is more important than **Low Load**	B	

Ranking Category	Category Definition by Expert
High Load	Ten days or less into an outage
Medium Load	Between the two.
Low Load	1) after 30 days or 2) after 20 days and after core reload

SCR4

Availability of Reactor Cooling Systems Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the availability of reactor cooling systems?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Not Very Available is more important than **Available**	D	Only 1 train RHR without SG cooling is high risk condition; standby RHR can be manually started if only the running train is lost
Not Very Available is more important than **Very Available**	E	Only 1 train RHR without SG cooling is high risk condition; SG cooling is redundant and RHR may still work if RCS depressurization is successful
Available is more important than **Very Available**	C	Second train of RHR provides redundancy but Very available has SG cooling which also has redundancy plus possibility of RCS depressurization for RHR

Ranking Category	Category Definition by Expert
Not Very Available	1 RHR Train running
Available	2 RHR Trains running or 1 running and other in standby with suction MOV closed, or 1 RHR Train aligned for gravity feed
Very Available	SG cooling with or without RHR operating; not crediting AFWPs if only aligned for mini-flow and not injection with relief through just the ARVs

SCR5

RCS Isolation Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the level of vulnerability to maintaining RCS loop isolation?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**	E	These openings can not be quickly closed in response to inventory loss.
Very Vulnerable is more important than **Normal Vulnerability**	E	See above
Vulnerable is more important than **Normal Vulnerability**	C	These opening can be closed relatively quickly with controlling procedures and alignment checklist.

Ranking Category	Category Definition by Expert
Very Vulnerable	RCS opening(s) for maintenance activities that require manual action to close
Vulnerable	RCS opening(s) required by outage procedures. (vents, drains, etc)
Normal Vulnerability	RCS opening(s) that can be isolated from the Control Room

SCR6

RCS Pressure Relief Capability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the level of vulnerability to over-pressurization of the RCS?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**	C	Plant condition challenges COMs with strict control needed for temperature and pressure.
Very Vulnerable is more important than **Normal Vulnerability**	C	Normal shutdown control with RHR utilizing COMS as overpressure protection
Vulnerable is more important than **Normal Vulnerability**	E	Normal configuration overpressurization controls contain tremendous redundancy to protect from overpressure conditions.

Ranking Category	Category Definition by Expert
Very Vulnerable	Solid plant degas and vacuum fill operations.
Vulnerable	COMs in service while RCS is intact and vent path is not established. Head vents available.
Normal Vulnerability	RCS pressurizer pressure control system with back ups from S/G Atmospheric dumps/safety's and RCS PORV's and safety's. When on line, the system is controlled by OP Delta T trips with AMSAC back up.

SCR7

Human Initiated Errors Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the level of opportunity for accident sequences initiated by human errors?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Opportunity for Errors is more important than **Medium Opportunity for Errors**	C	Historically mid-loop POS has seen problems with RCS level control; while maintenance actions unrelated to POS transitions have been the source of errors, its expected these would be amplified when transitions are also underway. RCS level indication not expected to be significant since instrument redundancy has been added over the years and procedures formalized.
High Opportunity for Errors is more important than **Low Opportunity for Errors**	D	Historically mid-loop POS has seen problems with RCS level control; while maintenance actions unrelated to POS transitions have been the source of errors, its expected that they would be less of concern when there are few POS Transitions underway.
Medium Opportunity for Errors is more important than **Low Opportunity for Errors**	B	Maintenance actions have some risk but likely only slight increase as compared to POSs with no maintenance of POS transition steps

Ranking Category	Category Definition by Expert
High Opportunity for Errors	Maintenance on Unprotected train with (appreciable transition activities and POS duration >10 hours); or unplanned maintenance required, or draining to mid-loop; i.e. <192'
Moderate Opportunity for Errors	Maintenance on unprotected train and (no appreciable transition activities or POS duration less than 10 hours)
Low Opportunity for Errors	Both trains protected and (no appreciable transition activities or POS duration less than 10 hours)

SCR8

Important Equipment Failures Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the level of opportunity for accident sequences initiated by equipment failures?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Opportunity for Failures is more important than **Medium Opportunity for Failures**	D	Use appropriate Testing procedures, Subject Matter Expert available.
High Opportunity for Failures is more important than **Low Opportunity for Failures**	D	See Above
Medium Opportunity for Failures is more important than **Low Opportunity for Failures**	B	Seasoned equipment. Maintenance and Operation familar with all aspects of startup and operation

Ranking Category	Category Definition by Expert
High Opportunity for Failures	New equipment being placed in service for first time
Moderate Opportunity for Failures	Equipment being returned to service following maintenance
Low Opportunity for Failures	Reliable equipment good operating record

SCR9

Containment Isolation Capability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the time required to close containment versus time available?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Short Time is more important than **Moderate Time**	D	
Short Time is more important than **Long Time**	D	Less than 30 minutes would make successful containment isolation a major challenge.
Moderate Time is more important than **Long Time**	C	

Ranking Category	Category Definition by Expert
Short Time	Local Action Required and Time Margin < 2
Moderate Time	Local Action Required and Time Margin > 2 but < 10
Long Time	Automatic or manual isolation from MCR or Time Margin > 10; Time Margin = TimeAvailable / TimeRequired

SCR10

Availability of Radionuclide Suppression Systems Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the availability of radionuclide suppression systems?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Not Very Available is more important than Available	C	In general, Containment Spray Injection has limited availablity during the outage.
Not Very Available is more important than Very Available	D	Very available can be orders of magnitude improvement in unavailability.
Available is more important than Very Available	B	Once CS is available, the HEP is the limiting factor over the redundancy in the number of trains available.

Ranking Category	Category Definition by Expert
Not Very Available	Typically, only 1 train available, powered by an EDG; short time available for operator response. UA at or above 0.1. Radionuclide removal is primarily Containment Spray, and during the refueling outage CS injection is unavailable when the RWST is empty; and recirculation is unavailable when the sump covers are installed.
Available	1 train available, powered by offsite power; short time available for operator response. UA ~E-2. Radionuclide removal is primarily Containment Spray, and during the refueling outage CS injection is available when the RWST is full; and recirculation is available when the sump covers are removed.
Very Available	2 or more trains available, powered by offsite power; longer time available for operator response. UA at or below E-3. Radionuclide removal is primarily Containment Spray, and during the refueling outage CS injection is available when the RWST is full; and recirculation is available when the sump covers are removed.

SCR11

Fire Frequency Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the fire frequency for this plant configurations and set of activities?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Frequency is more important than **Medium Frequency**	C	I have no clue what the mean frequency is or the shape of the distribution. Therefore, it is very hard for me to add much input here.
High Frequency is more important than **Low Frequency**	E	I have no clue what the mean frequency is or the shape of the distribution. Therefore, it is very hard for me to add much input here.
Medium Frequency is more important than **Low Frequency**	C	I have no clue what the mean frequency is or the shape of the distribution. Therefore, it is very hard for me to add much input here.

Ranking Category	Category Definition by Expert
High Frequency	Ten time increase on mean frequency
Moderate Frequency	Mean frequency
Low Frequency	One tenth of mean frequency

SCR12

Fire Damage Vulnerability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the chance that fire damage initiates an accident sequence?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**	C	Extensive maintenance during refueling may have multiple fire barriers disabled
Very Vulnerable is more important than **Normal Vulnerability**	D	Expect maintenance during refueling would involve more fire barrier degradations than for unscheduled outages
Vulnerable is more important than **Normal Vulnerability**	B	Unscheduled maintenance events are not expected to affect many fire barriers while fuel is in vessel

Ranking Category	Category Definition by Expert
Very Vulnerable	Fire barriers degraded for maintenance
Vulnerable	Limited barriers degraded
Normal Vulnerability	No barriers degraded

SCR13

Internal Flooding Frequency Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the internal flooding event frequency for this plant configurations and set of activities?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Frequency is more important than **Medium Frequency**	C	Large bore pipe flooding is much more severe that a postulated nozzle dam failure. However, failures involving containment flooding can have adverse effects and reduce margin to core damage.
High Frequency is more important than **Low Frequency**	D	Flooding due to a large bore piping system has occurred at multiple US and International facilities. These systems are removed from service when shut down.
Medium Frequency is more important than **Low Frequency**	B	RCS intact with minimal fluid system support in operation is the least likely flood scenario at a power plant.

Ranking Category	Category Definition by Expert
High Frequency	Full large bore piping fluid systems in operation to support plant power operations.
Moderate Frequency	Cavity flooded/mid loop operations and RCS heat up/cooldown fluid system operations.
Low Frequency	Plant fluid systems predominantly secured with the RCS intact/head tensioned.

SCR14

Internal Flooding Damage Vulnerability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the chance that internal flooding damage initiates an accident sequence?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**	C	Accident sequence occurrence higher in this condition.
Very Vulnerable is more important than **Normal Vulnerability**	D	Accident sequence occurrence much higher in this condition.
Vulnerable is more important than **Normal Vulnerability**	B	Accident sequence occurrence much lower in this condition.

Ranking Category	Category Definition by Expert
Very Vulnerable	If numerous systems being drained for outage activities
Vulnerable	If several systems being drained for outage activities
Normal Vulnerability	Only one system being drained for maintenance activities

SCR15

Seismic Frequency Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the seismic event frequency for this plant?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
High Frequency is more important than **Medium Frequency**	C	High frequency is roughly 2 times the moderate frequency due to effective POS duration
High Frequency is more important than **Low Frequency**	D	High frequency is roughly 5 times or greater than low frequency POS durations
Medium Frequency is more important than **Low Frequency**	C	Medium frequency is roughly 3 times or greater than low frequency POS durations

Ranking Category	Category Definition by Expert
High Frequency	POS duration >48 hours
Moderate Frequency	POS duration 10 -48 hours
Low Frequency	POS duration 0-10 hours

SCR16

Seismic Damage Vulnerability Sub-Criteria Ranking Elicitation Form

Evaluation Question: What is the chance that seismic damage initiates an accident sequence?

Pair-wise Comparison of Two Categories (X and Y)	Comparison Result [Enter A, B, C, D, or E in just one of the two rows for each pair-wise category comparison]	Expert Comments
Very Vulnerable is more important than **Vulnerable**	C	
Very Vulnerable is more important than **Normal Vulnerability**	C	Moderate distinction is expected with RCS not intact, since that includes a number of plant configurations that are different from at-power.
Vulnerable is more important than **Normal Vulnerability**	B	

Ranking Category	Category Definition by Expert
Very Vulnerable	RHR in service with RCS not intact, additional deadweight loads (tank water levels, lead blankets, etc)
Vulnerable	RHR in service with RCS intact
Normal Vulnerability	Plant configured similar to Mode 1

Plant Operating State (POS) Importance Ranking Elicitation Form (PR1)

POS		Technical Specification (TS)	POS Duration (hours)	Evaluation Criteria Rank															
No.	Description	TS Mode	Range over POTs	Heat Load	Availability of Systems to Make-up RCS Inventory	RCS Water Level	Availability of Reactor Cooling Systems	RCS Isolation	RCS Pressure Relief Capability	Human Initiated Errors	Important Equipment Failures	Containment Isolation Capability	Availability of Radionuclide Suppression Systems	Fire Frequency	Vulnerability to Fire Damage	Internal Flooding Frequency	Vulnerability to Internal Flooding Damage	Seismic Frequency	Vulnerability to Seismic Damage
				SCR3	SCR2	SCR1	SCR4	SCR5	SCR6	SCR7	SCR8	SCR9	SCR10	SCR11	SCR12	SCR13	SCR14	SCR15	SCR16
				High Load (H) Medium Load (M) Low Load (L)	Not Very Available (H) Available (M) Very Available (L)	Low Water Level (H) Medium Water Level (M) High Water Level (L)	Not Very Available (H) Available (M) Very Available (L)	Not Very Available (H) Available (M) Very Available (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)	High Opportunity (H) Moderate Opportunity (M) Low Opportunity (L)	High Opportunity (H) Moderate Opportunity (M) Low Opportunity (L)	Short Time (H) Moderate Time (M) Long Time (L)	Not Very Available (H) Available (M) Very Available (L)	High Frequency (H) Moderate Frequency (M) Low Frequency (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)	High Frequency (H) Moderate Frequency (M) Low Frequency (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)	High Frequency (H) Moderate Frequency (M) Low Frequency (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)
1	Low power and reactor shutdown	1,2	6	H	L	M	L	L	L	L	M	H	L	L	L	M	L	L	L
			Expert Comments:	Max Decay Heat	All ECCS systems and charging available.	Normal inventory	All normal supplies for feedwater, aux feedwater, and ECCS/charging are available.	No Challenges to RCS integrity.	Normal pressure control via pressurizer heaters/spray and back up PORV's and safety's.	Activities bounded within normally utilized procedures. JIT training conducted for the shutdown/cooldown crew.	Typical issues during this transition period are associated with secondary plant challenges.	No changes to normal purge/cooling or containment spray.	Containment spray and recirc sumps are available.	Normal Appendix R procedures in use.	Normal processes and control of combustible materials and ignition sources in full effect.	Leakage of large bore plant equipment is most likely during large plant pressure and temperature transients.	Plant flood barriers and processes are in full effect.	Seismic Frequency is a constant that is not affected by outage activities.	Normal plant seismic restraints fully operable.
2	Cooldown with steam generators to 350°F	3	5	H	L	M	L	L	L	L	L	H	L	L	L	M	L	L	L
			Expert Comments:	Max Decay Heat	All ECCS systems and charging available.	Normal inventory	All normal supplies for feedwater, aux feedwater, and ECCS/charging are available.	No Challenges to RCS integrity.	Normal pressure control via pressurizer heaters/spray and back up PORV's and safety's.	Activities bounded within normally utilized procedures. JIT training conducted for the shutdown/cooldown crew.	Typical equipment failures in this state generally involve instrumentation such as Nis or SSPS permissives and coincidence circuits.	Containment purge and cooling fans remain in service but Containment Spray is defeated while cooling down.	Containment Spray is defeated while cooling down.	Normal Appendix R procedures in use.	Normal processes and control of combustible materials and ignition sources in full effect.	Leakage of large bore plant equipment is most likely during large plant pressure and temperature transients.	Plant flood barriers and processes are in full effect.	Seismic Frequency is a constant that is not affected by outage activities.	Normal plant seismic restraints fully operable.
2-P1	Cooldown with steam generators to 350°F	3	16	H	L	M	L	L	L	L	L	H	L	L	L	M	M	L	L
			Expert Comments:	Max Decay Heat	All ECCS systems and charging available.	Normal inventory	All normal supplies for feedwater, aux feedwater, and ECCS/charging are available. Note that ECCS SI is isolated during this evolution.	No Challenges to RCS integrity.	Normal pressure control via pressurizer heaters/spray and back up PORV's and safety's.	Activities bounded within normally utilized procedures. JIT training conducted for the shutdown/cooldown crew.	Typical equipment failures in this state generally involve instrumentation such as Nis or SSPS permissives and coincidence circuits.	Containment purge and cooling fans remain in service but Containment Spray is defeated while cooling down.	Containment Spray is defeated while cooling down.	Normal Appendix R procedures in use.	Normal processes and control of combustible materials and ignition sources in full effect.	Leakage of large bore plant equipment is most likely during large plant pressure and temperature transients.	Plant flood barriers and processes are in full effect.	Seismic Frequency is a constant that is not affected by outage activities.	Normal plant seismic restraints fully operable.
3	Cooldown with RHR system to 200°F	4	3	H	M	M	M	L	M	M	M	H	H	L	M	L	M	L	M
			Expert Comments:	Max Decay Heat. Most significant challenge to the RHR pumps, Heat Exchangers and Ultimate Heat Sink.	ECCS bypassed. Charging tagged out when RCP is secured. Inventory control via RHR, Purification/cleanup and the blender via chem add.	Normal inventory	The S/Gs are still available for use with existing inventory although aux. feedwater is isolated.	No Challenges to RCS integrity.	Coms is placed into service prior to establishing a high point vent to atmosphere.	Transfer of cooling from the S/Gs to RHR is a complex and infrequently performed evolution. Opportunities for error are elevated.	Failures in this area will be associated with RHR flow and pressure control. Off site power and diesel testing also occur during this window.	Containment mobilization is fully underway with the personnel airlock interlocks bypassed and the equipment hatch removed.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	Containment and critical work areas mobilizing but ignition sources are not yet in use.	Breach of fire barriers begins with compensatory posting as required..	Large bore cooling systems such as Circulation Water, Open and Closed Loop Turbine Cooling are secured reducing the possibility of flooding.	Flood barriers are degraded as large bore systems are removed from service following Mode 4 entry.	Seismic Frequency is a constant that is not affected by outage activities.	Small snubber removal and hanger/support removal begin. Plant seismic integrity is impacted.
4	Cooldown to ambient temperature with RHR only	5	4	H	M	M	M	L	M	M	L	H	H	M	M	L	M	L	M
			Expert Comments:	Max Decay Heat. Most significant challenge to the RHR pumps, Heat Exchangers and Ultimate Heat Sink.	ECCS bypassed. Charging tagged out when RCP is secured. Inventory control via RHR, Purification/cleanup and the blender via chem add.	Normal inventory	Steam generators are no longer capable of removing heat. Completely reliant on RHR.	No Challenges to RCS integrity.	Coms is placed into service prior to establishing a high point vent to atmosphere.	Transfer of cooling from the S/Gs to RHR is a complex and infrequently performed evolution. Opportunities for error are elevated.	Failures in this area will be associated with RHR flow and pressure control. Off site power and diesel testing also occur during this window.	Containment mobilization is fully underway with the personnel airlock interlocks bypassed and the equipment hatch removed.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	Hot work begins in earnest and ignition sources are introduced across the site. This will remain constant through POS 13.	Breach of fire barriers begins with compensatory posting as required..	In addition to the large bore systems previously mentioned, portions of CCW, all of ECCS and most of CVCS flow paths are eliminated.	Flood barriers are now fully degraded for equipment movement.	Seismic Frequency is a constant that is not affected by outage activities.	Small snubber removal and hanger/support removal begin. Plant seismic integrity is impacted.
4-P2	Cooldown to ambient temperature with RHR only	5	116	H	M	M	M	L	M	M	L	H	H	M	M	L	M	L	M
			Expert Comments:	Max Decay Heat. Most significant challenge to the RHR pumps, Heat Exchangers and Ultimate Heat Sink.	ECCS bypassed. Charging tagged out when RCP is secured. Inventory control via RHR, Purification/cleanup and the blender via chem add.	Normal inventory	Steam generators are no longer capable of removing heat. Completely reliant on RHR.	No Challenges to RCS integrity.	Coms is placed into service prior to establishing a high point vent to atmosphere.	Transfer of cooling from the S/Gs to RHR is a complex and infrequently performed evolution. Opportunities for error are elevated.	Failures in this area will be associated with RHR flow and pressure control. Off site power and diesel testing also occur during this window.	Containment mobilization is fully underway with the personnel airlock interlocks bypassed and the equipment hatch removed.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	Hot work begins in earnest and ignition sources are introduced across the site. This will remain constant through POS 13.	Breach of fire barriers begins with compensatory posting as required..	No change	No change	Seismic Frequency is a constant that is not affected by outage activities.	Small snubber removal and hanger/support removal begin. Plant seismic integrity is impacted.
5A	Pressurizer water solid for degassing	5	20-30	H	H	M	M	L	M	M	L	H	H	M	M	L	M	L	M
			Expert Comments:	Continued high heat rejection rate but decaying.	All high pressure sources secured and tagged. COMs instated. Make up extremely limited.	Slightly greater than normal inventory.	Steam generators are no longer capable of removing heat. Completely reliant on RHR.	No Challenges to RCS integrity.	Coms is placed into service prior to establishing a high point vent to atmosphere.	Another infrequently performed evolution coupled with placing COMS in service. Opportunities for error are increased.	COMS and head vents are available.	Containment mobilization is fully underway with the personnel airlock interlocks bypassed and the equipment hatch removed.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	Hot work begins in earnest and ignition sources are introduced across the site. This will remain constant through POS 13.	Breach of fire barriers begins with compensatory posting as required..	No change	No change	Seismic Frequency is a constant that is not affected by outage activities.	Small snubber removal and hanger/support removal begin. Plant seismic integrity is impacted.
5B	Draining the RCS to reduce inventory, RCS is vented	5	20-40	H	M	H	M	M	L	M	M	H	H	M	M	L	M	L	M
			Expert Comments:	Worst case situation for RHR where the highest flow rate is needed due to high decay heat and RCS level approaches vortexing of RHR at mid-loop.	Normal low pressure make up capabilities.	Inventory reduced and heading toward minimum coverage over active fuel.	RHR only.	High point atmospheric vent established.	RCS is continuously vented to atmosphere.	Typical errors during this evolution are associated with valving in and monitoring vessel level and RHR flow.	Proper performance of Reactor Vessel level instrumentation is critical in this evolution. Note that the equipment hatch closure hoist operability is critical in this state through POS 13.	RCS vent path is established and containment is open. Only RHR cooling and time to boil stand in the way of an un-monitored release.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	No change	No change	Seismic Frequency is a constant that is not affected by outage activities.	Large structural snubber removal for testing begins when the RCS reaches "cold iron" condition. The plant is at minimum seismic integrity.
6	Mid-loop operation prior to refueling	5	14-20	H	M	H	M	H	L	M	L	H	H	L	M	L	M	L	M
			Expert Comments:	Worst case situation for RHR where the highest flow rate is needed due to high decay heat and RCS level approaches vortexing of RHR at mid-loop.	Normal low pressure make up capabilities.	Worst case procedurally driven RCS water level.	RHR only.	Worst case scenario for loss of inventory with S/G manways off and no nozzle dams installed.	RCS is continuously vented to atmosphere.	Operator control of RCS flow is critical during this process.	RCS flow control failure can result in air binding.	RCS vent path is established and containment is open. Only RHR cooling and time to boil stand in the way of an un-monitored release.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	No change	No change	Seismic Frequency is a constant that is not affected by outage activities.	Large structural snubber removal for testing begins when the RCS reaches "cold iron" condition. The plant is at minimum seismic integrity.
6-P3	Mid-loop operations	5	greater than 20	H	M	H	M	H	L	M	L	H	H	L	M	L	M	L	M

Plant Operating State (POS) Importance Ranking Elicitation Form (PR1)

POS		Technical Specification (TS)	POS Duration (hours)	Evaluation Criteria Rank															
				Heat Load	Availability of Systems to Make-up RCS Inventory	RCS Water Level	Availability of Reactor Cooling Systems	RCS Isolation	RCS Pressure Relief Capability	Human Initiated Errors	Important Equipment Failures	Containment Isolation Capability	Availability of Radionuclide Suppression Systems	Fire Frequency	Vulnerability to Fire Damage	Internal Flooding Frequency	Vulnerability to Internal Flooding Damage	Seismic Frequency	Vulnerability to Seismic Damage
No.	Description	TS Mode	Range over POTs	SCR3	SCR2	SCR1	SCR4	SCR5	SCR6	SCR7	SCR8	SCR9	SCR10	SCR11	SCR12	SCR13	SCR14	SCR15	SCR16
				High Load (H) Medium Load (M) Low Load (L)	Not Very Available (H) Available (M) Very Available (L)	Low Water Level (H) Medium Water Level (M) High Water Level (L)	Not Very Available (H) Available (M) Very Available (L)	Not Very Available (H) Available (M) Very Available (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)	High Opportunity (H) Moderate Opportunity (M) Low Opportunity (L)	High Opportunity (H) Moderate Opportunity (M) Low Opportunity (L)	Short Time (H) Moderate Time (M) Long Time (L)	Not Very Available (H) Available (M) Very Available (L)	High Frequency (H) Moderate Frequency (M) Low Frequency (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)	High Frequency (H) Moderate Frequency (M) Low Frequency (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)	High Frequency (H) Moderate Frequency (M) Low Frequency (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)
			Expert Comments:	Worst case situation for RHR where the highest flow rate is needed due to high decay heat and RCS level approaches vortexing of RHR at mid-loop.	Normal low pressure make up capabilities.	Worst case procedurally driven RCS water level.	RHR only.	Worst case scenario for loss of inventory with S/G manways off and no nozzle dams installed.	RCS is continuously vented to atmosphere.	Operator control of RCS flow is critical during this process.	RCS flow control failure can result in air binding.	RCS vent path is established and containment is open. Only RHR cooling and time to boil stand in the way of an un-monitored release.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	No change	No change	Seismic Frequency is a constant that is not affected by outage activities.	Large structural snubber removal for testing begins when the RCS reaches "cold iron" condition. The plant is at minimum seismic integrity.
7	Filling refueling cavity for refueling operation	6	45	H	M	L	M	M	L	L	L	M	H	M	M	L	M	L	M
			Expert Comments:	Continued high heat rejection rate but decaying.	Normal low pressure make up capabilities.	Risk due to inventory greatly reduced as water level increases to 23' over the vessel flange.	RHR only.	Exposure due to nozzle dams and cavity seal. Note that nozzle dams have compensatory measures.	RCS is continuously vented to atmosphere.	Operator actions are procedurally controlled with multiple inputs and communications in place to prevent cavity overfill.	No significant failure mechanisms are present.	Containment is open. Only RHR cooling and time to boil stand in the way of an un-monitored release.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	Hot work resumes and ignition sources are introduced across the site. This will remain constant through POS 13.	Vulnerability is increased as safety train backup availability is typically reduced by scheduled bus outages.	No change	During flood up activities with nozzle dams installed, the possibility for dam failure and the loss of a large volume of the cavity with associated lower containment flooding exists.	Seismic Frequency is a constant that is not affected by outage activities.	Snubber removal and installation continue. Plant is still at minimum seismic integrity.
8E	Refueling operation (offloading old core)	6	68	M	M	L	M	M	L	M	L	M	H	M	M	L	M	L	M
			Expert Comments:	Continued high heat rejection rate but decaying. Removal of thrice burned assemblies to the fuel pool also diminishes the heat load.	Normal low pressure make up capabilities.	23" minimum over the flange. High time to boil with maximum margin to fuel damage in mode 6.	RHR with entire cavity inventory available for boil off.	Exposure due to nozzle dams and cavity seal. Note that nozzle dams have compensatory measures.	RCS is continuously vented to atmosphere.	All actions involving the potential for operator error surround reactivity management.	All refueling equipment is "fail safe". No significant equipment malfunctions can affect the core during this state.	Containment is open. Only RHR cooling and time to boil stand in the way of an un-monitored release.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	Hot work continues and ignition sources are introduced across the site. This will remain constant through POS 13.	Vulnerability is increased as safety train backup availability is typically reduced by scheduled bus outages.	No change	During flood up activities with nozzle dams installed, the possibility for dam failure and the loss of a large volume of the cavity with associated lower containment flooding exists.	Seismic Frequency is a constant that is not affected by outage activities.	Snubber removal and installation continue. Plant is still at minimum seismic integrity.
DF	Defueled	n/a	120																
8L	Refueling operation (loading new core)	6	52	L	M	L	M	M	L	M	L	L	H	M	M	L	M	L	M
			Expert Comments:	Hat load has greatly diminished due to decay of shorter lived radionuclides coupled with removal of 1/3 of the active fuel and addition of 1/3 core new assemblies.	Normal low pressure make up capabilities.	23" minimum over the flange. High time to boil with maximum margin to fuel damage in mode 6.	RHR with entire cavity inventory available for boil off.	Exposure due to nozzle dams and cavity seal. Note that nozzle dams have compensatory measures.	RCS is continuously vented to atmosphere.	All actions involving the potential for operator error surround reactivity management.	All refueling equipment is "fail safe". No significant equipment malfunctions can affect the core during this state.	Containment is open. Only RHR cooling and time to boil stand in the way of an un-monitored release.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	Hot work continues and ignition sources are introduced across the site. This will remain constant through POS 13.	Train outage work is usually complete immediately prior to core reload so vulnerability is reduced although breeches still exist.	No change	During flood up activities with nozzle dams installed, the possibility for dam failure and the loss of a large volume of the cavity with associated lower containment flooding exists.	Seismic Frequency is a constant that is not affected by outage activities.	Snubber removal and installation continue. Plant is still at minimum seismic integrity.
9	Draining the RCS refueling operation	6	120	L	M	H	M	M	L	M	L	M	H	M	M	L	M	L	M
			Expert Comments:	Challenging situation for RHR but required flow rates are significantly reduced and the time to boil is much higher should the RHR train be lost.	Normal low pressure make up capabilities.	Worst case procedurally driven RCS water level.	RHR only.	Exposure due to nozzle dams and cavity seal. Note that nozzle dams have compensatory measures.	RCS is continuously vented to atmosphere.	Operator actions are procedurally controlled with multiple inputs and communications in place to monitor and control drain down.	RHR flow control.	Although time to boil is much longer, containment is still vulnerable to the need to close up in a short time period.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	No change	During flood up activities with nozzle dams installed, the possibility for dam failure and the loss of a large volume of the cavity with associated lower containment flooding exists.	Seismic Frequency is a constant that is not affected by outage activities.	Snubber removal and installation continue. Plant is still at minimum seismic integrity.
10	Mid-loop operation after refueling	5,6	25	L	M	H	M	H	L	M	L	M	H	M	M	L	M	L	M
			Expert Comments:	Challenging situation for RHR but required flow rates are significantly reduced and the time to boil is much higher should the RHR train be lost.	Normal low pressure make up capabilities.	Worst case procedurally driven RCS water level.	RHR only.	Worst case scenario for loss of inventory with S/G manways off and no nozzle dams installed.	RCS is continuously vented to atmosphere.	Operator control of RCS flow is critical during this process.	RHR flow control.	Although time to boil is much longer, containment is still vulnerable to the need to close up in a short time period.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	As with most plants, administrative controls are implemented to stop work, review and re-authorize activities which pose no threat to mid loop support. Hot work is typically suspended with the exception of the turbine Building and certain yard areas.	No change	After manway installation, any threat of flooding from the RCS is greatly reduced.	Seismic Frequency is a constant that is not affected by outage activities.	Snubber removal and installation continue. Plant is still at minimum seismic integrity.
11	Refill RCS, reactor vents are closed	5,6	20-24	L	M	M	M	M	L	M	L	L	H	M	M	L	M	L	M
			Expert Comments:	Minimum RHR heat load with normal plant intact.	Normal low pressure make up capabilities.	Normal inventory.	RHR only.	Manways installed with high point vent open.	Coms instated. High point vent isolated during this evolution.	Operator errors involving initiation of COMS can result in overpressure. Also, lack of familiarity with vacuum fill processes can cause issues.	RHR flow control.	Low decay heat and long time to boil.	Containment Spray is defeated. Letdown and CVCS cleanup are secured. Purification system is available.	Hot work begins in earnest and ignition sources are introduced across the site. This will remain constant through POS 13.				Seismic Frequency is a constant that is not affected by outage activities.	Snubber removal and installation continue. Plant is still at minimum seismic integrity.
12	RCS heatup/draw	5	48-54	L	M	M	M	L	M	M	L	L	H	L	M	L	M	L	L

Plant Operating State (POS) Importance Ranking Elicitation Form (PR1)

POS		Technical Specification (TS)	POS Duration (hours)	Evaluation Criteria Rank															
No.	Description	TS Mode	Range over POTs	Heat Load	Availability of Systems to Make-up RCS Inventory	RCS Water Level	Availability of Reactor Cooling Systems	RCS Isolation	RCS Pressure Relief Capability	Human Initiated Errors	Important Equipment Failures	Containment Isolation Capability	Availability of Radionuclide Suppression Systems	Fire Frequency	Vulnerability to Fire Damage	Internal Flooding Frequency	Vulnerability to Internal Flooding Damage	Seismic Frequency	Vulnerability to Seismic Damage
				SCR3	SCR2	SCR1	SCR4	SCR5	SCR6	SCR7	SCR8	SCR9	SCR10	SCR11	SCR12	SCR13	SCR14	SCR15	SCR16
				High Load (H) Medium Load (M) Low Load (L)	Not Very Available (H) Available (M) Very Available (L)	Low Water Level (H) Medium Water Level (M) High Water Level (L)	Not Very Available (H) Available (M) Very Available (L)	Not Very Available (H) Available (M) Very Available (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)	High Opportunity (H) Moderate Opportunity (M) Low Opportunity (L)	High Opportunity (H) Moderate Opportunity (M) Low Opportunity (L)	Short Time (H) Moderate Time (M) Long Time (L)	Not Very Available (H) Available (M) Very Available (L)	High Frequency (H) Moderate Frequency (M) Low Frequency (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)	High Frequency (H) Moderate Frequency (M) Low Frequency (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)	High Frequency (H) Moderate Frequency (M) Low Frequency (L)	Very Vulnerable (H) Vulnerable (M) Normal (L)
	bubble in pressurizer		Expert Comments:	Minimum RHR heat load with normal plant intact.	Make up shifted from low pressure control to charging system via seal injection. Charging and letdown still isolated but available.	Normal inventory.	RHR only.	RCS integrity is intact.	Pressure relief capability shifted back to PORV's and Pressurizer.	Infrequent evolution for plant operators. This evolution is typically covered in simulator under JIT training	Typical failures in this POS include RCP Seals, seal injection, and instrumentation errors associated with improper fill and venting.	Low decay heat and long time to boil.	CVCS letdown demins are placed in service late in this evolution.	Hot work is concluding and breeches are in closure and ignition sources are diminishing across the site.	Normal Appendix R procedures are still not in place and compensatory measures are still in effect.	Restoration of CCW, CVCS and other fluid systems in containment increases the possible frequency, particularly if these systems have been overhauled during the outage.	Flood barriers are beginning to be reinstalled and the vulnerability diminishes.	Seismic Frequency is a constant that is not affected by outage activities.	All structural snubbers are installed with small snubber and hanger installation in the process of completion.
13	RCS heatup to 350°F	4	20	L	M	M	M	L	M	M	M	M	M	L	L	L	L	L	L
			Expert Comments:	Minimum RHR heat load with normal plant intact.	Make up shifted from low pressure control to charging system via seal injection. Charging and letdown still isolated but available.	Normal inventory.	Charging restored. Aux. Feedwater testing is conducted and steam generators are available if needed utilizing atmospheric steam dumps.	Normal plant integrity	Normal pressure control.	See comment above concerning JIT training	Typical issues are associated with instrumentation.	Containment closure is in progress during this state.	CVCS demins are in service. Purification is secured.	Hot work is concluded. Normal plant procedures for Appendix R are placed into use.	All barriers are typically restored prior to Mode 4 entry and are required to be in effect in Mode 3.	Large bore systems are not yet started up but all containment support fluid systems are fully operational.	Barriers are reinstalled and tested by 350 degrees.	Seismic Frequency is a constant that is not affected by outage activities.	Normal plant seismic restraints fully operable.
14	Startup with steam generators to Hot Standby	3	40	L	L	M	L	L	L	L	M	L	L	L	L	M	L	L	L
			Expert Comments:	Lowest heat load of the outage.	All ECCS systems restored. Normal charging and letdown in service.	Normal inventory.	All mechanisms available with the exception of normal feedwater which is precluded by temperature interlocks.	Normal integrity	Normal RCS pressure control and relief capability.	Returning to normal operating procedures coupled with the "start up" crew attending JIT simulator training.	S/G atmospheric dump valves and AFW flow control issues pose the greatest threat to heat removal. Additionally, ECCS functions are restored on the mode 4 to 3 transition.	ECCS is restored including containment isolation capability.	Containment spray is restored.	Normal Appendix R procedures in effect.	Normal Appendix R procedures in effect.	Large bore fluid systems return to service and the plant is in normal configuration.	All normal flood barriers and processes are in effect.	Seismic Frequency is a constant that is not affected by outage activities.	Normal plant seismic restraints fully operable.
15A	Reactor startup and low power operation (0=Power<5%)	2	20	M	L	M	L	L	L	L	M	H	L	L	L	M	L	L	L
			Expert Comments:	Heat load begins to increase.	Normal plant on-line make up capability.	Normal inventory.	All mechanisms available including normal feedwater dumping to the main condenser.	Normal integrity	Normal RCS pressure control and relief capability.	Returning to normal operating procedures coupled with the "start up" crew attending JIT simulator training.	Always a critical exposure for operators. Compensatory measures include JIT training on simulator, second party concurrence on all actions and independent review of startup pull/dilution data in progress.	ECCS is restored including containment isolation capability.	Containment spray is restored.	Normal Appendix R procedures in effect.	Normal Appendix R procedures in effect.	Large bore fluid systems return to service and the plant is in normal configuration.	All normal flood barriers and processes are in effect.	Seismic Frequency is a constant that is not affected by outage activities.	Normal plant seismic restraints fully operable.
15B	Reactor startup and low power operation (5<Power<50 %)	1,2	20-36	M	L	M	L	L	L	L	M	H	L	L	L	M	L	L	L
			Expert Comments:	Heat load begins to increase.	Normal plant on-line make up capability.	Normal inventory.	All mechanisms available including normal feedwater dumping to the main condenser.	Normal integrity	Normal RCS pressure control and relief capability.	Returning to normal operating procedures coupled with the "start up" crew attending JIT simulator training.	Always a critical exposure for operators. Compensatory measures include JIT training on simulator, second party concurrence on all actions and independent review of startup pull/dilution data in progress.	ECCS is restored including containment isolation capability.	Containment spray is restored.	Normal Appendix R procedures in effect.	Normal Appendix R procedures in effect.	Large bore fluid systems return to service and the plant is in normal configuration.	All normal flood barriers and processes are in effect.	Seismic Frequency is a constant that is not affected by outage activities.	Normal plant seismic restraints fully operable.

LOK1

Self Evaluation of Level of Knowledge

Form	Description of Form	Level (H, M, L)	Comments
TCL1	Comparison of TLC for Internal Events Core Damage	H	This risk is generally understood, although there are aspects that have not been fully investigated.
TCL2	Comparison of TLC for Internal Events Core Release	H	This risk is generally understood, although there are aspects that have not been fully investigated. This risk has the benefit of both the decay heat and source term decreasing with increasing time after shutdown, so the back-end of long outages tend to be no be so important to risk.
TCL3	Comparison of TLC for Fire Core Damage	M	Fire risk at LPSD is the intersection of two complicated analyses. I am familiar with fire risk at power and somewhat with fire frequency and fire protection controls at shutdown, so it is possible to make some judgments about fire risk at shutdown.
TCL4	Comparison of TLC for Fire Core Release	M	
TCL5	Comparison of TLC for Internal Flooding Core Damage	M	Flood risk is somewhat less complicated than fire and the at-power flood risk can give a good picture of flood risk at shutdown. However, we don't have a good handle on the increase in frequency that may occur in some POSs and the potential for flood barriers to be removed in a way that would not allow prompt reinstallation (e.g., a door held open).
TCL6	Comparison of TLC for Internal Flooding Core Release	M	
TCL7	Comparison of TLC for Seismic Events Core Damage	M	Seismic risk is expected to be not significant because of (a) the short duration of most POSs and (b) the Seismic-Cat 1 equipment that is performing a function during shutdown. However, there are short-duration events (e.g., reactor head on the polar crane) where the conditional risk may be higher. Also, the change in tank levels may change the tank fragility analysis.
TCL8	Comparison of TLC for Sismic Events Core Release	M	
SCL1	Sub-Criteria Elicitaion Form	H	
SCR1	RCS Water Level Ranking	H	
SCR2	Availability of Systems to Make-up Inventory Ranking	H	
SCR3	Heat Load Ranking	H	
SCR4	Availability of Reactor Cooling Systems Ranking	H	
SCR5	RCS Isolation Ranking	H	
SCR6	Pressure Relief Capability Ranking	H	
SCR7	Containment Isolation Capability	H	
SCL8	Availability of Radionuclide Suppression Systems Ranking	H	
SCR9	Operator Initiated Events	H	
SCR10	Important Equipment Failures	H	
SCR11	Fire Frequency	M	
SCR12	Fire Damage Vulnerability	M	
SCR13	Internal Flooding Event Frequency	M	
SCR14	Internal Flooding Damage Vulnerability	M	
SCR15	Seismic Event Frequency	M	
SCR16	Seismic Damage Vulnerability	M	
PR1	POS Importance Ranking Elicitation Form	H	

APPENDIX K RESULTS OF GROUP ELICITATION SESSION

This appendix provides the compiled results from the completed individual elicitation forms provided by each expert after the group elicitation meeting.

Section K.1 provides the aggregated results for all experts (geometric mean) and comparative results that show how each experts' input compared to each other. The following diagrams/figures are included in section K.1:

- One diagram of aggregated POS priorities for all of the goals (one page).
- Eight diagrams comparing final POS priorities for each expert – one for each goal (eight pages).
- Eight figures showing the spread of the calculated weights for each expert for all of the top-level criteria – one for each goal (eight pages).
- Eight figures showing the spread of the calculated weights for each expert for all of the sub-criteria – one for each top-level criteria (two pages).
- Eight diagrams comparing POS priorities for each sub-criteria – one for each goal (eight pages).

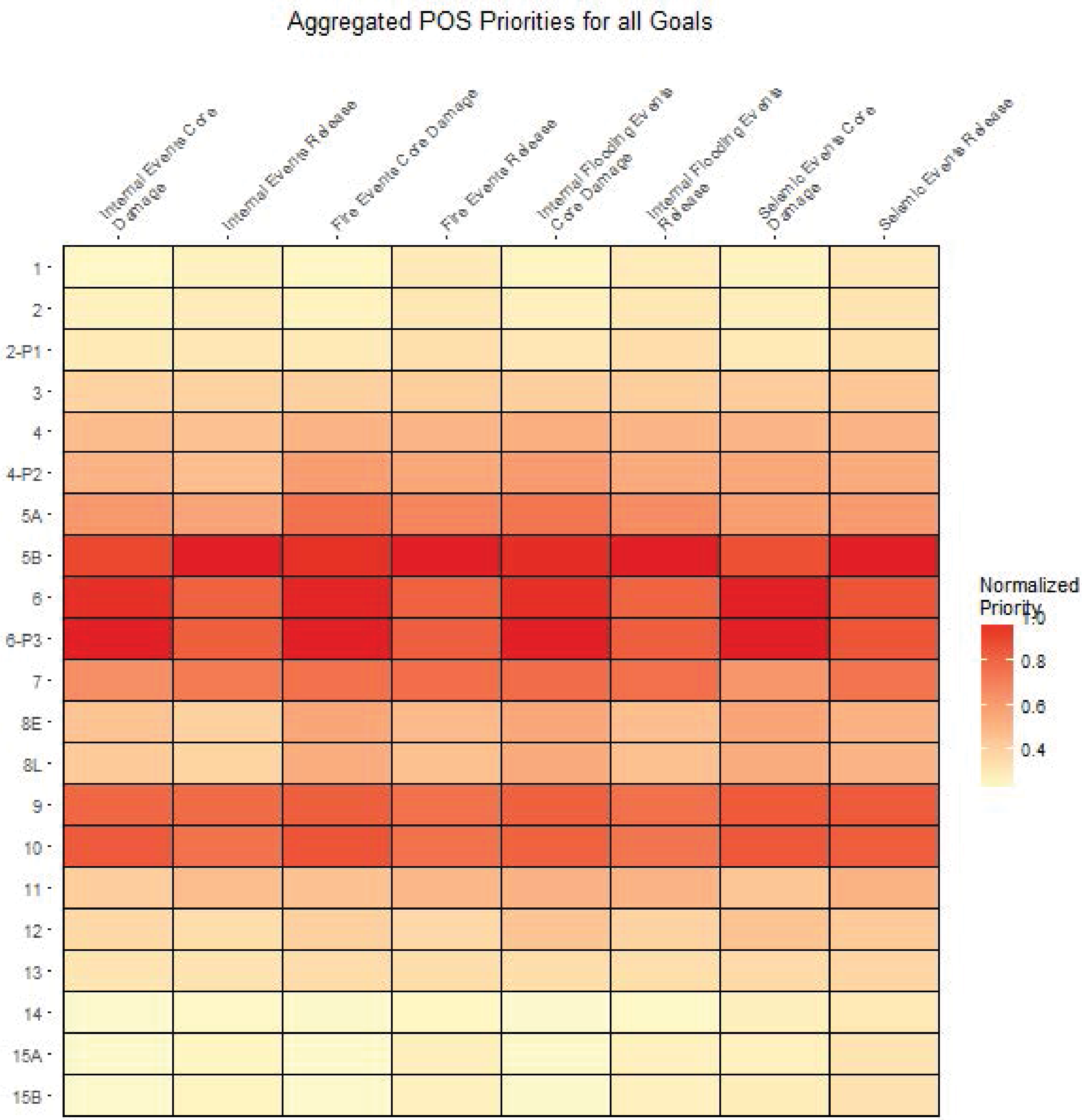
Sections K.2 through K.8 provide the specific results for each expert. The following diagrams/tables are included in of these sections:

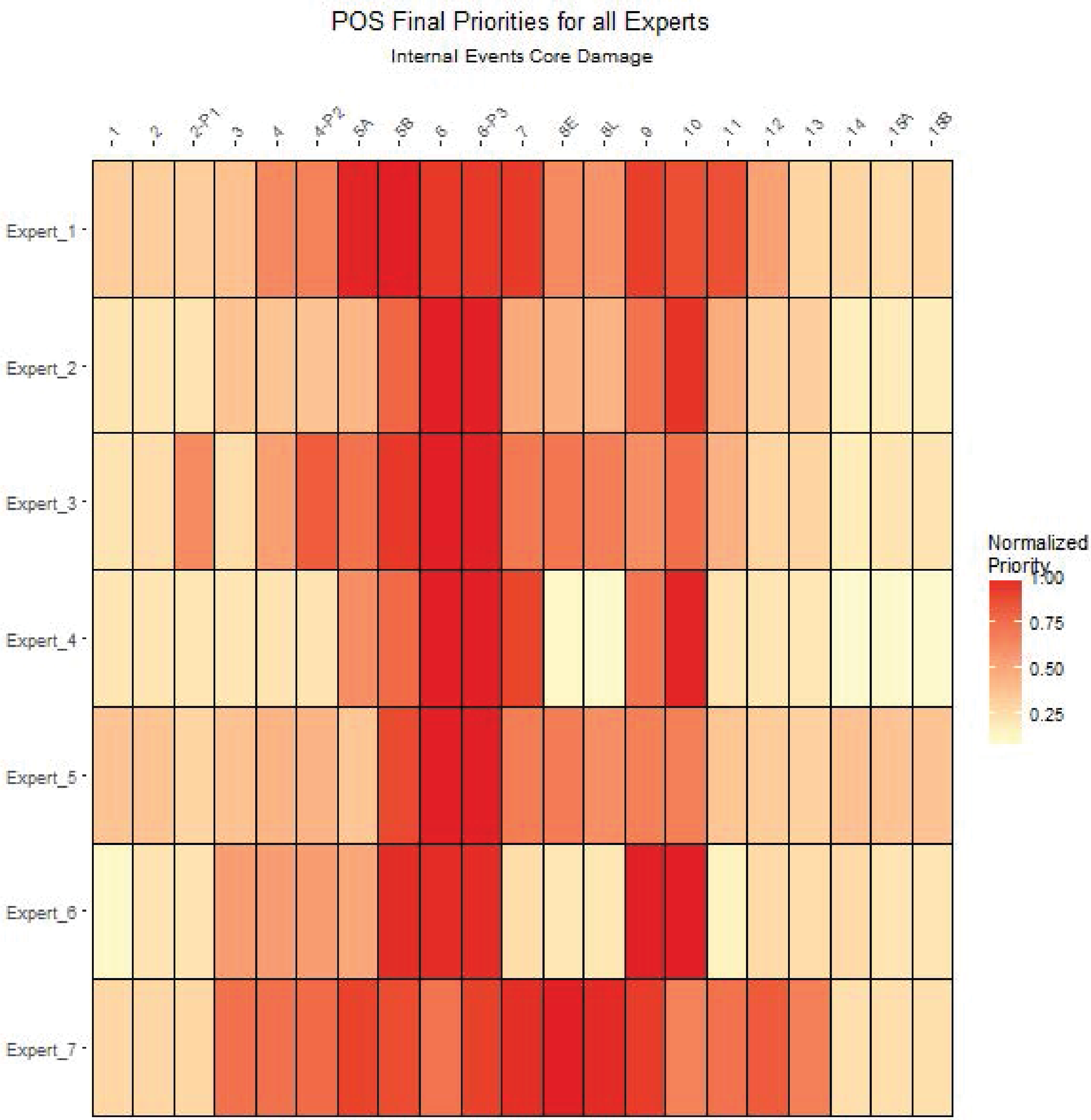
- Eight diagrams of the normalized weights for each sub-criteria – one for each goal (eight pages)
- One table of the normalized weights for each top-level criteria for each of the goals (one page)
- One table of the normalized weights calculated for each of the sub-criteria for each of the top-level criteria (one page).

The reported weights were calculated using the scale values assigned by each expert in their elicitation forms.

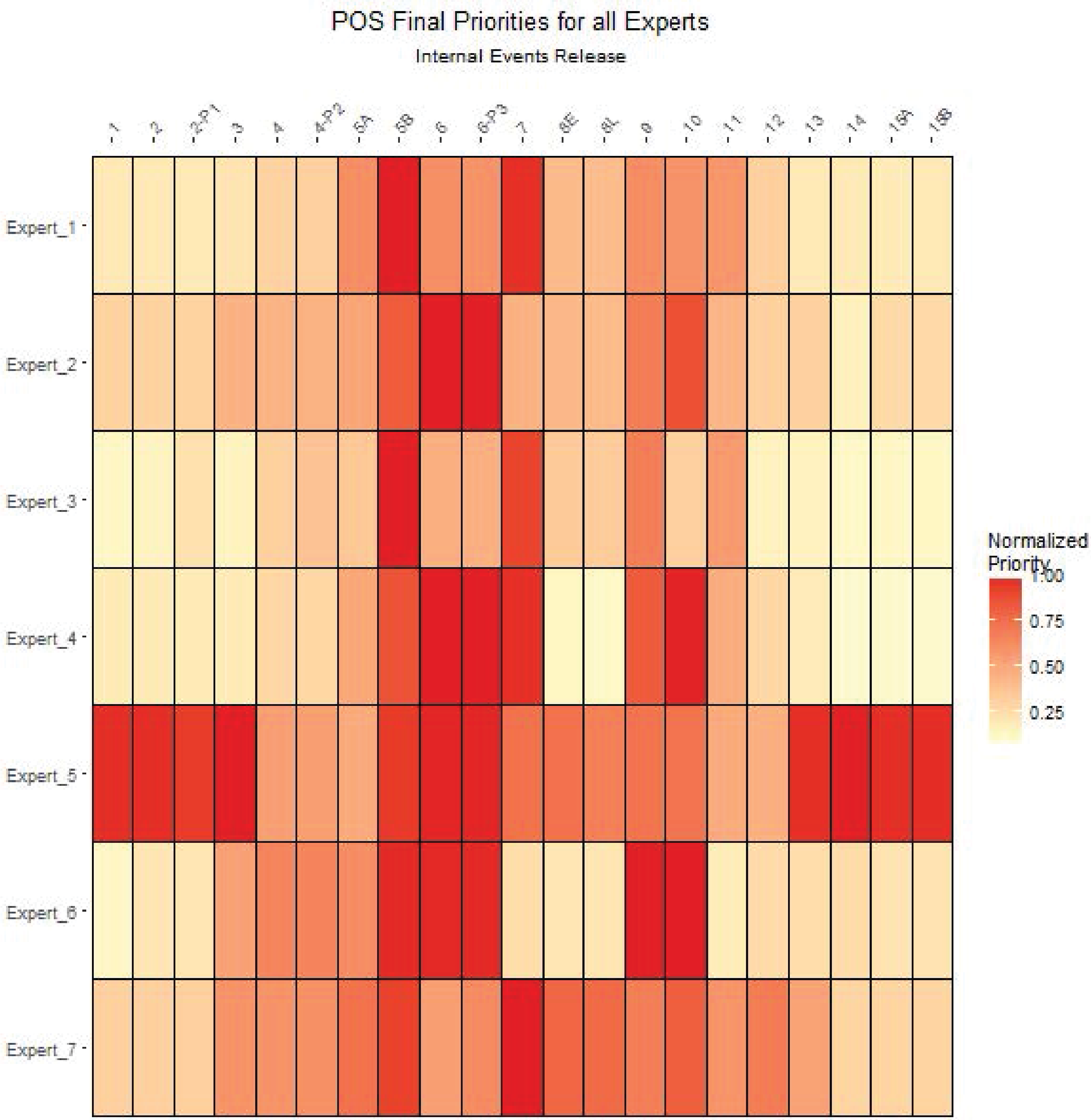
The eight sections are as follows:

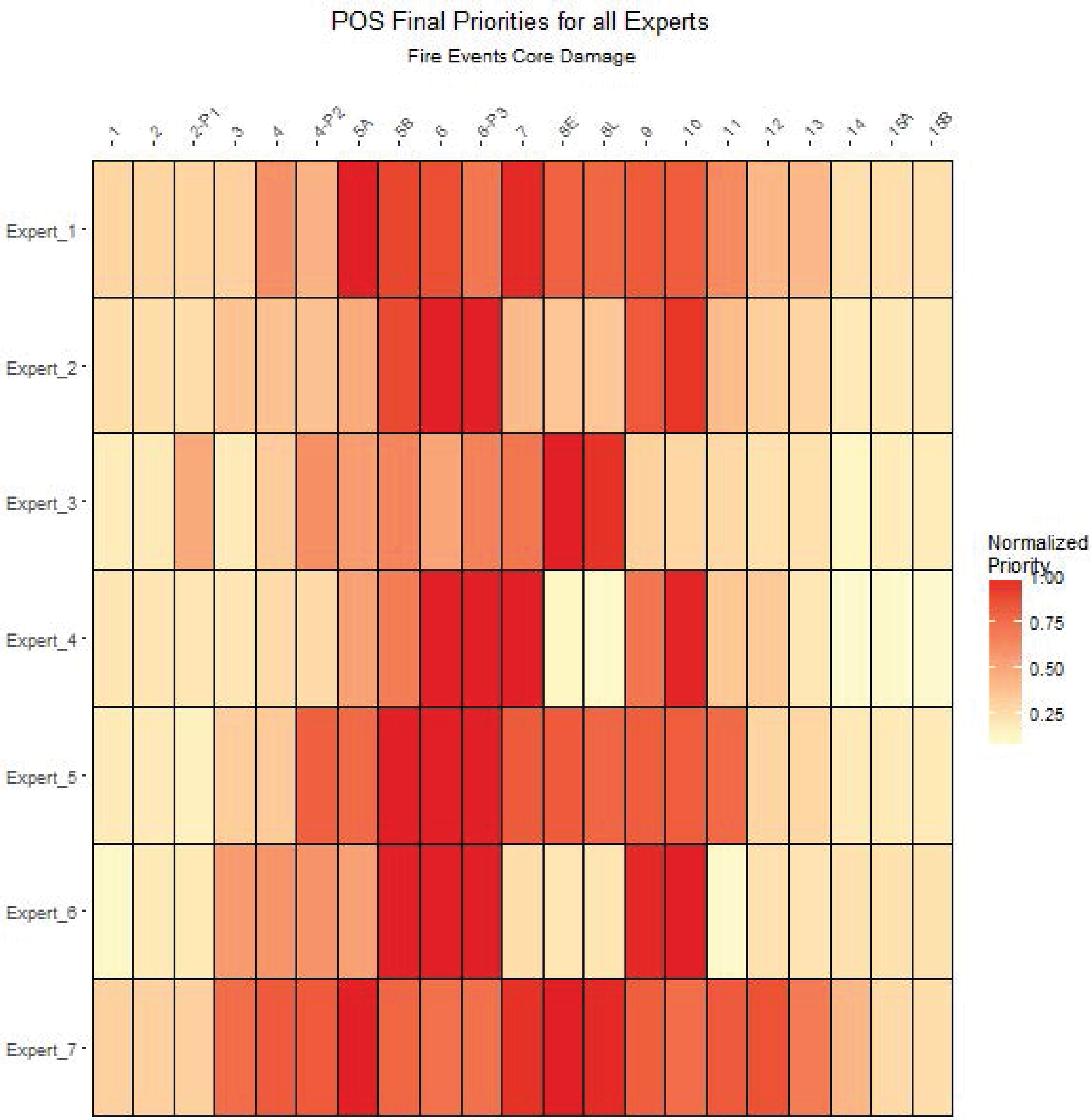
K.1 Results Provided to All Experts



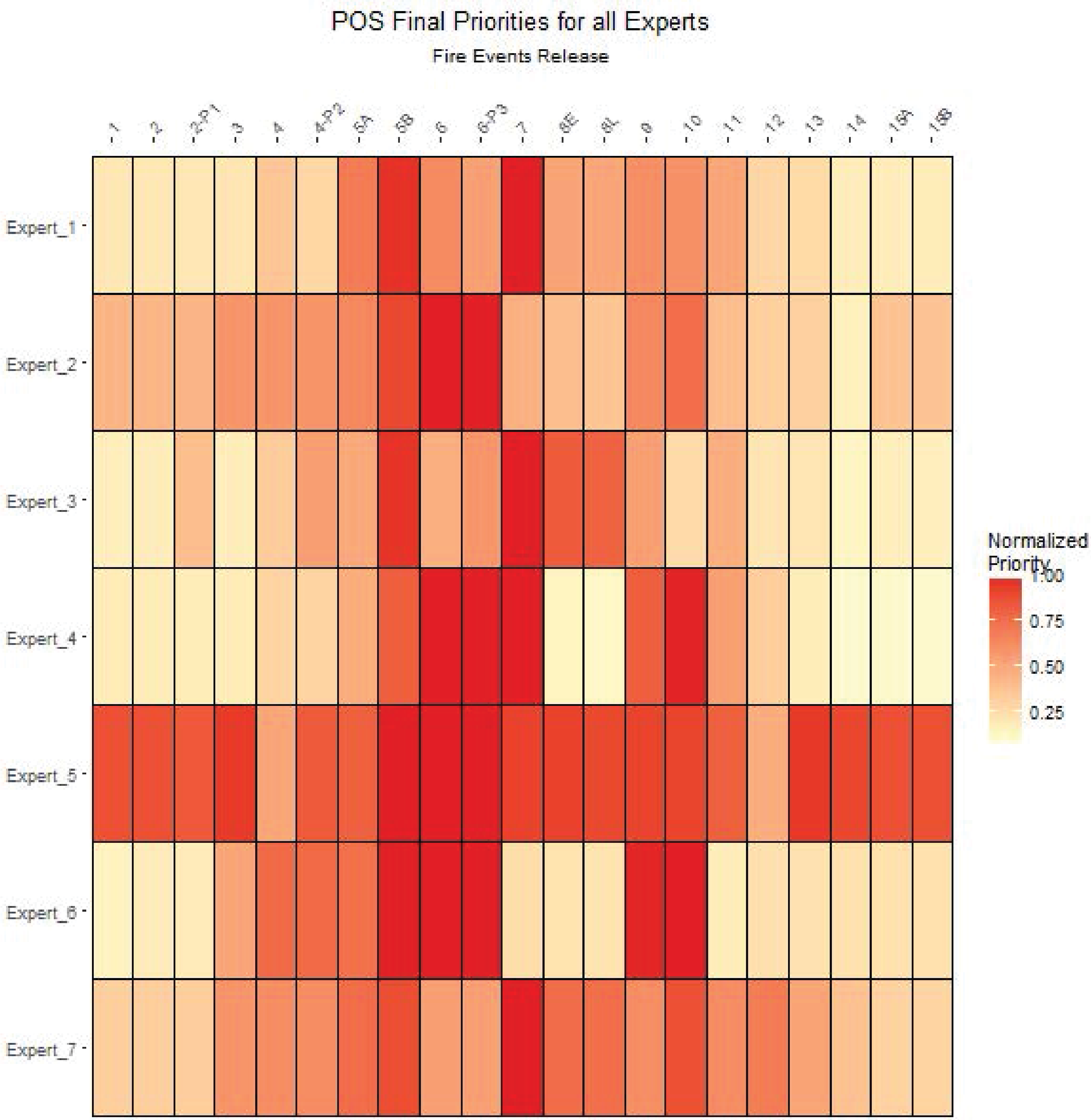


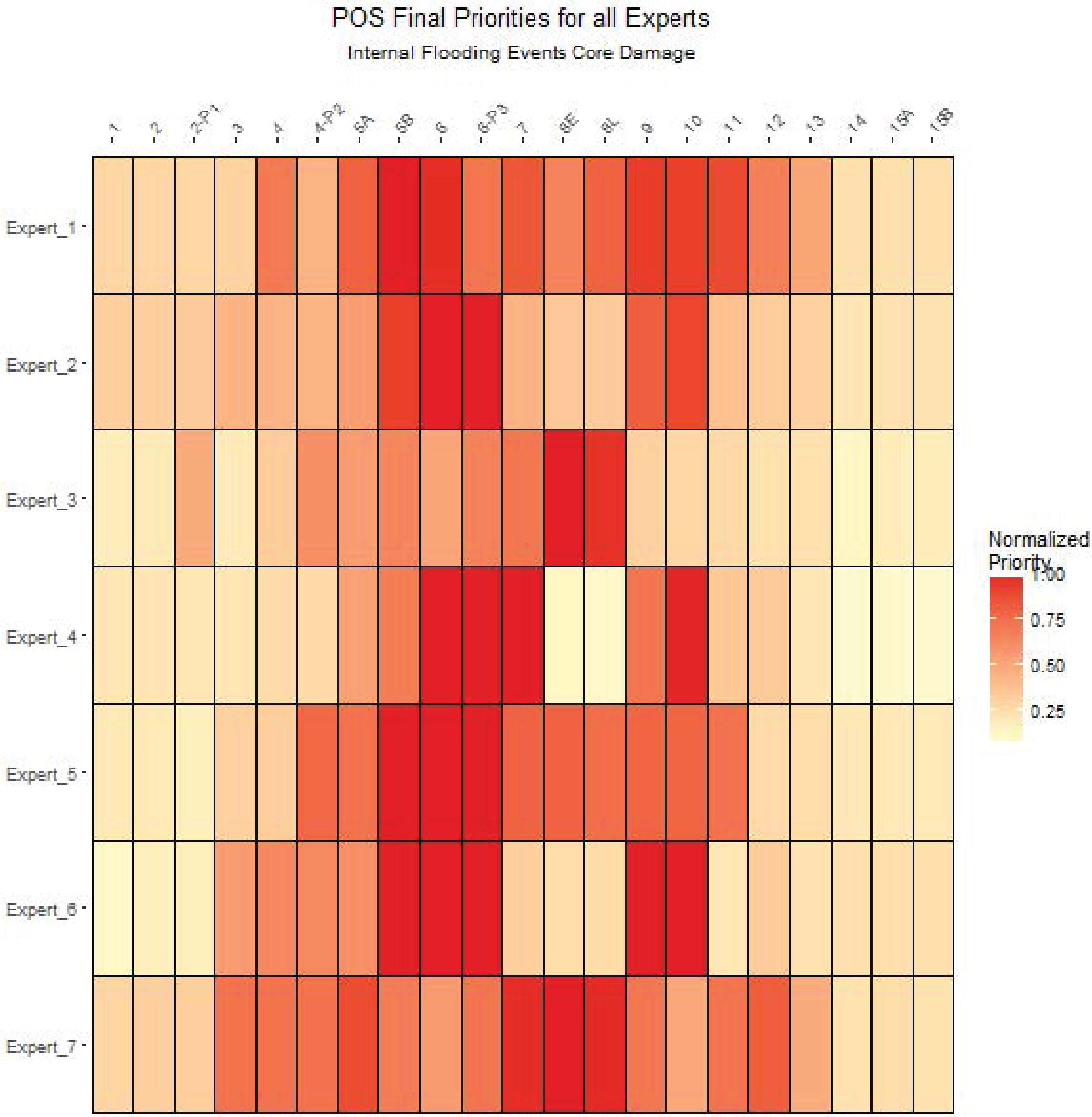
POS Final Priorities for all Experts
Internal Events Core Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Expert_1
Expert_2
Expert_3
Expert_4
Expert_5
Expert_6
Expert_7
Normalized Priority
1.00
0.75
0.50
0.25

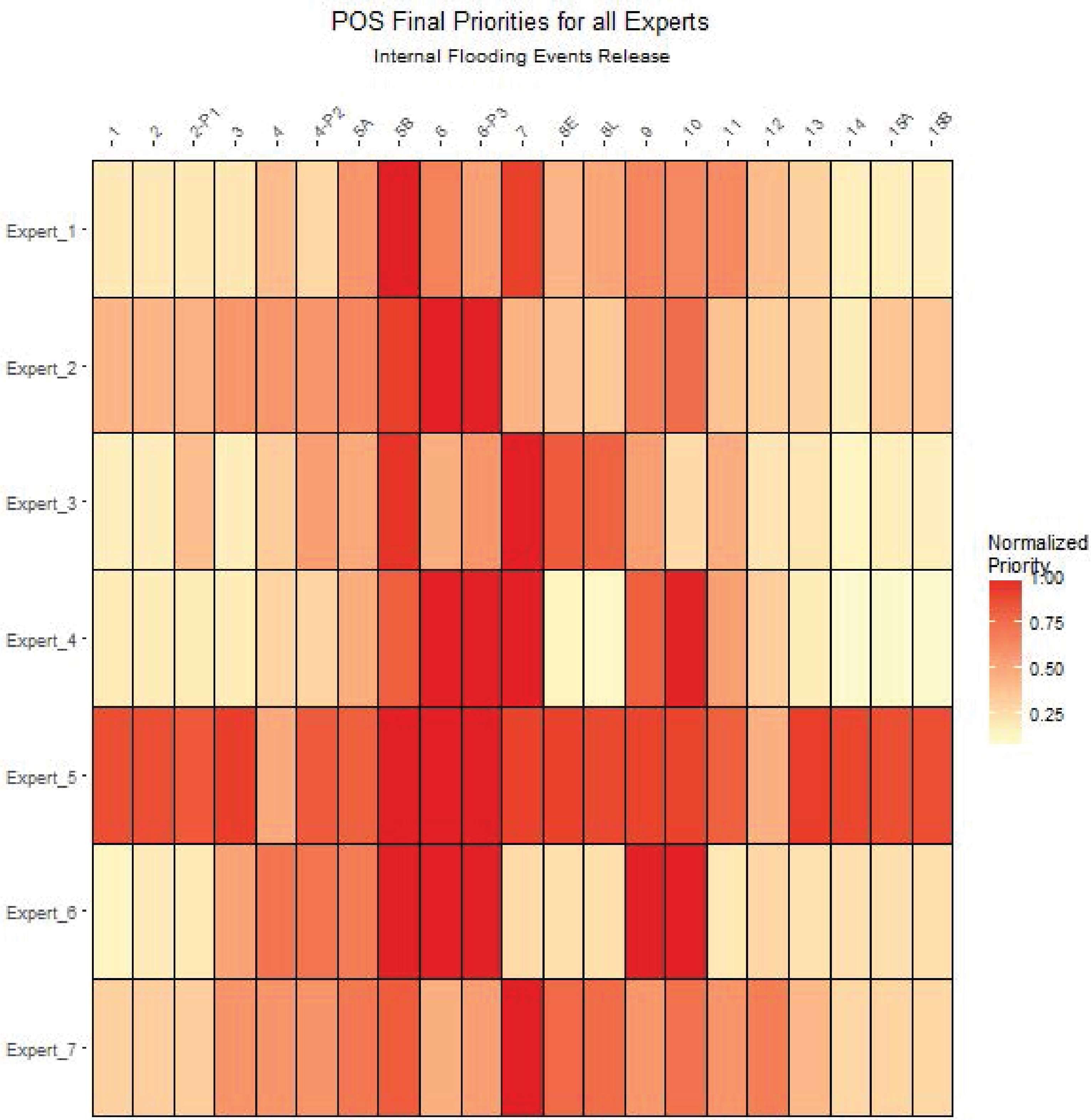

POS Final Priorities for all Experts
Internal Events Release
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Expert_1
Expert_2
Expert_3
Expert_4
Expert_5
Expert_6
Expert_7
Normalized Priority
1.00
0.75
0.50
0.25

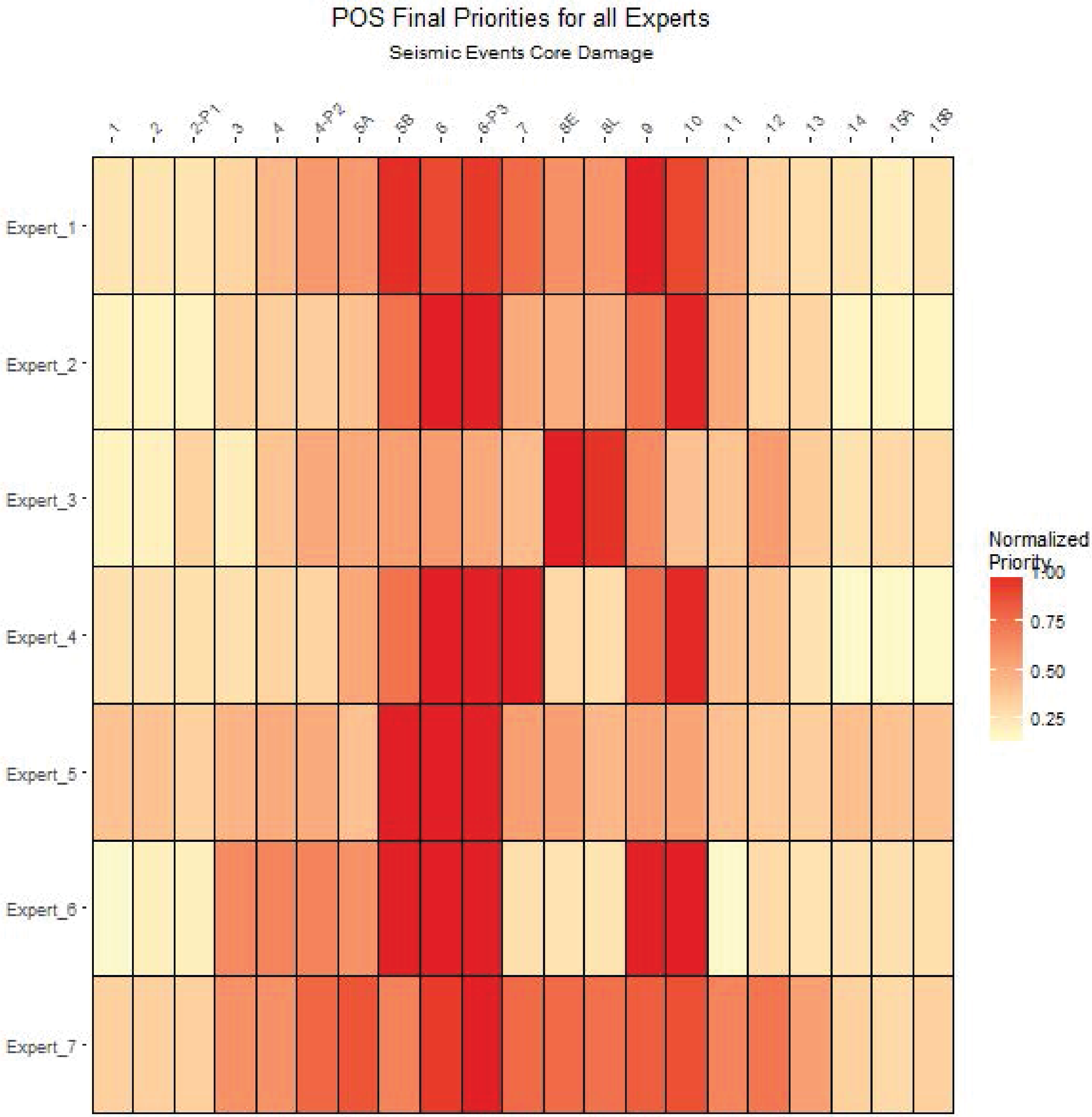


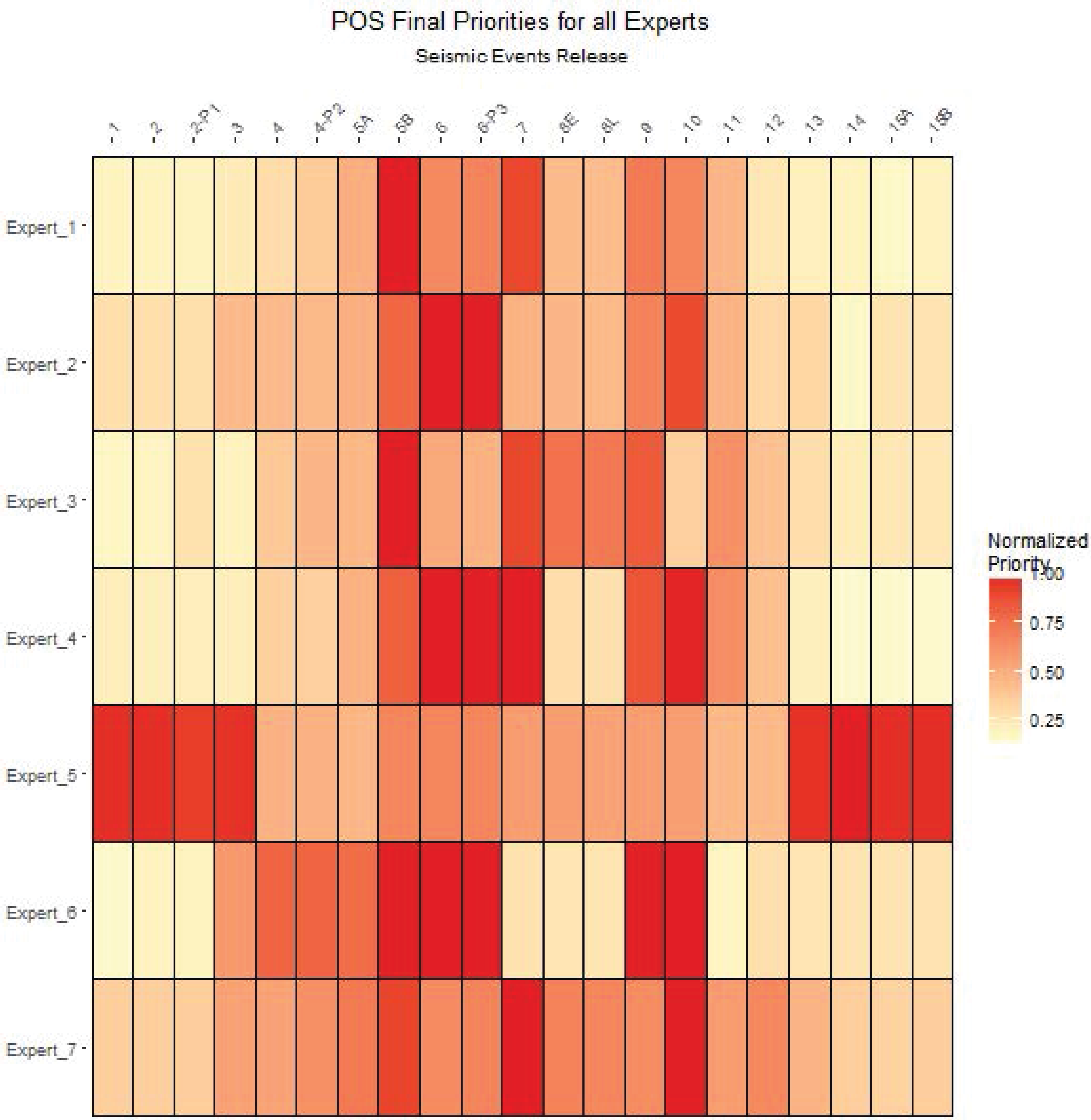
POS Final Priorities for all Experts
Fire Events Core Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Expert_1
Expert_2
Expert_3
Expert_4
Expert_5
Expert_6
Expert_7
Normalized Priority
1.00
0.75
0.50
0.25


POS Final Priorities for all Experts
Fire Events Release
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Expert_1
Expert_2
Expert_3
Expert_4
Expert_5
Expert_6
Expert_7
Normalized
Priority
1.00
0.75
0.50
0.25

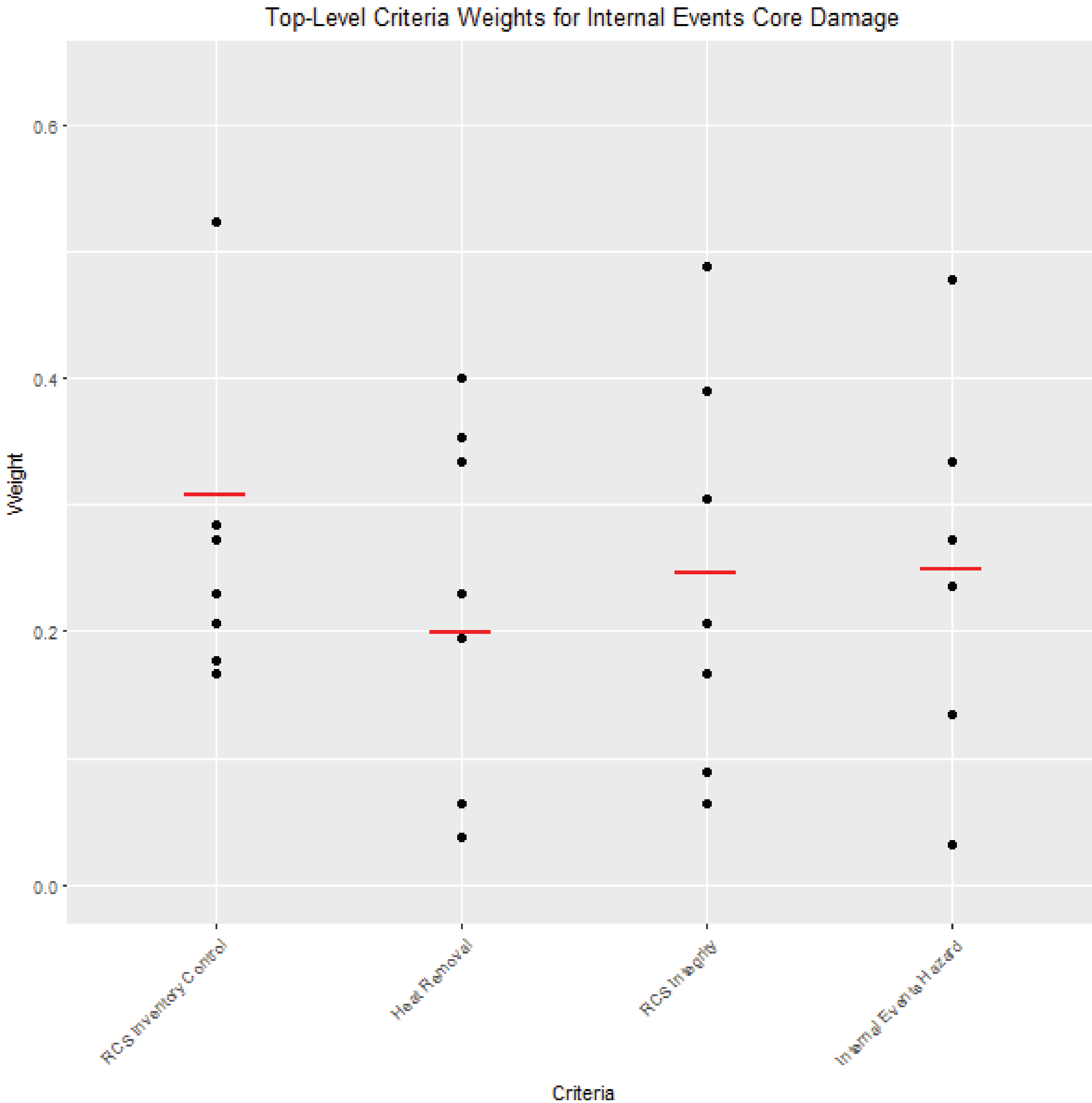

POS Final Priorities for all Experts
Internal Flooding Events Core Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Expert_1
Expert_2
Expert_3
Expert_4
Expert_5
Expert_6
Expert_7
Normalized Priority
1.00
0.75
0.50
0.25


POS Final Priorities for all Experts
Internal Flooding Events Release
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Expert_1
Expert_2
Expert_3
Expert_4
Expert_5
Expert_6
Expert_7
Normalized Priority
1.00
0.75
0.50
0.25


POS Final Priorities for all Experts
Seismic Events Core Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Expert_1
Expert_2
Expert_3
Expert_4
Expert_5
Expert_6
Expert_7
Normalized Priority
1.00
0.75
0.50
0.25



POS Final Priorities for all Experts
Seismic Events Release
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Expert_1
Expert_2
Expert_3
Expert_4
Expert_5
Expert_6
Expert_7
Normalized Priority
1.00
0.75
0.50
0.25


Top-Level Criteria Weights for Internal Events Core Damage
Weight
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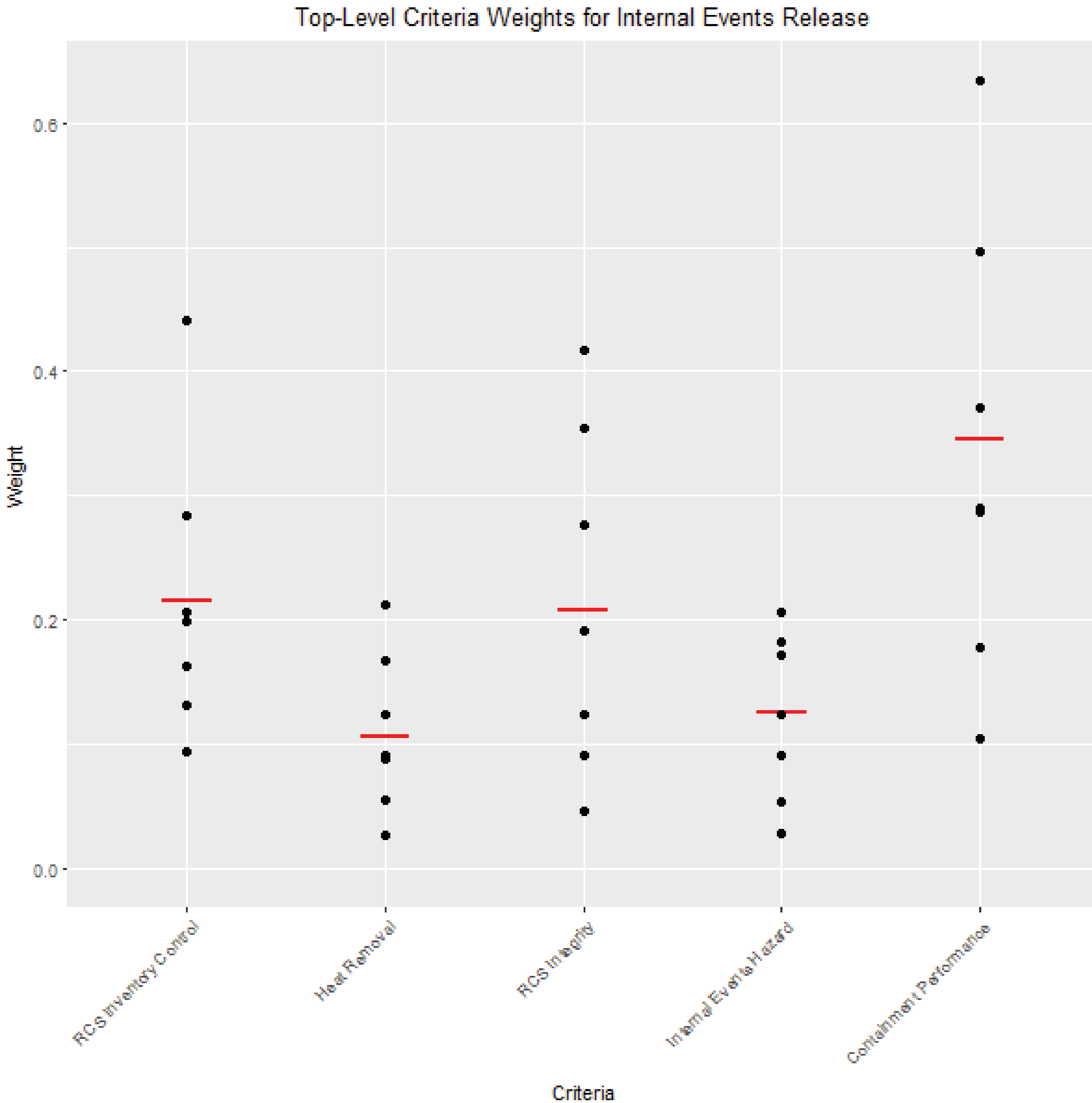
0.2
0.4
0.6
RCS Inventory Control
Heat Removal
RCS Integrity
Internal Events Hazard
Criteria


Top-Level Criteria Weights for Internal Events Release
Weight
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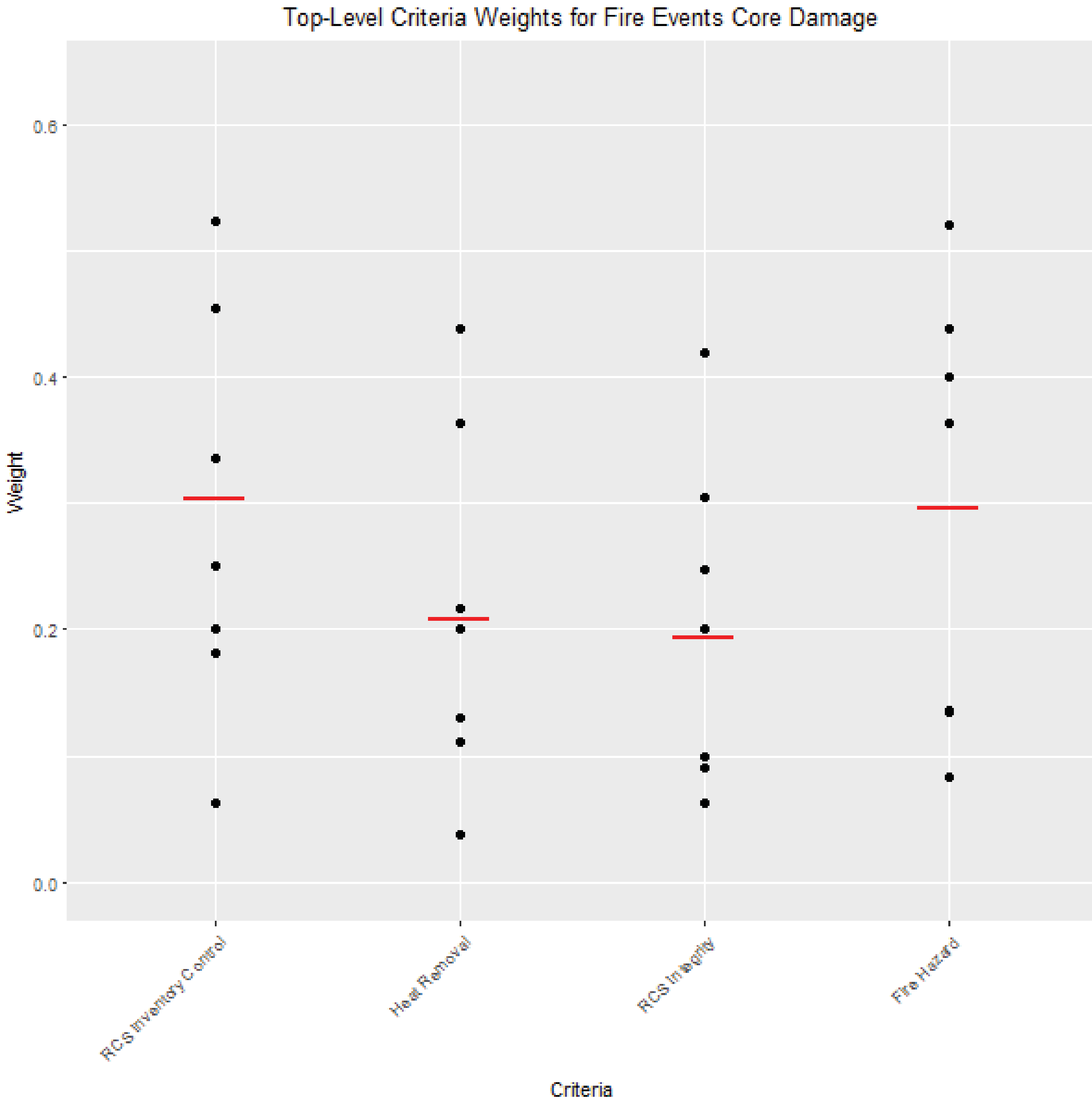
0.2
0.4
0.6
RCS Inventory Control
Heat Removal
RCS Integrity
Internal Events Hazard
Containment Performance
Criteria


Top-Level Criteria Weights for Fire Events Core Damage
Weight
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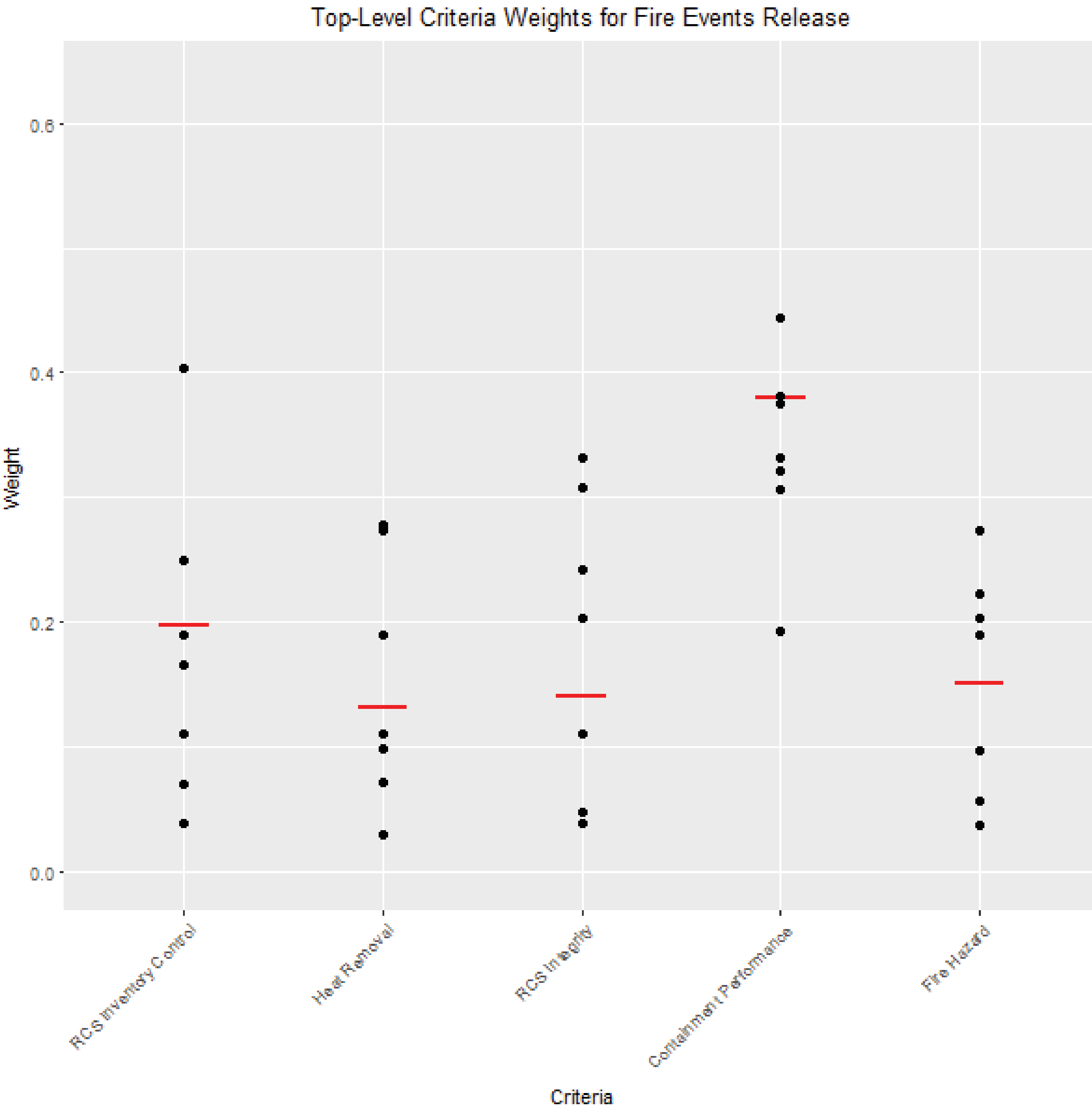
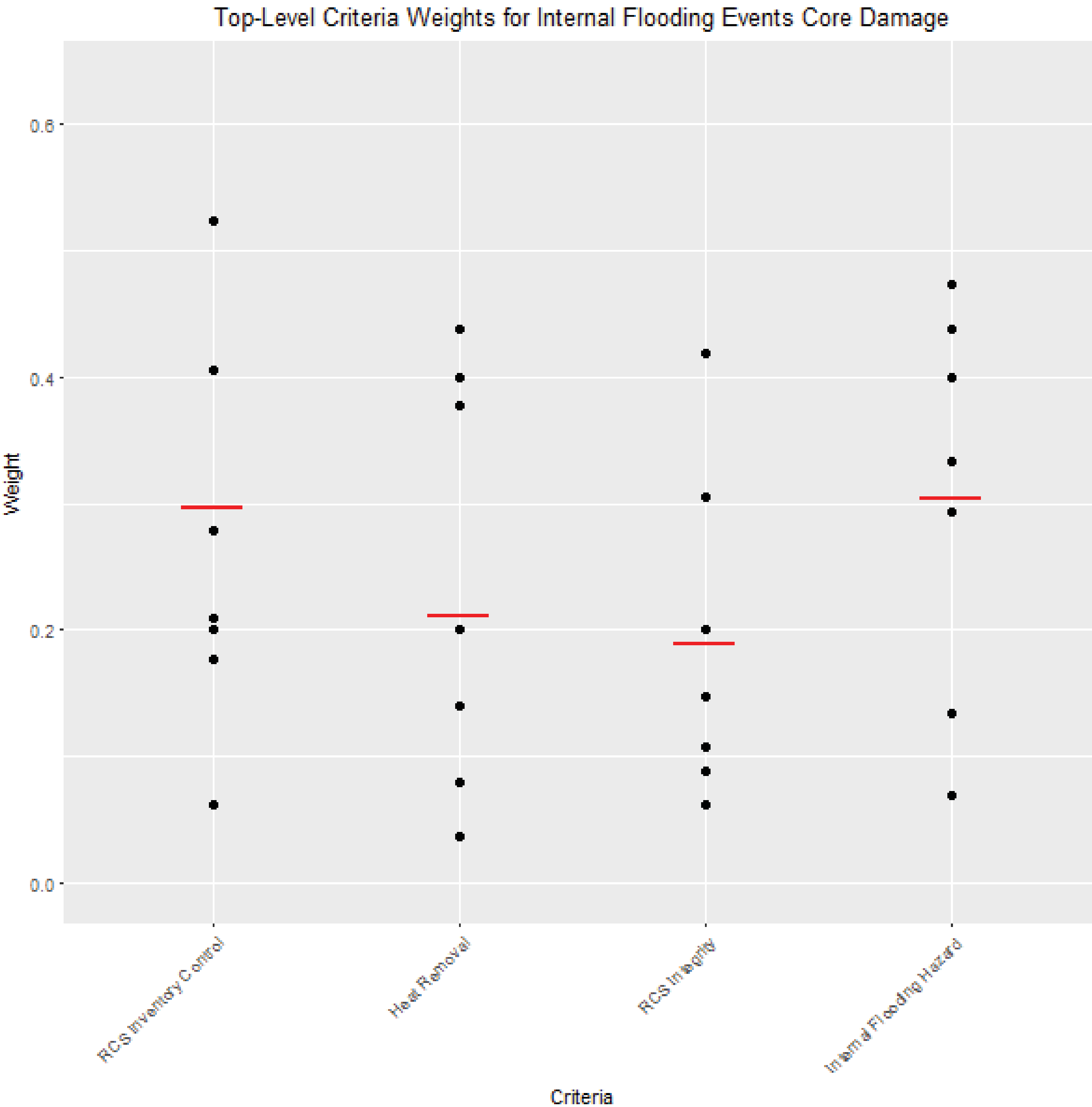
0.2
0.4
0.6
RCS Inventory Control
Heat Removal
RCS Integrity
Fire Hazard
Criteria



Top-Level Criteria Weights for Fire Events Release
Weight
0.6
0.4
0.2
0.0
RCS Inventory Control
Heat Removal
RCS Integrity
Containment Performance
Fire Hazard
Criteria



Top-Level Criteria Weights for Internal Flooding Events Core Damage
Weight
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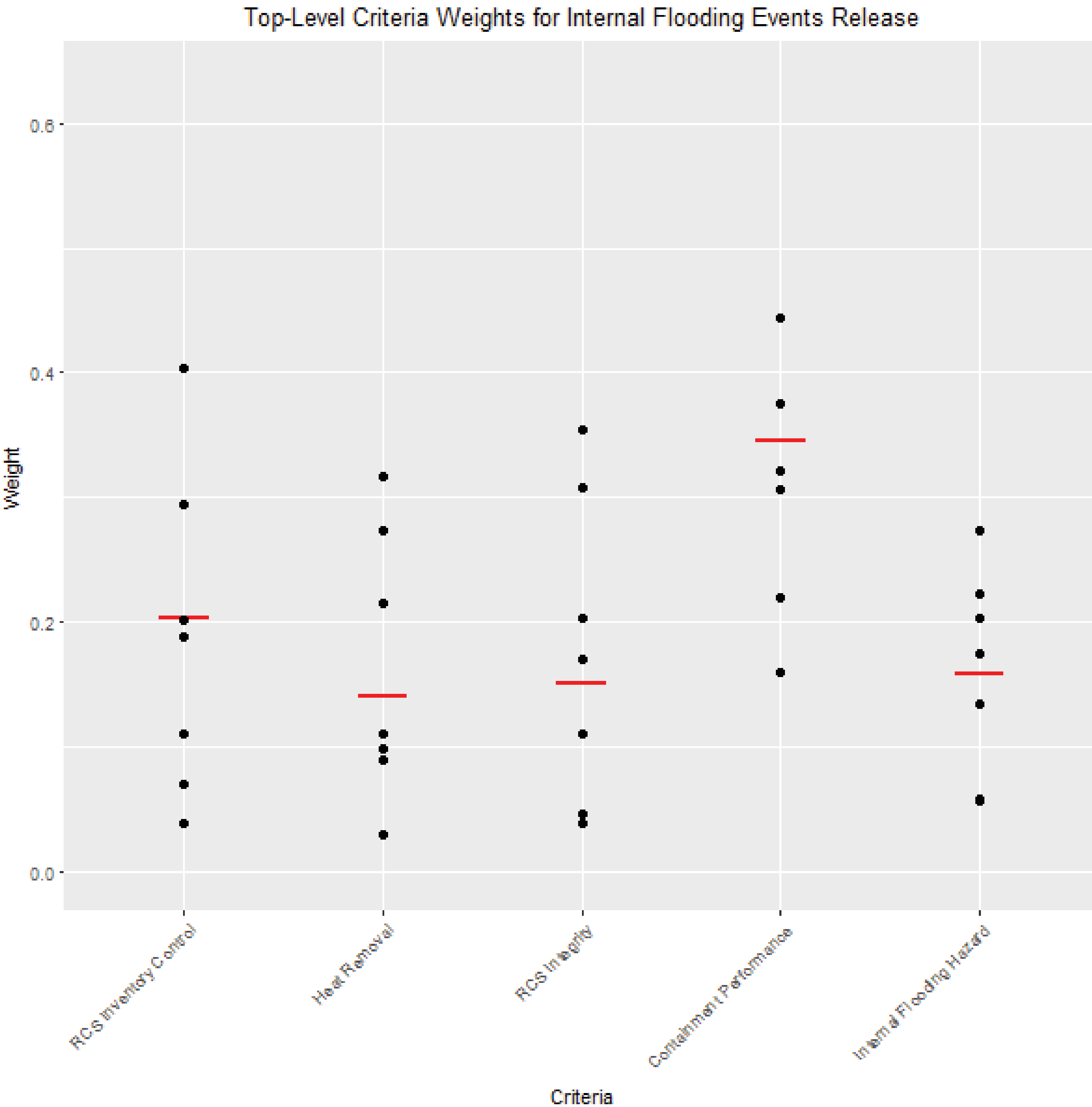
0.2
0.4
0.6
RCS Inventory Control
Heat Removal
RCS Integrity
Internal Flooding Hazard
Criteria


Top-Level Criteria Weights for Internal Flooding Events Release
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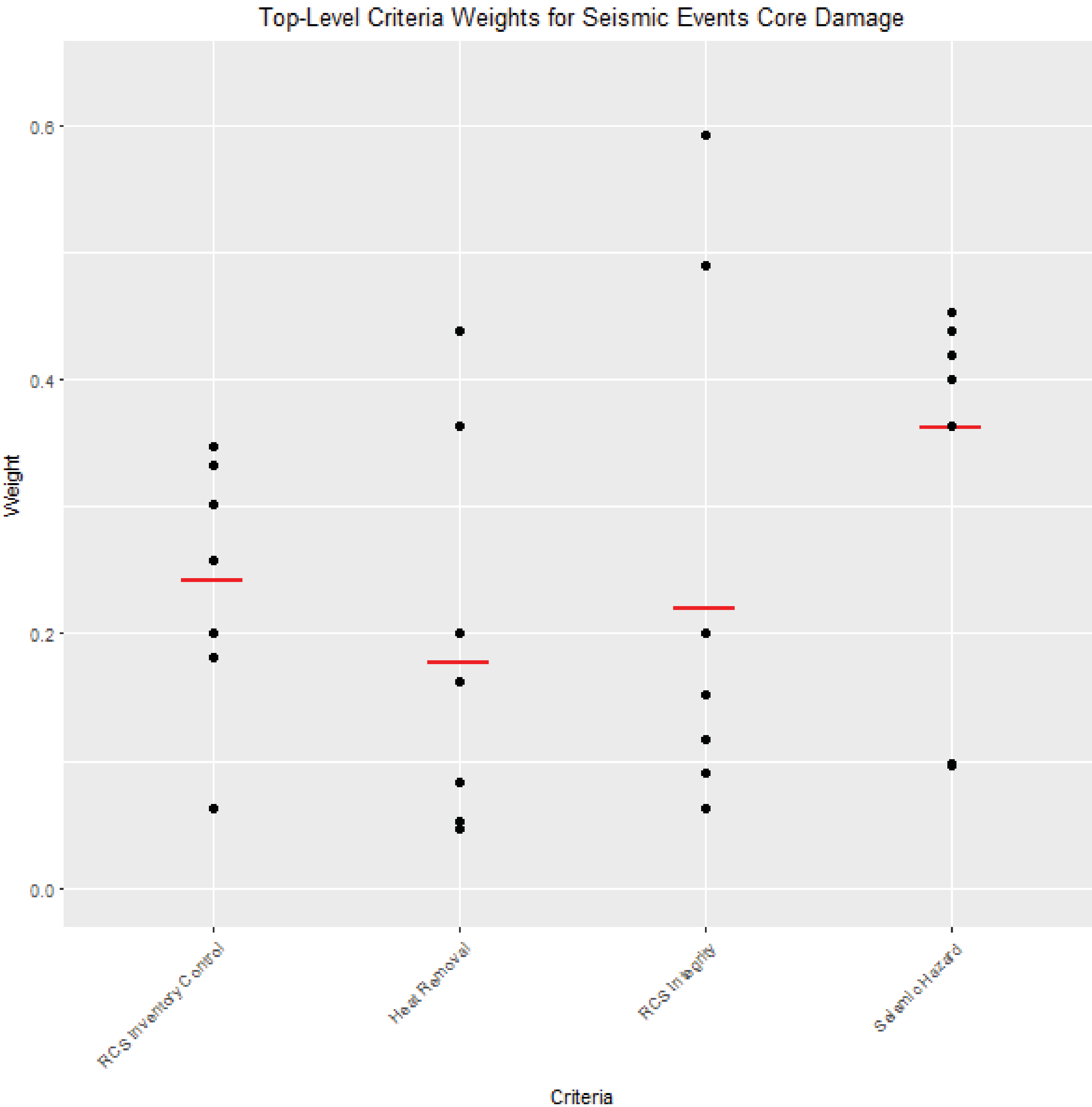
0.2
0.4
0.6
RCS Inventory Control
Heat Removal
RCS Integrity
Containment Performance
Internal Flooding Hazard
Criteria


Top-Level Criteria Weights for Seismic Events Core Damage
Weight
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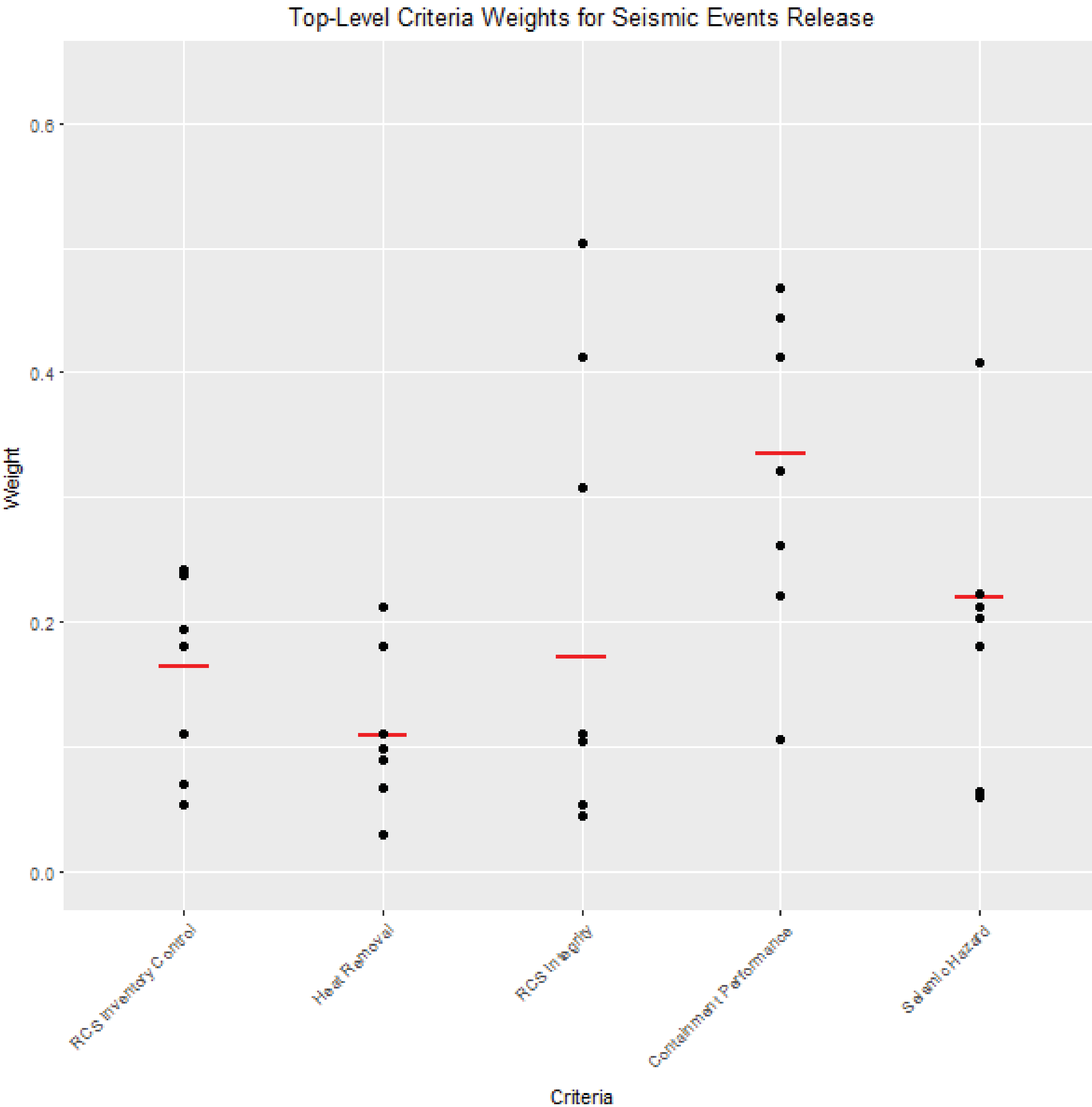
0.4
0.2
0.0
RCS Inventory Control
Heat Removal
RCS Integrity
Seismic Hazard
Criteria


Top-Level Criteria Weights for Seismic Events Release
Weight
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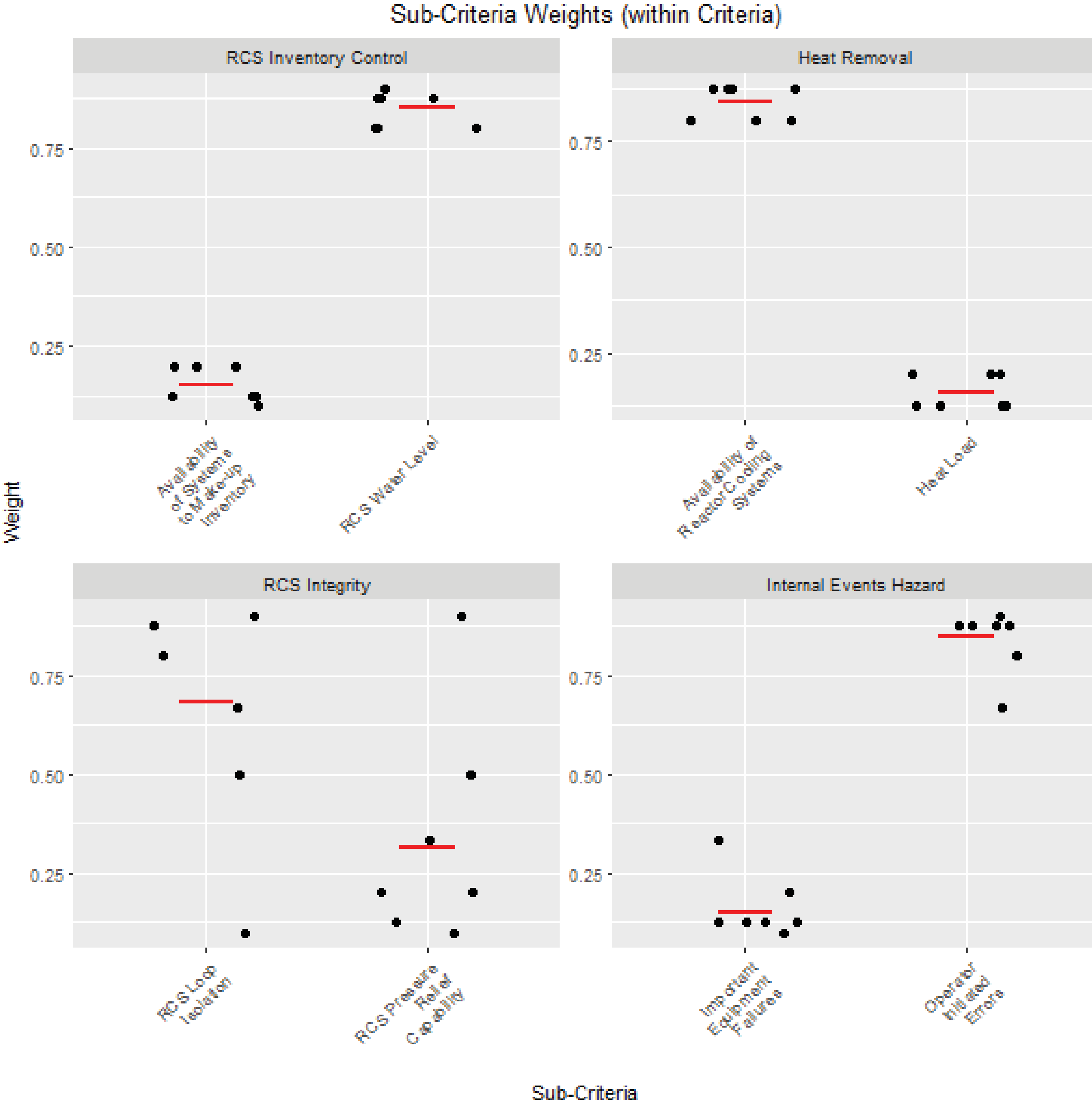
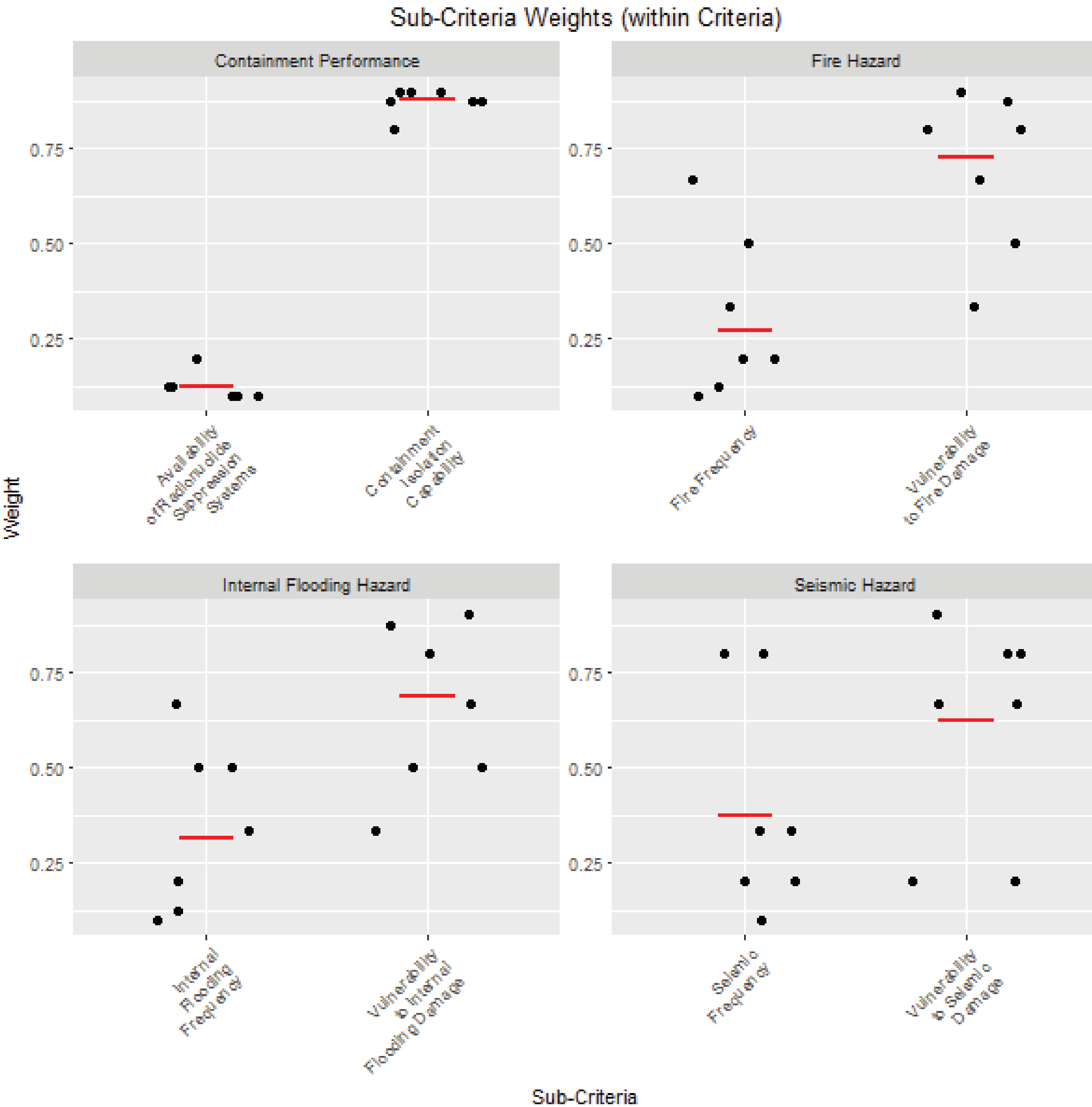
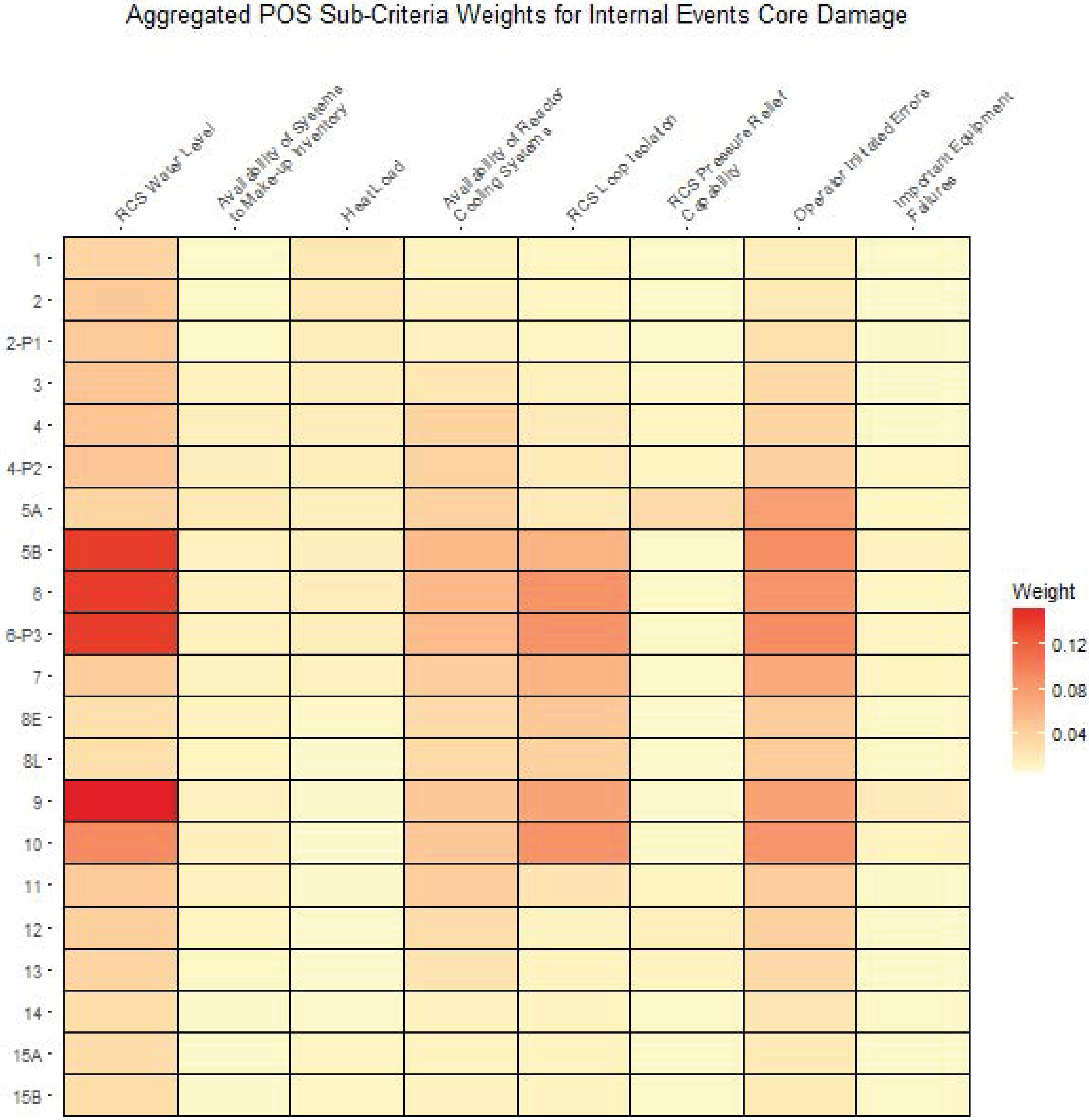
0.4
0.2
0.0
RCS Inventory Control
Heat Removal
RCS Integrity
Containment Performance
Seismic Hazard
Criteria



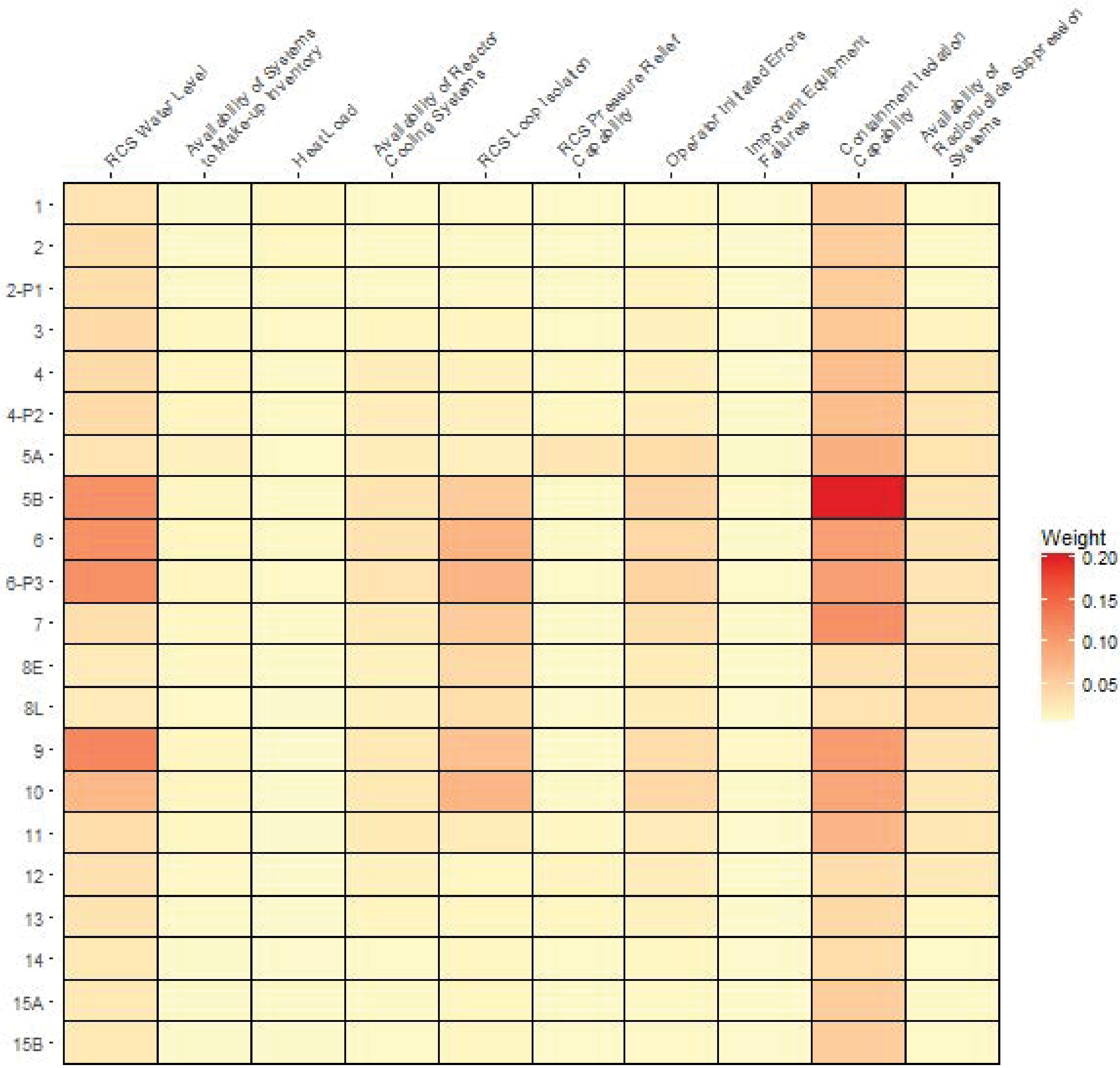
Sub-Criteria Weights (within Criteria)
RCS Inventory Control
Heat Removal
RCS Integrity
Internal Events Hazard
Weight
0.25
0.50
0.75
Availability of Systems to Make-up Inventory
RCS Water Level
Availability of Reactor Cooling Systems
Heat Load
RCS Loop Isolation
RCS Pressure Relief Capability
Important Equipment Failures
Operator Initiated Errors
Sub-Criteria



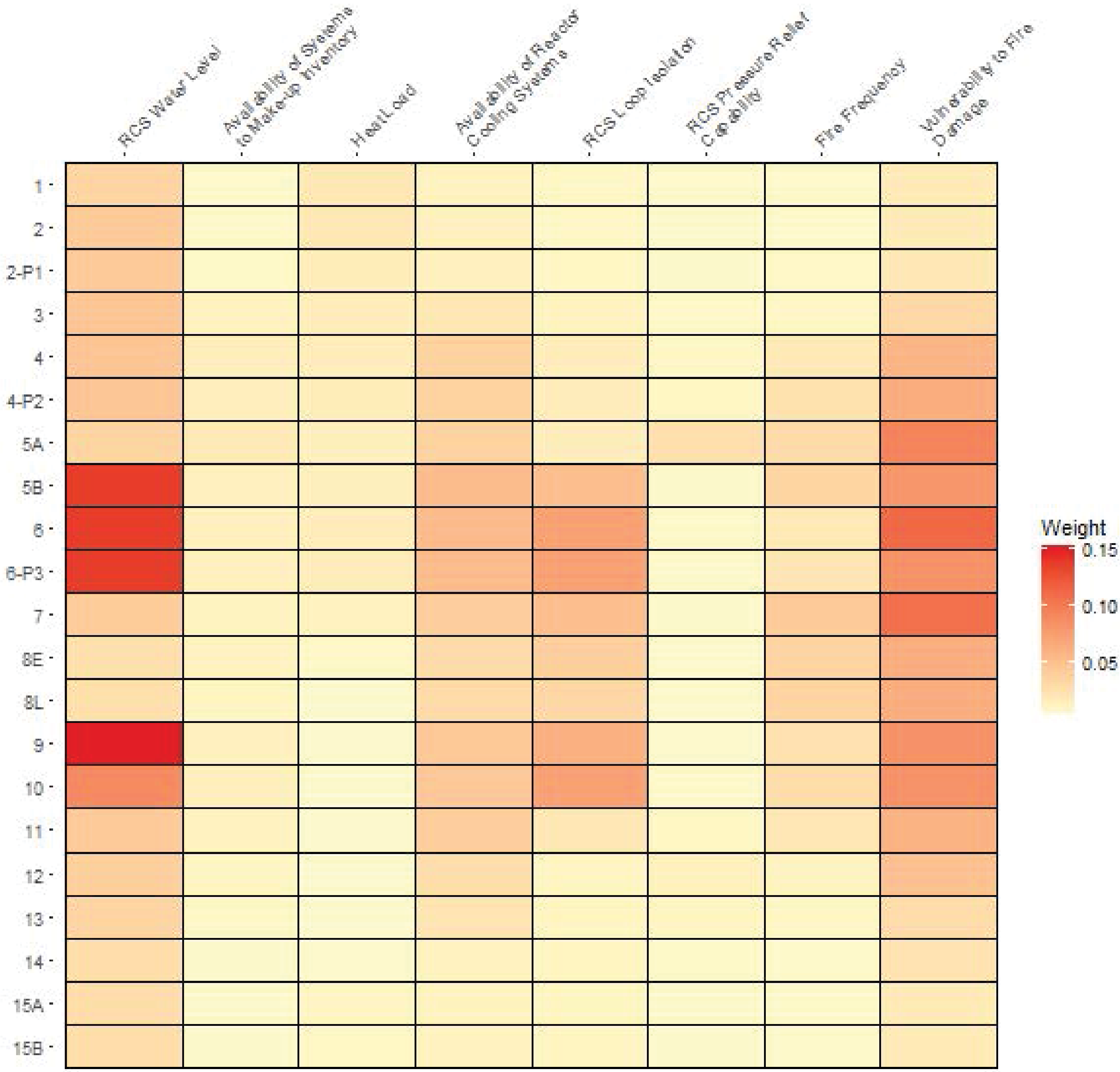
Sub-Criteria Weights (within Criteria)
Containment Performance
Fire Hazard
Internal Flooding Hazard
Seismic Hazard
Weight
0.25
0.50
0.75
Availability of Radiological Suppression Systems
Containment Isolation Capability
Fire Frequency
Vulnerability to Fire Damage
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
Seismic Frequency
Vulnerability to Seismic Damage
Sub-Criteria

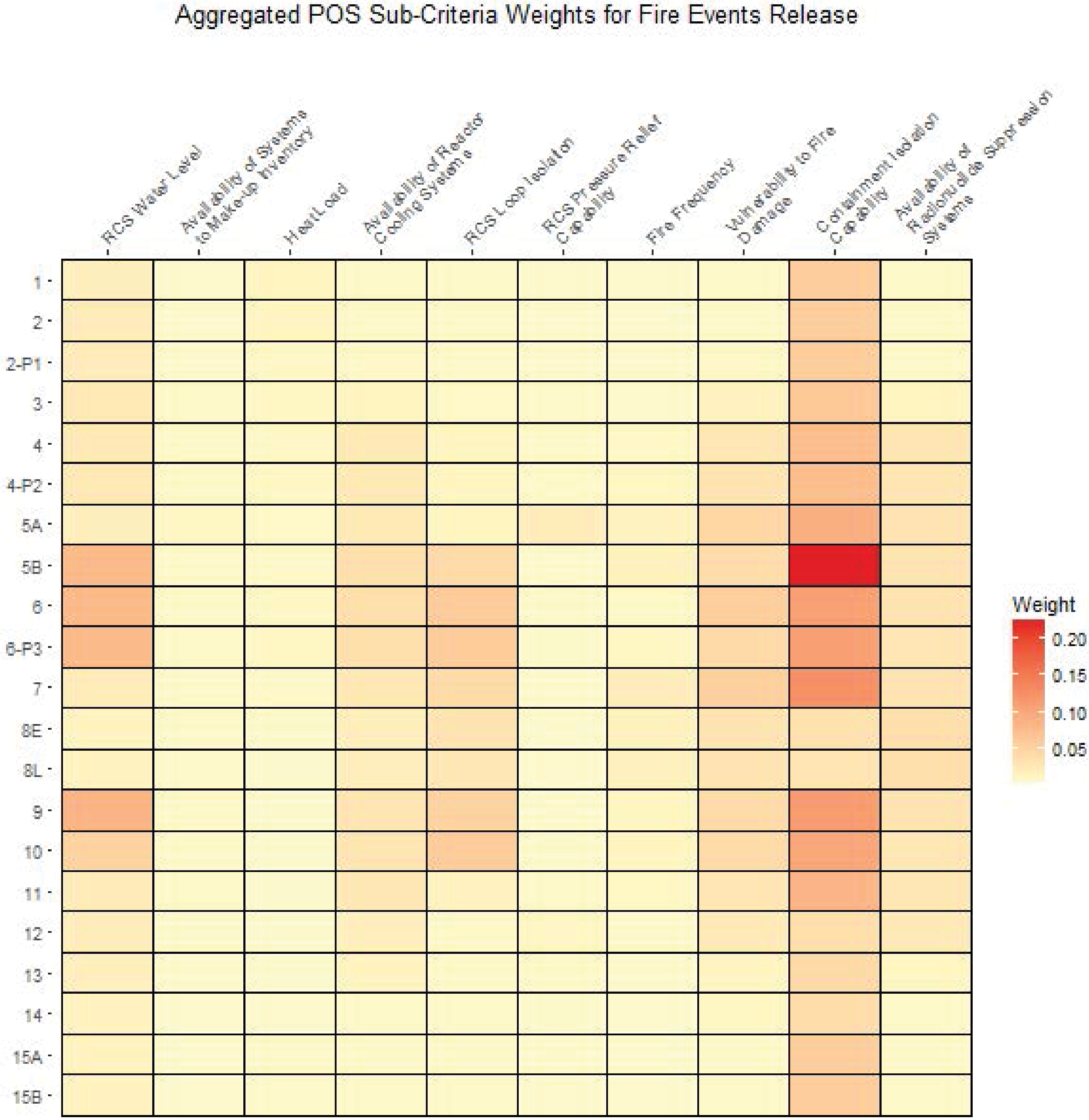

Aggregated POS Sub-Criteria Weights for Internal Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Operator Initiated Errors
Important Equipment Failures
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Weight
0.12
0.08
0.04

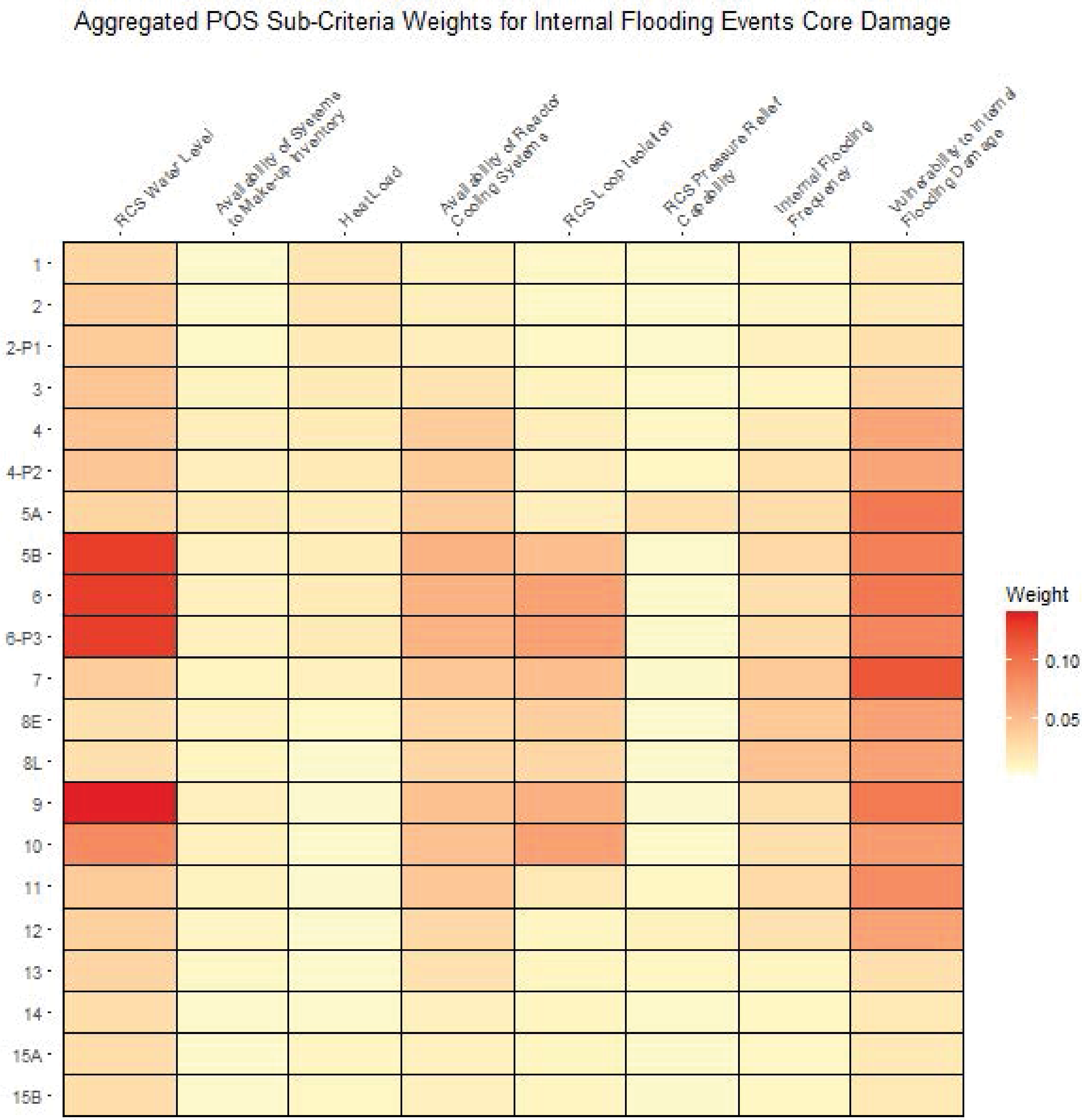
Aggregated POS Sub-Criteria Weights for Internal Events Release



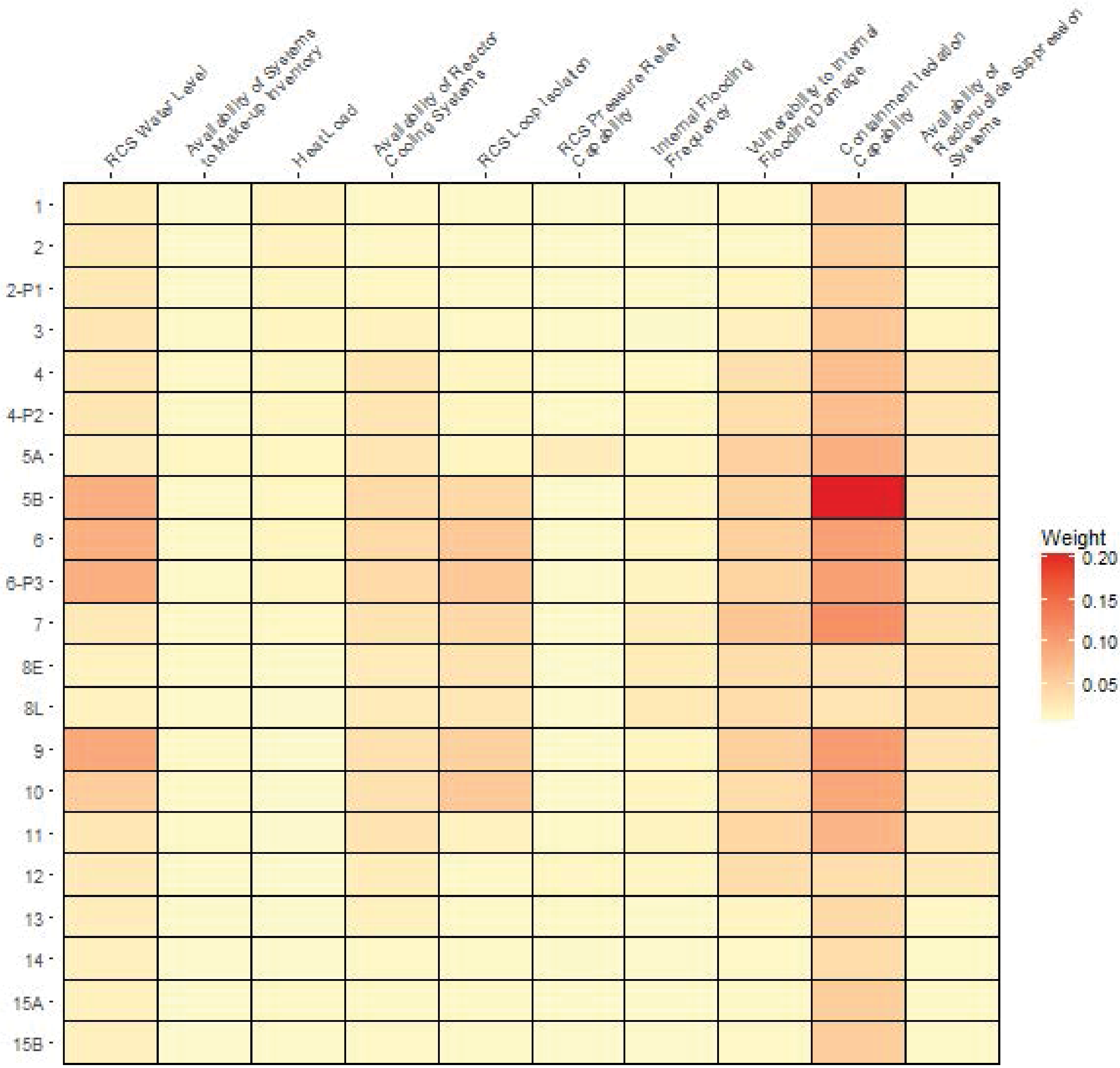
Aggregated POS Sub-Criteria Weights for Fire Events Core Damage



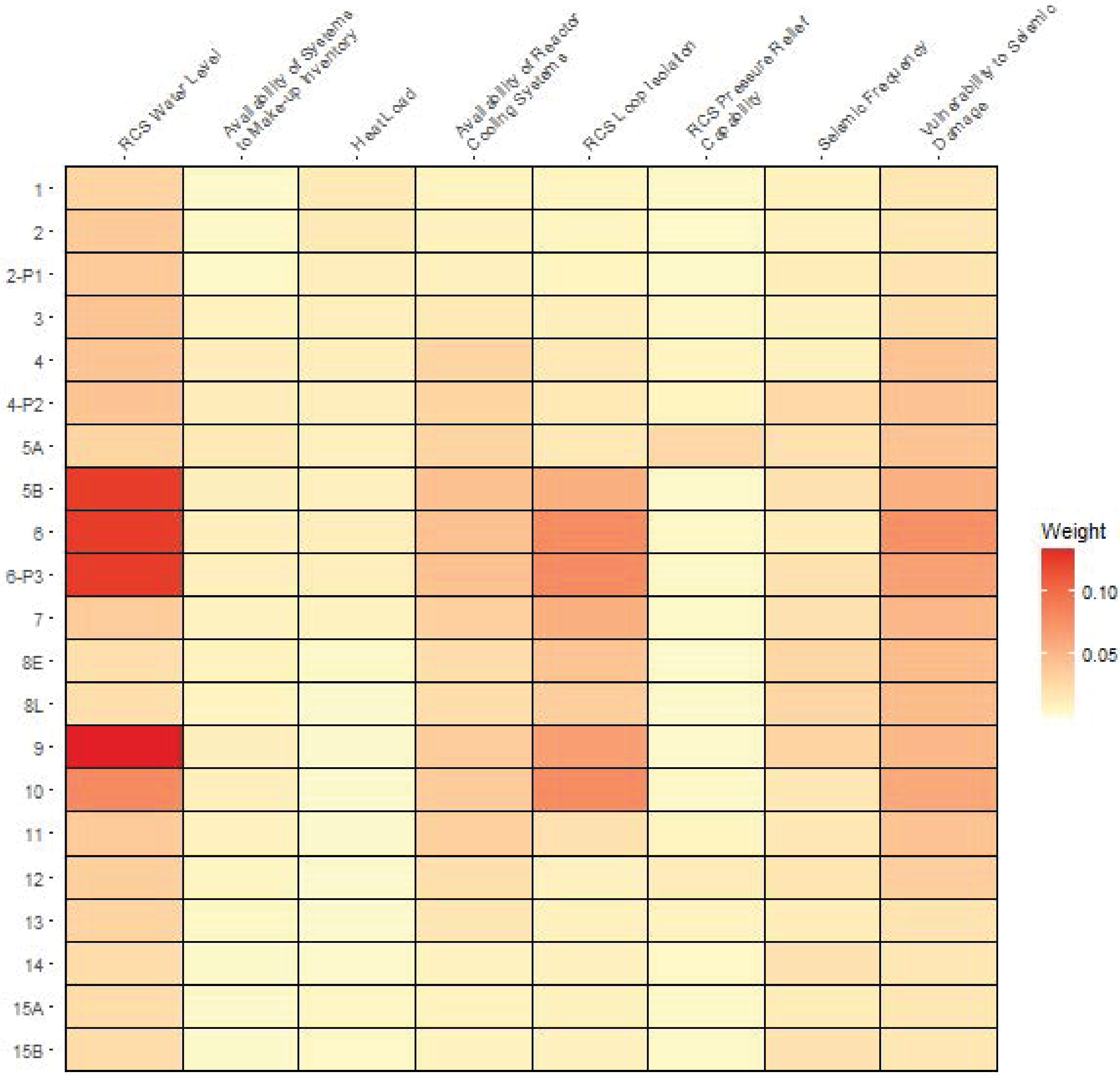

Aggregated POS Sub-Criteria Weights for Fire Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Weight
0.20
0.15
0.10
0.05


Aggregated POS Sub-Criteria Weights for Internal Flooding Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Weight
0.10
0.05

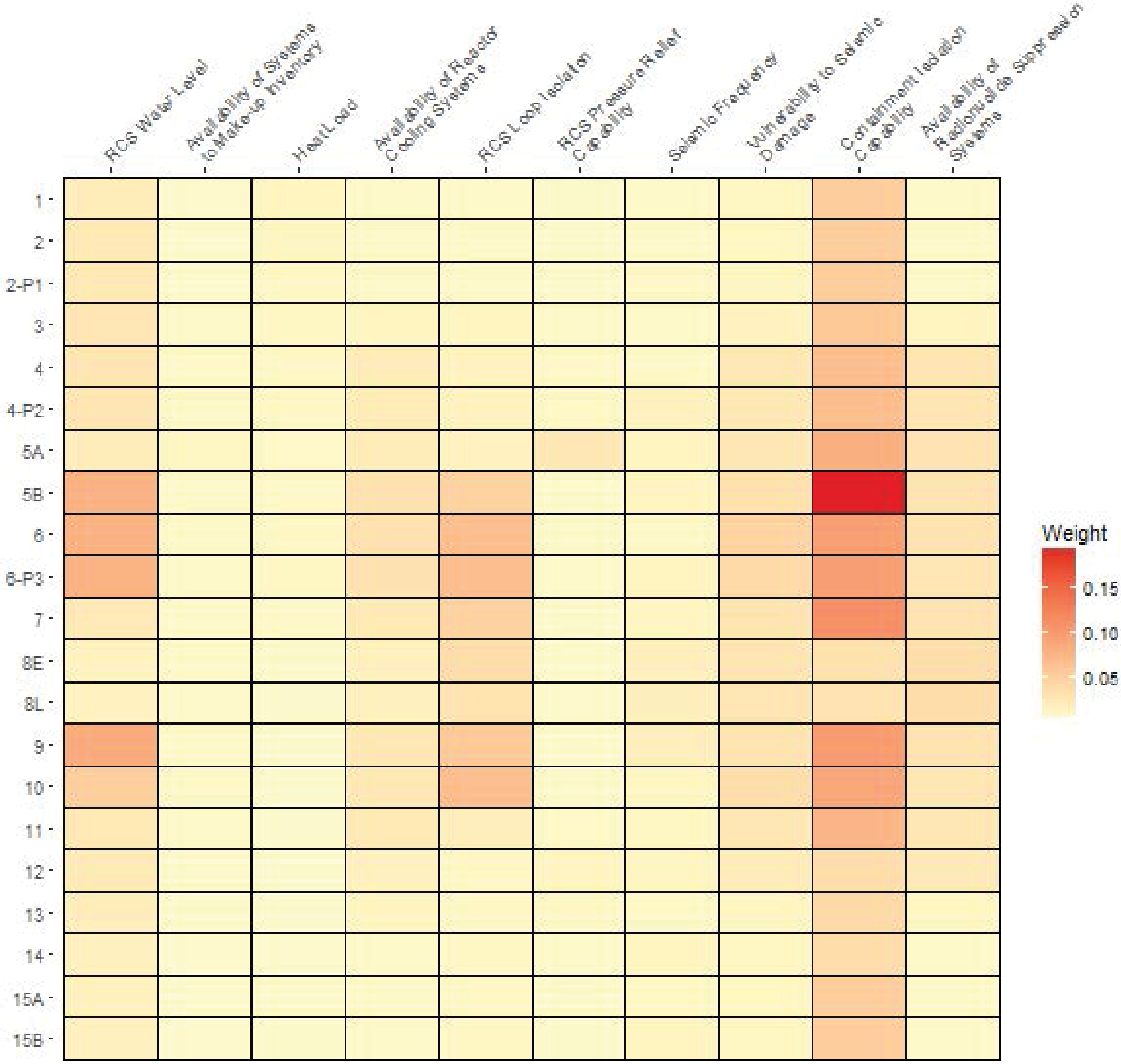
Aggregated POS Sub-Criteria Weights for Internal Flooding Events Release



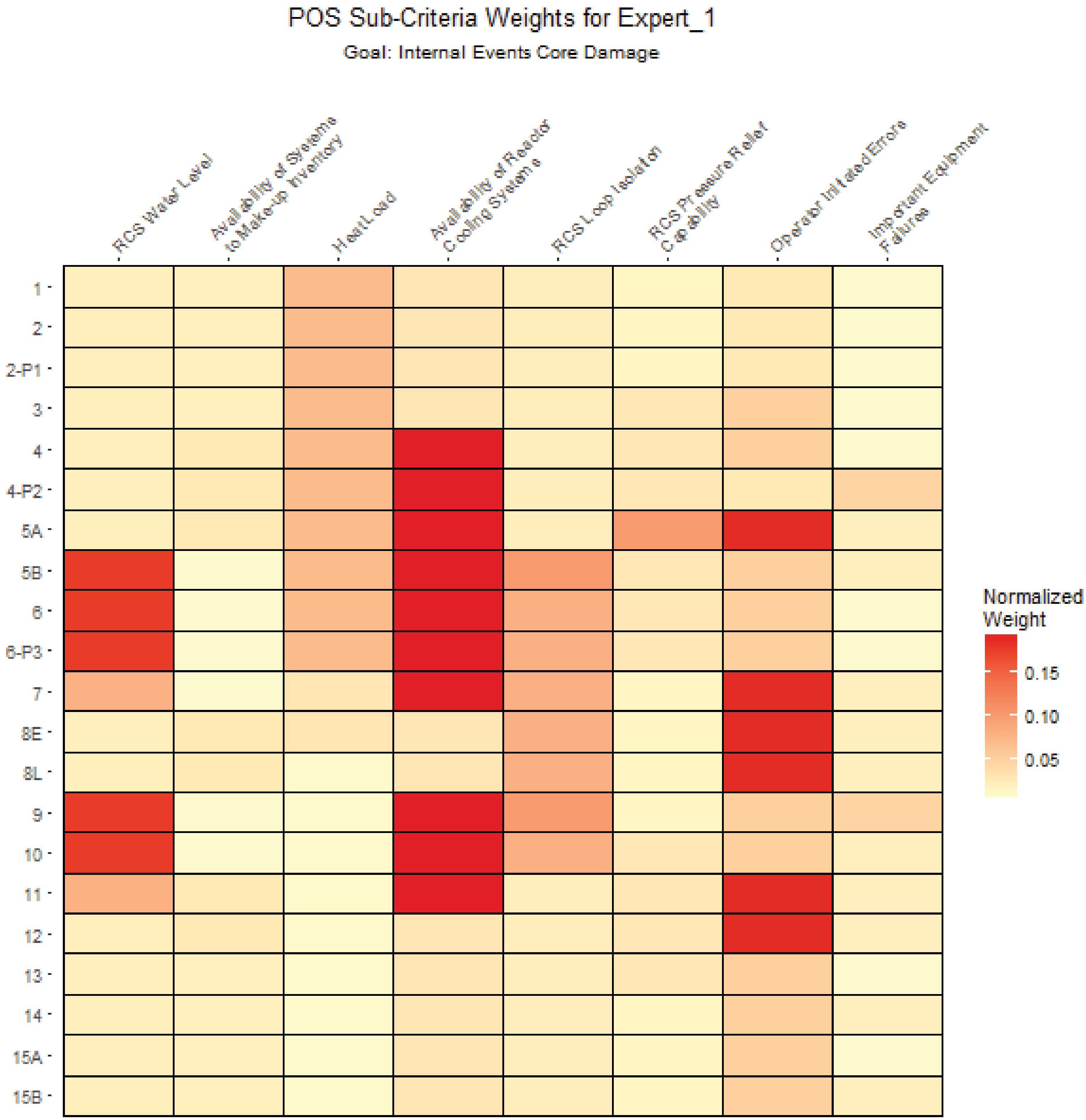
Aggregated POS Sub-Criteria Weights for Seismic Events Core Damage

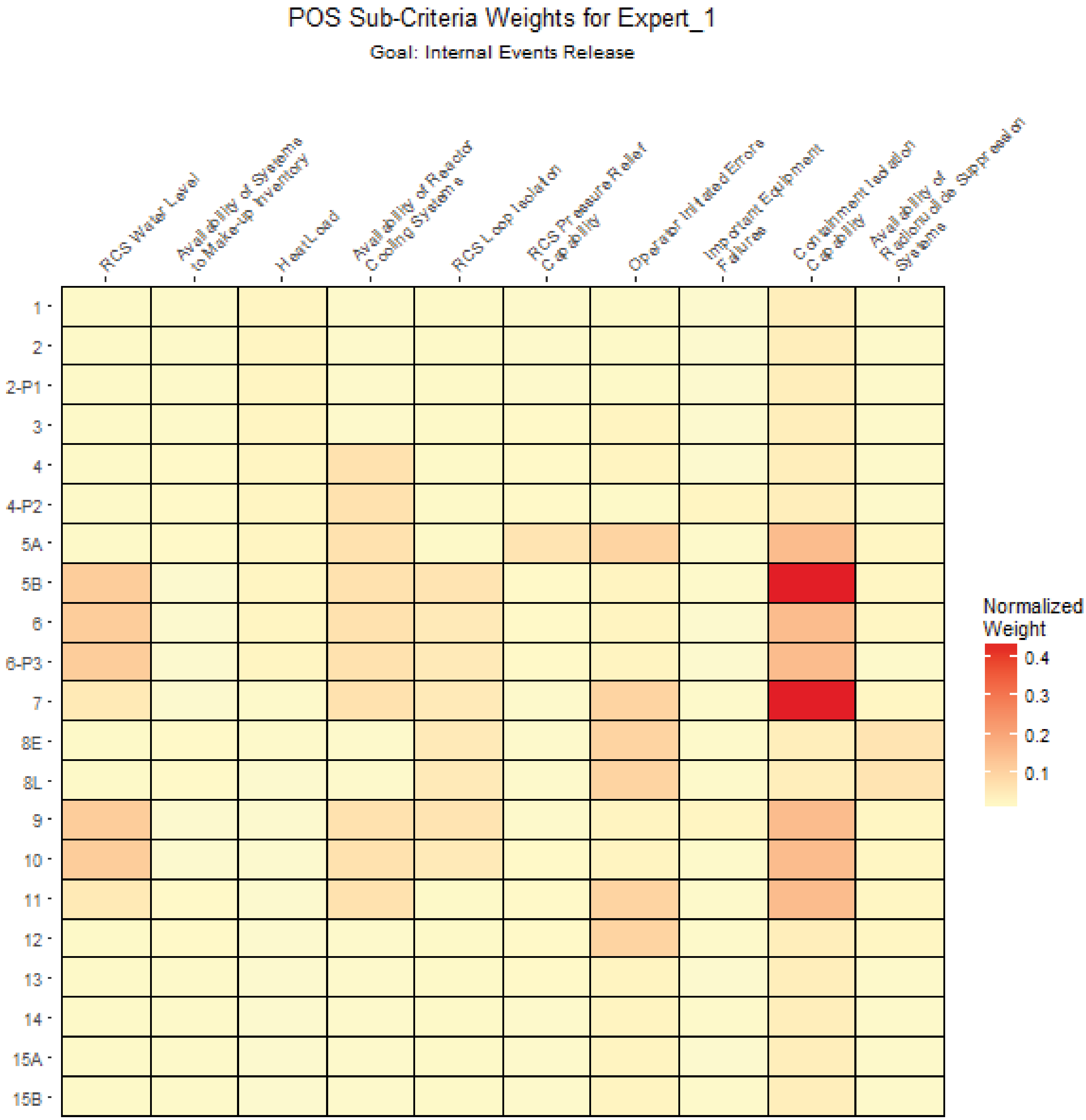


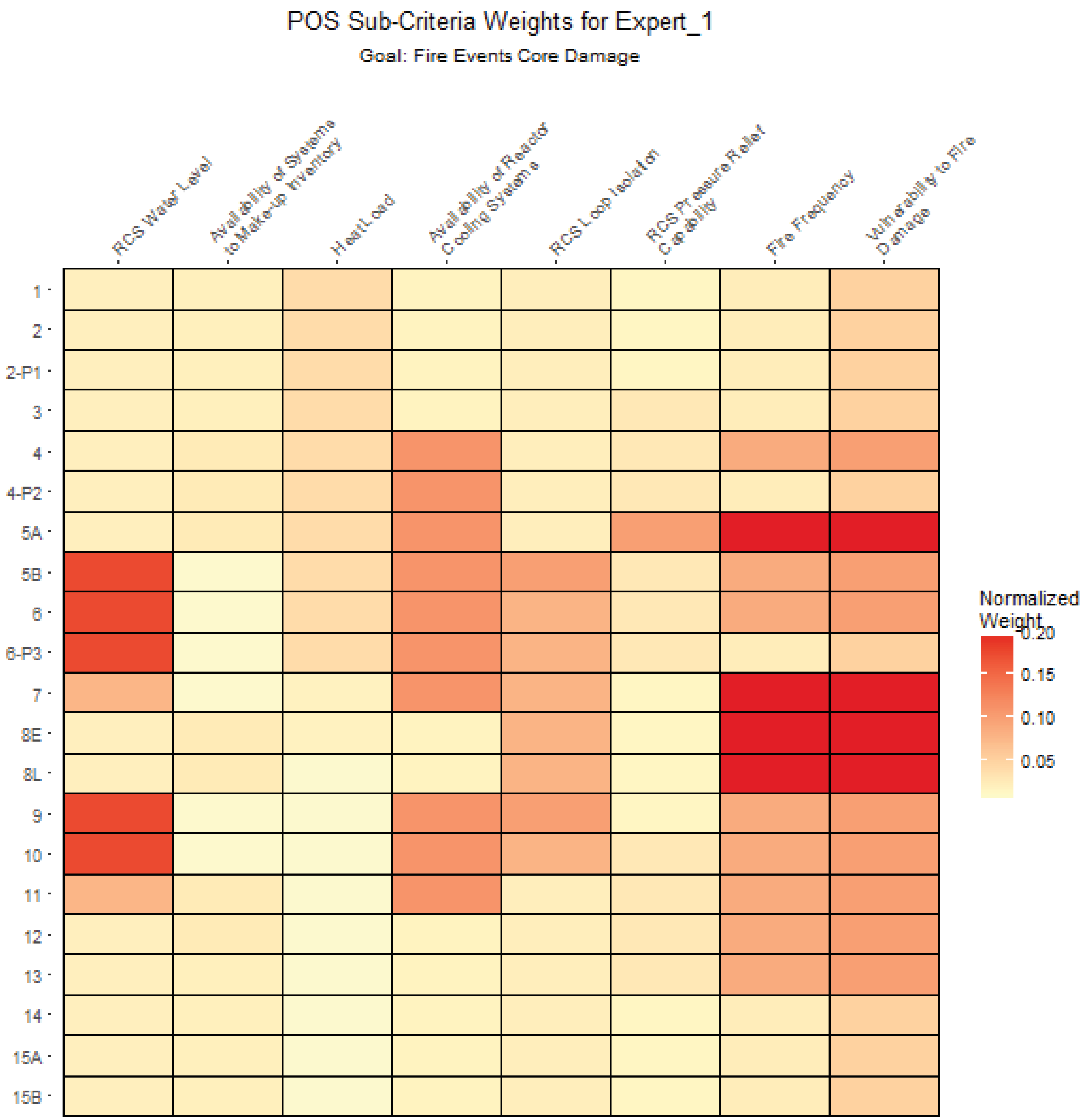
Aggregated POS Sub-Criteria Weights for Seismic Events Release

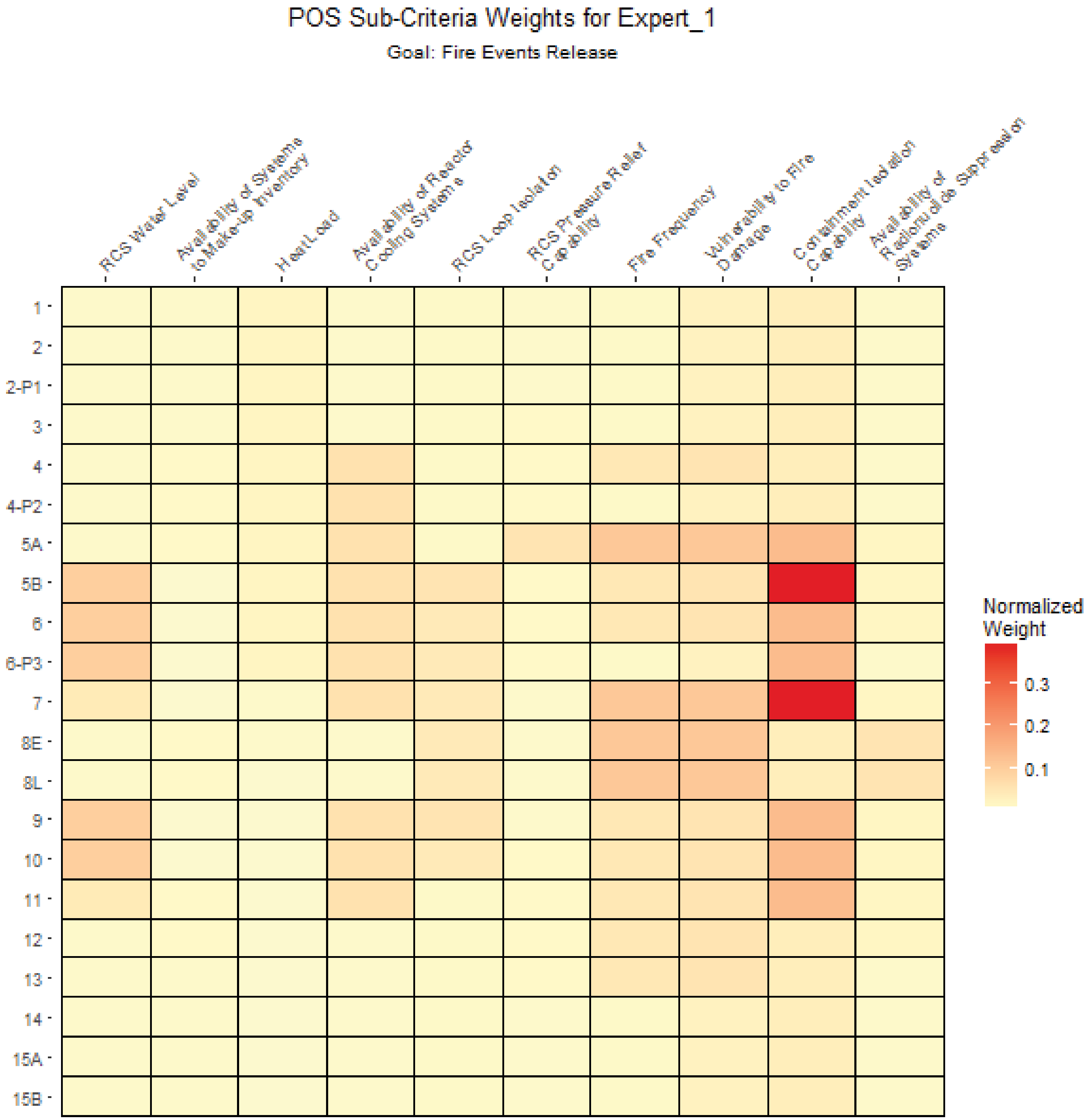


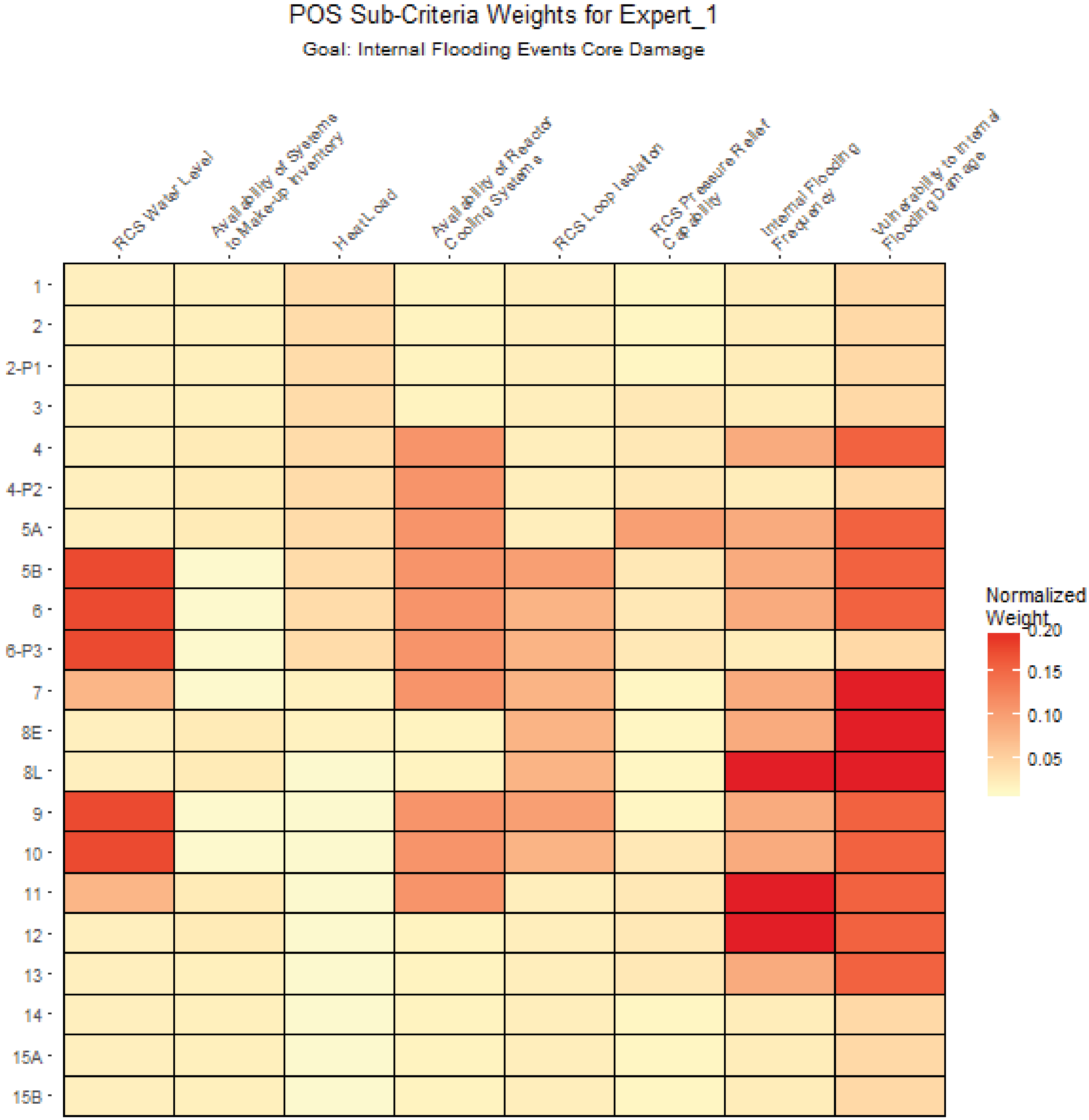
K.2 Expert 1 Results



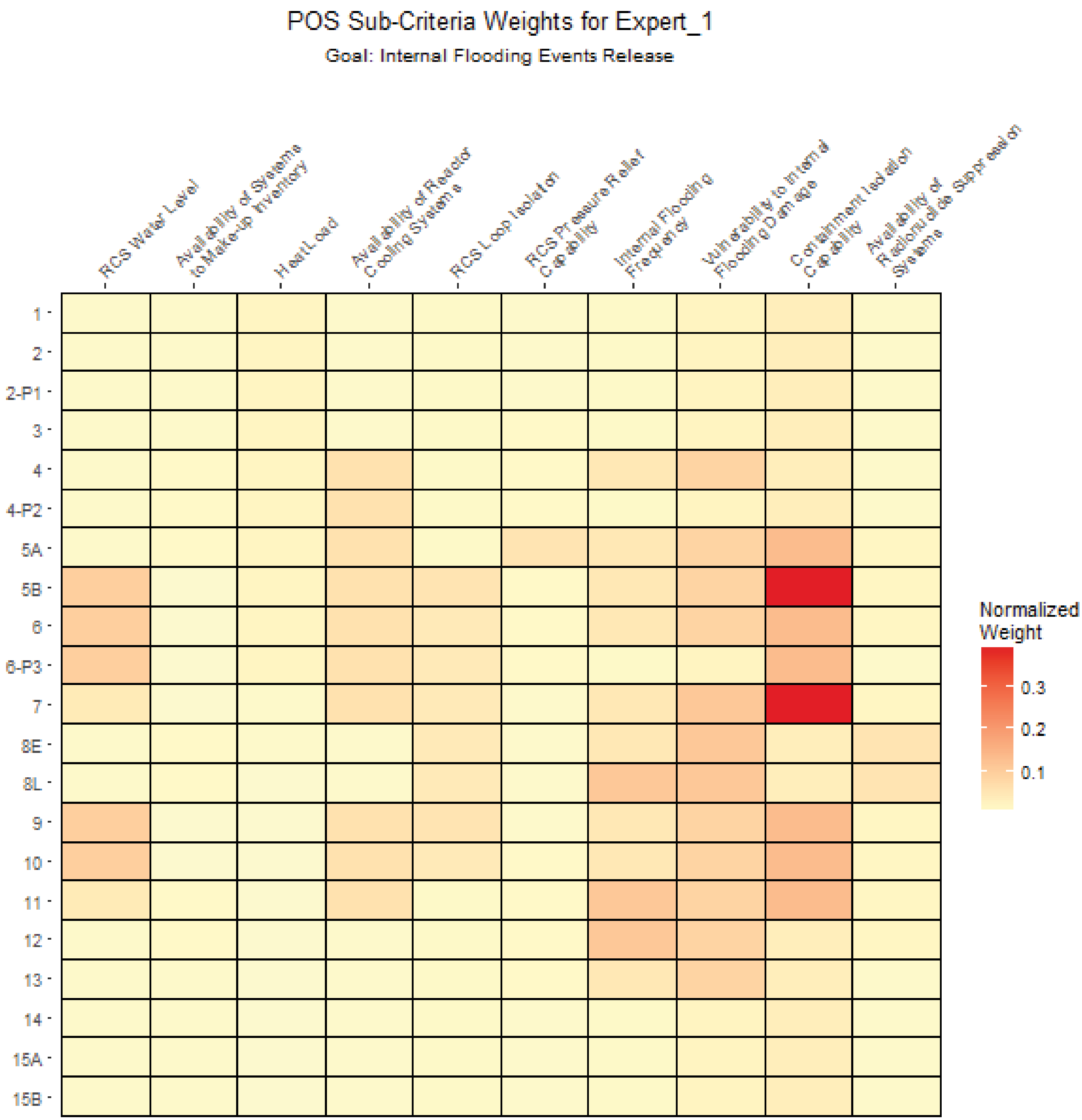

POS Sub-Criteria Weights for Expert_1
Goal: Internal Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Operator Initiated Errors
Important Equipment Failures
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.4
0.3
0.2
0.1

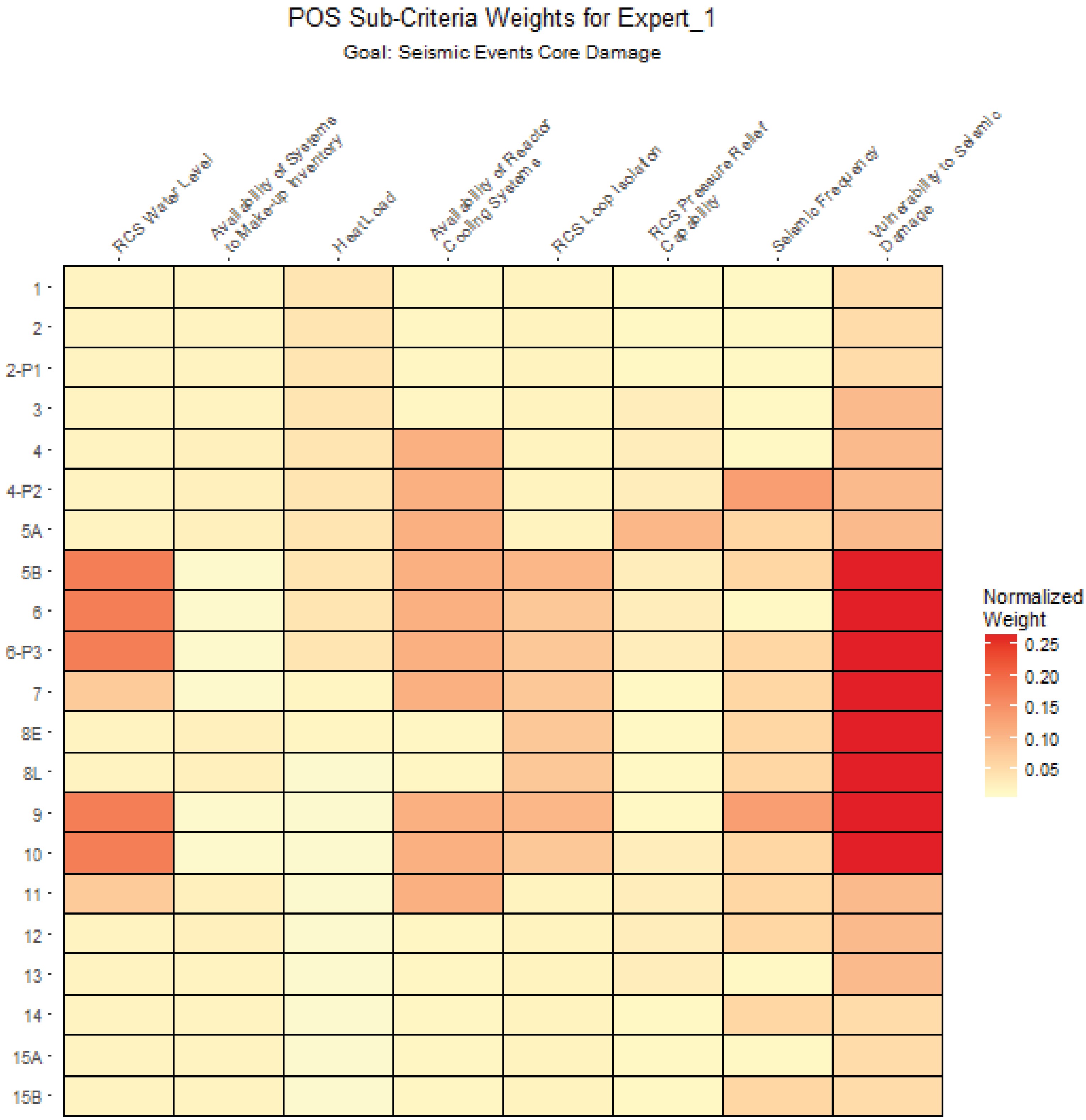

POS Sub-Criteria Weights for Expert_1
Goal: Fire Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.20
0.15
0.10
0.05


POS Sub-Criteria Weights for Expert_1
Goal: Fire Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1



POS Sub-Criteria Weights for Expert_1
Goal: Internal Flooding Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.20
0.15
0.10
0.05

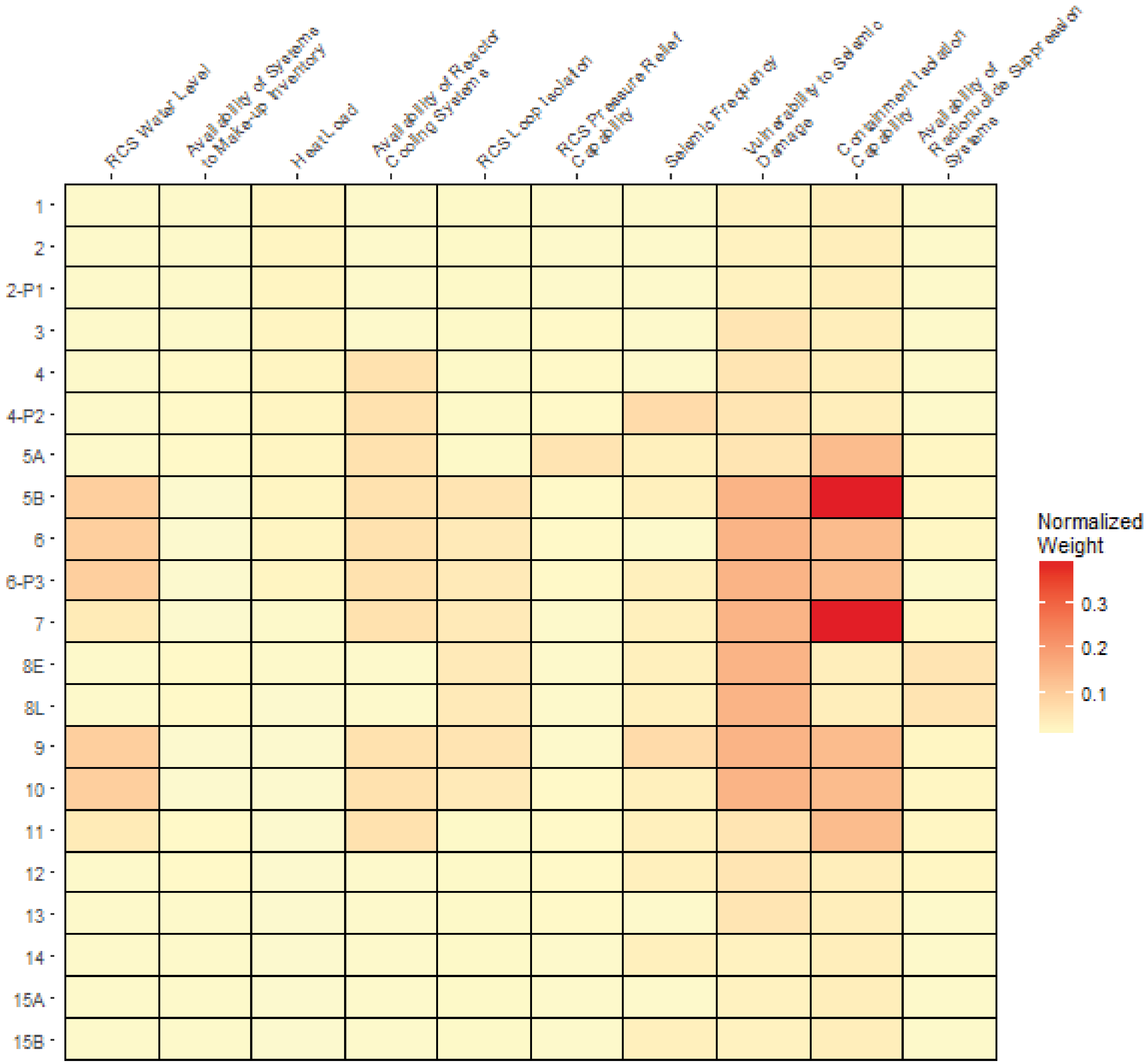

POS Sub-Criteria Weights for Expert_1
Goal: Internal Flooding Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1



POS Sub-Criteria Weights for Expert_1
Goal: Seismic Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.25
0.20
0.15
0.10
0.05

POS Sub-Criteria Weights for Expert_1

Goal: Seismic Events Release



Expert 1

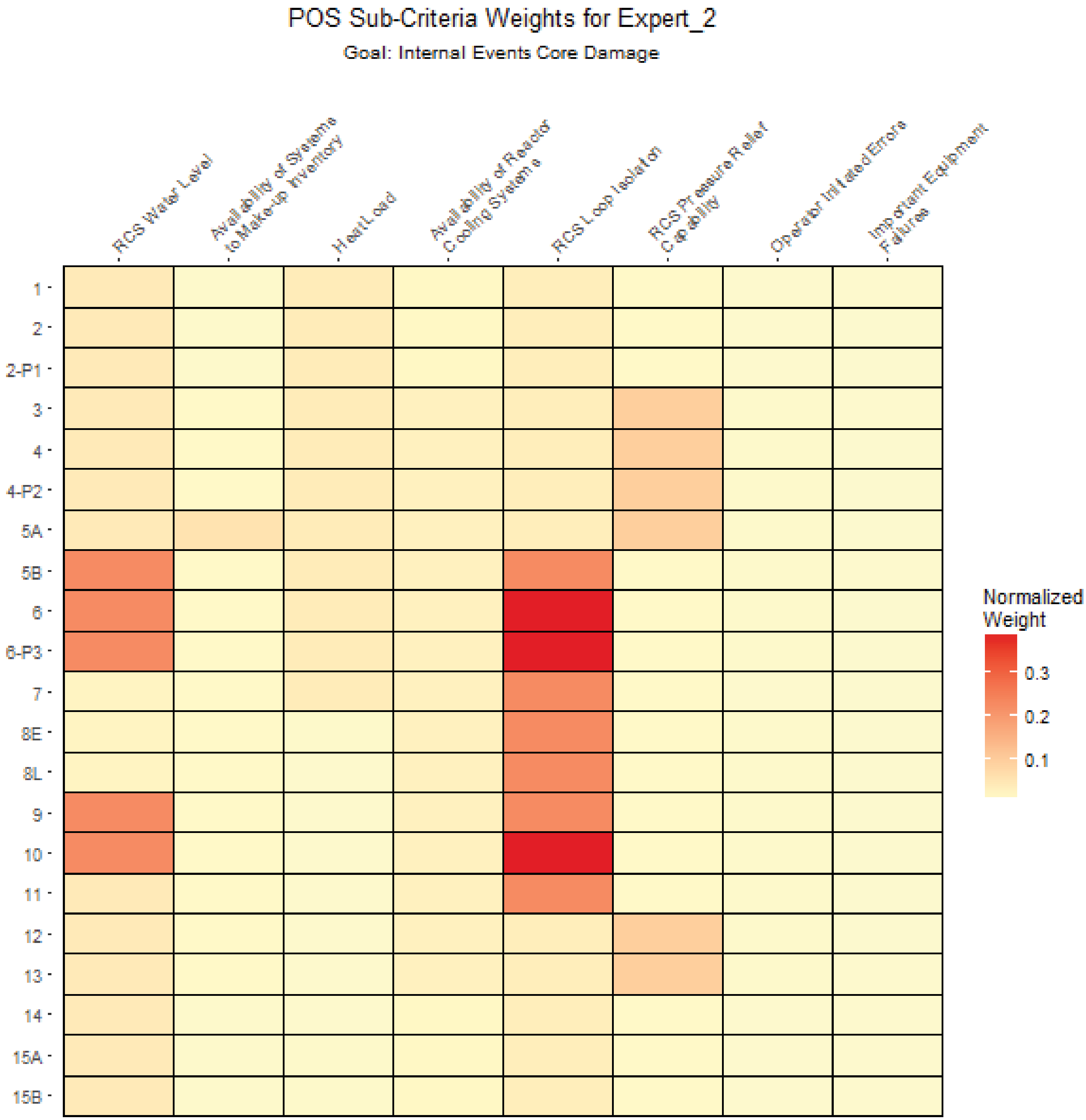
Top-Level Criteria Weights

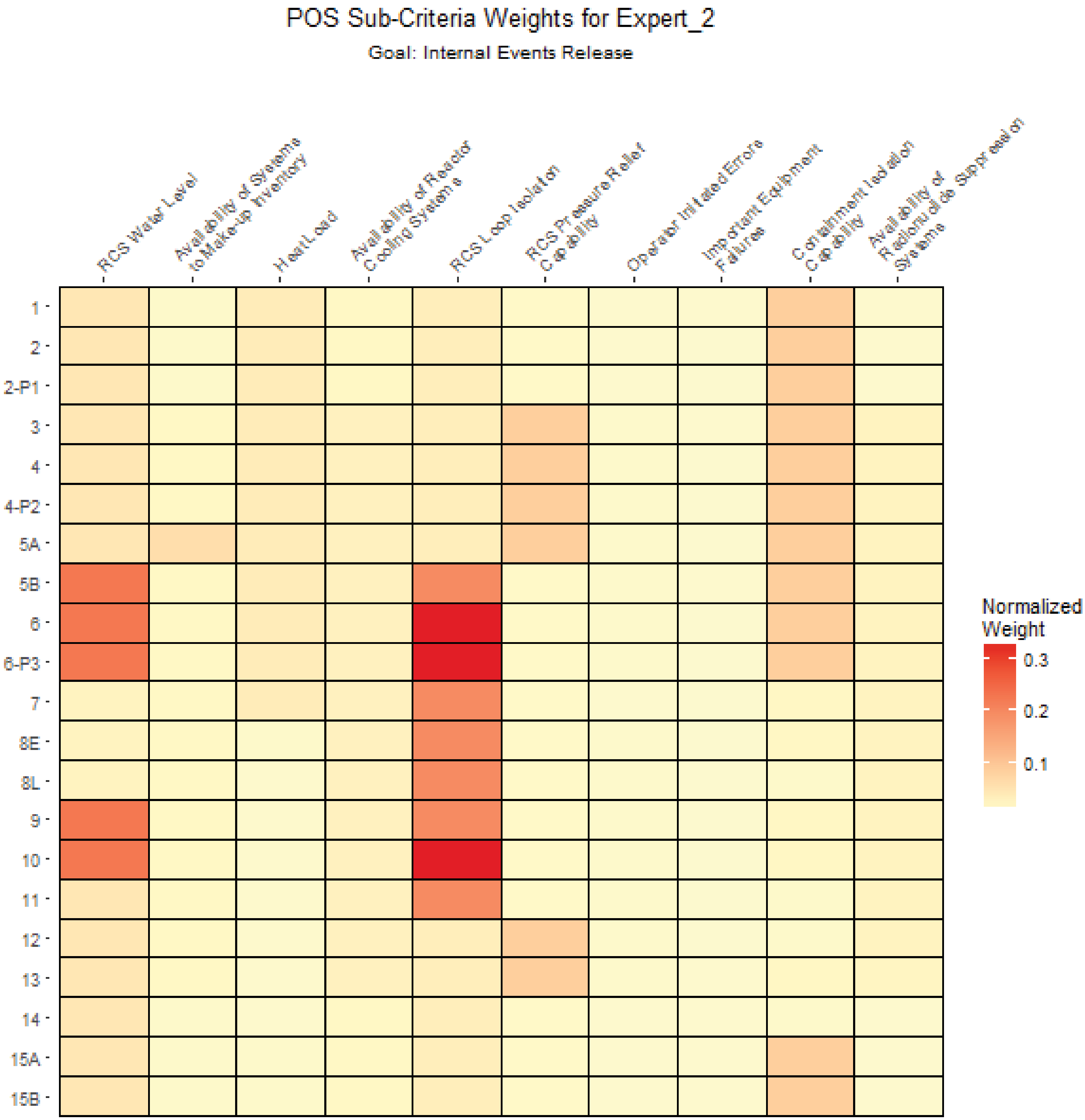
	Normalized Weights							
	Internal Events		Fire Events		Internal Flooding Events		Seismic Events	
Top-Level Criteria	Core Damage	Release	Core Damage	Release	Core Damage	Release	Core Damage	Release
RCS Inventory Control	0.21	0.13	0.20	0.11	0.20	0.11	0.20	0.11
Heat Removal	0.35	0.12	0.20	0.11	0.20	0.11	0.20	0.11
RCS Integrity	0.21	0.12	0.20	0.11	0.20	0.11	0.20	0.11
Hazard	0.24	0.12	0.40	0.22	0.40	0.22	0.40	0.22
Containment Performance		0.50		0.44		0.44		0.44
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

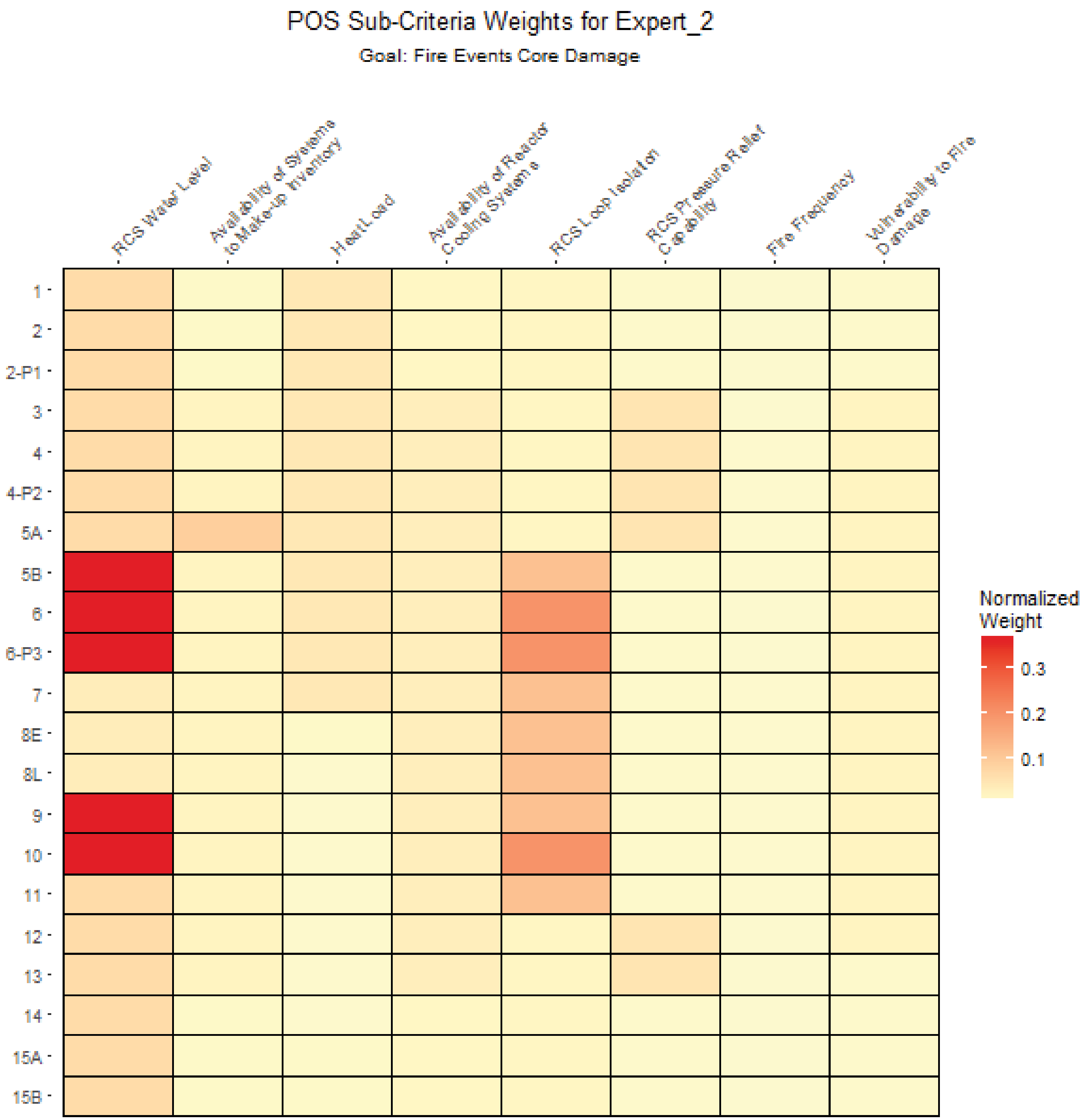
Sub-Criteria Weights

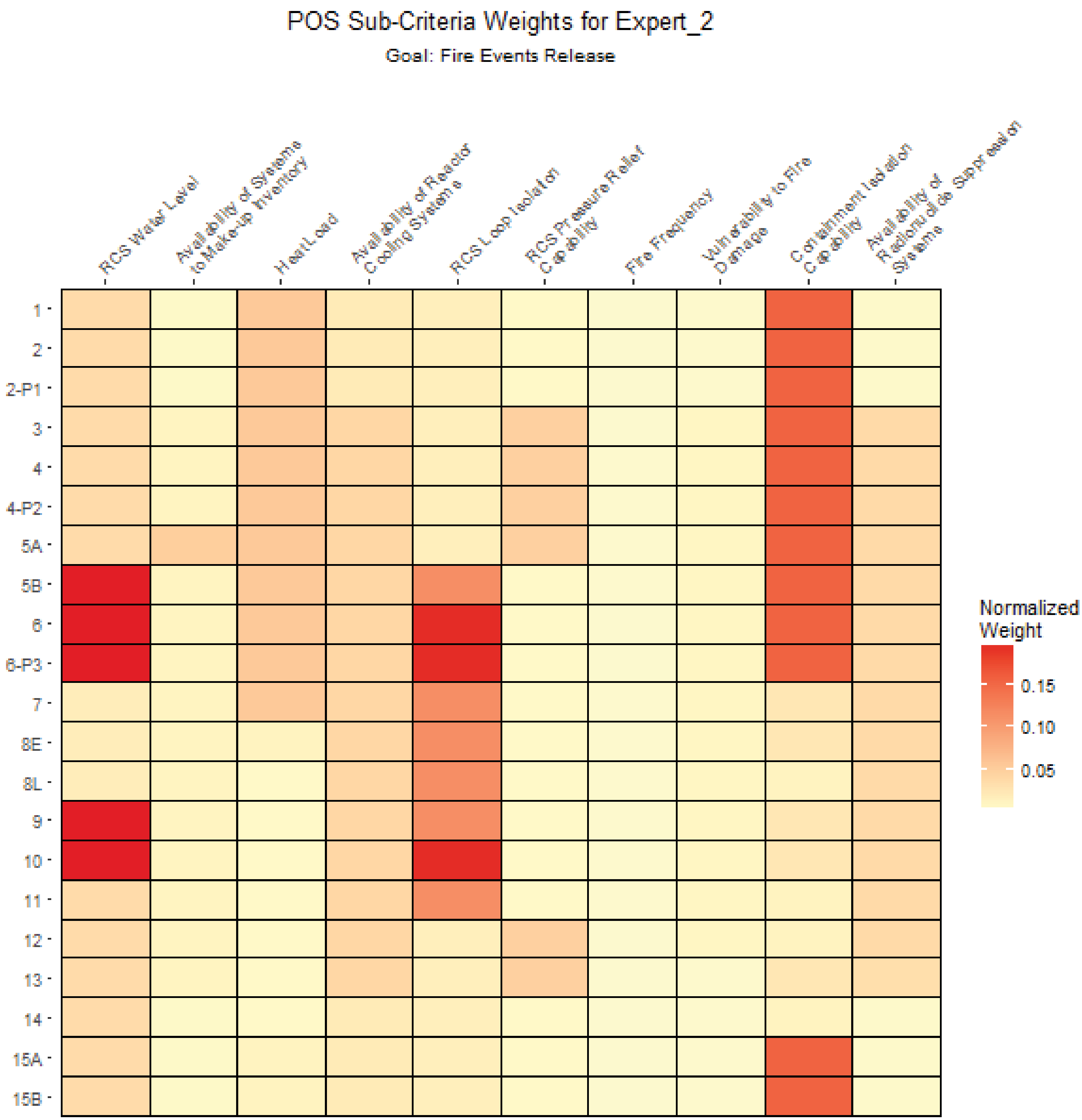
Sub-Criteria	Normalized Weights							
	RCS Inventory Control	Heat Removal	RCS Integrity	Internal Events Hazard	Containment Performance	Fire Events Hazard	Internal Flooding Hazard	Seismic Events Hazard
RCS Water Level	0.88							
Availability of Systems to Make-up Inventory	0.13							
Heat Load		0.20						
Availability of Reactor Cooling Systems		0.80						
RCS Loop Isolation			0.50					
RCS Pressure Relief Capability			0.50					
Operator Initiated Errors				0.80				
Important Equipment Failures				0.20				
Containment Isolation Capability					0.88			
Availability of Radionuclide Suppression Systems					0.13			
Fire Frequency						0.50		
Vulnerability to Fire Damage						0.50		
Internal Flooding Frequency							0.50	
Vulnerability to Internal Flooding Damage							0.50	
Seismic Frequency								0.33
Vulnerability to Seismic Damage								0.67
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

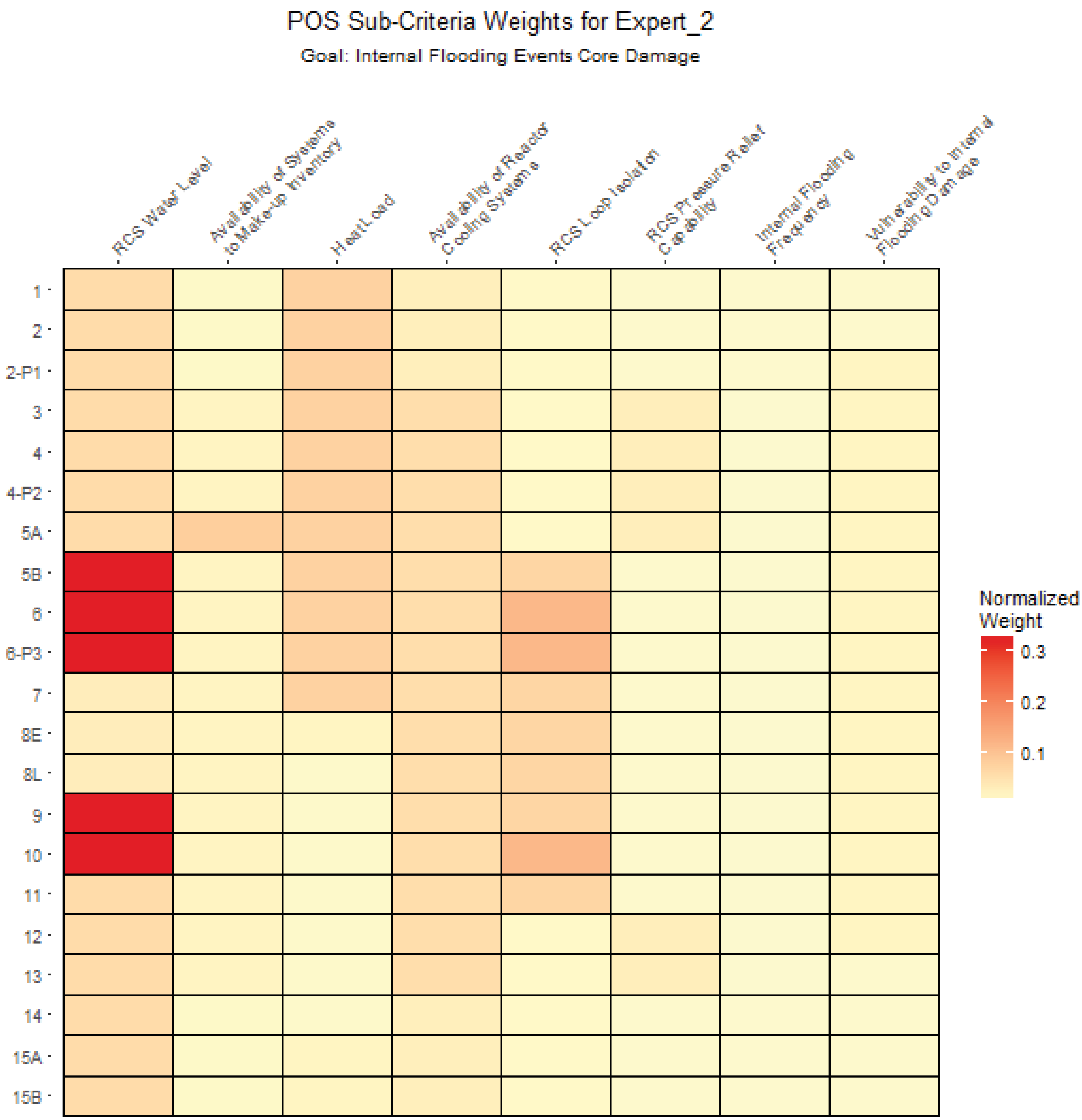
K.3 Expert 2 Results



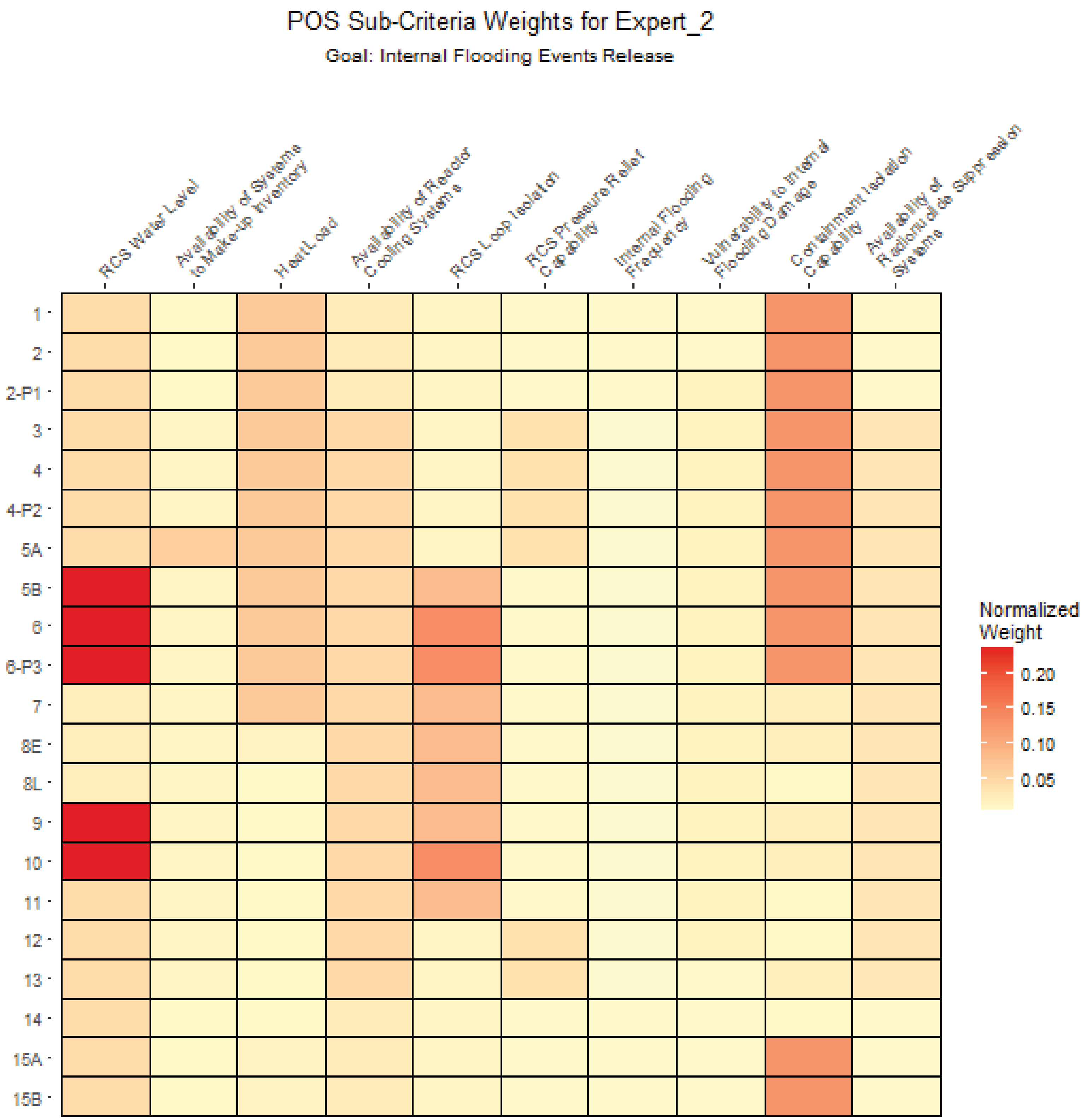

POS Sub-Criteria Weights for Expert_2
Goal: Internal Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Operator Initiated Errors
Important Equipment Failures
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1

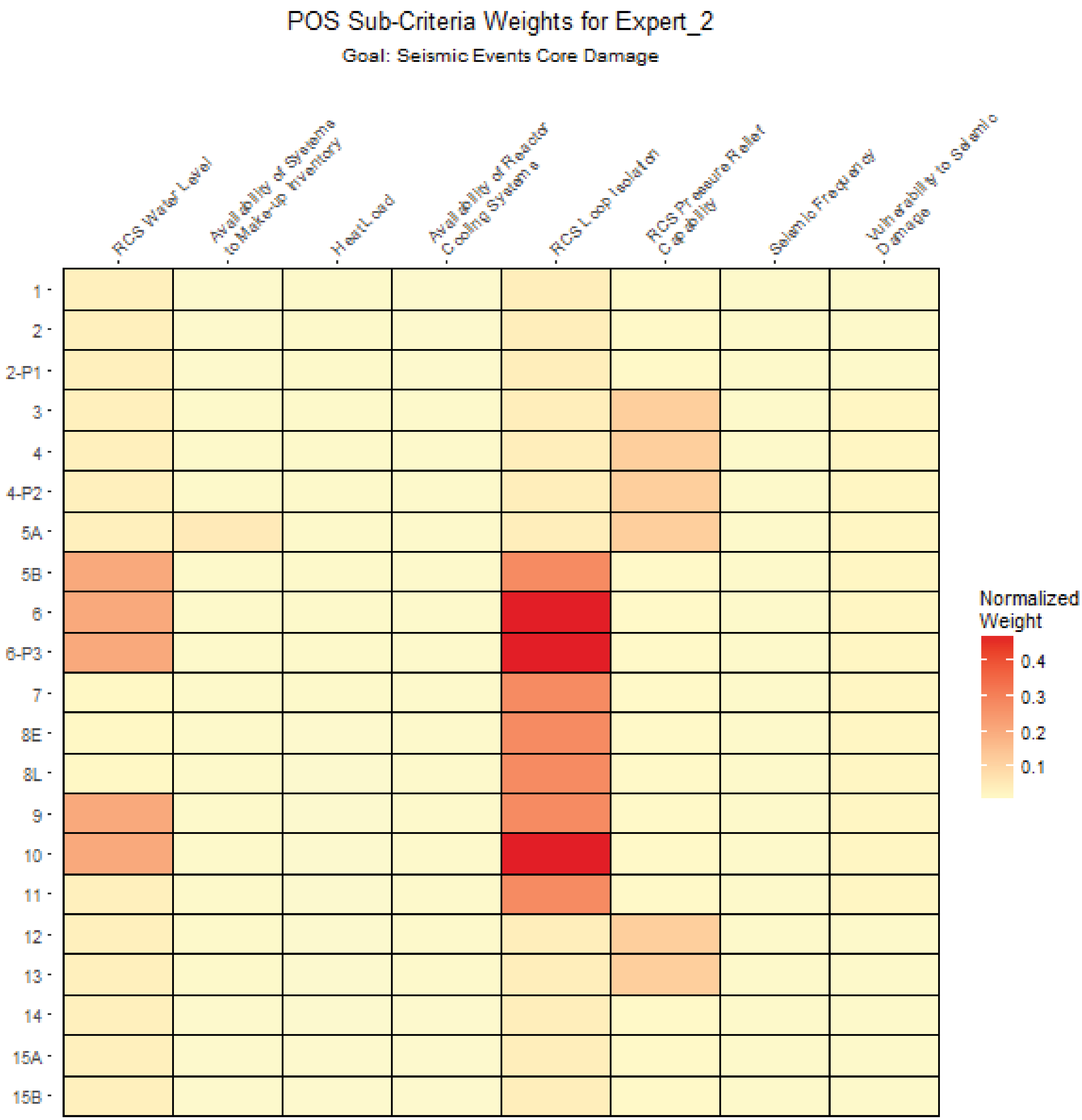

POS Sub-Criteria Weights for Expert_2
Goal: Fire Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1


POS Sub-Criteria Weights for Expert_2
Goal: Fire Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.15
0.10
0.05

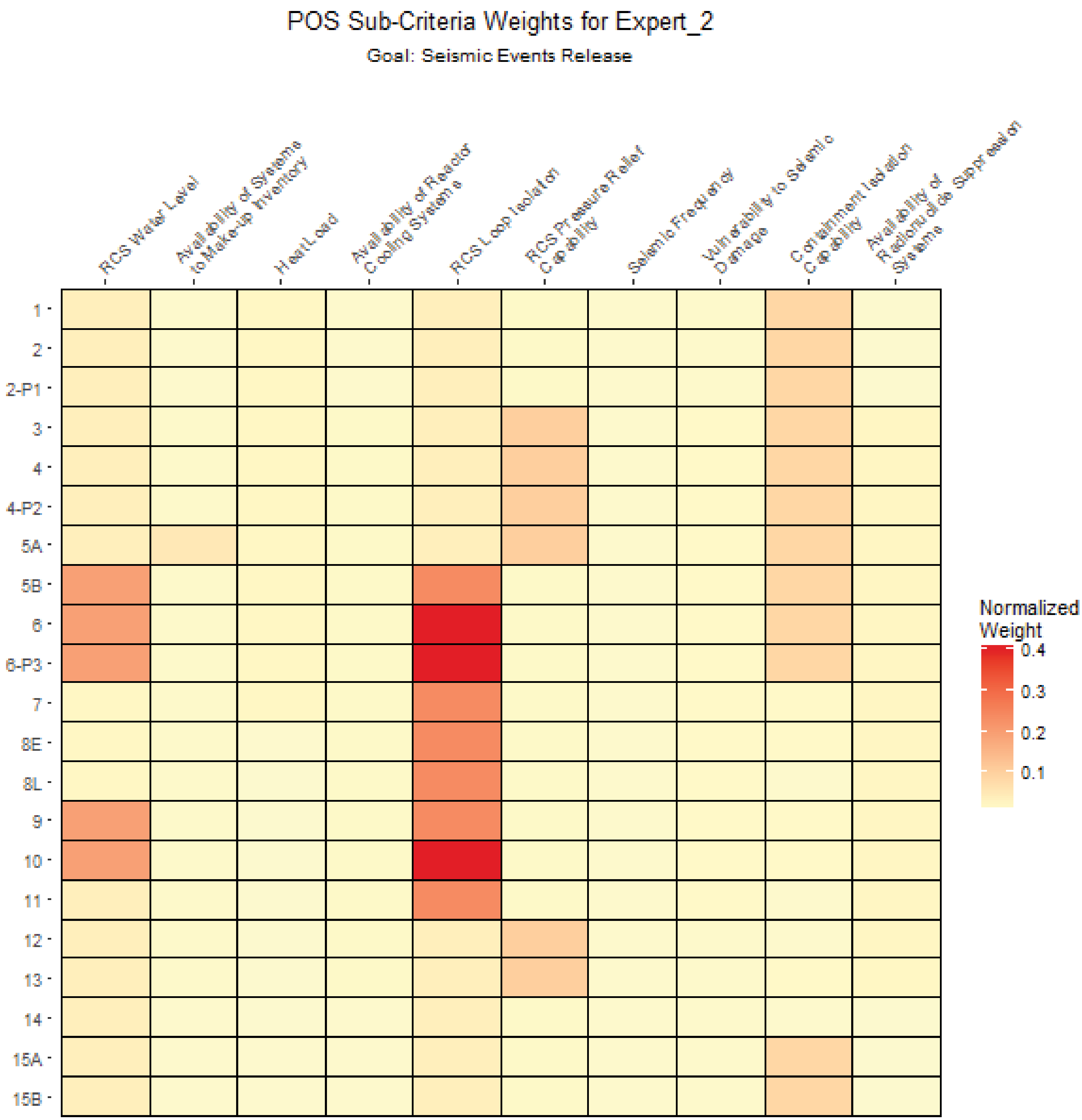


POS Sub-Criteria Weights for Expert_2
Goal: Internal Flooding Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1


POS Sub-Criteria Weights for Expert_2
Goal: Internal Flooding Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.20
0.15
0.10
0.05



POS Sub-Criteria Weights for Expert_2
Goal: Seismic Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.4
0.3
0.2
0.1



POS Sub-Criteria Weights for Expert_2
Goal: Seismic Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.4
0.3
0.2
0.1

Expert 2

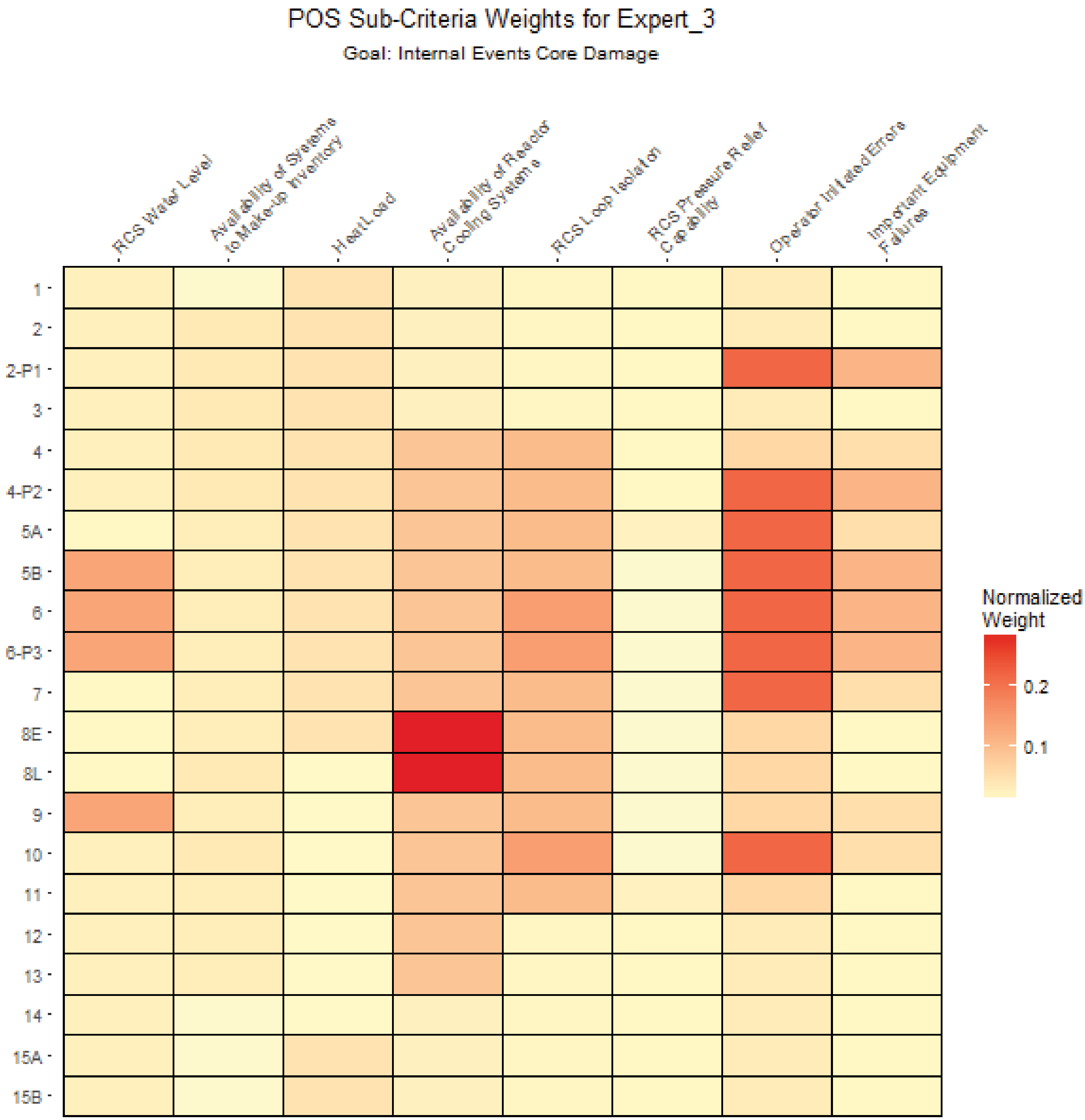
Top-Level Criteria Weights

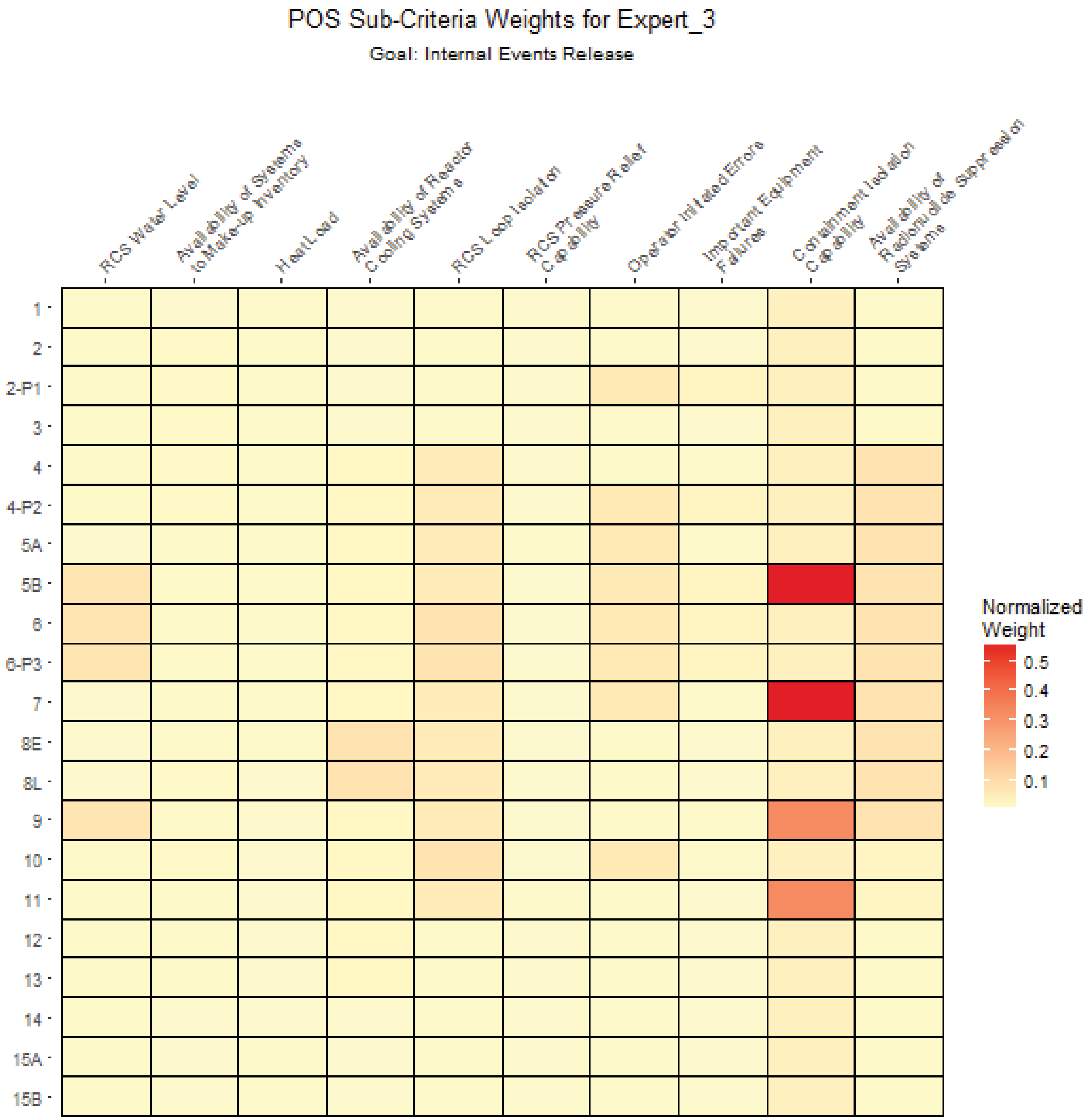
Top-Level Criteria	Normalized Weights							
	Internal Events		Fire Events		Internal Flooding Events		Seismic Events	
	Core Damage	Release	Core Damage	Release	Core Damage	Release	Core Damage	Release
RCS Inventory Control	0.28	0.28	0.45	0.25	0.41	0.29	0.26	0.24
Heat Removal	0.20	0.17	0.22	0.28	0.38	0.32	0.05	0.09
RCS Integrity	0.49	0.42	0.25	0.24	0.15	0.17	0.59	0.50
Hazard	0.03	0.03	0.08	0.04	0.07	0.06	0.10	0.06
Containment Performance		0.10		0.19		0.16		0.11
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Sub-Criteria Weights

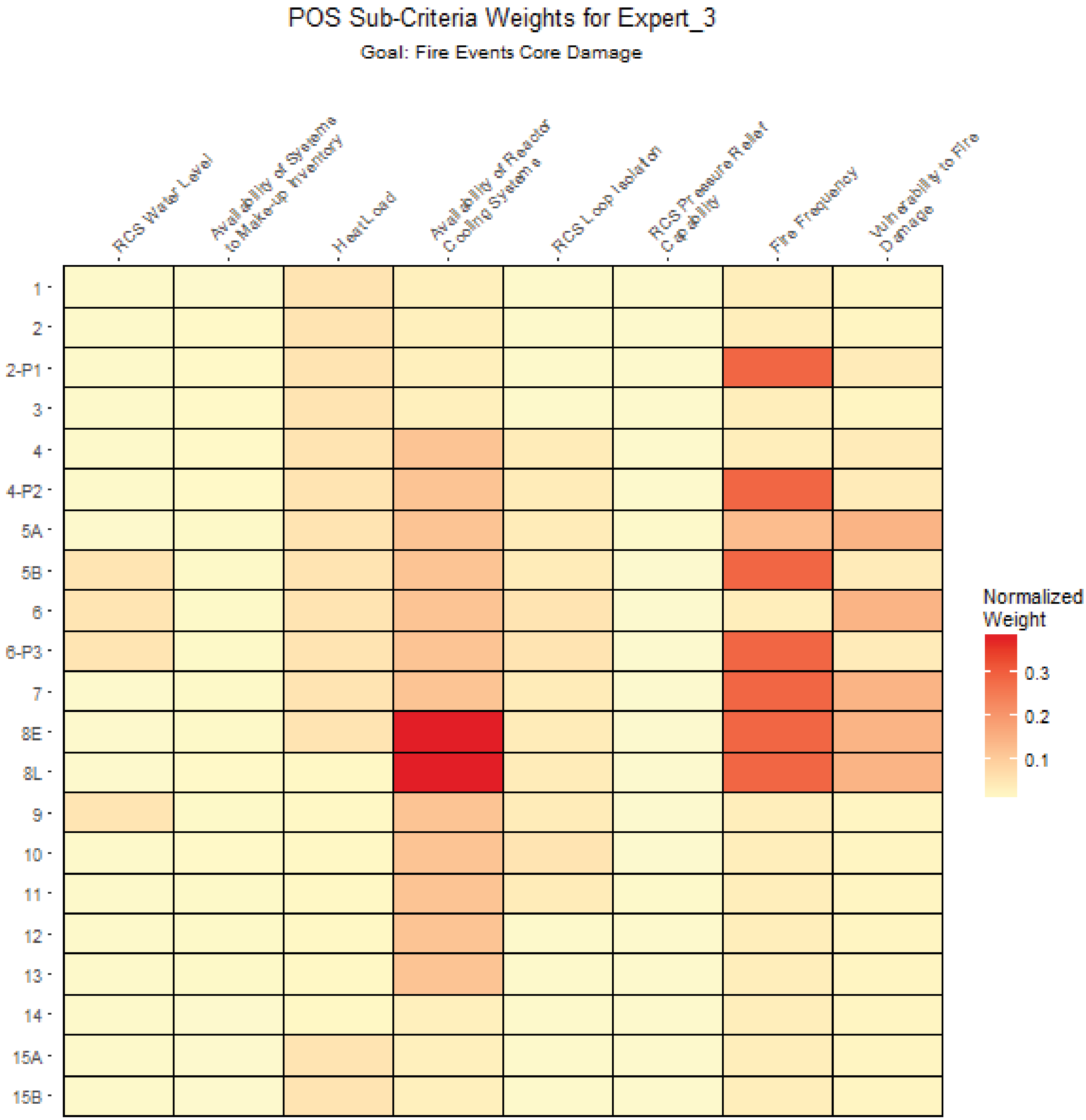
Sub-Criteria	Normalized Weights							
	RCS Inventory Control	Heat Removal	RCS Integrity	Internal Events Hazard	Containment Performance	Fire Events Hazard	Internal Flooding Hazard	Seismic Events Hazard
RCS Water Level	0.80							
Availability of Systems to Make-up Inventory	0.20							
Heat Load		0.20						
Availability of Reactor Cooling Systems		0.80						
RCS Loop Isolation			0.80					
RCS Pressure Relief Capability			0.20					
Operator Initiated Errors				0.88				
Important Equipment Failures				0.13				
Containment Isolation Capability					0.80			
Availability of Radionuclide Suppression Systems					0.20			
Fire Frequency						0.13		
Vulnerability to Fire Damage						0.88		
Internal Flooding Frequency							0.13	
Vulnerability to Internal Flooding Damage							0.88	
Seismic Frequency								0.10
Vulnerability to Seismic Damage								0.90
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

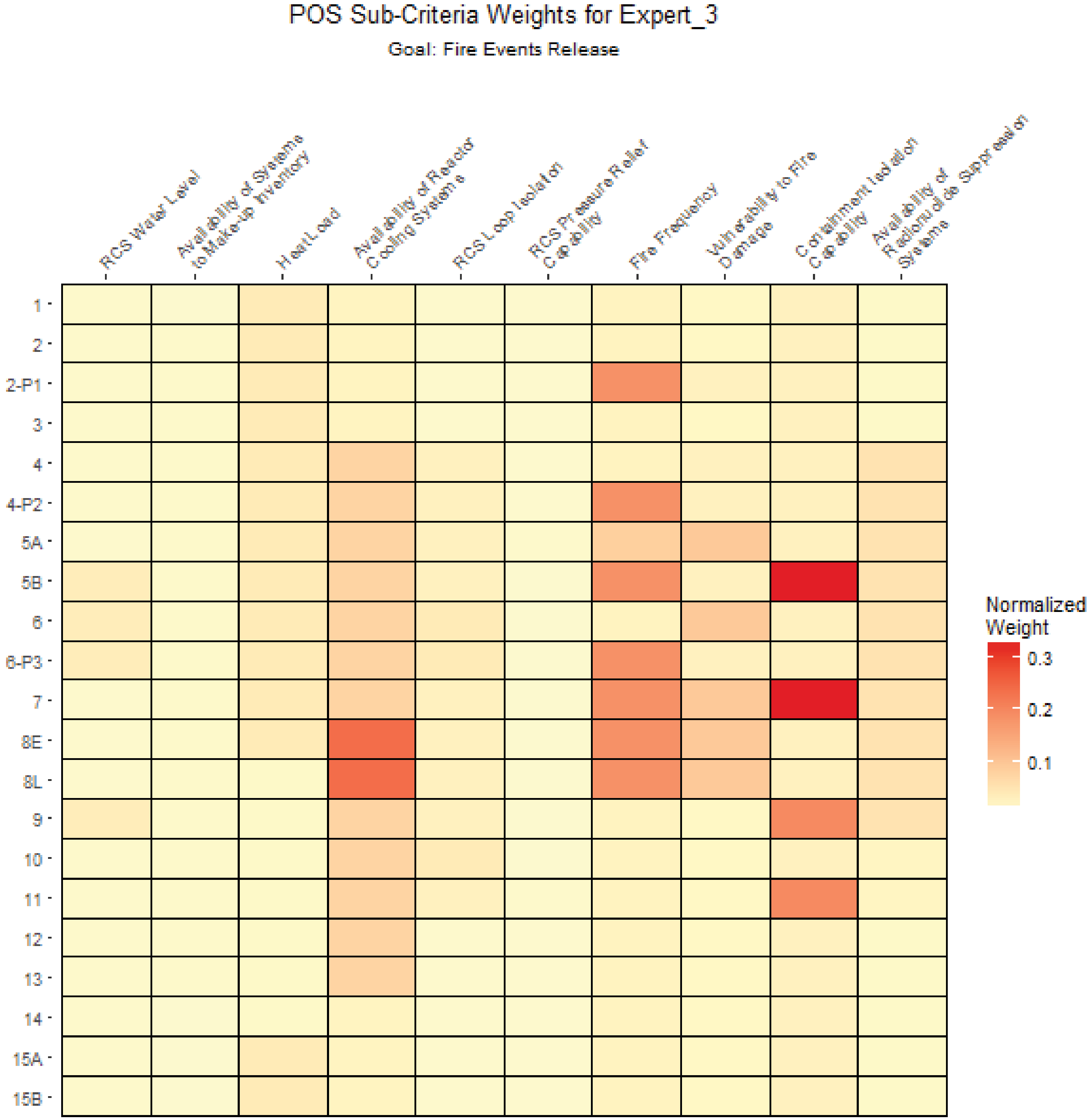
K.4 Expert 3 Results



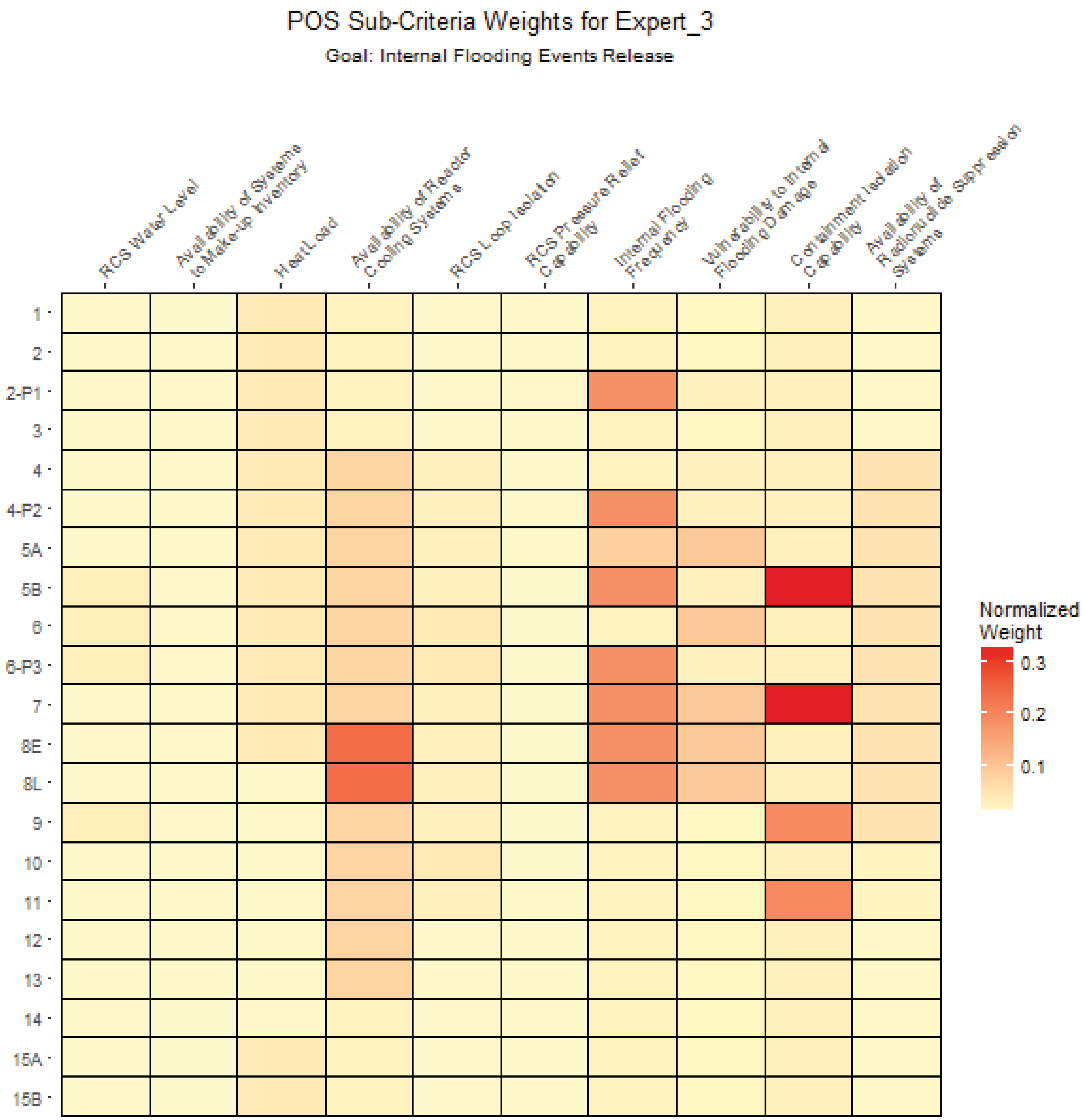


POS Sub-Criteria Weights for Expert_3
Goal: Internal Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Operator Initiated Errors
Important Equipment Failures
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.5
0.4
0.3
0.2
0.1

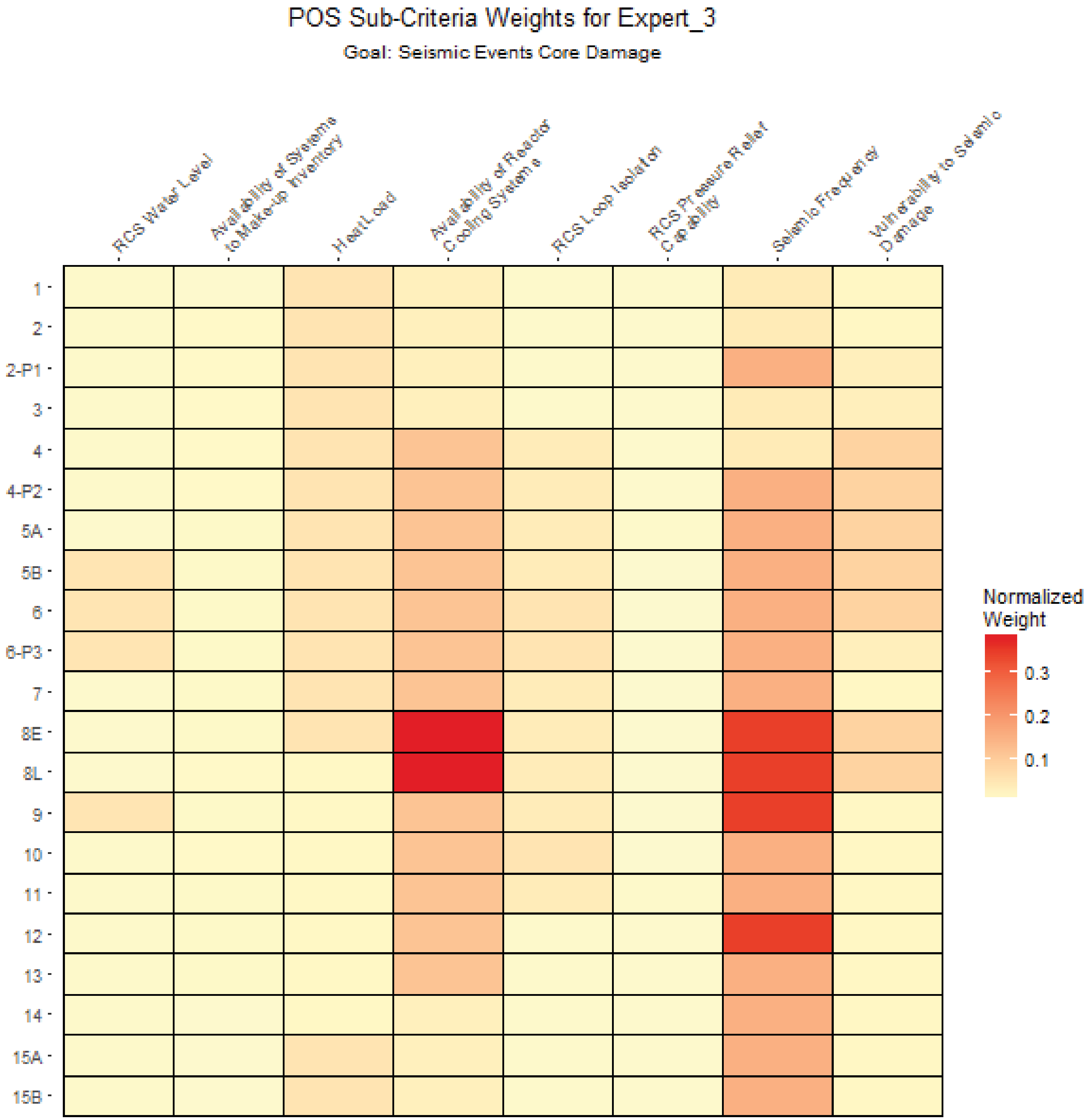

POS Sub-Criteria Weights for Expert_3
Goal: Fire Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1


POS Sub-Criteria Weights for Expert_3
Goal: Fire Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1


POS Sub-Criteria Weights for Expert_3
Goal: Internal Flooding Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1



POS Sub-Criteria Weights for Expert_3
Goal: Internal Flooding Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1


POS Sub-Criteria Weights for Expert_3
Goal: Seismic Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
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8E
8L
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10
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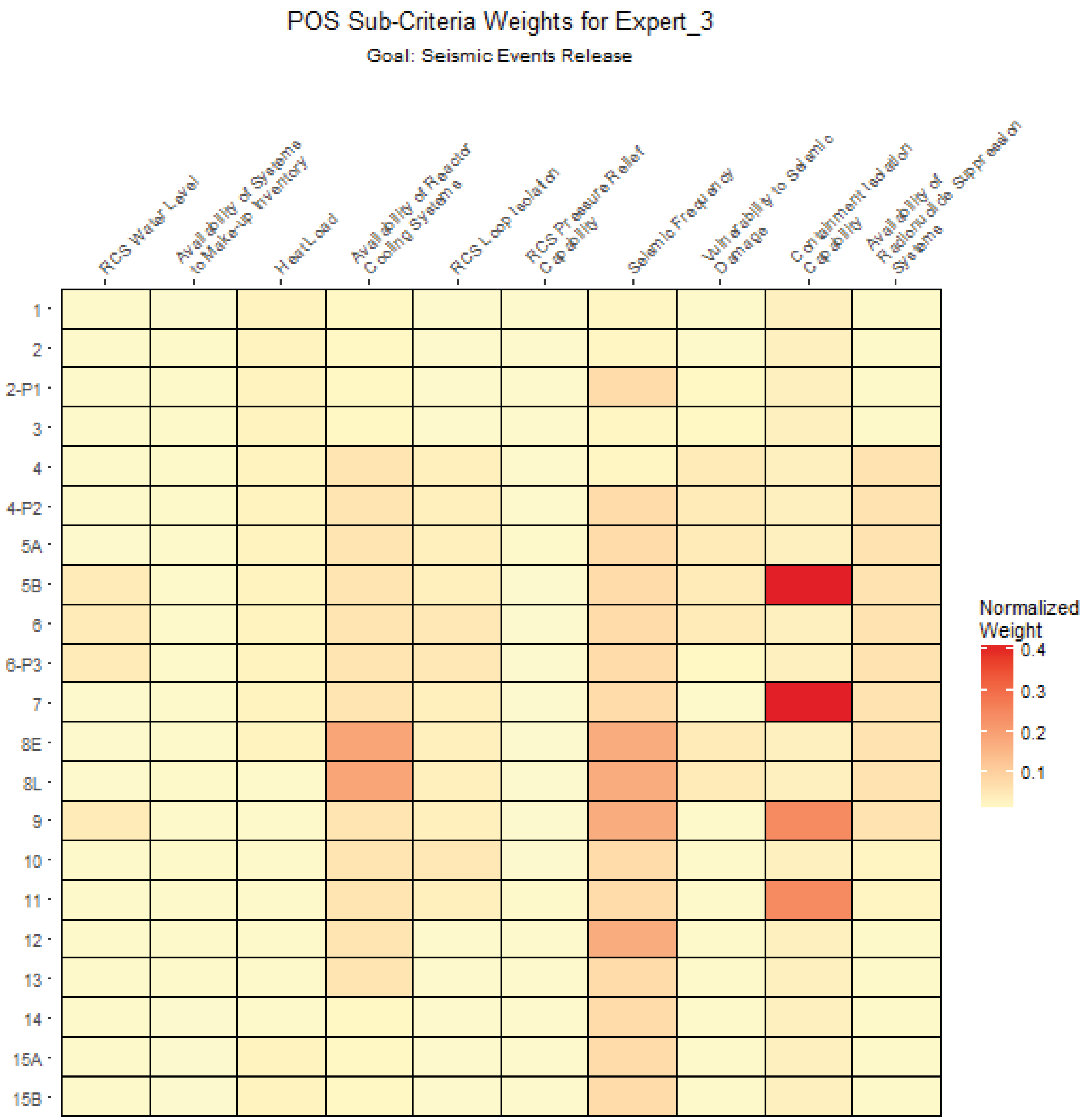
14
15A
15B
Normalized Weight
0.3
0.2
0.1


POS Sub-Criteria Weights for Expert_3
Goal: Seismic Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
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4-P2
5A
5B
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6-P3
7
8E
8L
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10
11
12
13
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15A
15B
Normalized Weight
0.4
0.3
0.2
0.1

Expert 3

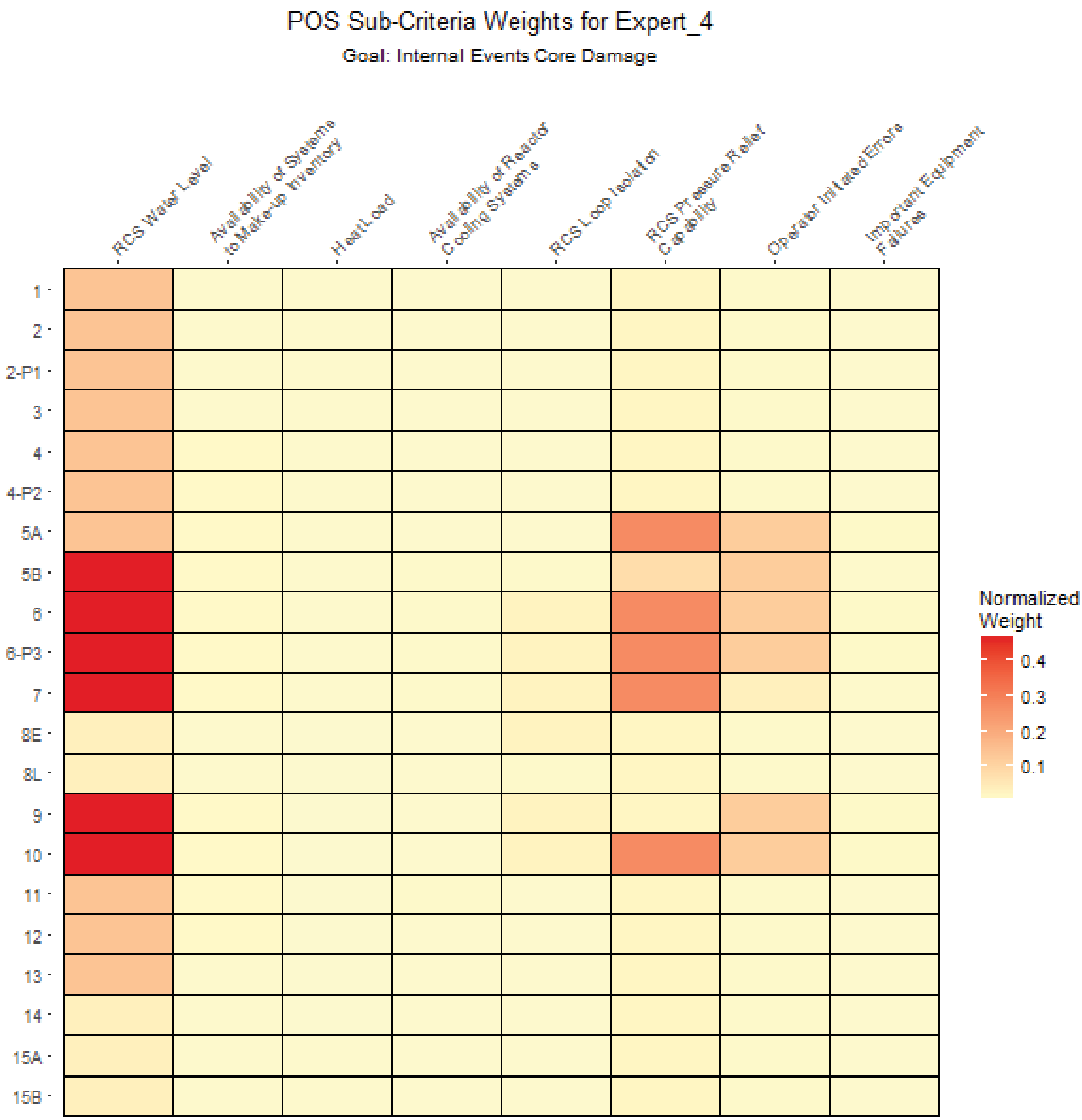
Top-Level Criteria Weights

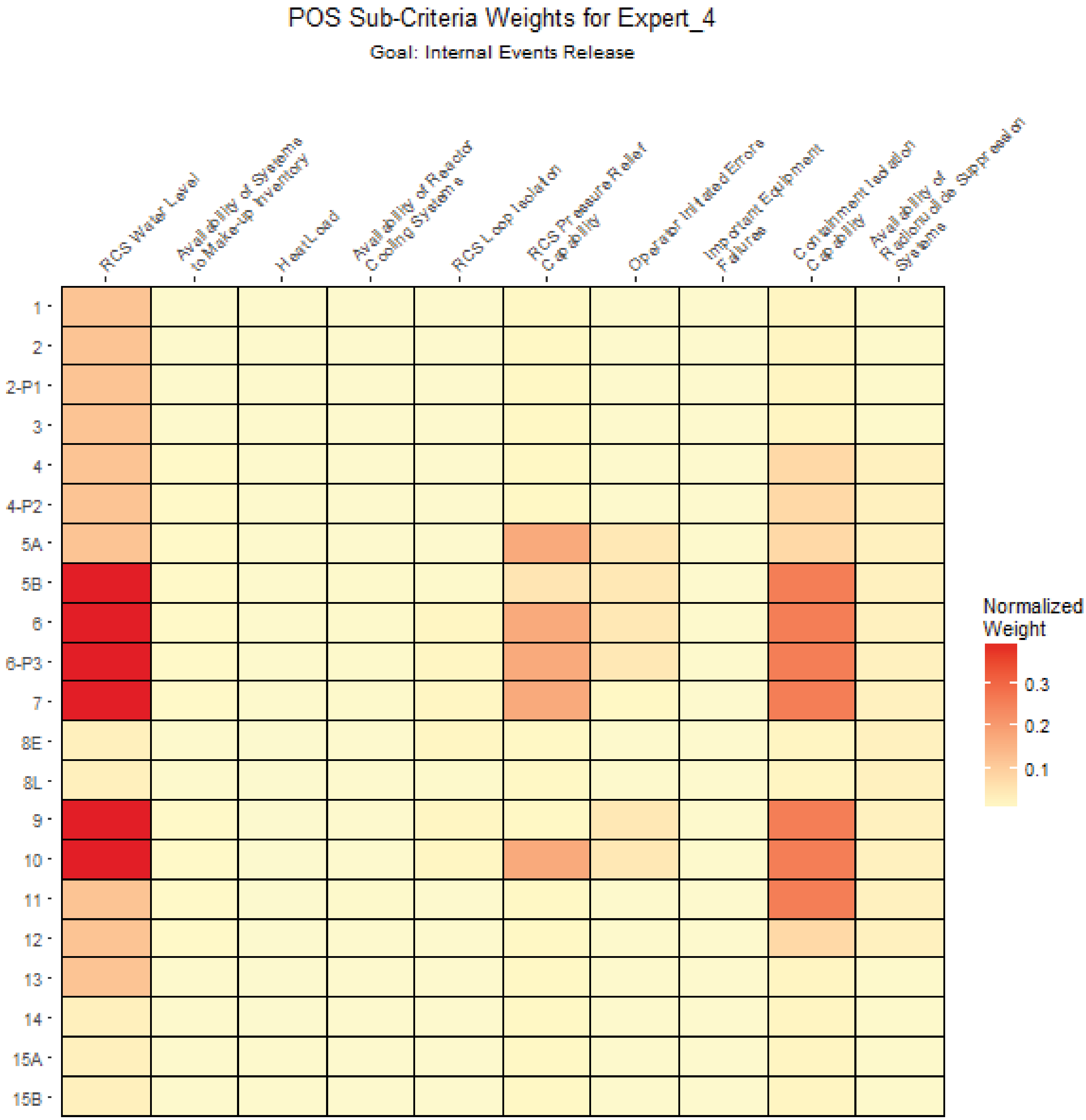
	Normalized Weights							
	Internal Events		Fire Events		Internal Flooding Events		Seismic Events	
Top-Level Criteria	Core Damage	Release	Core Damage	Release	Core Damage	Release	Core Damage	Release
RCS Inventory Control	0.17	0.09	0.06	0.04	0.06	0.04	0.06	0.05
Heat Removal	0.33	0.09	0.44	0.27	0.44	0.27	0.44	0.21
RCS Integrity	0.17	0.09	0.06	0.04	0.06	0.04	0.06	0.05
Hazard	0.33	0.09	0.44	0.27	0.44	0.27	0.44	0.21
Containment Performance		0.63		0.38		0.38		0.47
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Sub-Criteria Weights

Sub-Criteria	Normalized Weights							
	RCS Inventory Control	Heat Removal	RCS Integrity	Internal Events Hazard	Containment Performance	Fire Events Hazard	Internal Flooding Hazard	Seismic Events Hazard
RCS Water Level	0.80							
Availability of Systems to Make-up Inventory	0.20							
Heat Load		0.13						
Availability of Reactor Cooling Systems		0.88						
RCS Loop Isolation			0.88					
RCS Pressure Relief Capability			0.13					
Operator Initiated Errors				0.67				
Important Equipment Failures				0.33				
Containment Isolation Capability					0.88			
Availability of Radionuclide Suppression Systems					0.13			
Fire Frequency						0.67		
Vulnerability to Fire Damage						0.33		
Internal Flooding Frequency							0.67	
Vulnerability to Internal Flooding Damage							0.33	
Seismic Frequency								0.80
Vulnerability to Seismic Damage								0.20
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

K.5 Expert 4 Results





POS Sub-Criteria Weights for Expert_4
Goal: Internal Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Operator Initiated Errors
Important Equipment Failures
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
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2-P1
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4-P2
5A
5B
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6-P3
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8E
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15A
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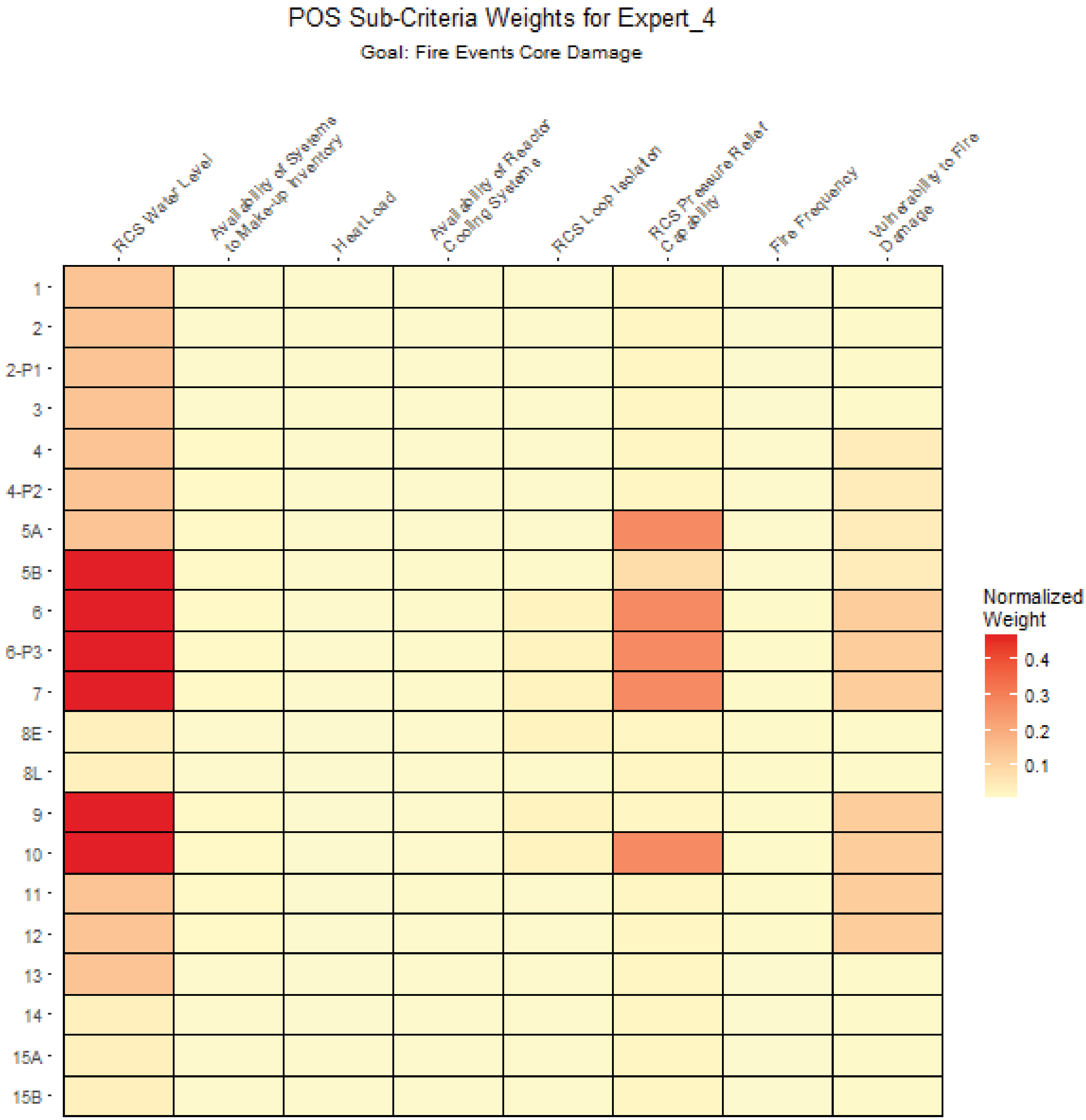
0.1



POS Sub-Criteria Weights for Expert_4
Goal: Fire Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
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8E
8L
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15A
15B
Normalized Weight
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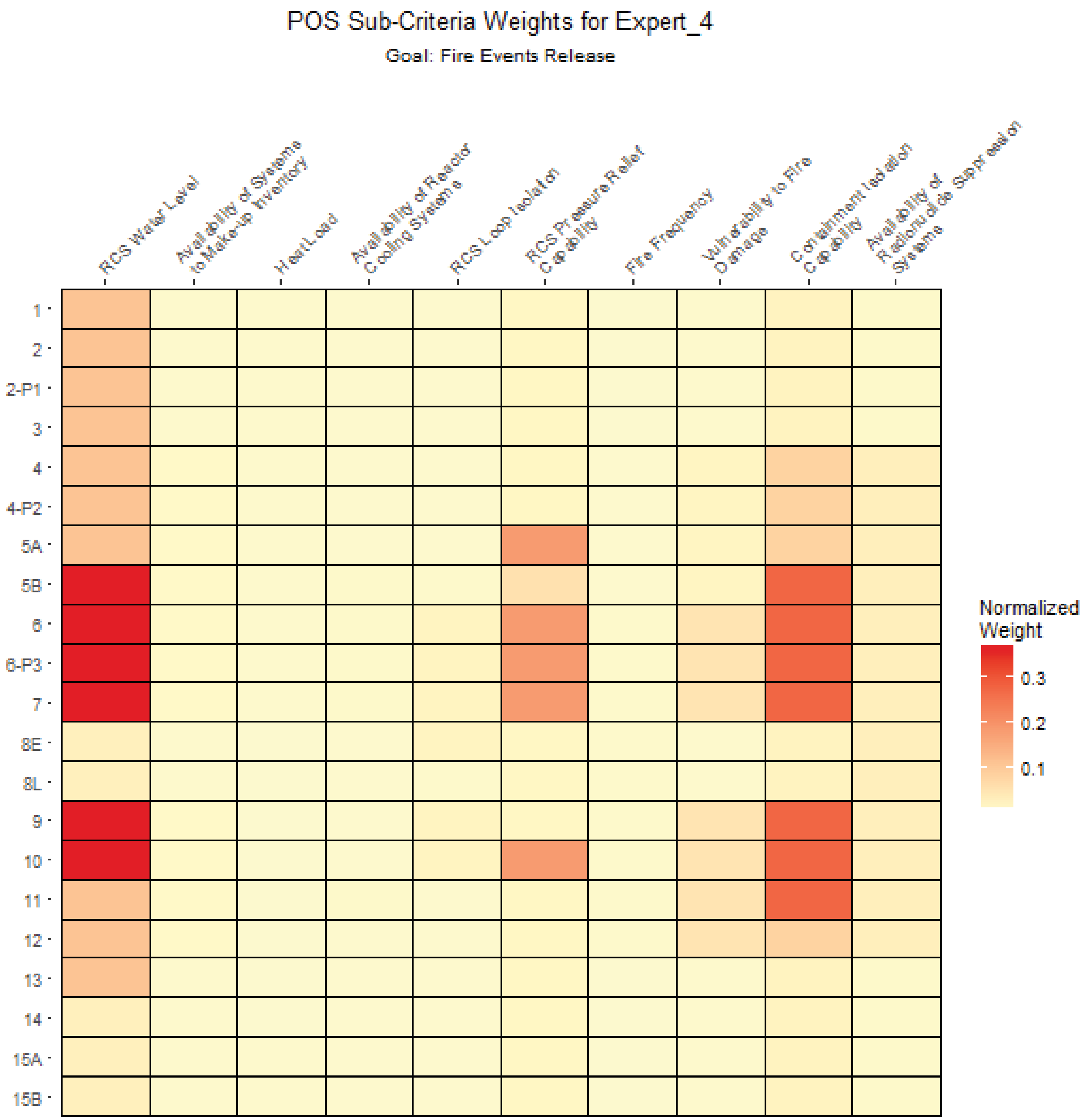
0.1



POS Sub-Criteria Weights for Expert_4
Goal: Fire Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
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2-P1
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4-P2
5A
5B
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6-P3
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8E
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15A
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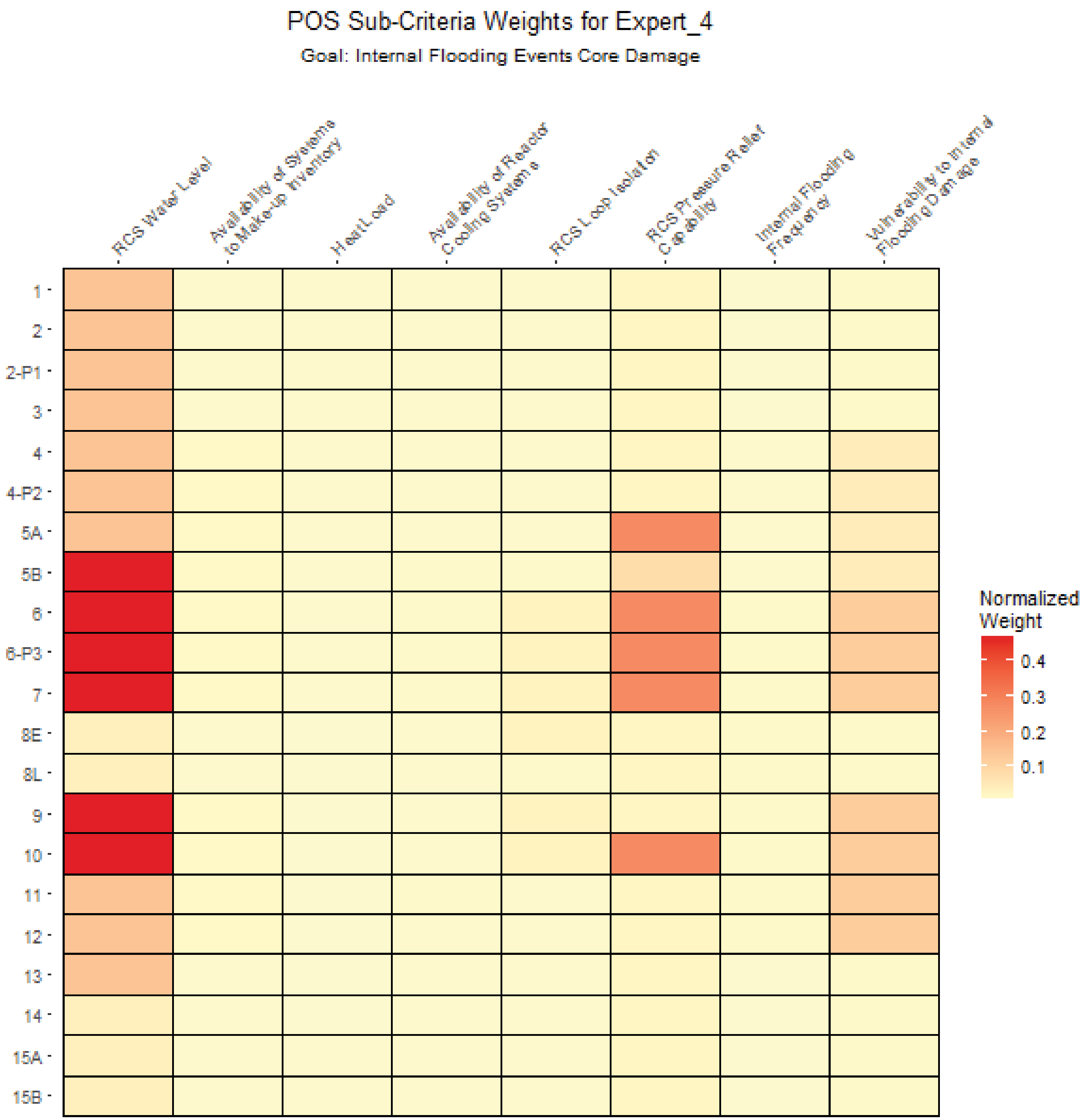
POS Sub-Criteria Weights for Expert_4
Goal: Internal Flooding Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
1
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2-P1
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4-P2
5A
5B
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6-P3
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8L
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15A
15B
Normalized Weight
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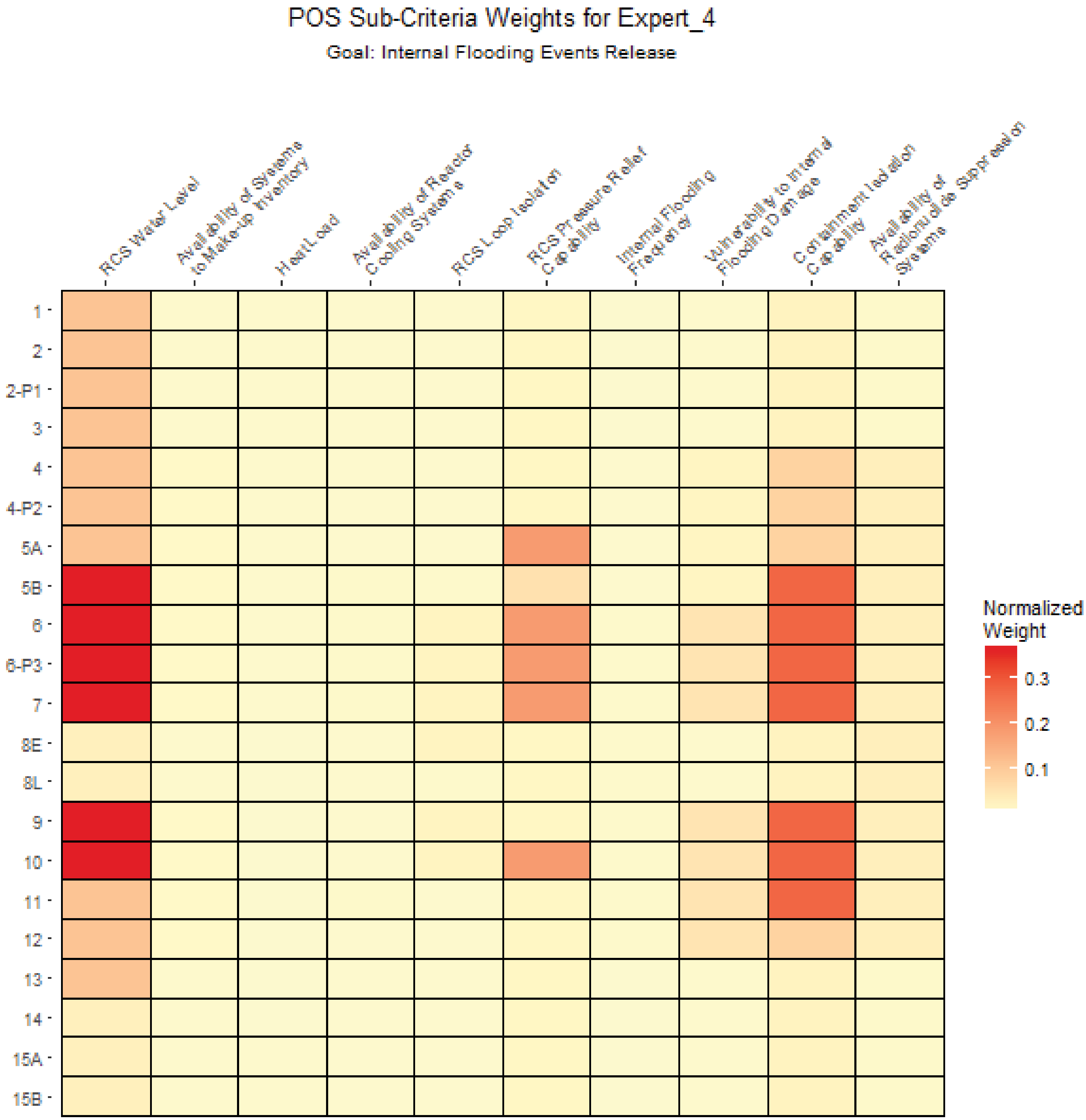
POS Sub-Criteria Weights for Expert_4
Goal: Internal Flooding Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
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2-P1
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4-P2
5A
5B
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6-P3
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8E
8L
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15A
15B
Normalized Weight
0.3
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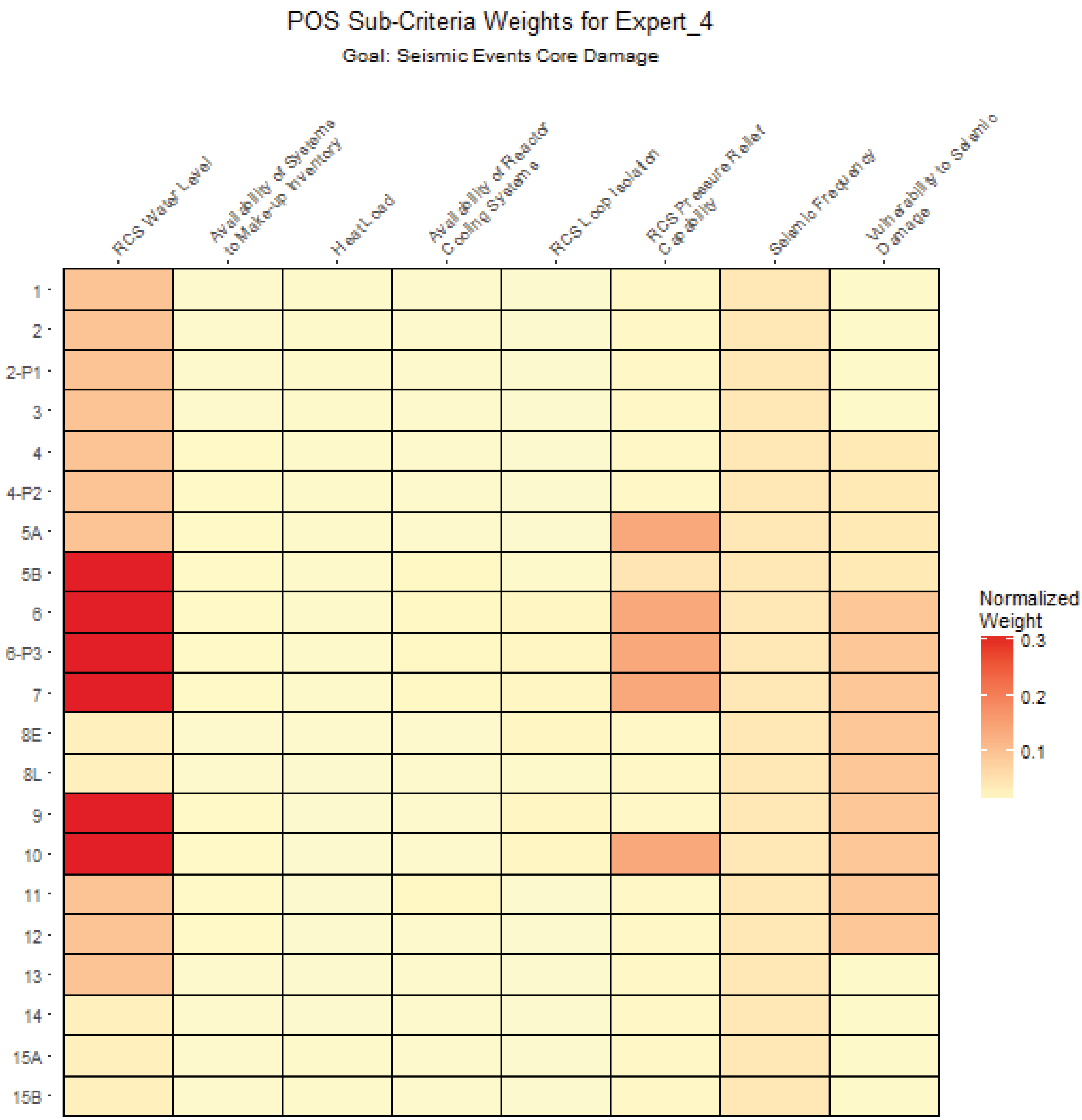
0.1


POS Sub-Criteria Weights for Expert_4
Goal: Seismic Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
1
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2-P1
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4-P2
5A
5B
6
6-P3
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8L
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15A
15B
Normalized Weight
0.3
0.2
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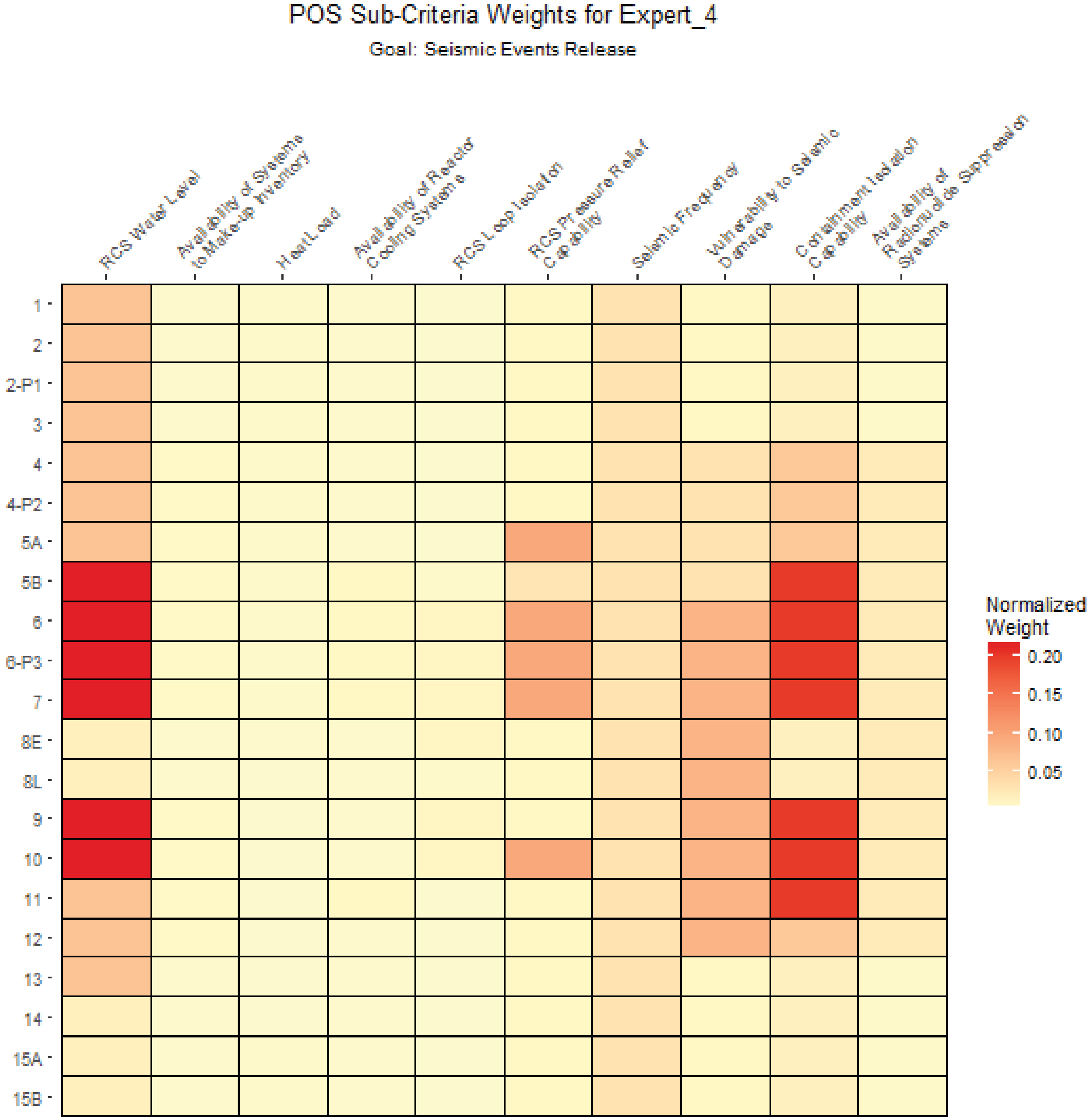
POS Sub-Criteria Weights for Expert_4
Goal: Seismic Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
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2-P1
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4
4-P2
5A
5B
6
6-P3
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8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.20
0.15
0.10
0.05

Expert 4

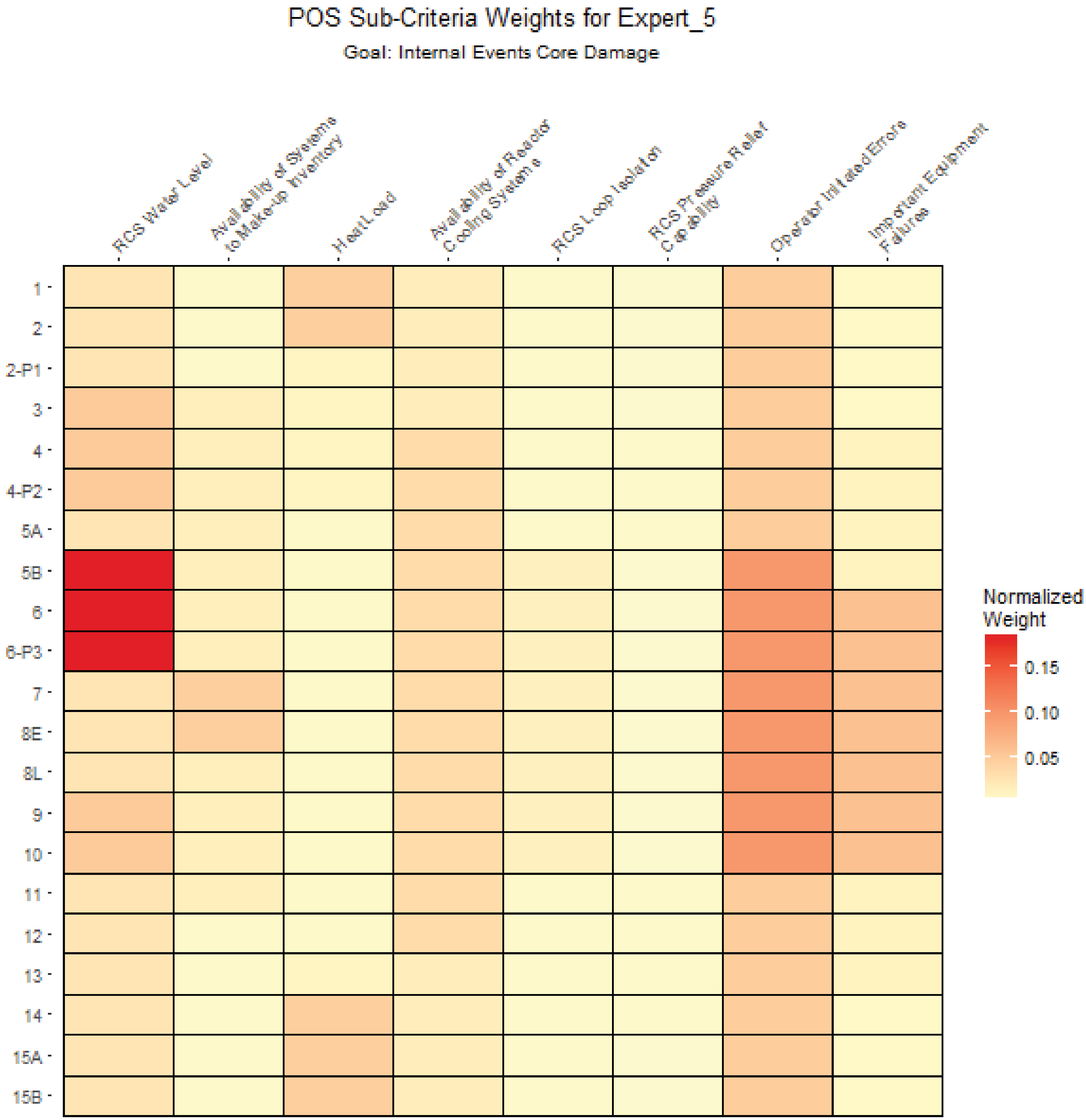
Top-Level Criteria Weights

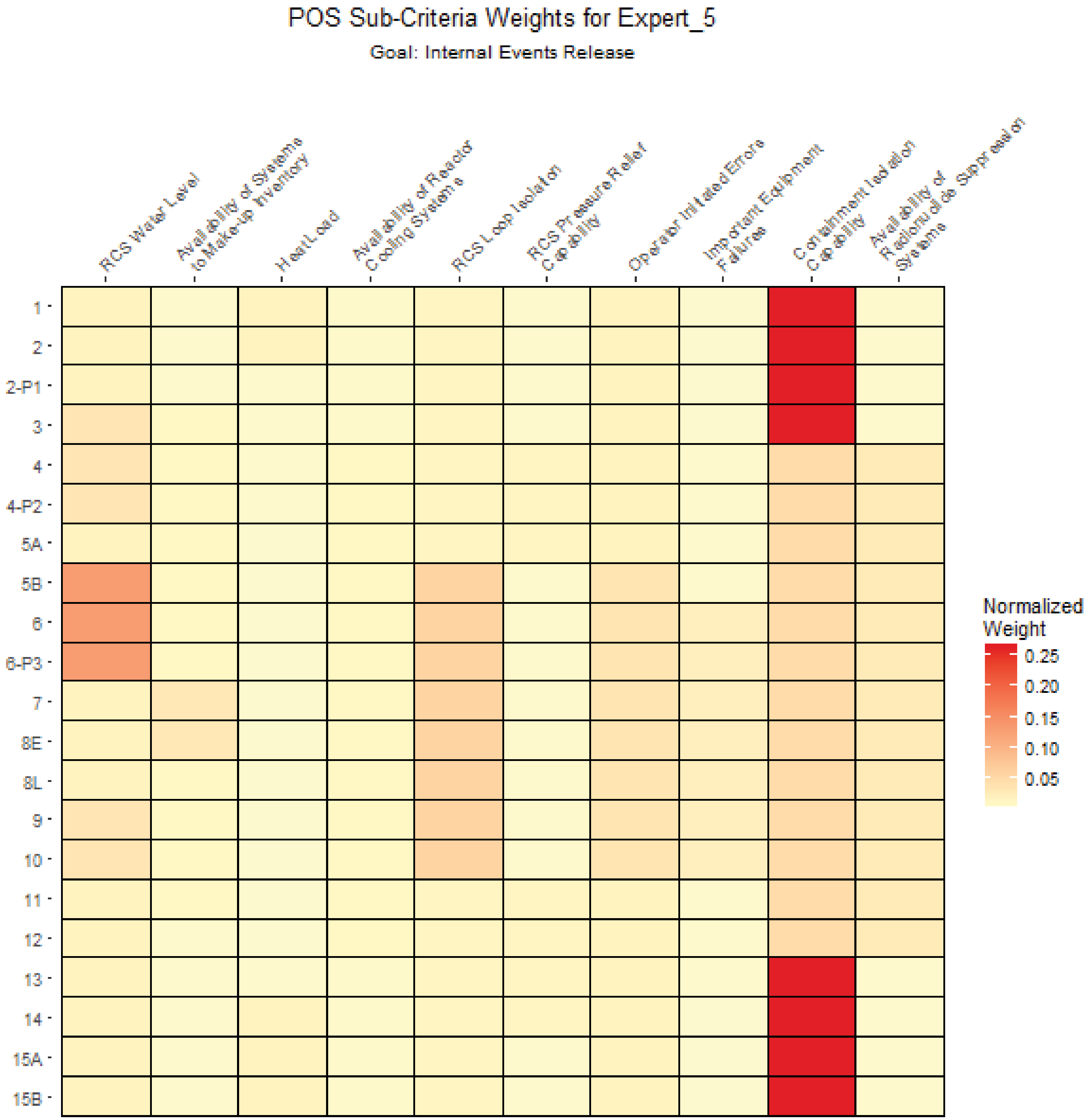
	Normalized Weights							
	Internal Events		Fire Events		Internal Flooding Events		Seismic Events	
Top-Level Criteria	Core Damage	Release	Core Damage	Release	Core Damage	Release	Core Damage	Release
RCS Inventory Control	0.52	0.44	0.52	0.40	0.52	0.40	0.35	0.24
Heat Removal	0.04	0.03	0.04	0.03	0.04	0.03	0.05	0.03
RCS Integrity	0.31	0.19	0.31	0.20	0.31	0.20	0.15	0.10
Hazard	0.13	0.05	0.13	0.06	0.13	0.06	0.45	0.41
Containment Performance		0.29		0.31		0.31		0.22
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Sub-Criteria Weights

Sub-Criteria	Normalized Weights							
	RCS Inventory Control	Heat Removal	RCS Integrity	Internal Events Hazard	Containment Performance	Fire Events Hazard	Internal Flooding Hazard	Seismic Events Hazard
RCS Water Level	0.90							
Availability of Systems to Make-up Inventory	0.10							
Heat Load		0.13						
Availability of Reactor Cooling Systems		0.88						
RCS Loop Isolation			0.10					
RCS Pressure Relief Capability			0.90					
Operator Initiated Errors				0.90				
Important Equipment Failures				0.10				
Containment Isolation Capability					0.90			
Availability of Radionuclide Suppression Systems					0.10			
Fire Frequency						0.10		
Vulnerability to Fire Damage						0.90		
Internal Flooding Frequency							0.10	
Vulnerability to Internal Flooding Damage							0.90	
Seismic Frequency								0.80
Vulnerability to Seismic Damage								0.20
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

K.6 Expert 5 Results





POS Sub-Criteria Weights for Expert_5
Goal: Internal Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Operator Initiated Errors
Important Equipment Failures
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
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2-P1
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4
4-P2
5A
5B
6
6-P3
7
8E
8L
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10
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12
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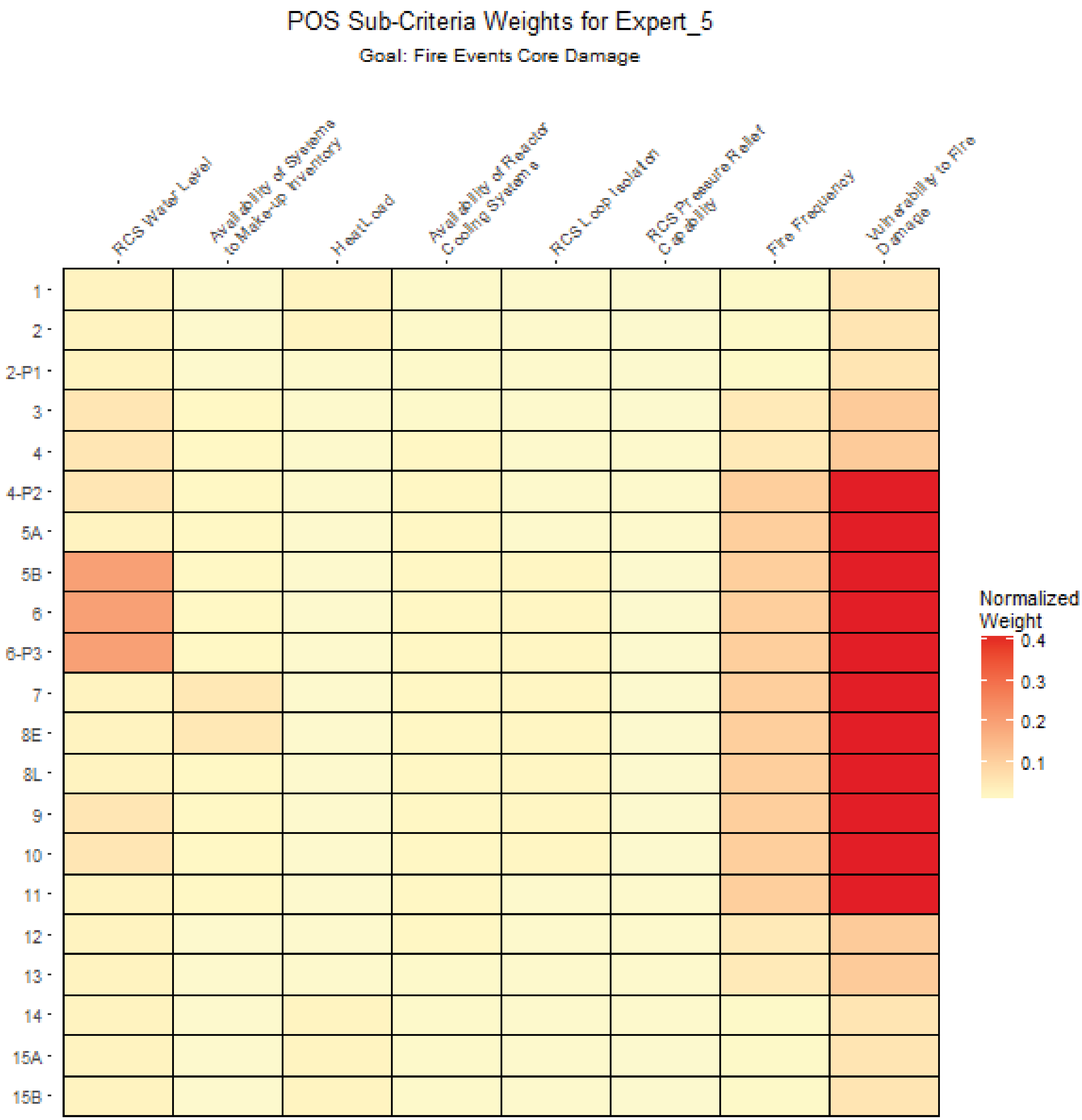
15A
15B
Normalized Weight
0.25
0.20
0.15
0.10
0.05


POS Sub-Criteria Weights for Expert_5
Goal: Fire Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
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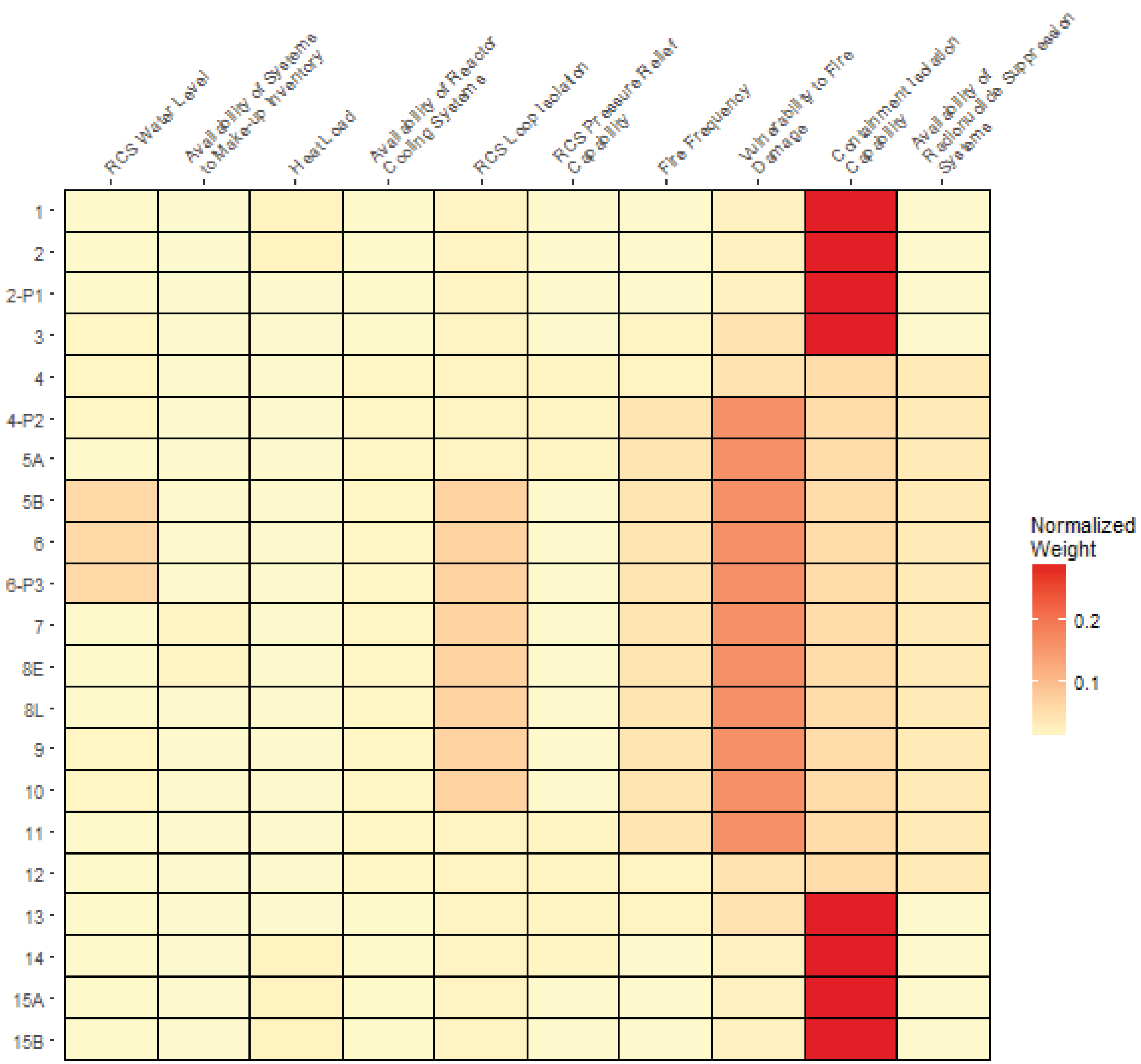
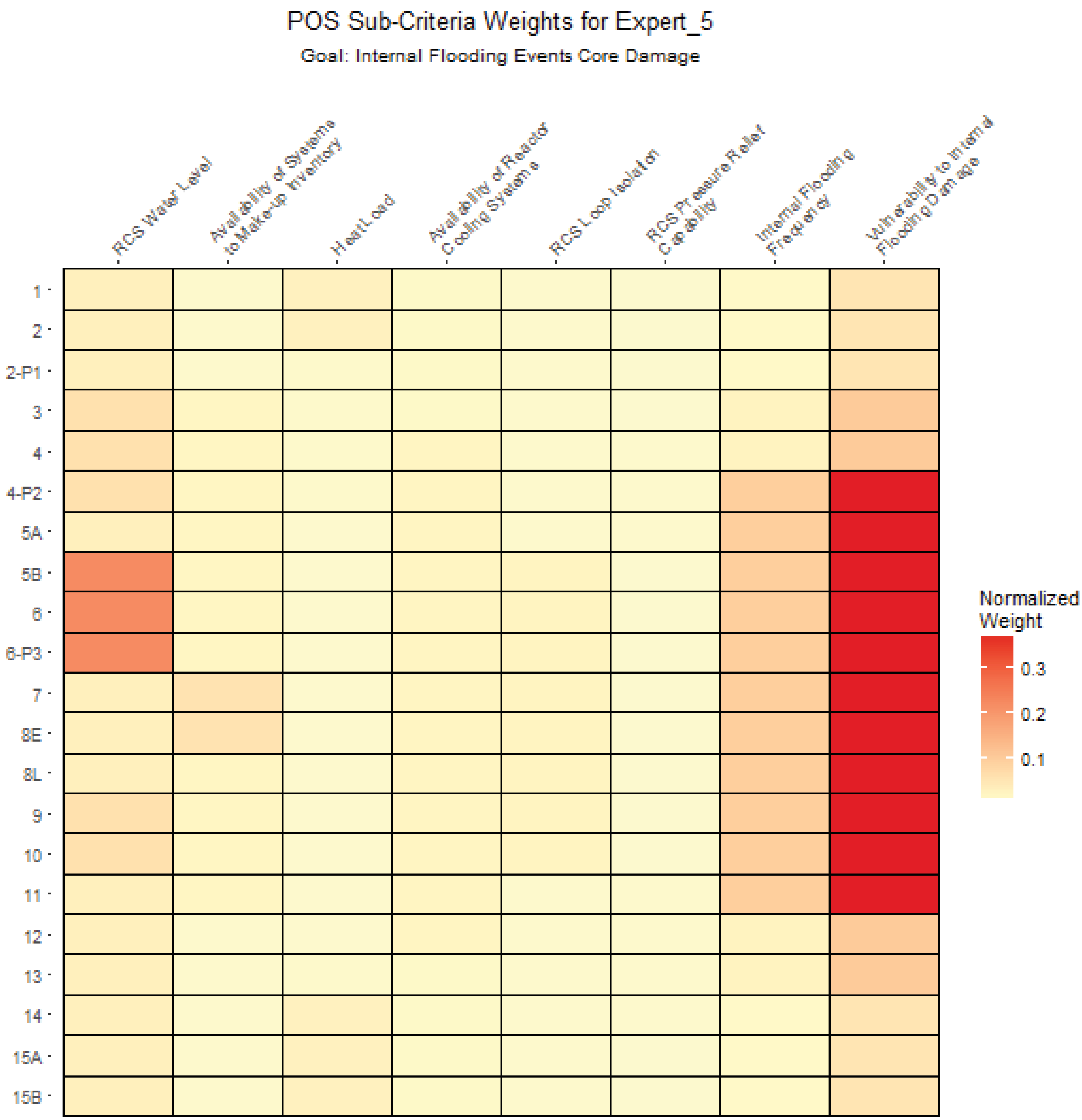
15A
15B
Normalized Weight
0.4
0.3
0.2
0.1

POS Sub-Criteria Weights for Expert_5

Goal: Fire Events Release





POS Sub-Criteria Weights for Expert_5
Goal: Internal Flooding Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
1
2
2-P1
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4-P2
5A
5B
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6-P3
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8E
8L
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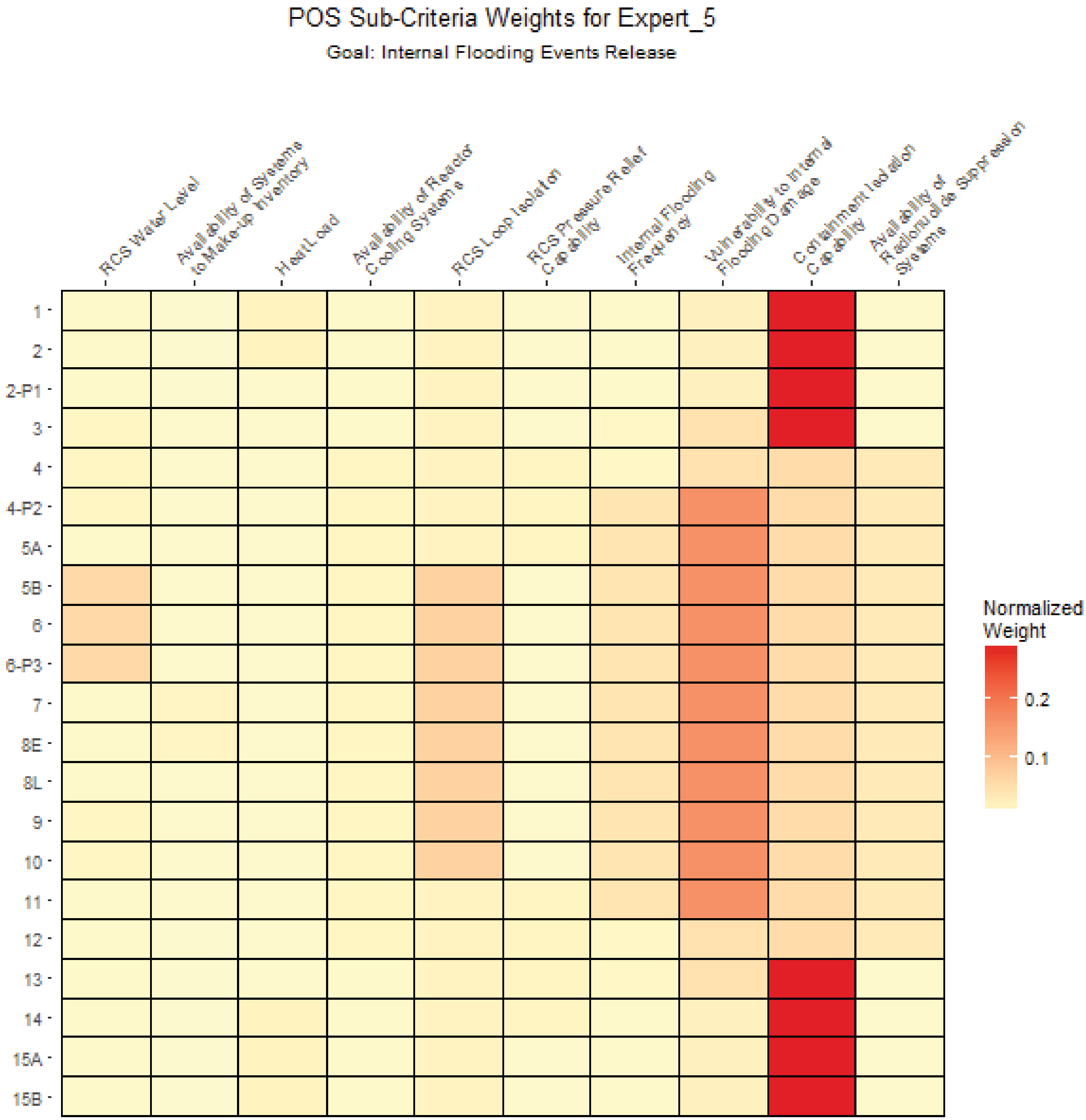
14
15A
15B
Normalized Weight
0.3
0.2
0.1



POS Sub-Criteria Weights for Expert_5
Goal: Internal Flooding Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
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2-P1
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4-P2
5A
5B
6
6-P3
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8E
8L
9
10
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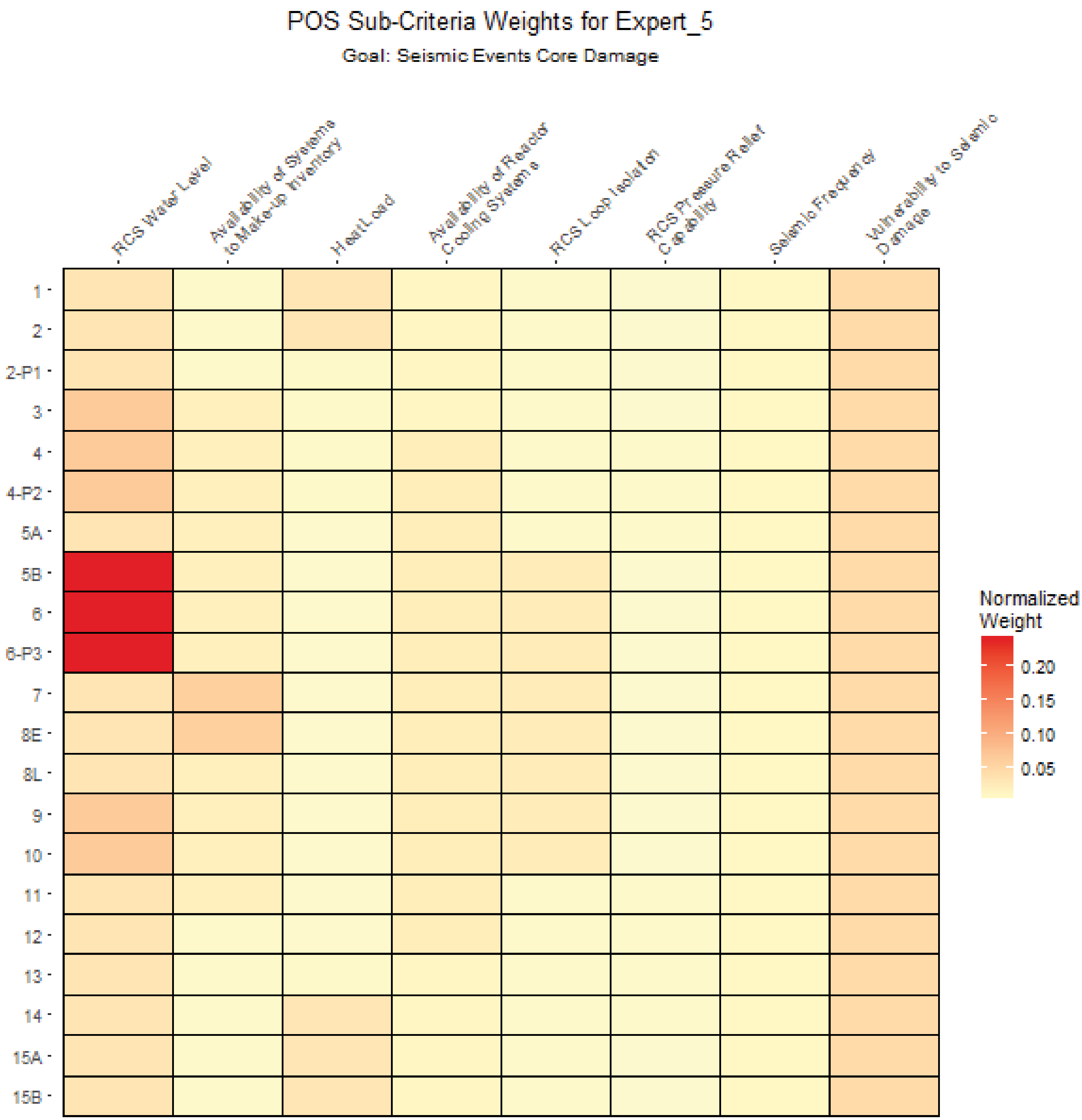
12
13
14
15A
15B
Normalized Weight
0.2
0.1



POS Sub-Criteria Weights for Expert_5
Goal: Seismic Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
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8E
8L
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10
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12
13
14
15A
15B
Normalized Weight
0.20
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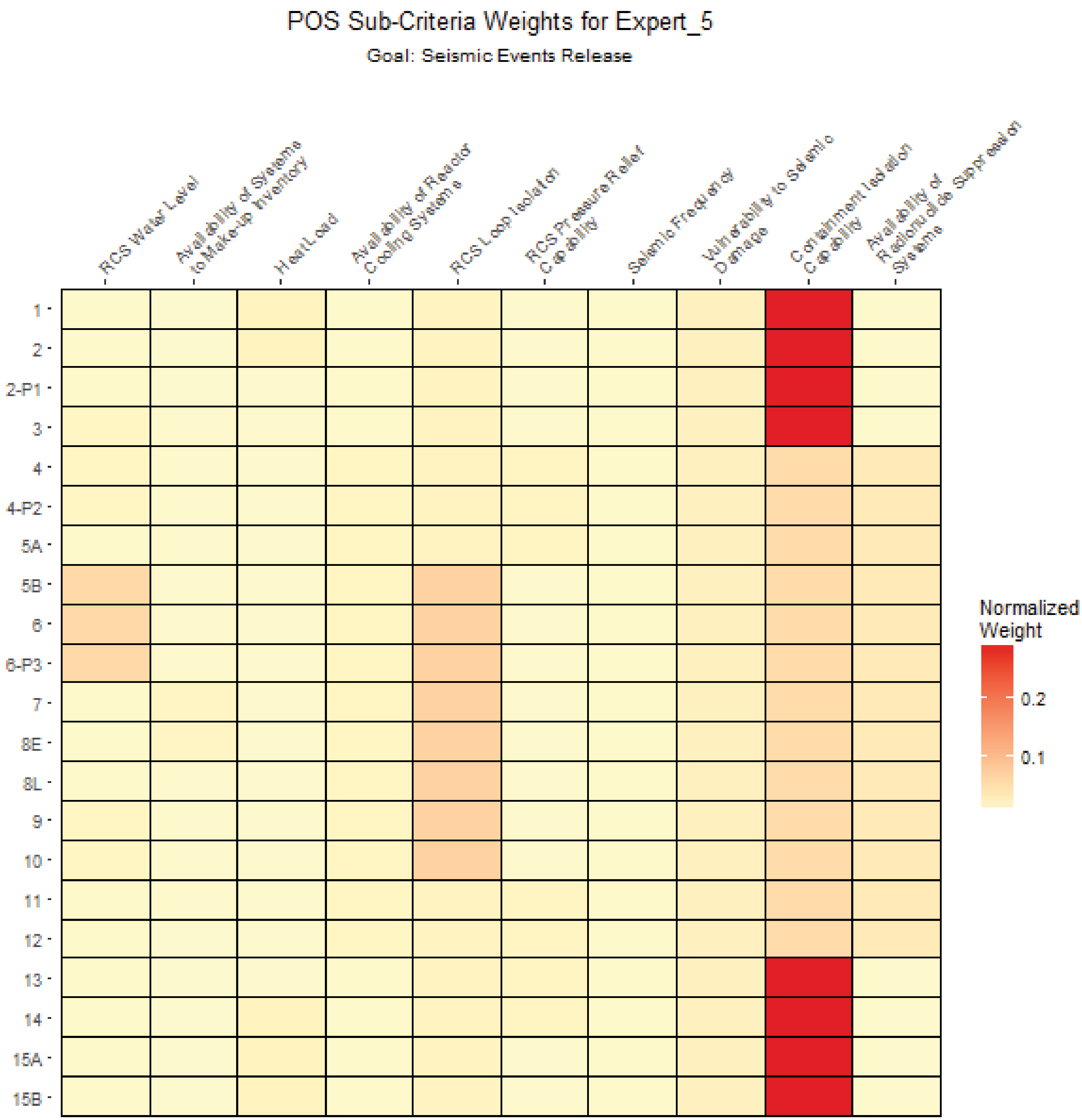
0.10
0.05


POS Sub-Criteria Weights for Expert_5
Goal: Seismic Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
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2-P1
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4-P2
5A
5B
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6-P3
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8E
8L
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13
14
15A
15B
Normalized Weight
0.2
0.1

Expert 5

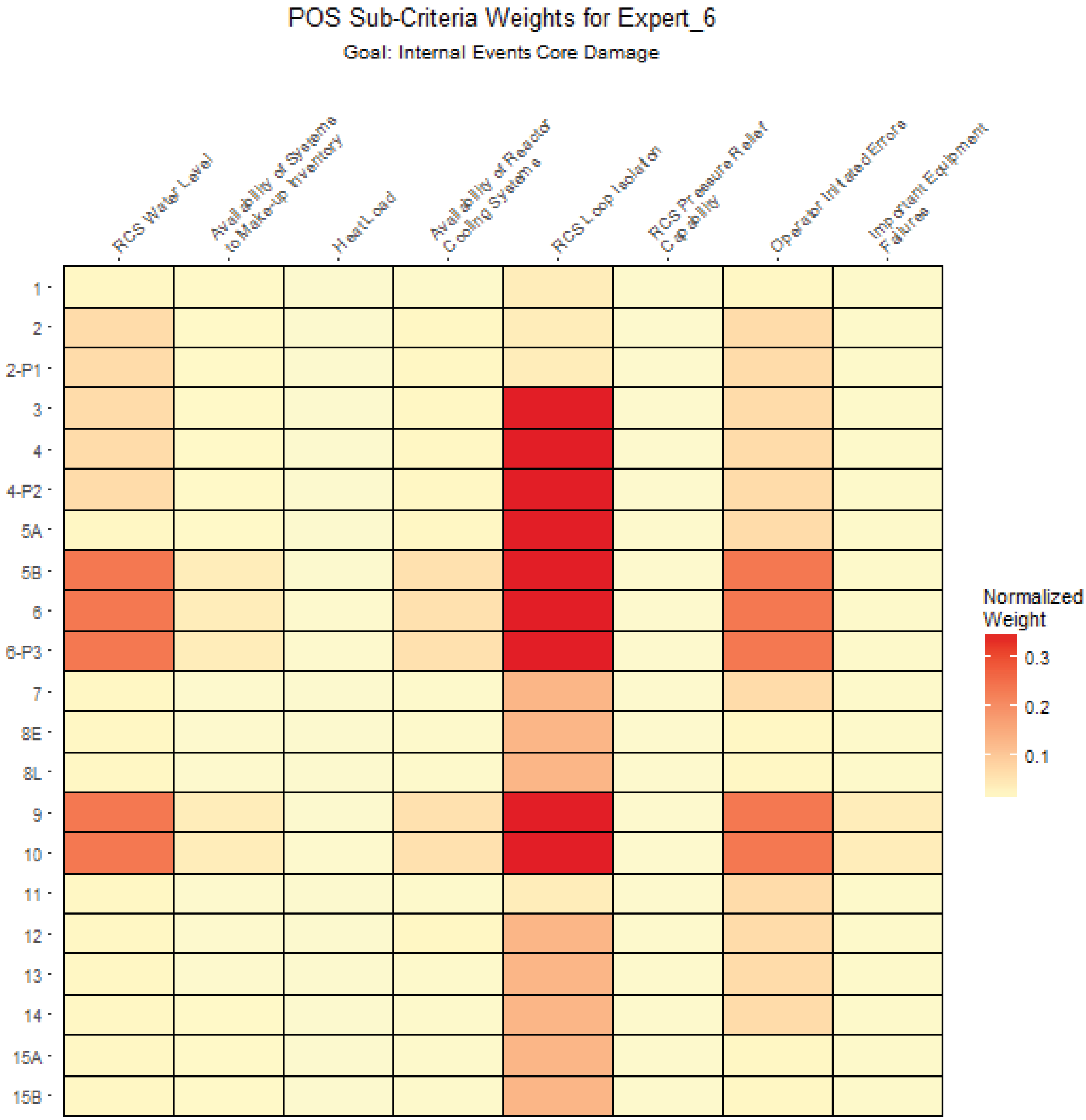
Top-Level Criteria Weights

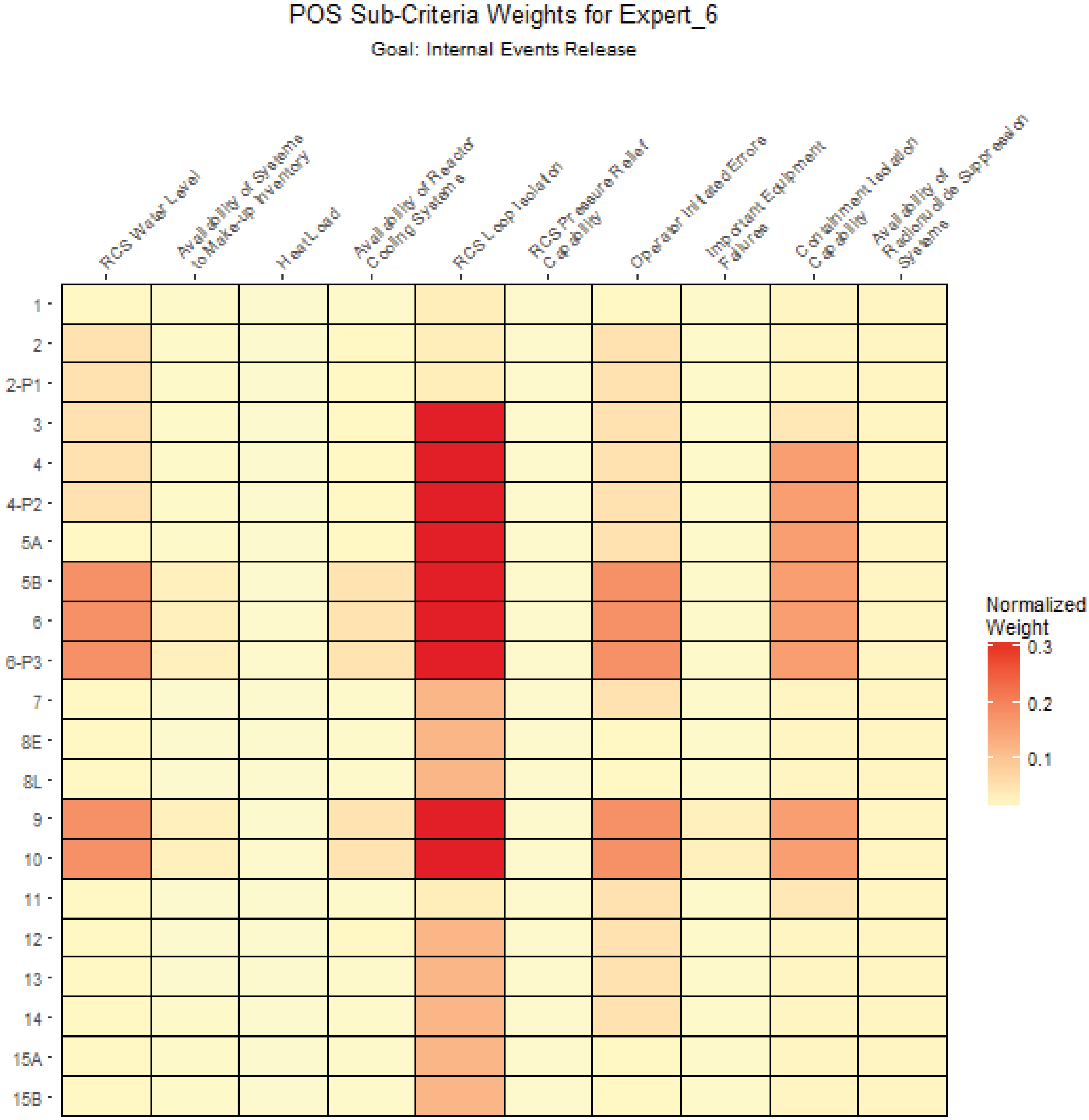
	Normalized Weights							
	Internal Events		Fire Events		Internal Flooding Events		Seismic Events	
Top-Level Criteria	Core Damage	Release	Core Damage	Release	Core Damage	Release	Core Damage	Release
RCS Inventory Control	0.23	0.16	0.25	0.07	0.28	0.07	0.30	0.07
Heat Removal	0.23	0.09	0.13	0.10	0.14	0.10	0.16	0.10
RCS Integrity	0.06	0.28	0.10	0.31	0.11	0.31	0.12	0.31
Hazard	0.48	0.18	0.52	0.20	0.47	0.20	0.42	0.20
Containment Performance		0.29		0.32		0.32		0.32
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Sub-Criteria Weights

Sub-Criteria	Normalized Weights							
	RCS Inventory Control	Heat Removal	RCS Integrity	Internal Events Hazard	Containment Performance	Fire Events Hazard	Internal Flooding Hazard	Seismic Events Hazard
RCS Water Level	0.80							
Availability of Systems to Make-up Inventory	0.20							
Heat Load		0.20						
Availability of Reactor Cooling Systems		0.80						
RCS Loop Isolation			0.80					
RCS Pressure Relief Capability			0.20					
Operator Initiated Errors				0.88				
Important Equipment Failures				0.13				
Containment Isolation Capability					0.90			
Availability of Radionuclide Suppression Systems					0.10			
Fire Frequency						0.20		
Vulnerability to Fire Damage						0.80		
Internal Flooding Frequency							0.20	
Vulnerability to Internal Flooding Damage							0.80	
Seismic Frequency								0.20
Vulnerability to Seismic Damage								0.80
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

K.7 Expert 6 Results




POS Sub-Criteria Weights for Expert_6
Goal: Internal Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Operator Initiated Errors
Important Equipment Failures
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
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2-P1
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4-P2
5A
5B
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6-P3
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8E
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15A
15B
Normalized Weight
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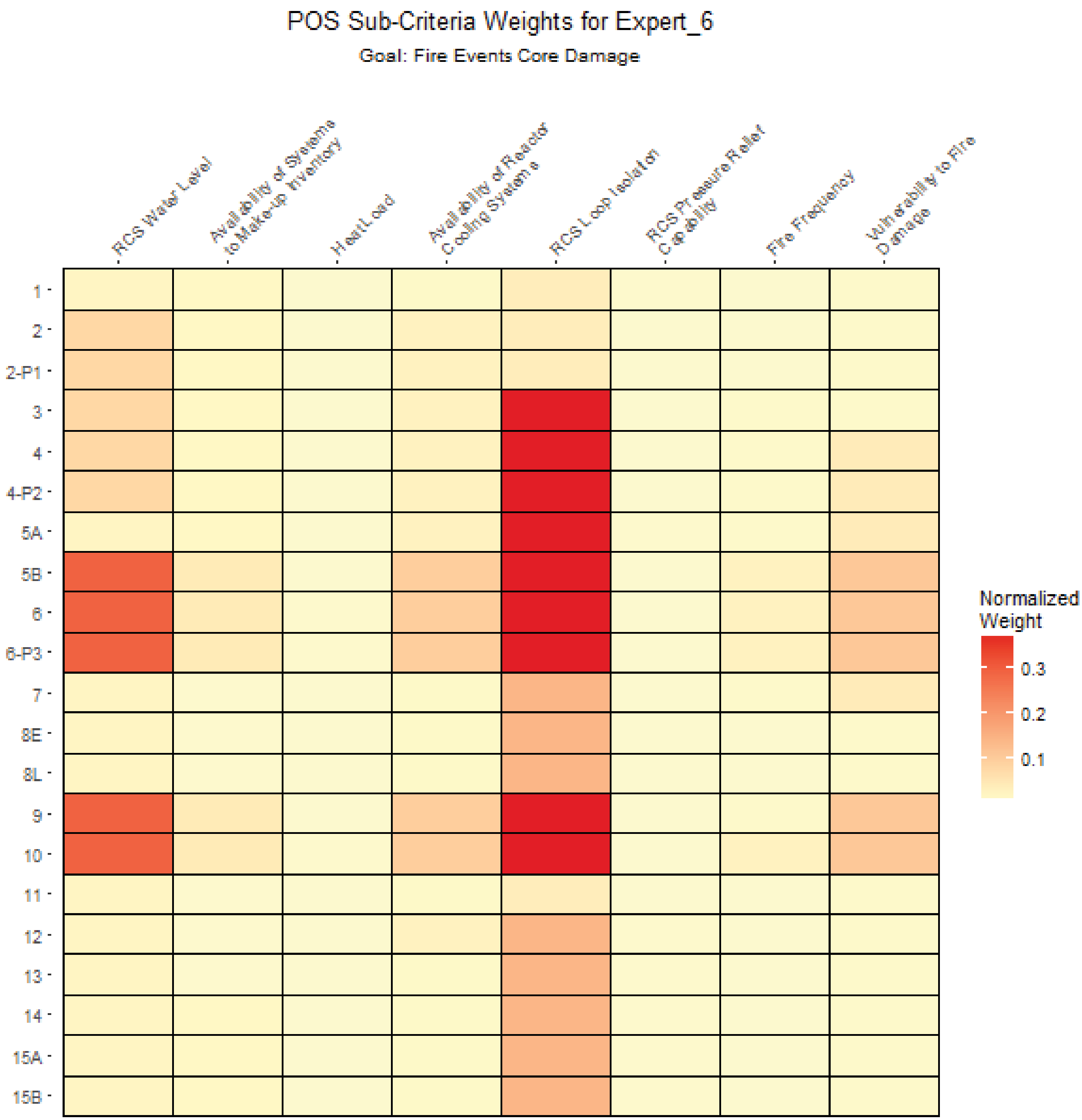
0.1



POS Sub-Criteria Weights for Expert_6
Goal: Fire Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
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8E
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15A
15B
Normalized Weight
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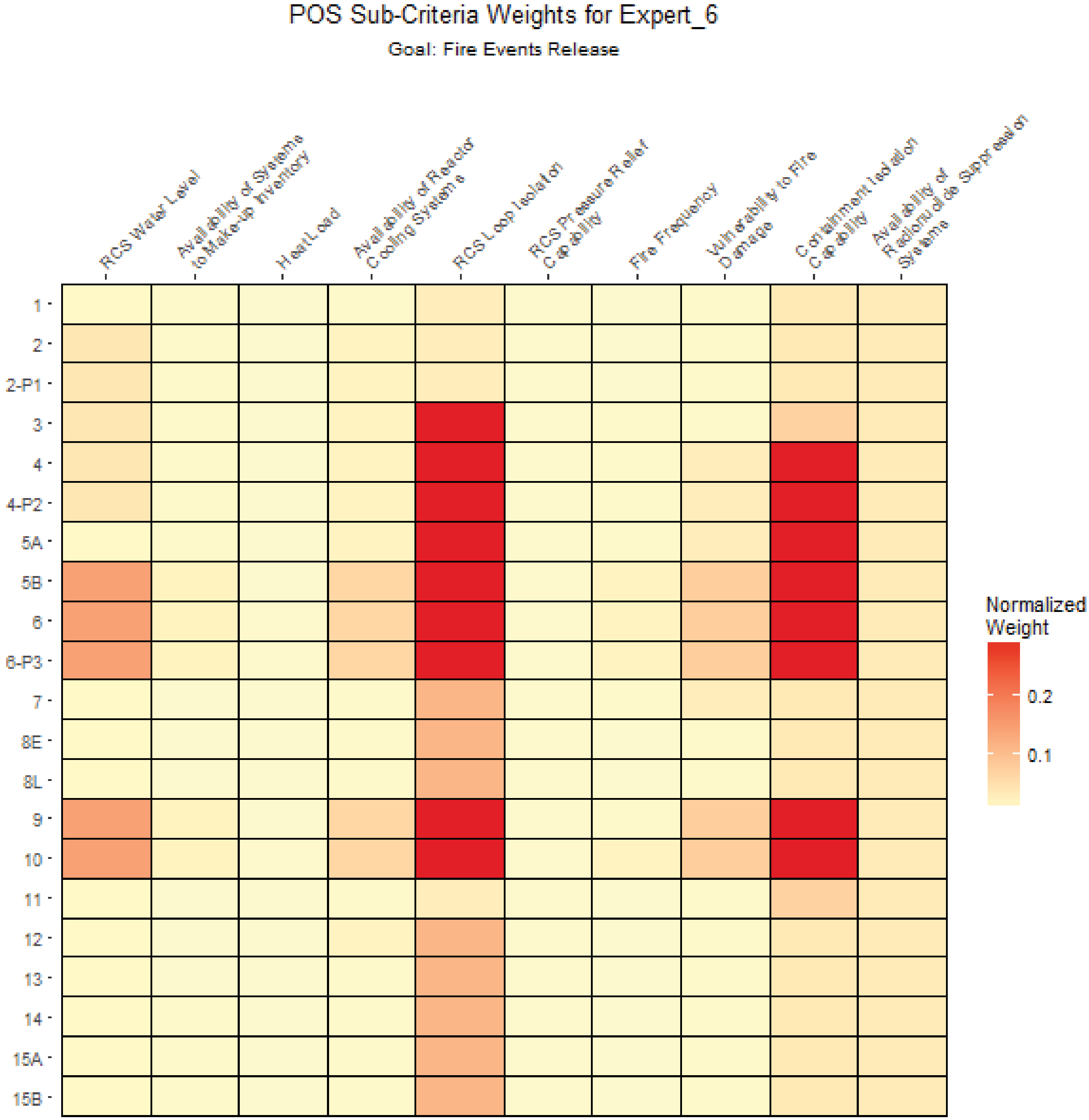

POS Sub-Criteria Weights for Expert_6
Goal: Fire Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
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4-P2
5A
5B
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6-P3
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8E
8L
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15A
15B
Normalized Weight
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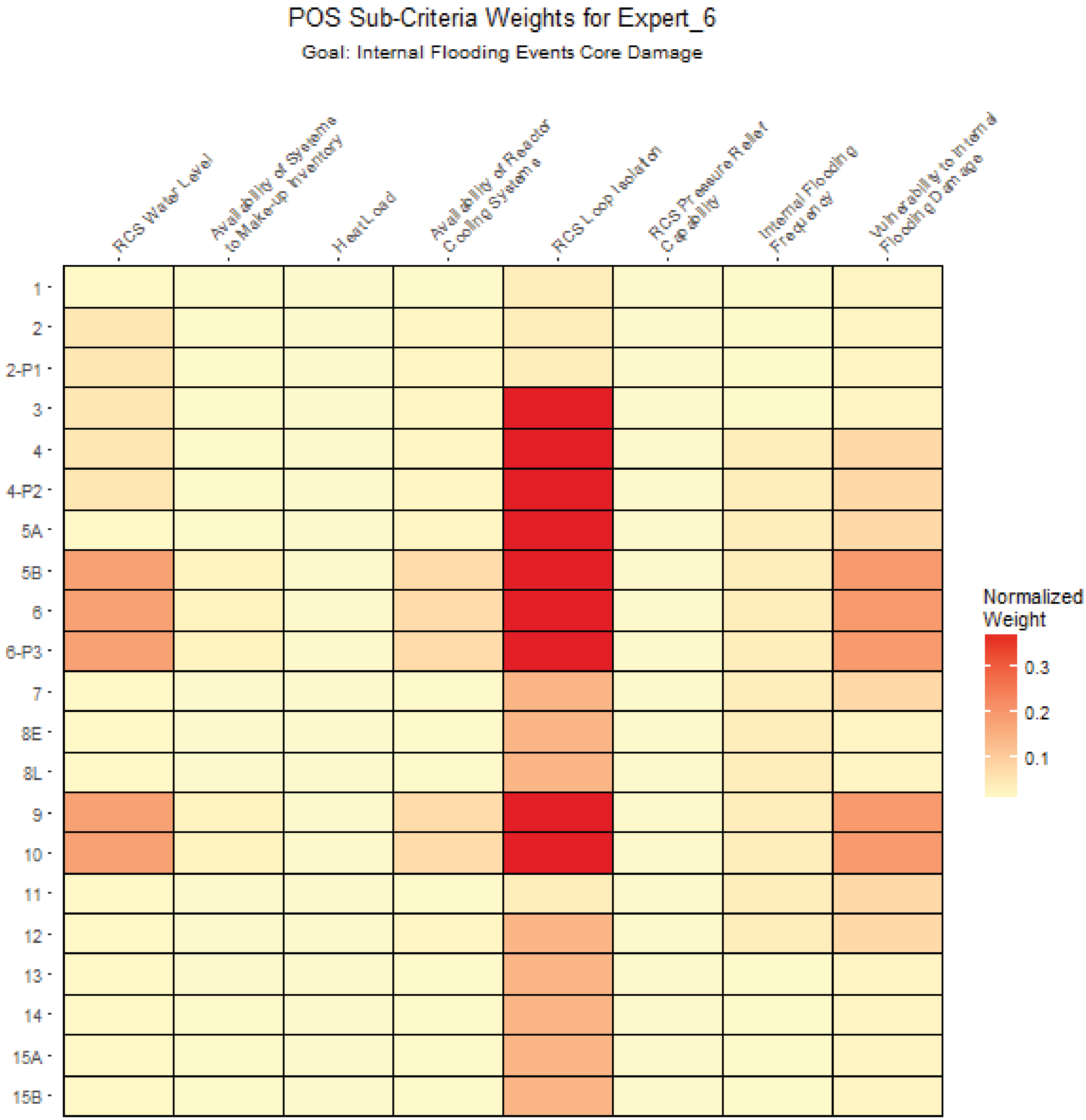
0.1


POS Sub-Criteria Weights for Expert_6
Goal: Internal Flooding Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
1
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2-P1
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4-P2
5A
5B
6
6-P3
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8E
8L
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15A
15B
Normalized Weight
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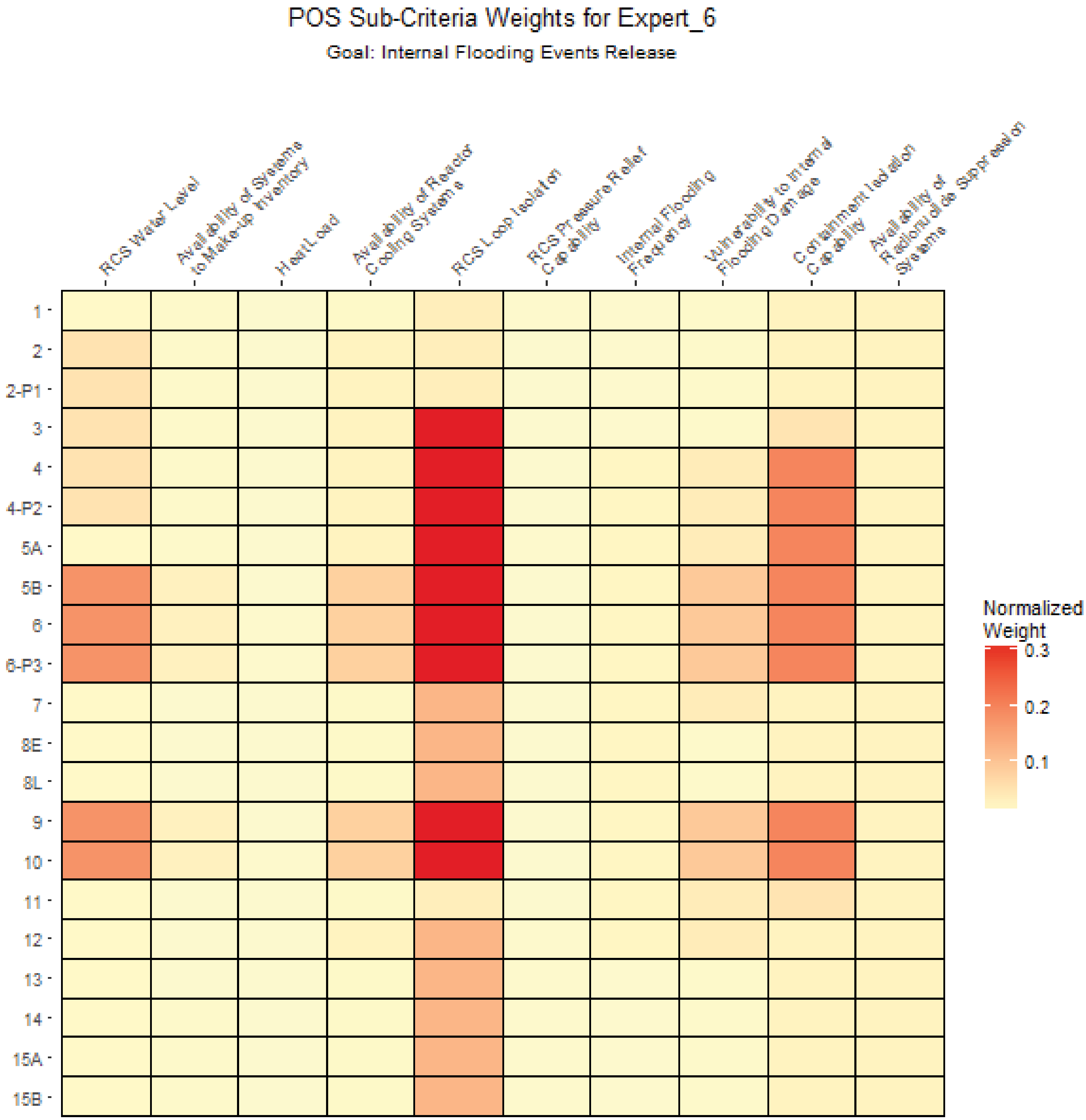
0.2
0.1


POS Sub-Criteria Weights for Expert_6
Goal: Internal Flooding Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
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2-P1
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4-P2
5A
5B
6
6-P3
7
8E
8L
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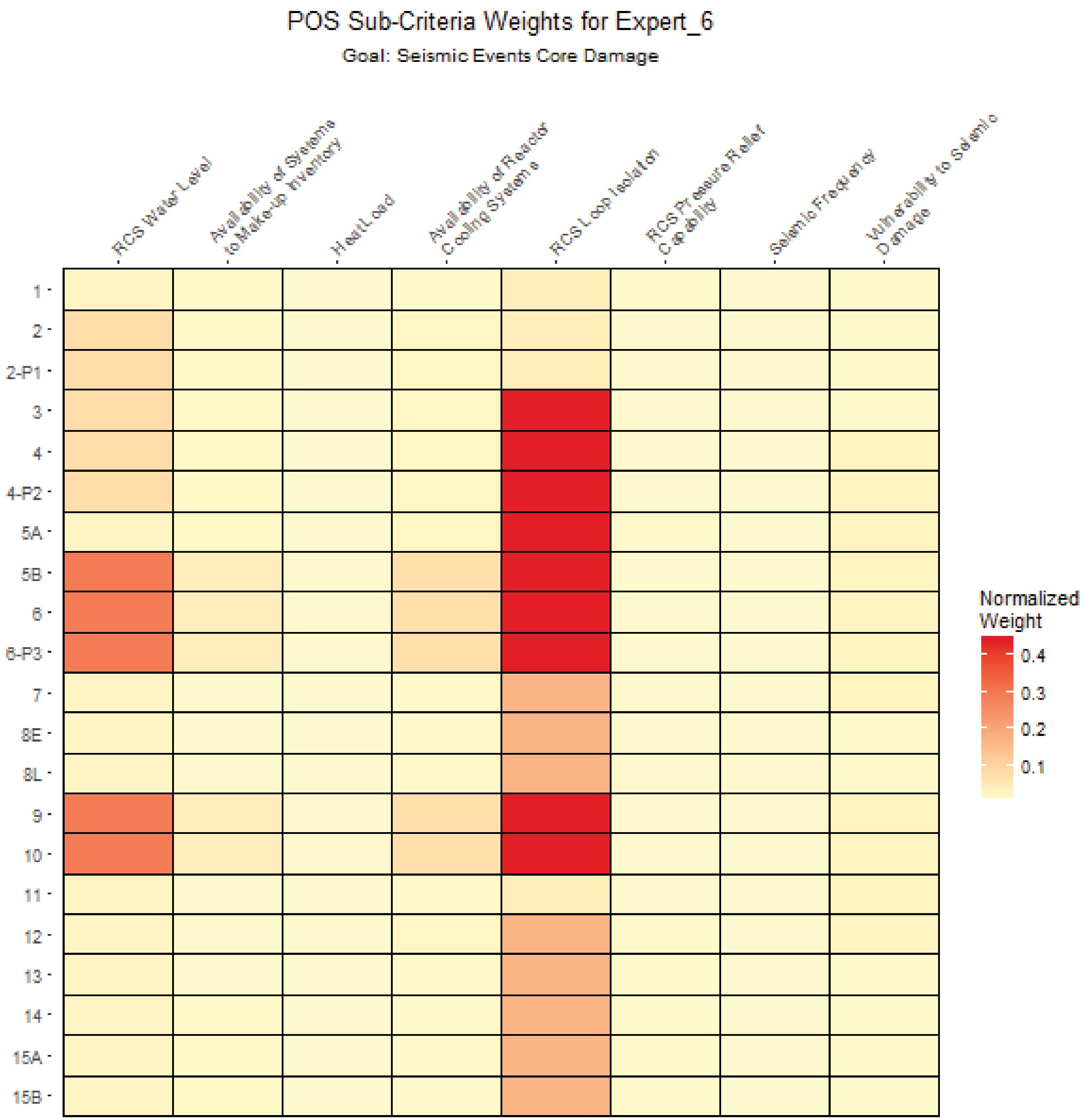
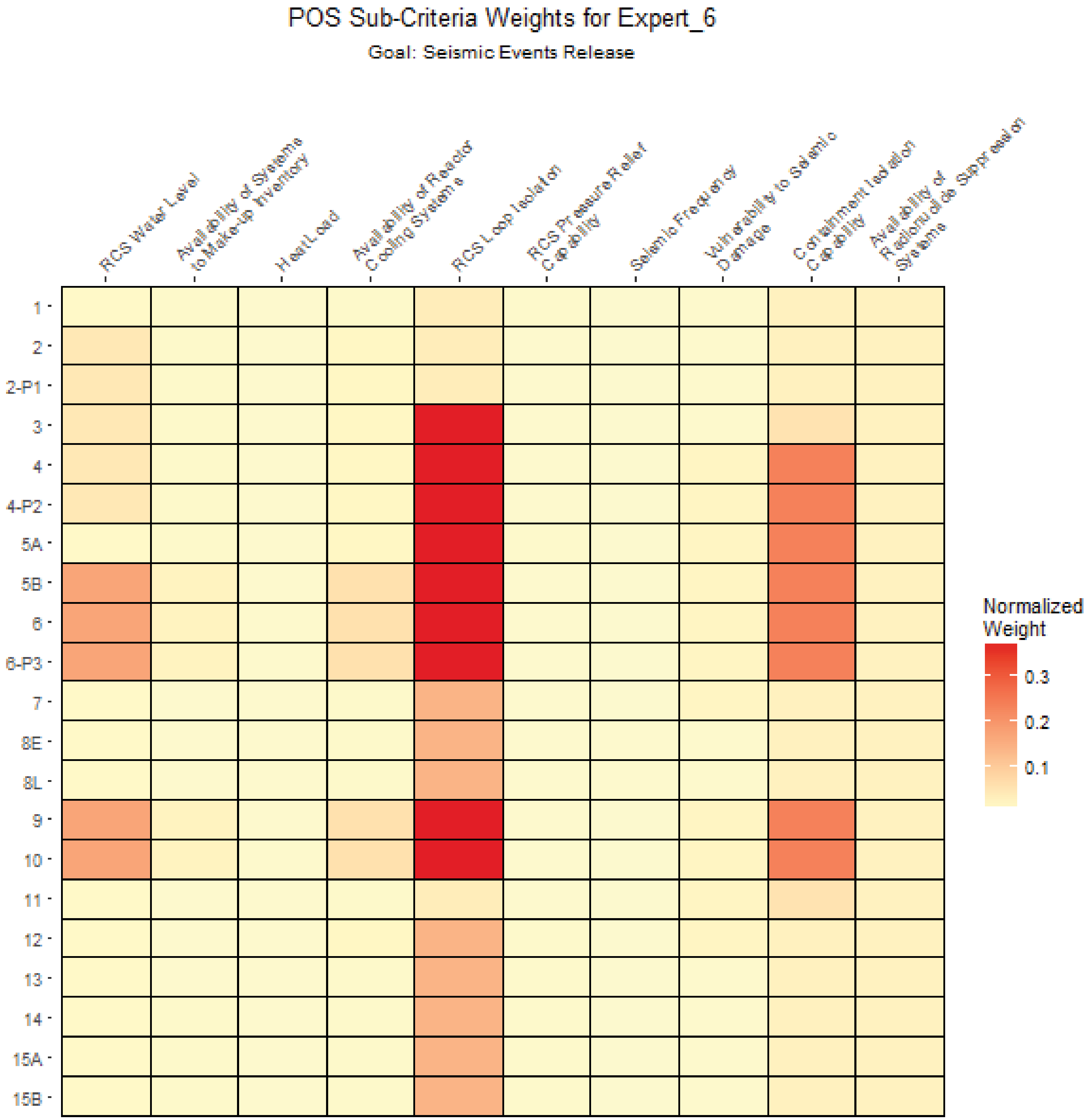
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1



POS Sub-Criteria Weights for Expert_6
Goal: Seismic Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.4
0.3
0.2
0.1


POS Sub-Criteria Weights for Expert_6
Goal: Seismic Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1

Expert 6

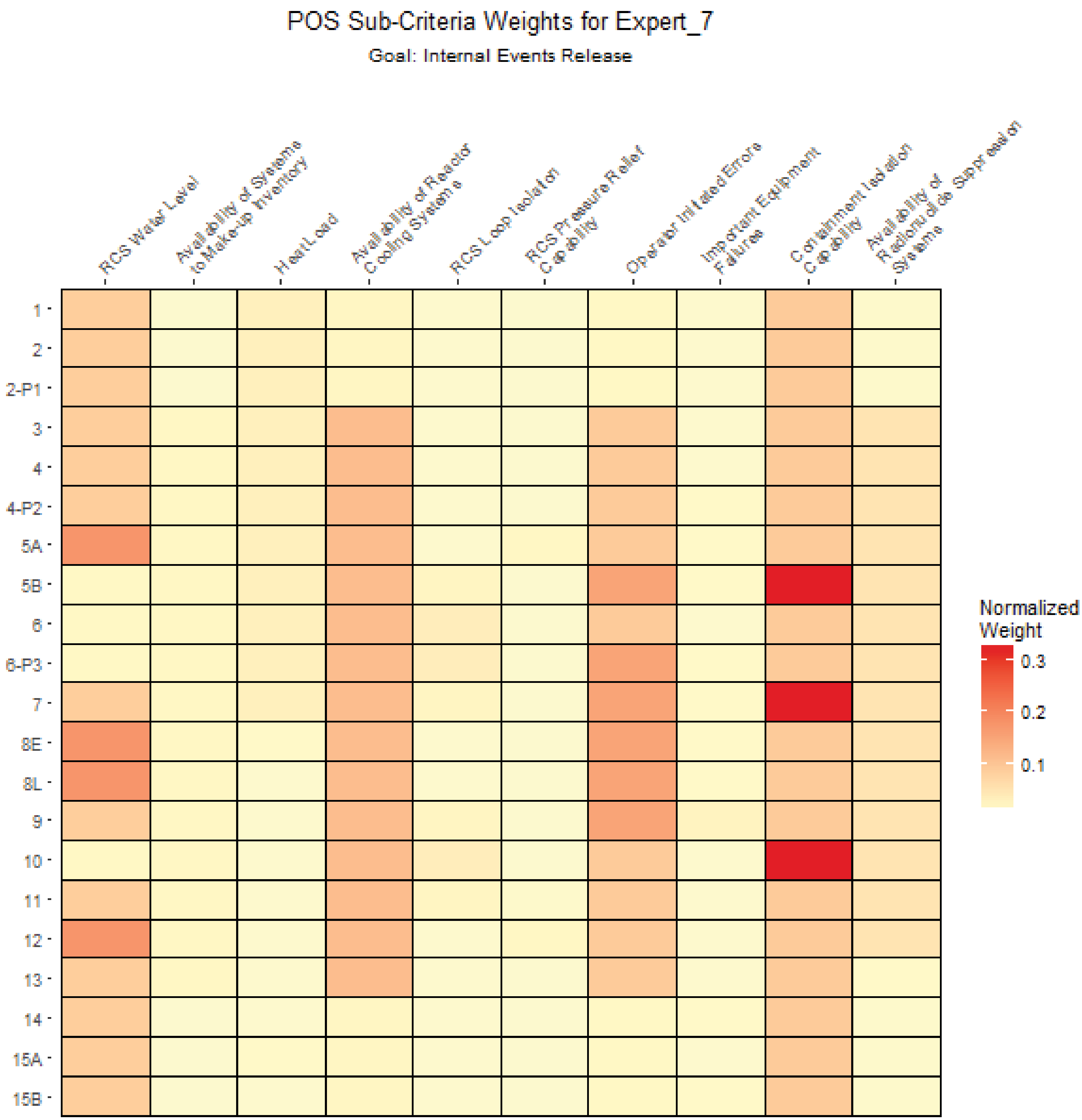
Top-Level Criteria Weights

	Normalized Weights							
	Internal Events		Fire Events		Internal Flooding Events		Seismic Events	
Top-Level Criteria	Core Damage	Release	Core Damage	Release	Core Damage	Release	Core Damage	Release
RCS Inventory Control	0.27	0.21	0.33	0.17	0.21	0.20	0.33	0.19
Heat Removal	0.06	0.05	0.11	0.07	0.08	0.09	0.08	0.07
RCS Integrity	0.39	0.35	0.42	0.33	0.42	0.35	0.49	0.41
Hazard	0.27	0.21	0.14	0.10	0.29	0.13	0.10	0.06
Containment Performance		0.18		0.33		0.22		0.26
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Sub-Criteria Weights

Sub-Criteria	Normalized Weights							
	RCS Inventory Control	Heat Removal	RCS Integrity	Internal Events Hazard	Containment Performance	Fire Events Hazard	Internal Flooding Hazard	Seismic Events Hazard
RCS Water Level	0.88							
Availability of Systems to Make-up Inventory	0.13							
Heat Load		0.13						
Availability of Reactor Cooling Systems		0.88						
RCS Loop Isolation			0.90					
RCS Pressure Relief Capability			0.10					
Operator Initiated Errors				0.88				
Important Equipment Failures				0.13				
Containment Isolation Capability					0.90			
Availability of Radionuclide Suppression Systems					0.10			
Fire Frequency						0.20		
Vulnerability to Fire Damage						0.80		
Internal Flooding Frequency							0.33	
Vulnerability to Internal Flooding Damage							0.67	
Seismic Frequency								0.20
Vulnerability to Seismic Damage								0.80
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

K.8 Expert 7 Results



POS Sub-Criteria Weights for Expert_7
Goal: Internal Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Operator Initiated Errors
Important Equipment Failures
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
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4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
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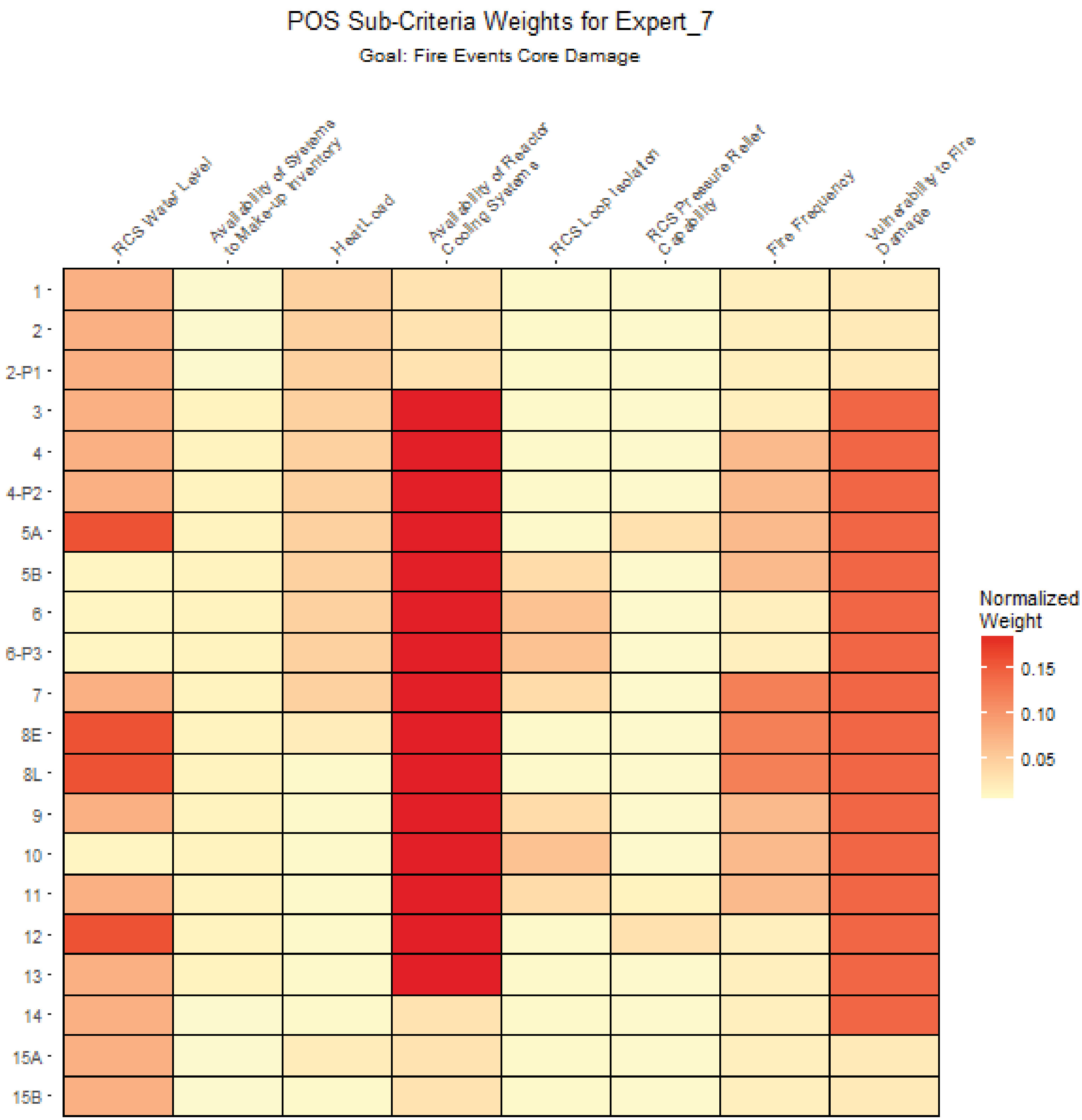
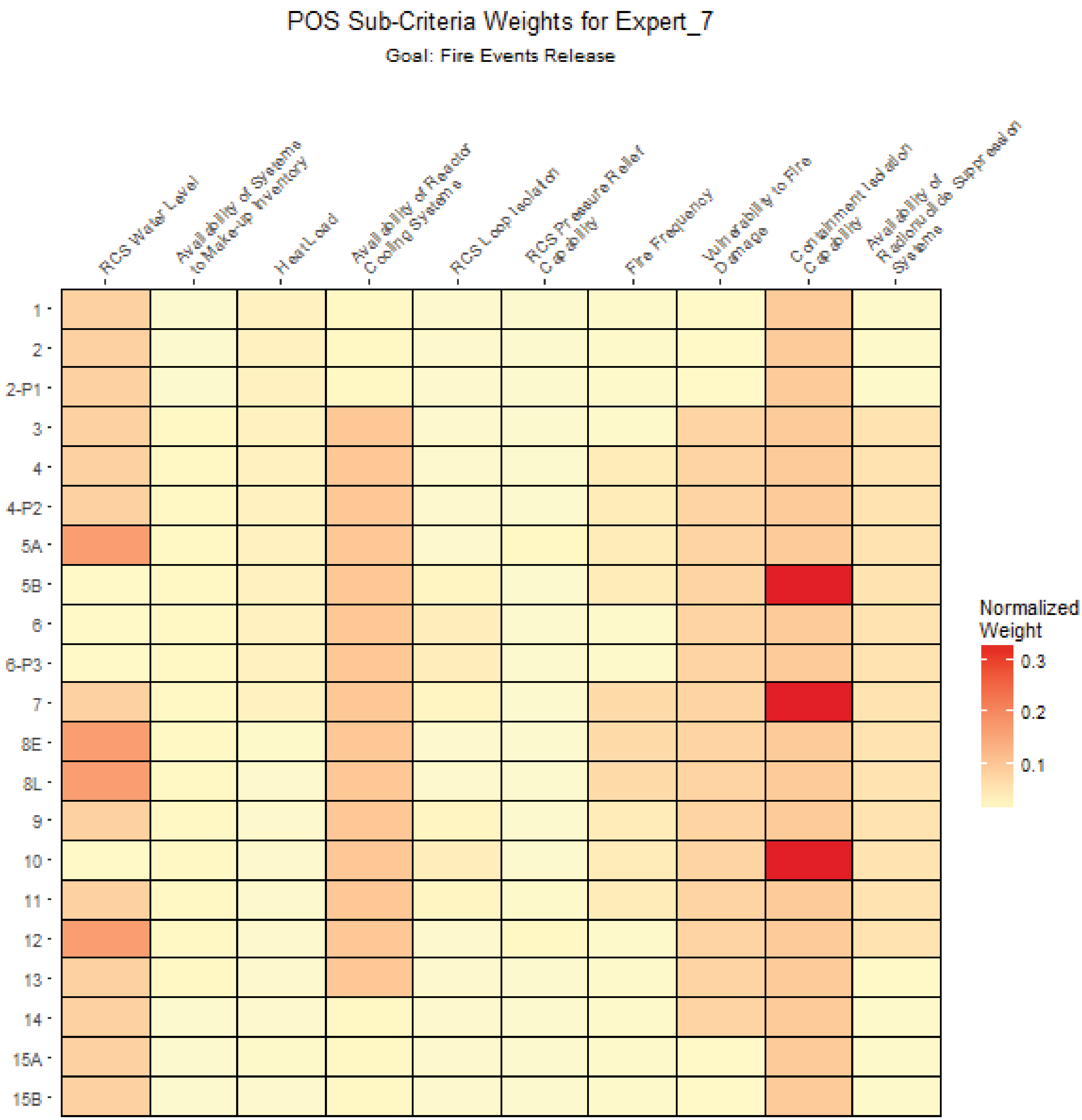
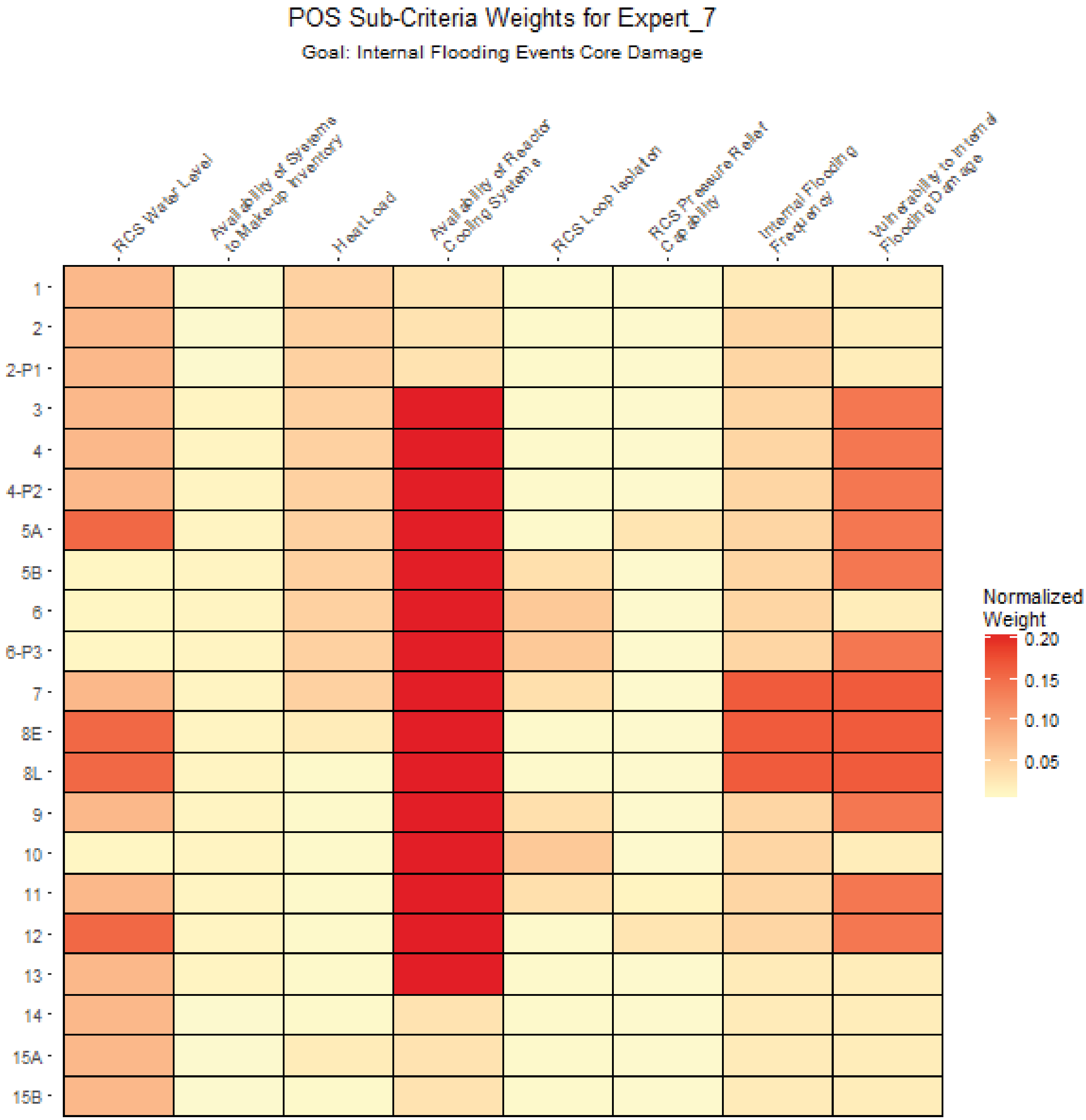
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1


POS Sub-Criteria Weights for Expert_7
Goal: Fire Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.15
0.10
0.05

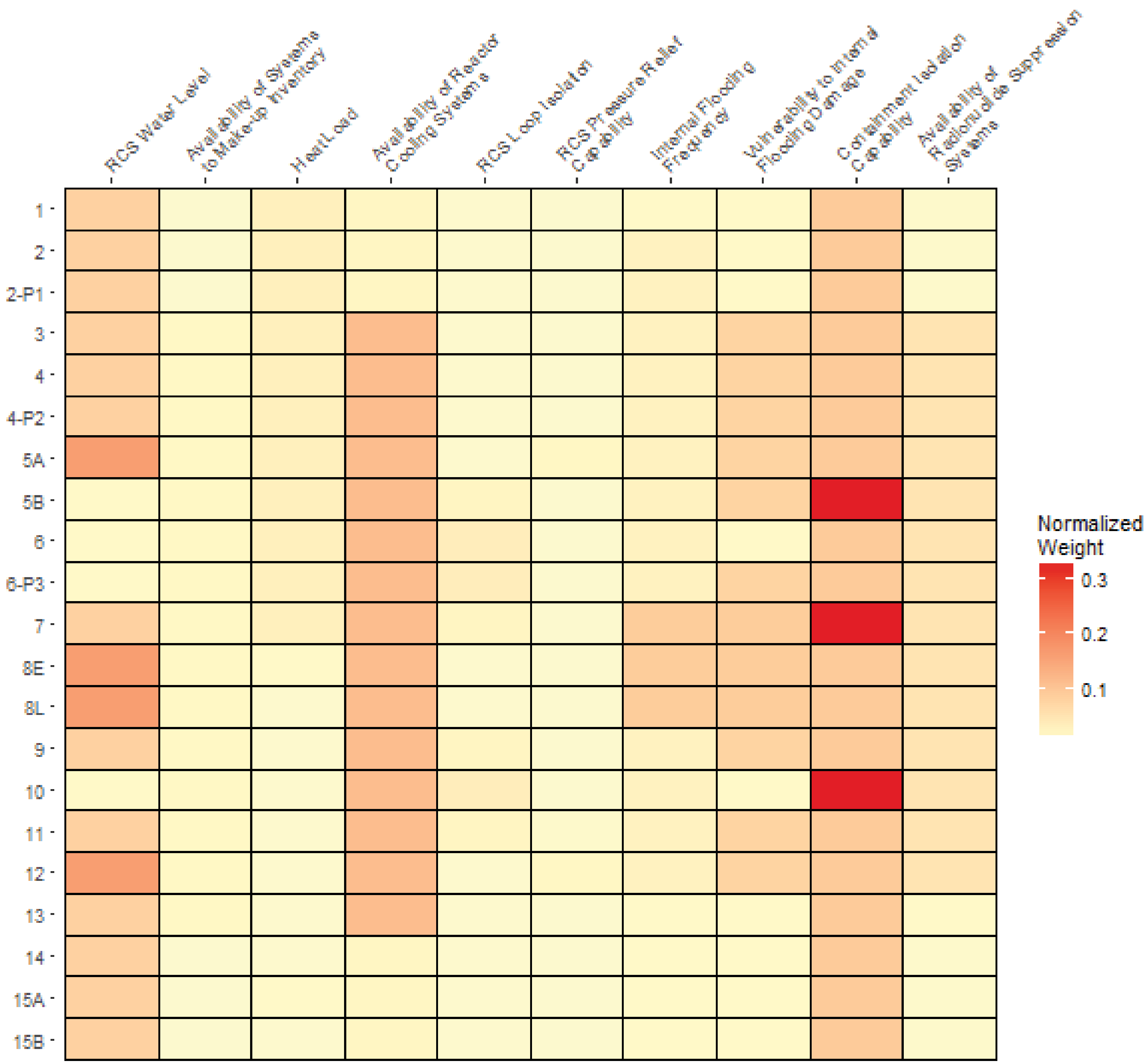


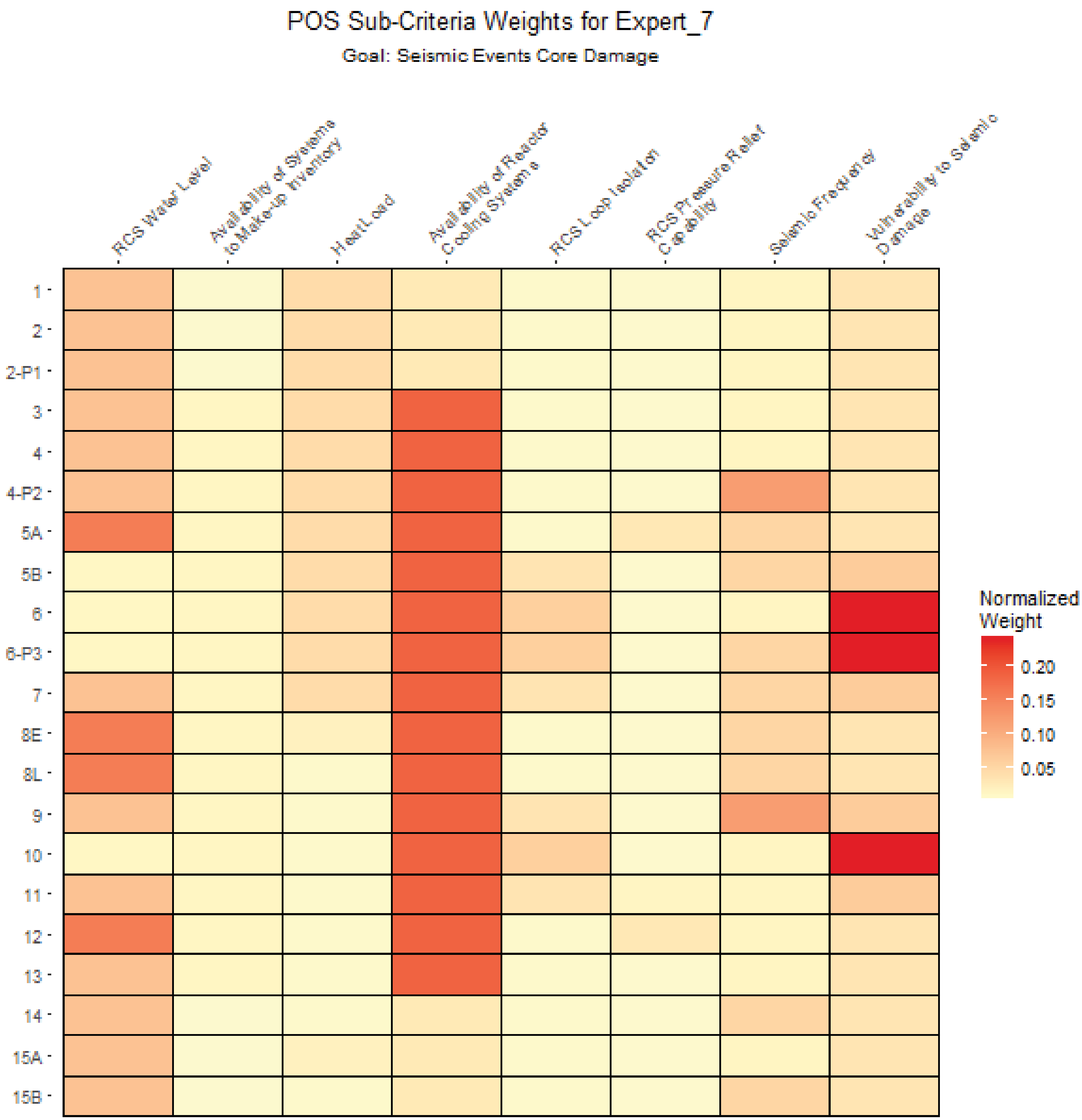
POS Sub-Criteria Weights for Expert_7
Goal: Fire Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Fire Frequency
Vulnerability to Fire Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.3
0.2
0.1


POS Sub-Criteria Weights for Expert_7
Goal: Internal Flooding Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Internal Flooding Frequency
Vulnerability to Internal Flooding Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
7
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.20
0.15
0.10
0.05

POS Sub-Criteria Weights for Expert_7

Goal: Internal Flooding Events Release





POS Sub-Criteria Weights for Expert_7
Goal: Seismic Events Core Damage
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
1
2
2-P1
3
4
4-P2
5A
5B
6
6-P3
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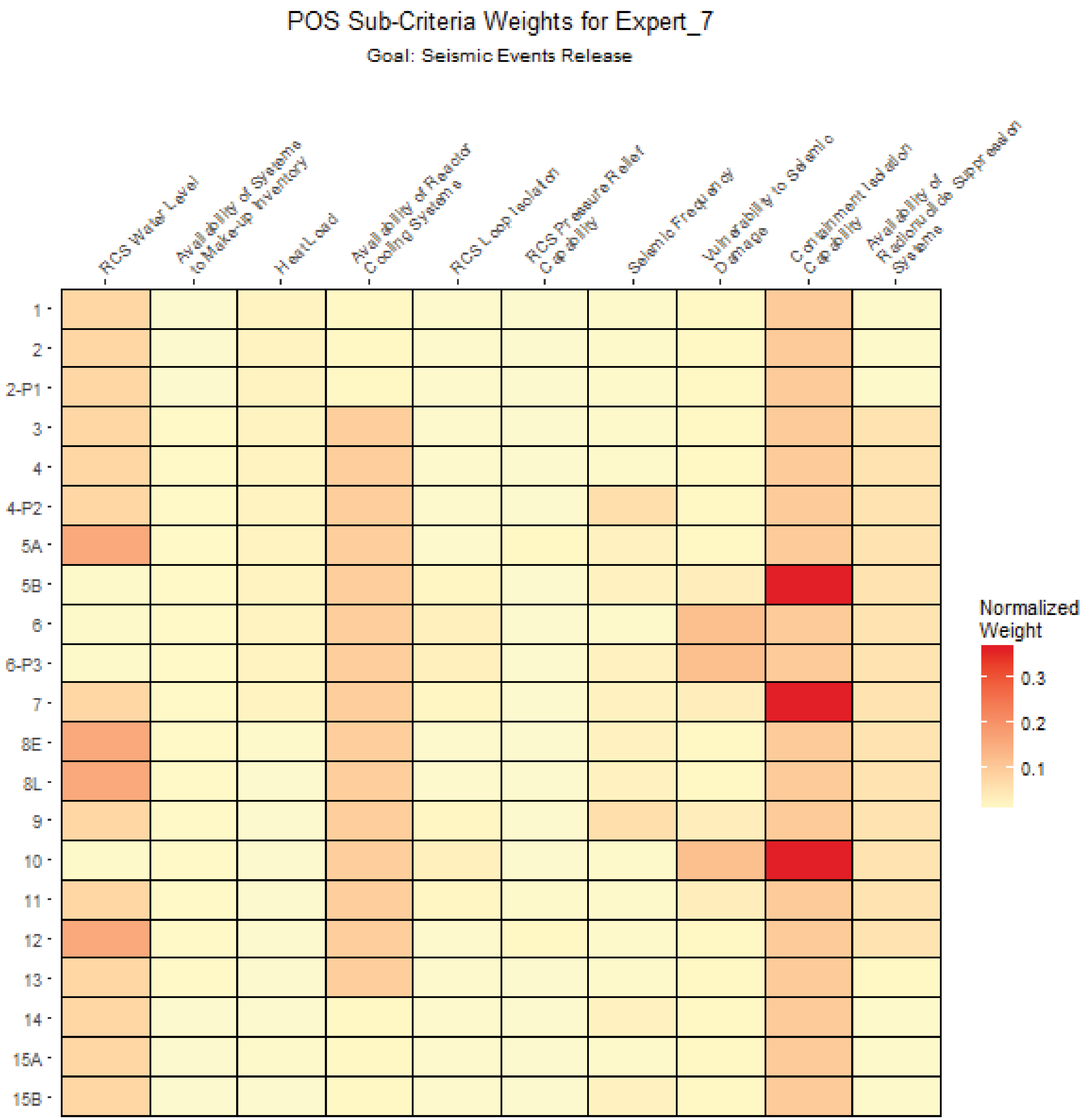
8E
8L
9
10
11
12
13
14
15A
15B
Normalized Weight
0.20
0.15
0.10
0.05


POS Sub-Criteria Weights for Expert_7
Goal: Seismic Events Release
RCS Water Level
Availability of Systems to Make-up Inventory
Heat Load
Availability of Reactor Cooling Systems
RCS Loop Isolation
RCS Pressure Relief Capability
Seismic Frequency
Vulnerability to Seismic Damage
Containment Isolation Capability
Availability of Radionuclide Suppression Systems
1
2
2-P1
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4-P2
5A
5B
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6-P3
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8E
8L
9
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12
13
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15A
15B
Normalized Weight
0.3
0.2
0.1

Expert 7

Top-Level Criteria Weights

	Normalized Weights							
	Internal Events		Fire Events		Internal Flooding Events		Seismic Events	
Top-Level Criteria	Core Damage	Release	Core Damage	Release	Core Damage	Release	Core Damage	Release
RCS Inventory Control	0.18	0.20	0.18	0.19	0.18	0.19	0.18	0.18
Heat Removal	0.40	0.21	0.36	0.19	0.40	0.21	0.36	0.18
RCS Integrity	0.09	0.05	0.09	0.05	0.09	0.05	0.09	0.05
Hazard	0.33	0.17	0.36	0.19	0.33	0.17	0.36	0.18
Containment Performance		0.37		0.38		0.38		0.41
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Sub-Criteria Weights

Sub-Criteria	Normalized Weights							
	RCS Inventory Control	Heat Removal	RCS Integrity	Internal Events Hazard	Containment Performance	Fire Events Hazard	Internal Flooding Hazard	Seismic Events Hazard
RCS Water Level	0.88							
Availability of Systems to Make-up Inventory	0.13							
Heat Load		0.13						
Availability of Reactor Cooling Systems		0.88						
RCS Loop Isolation			0.67					
RCS Pressure Relief Capability			0.33					
Operator Initiated Errors				0.88				
Important Equipment Failures				0.13				
Containment Isolation Capability					0.88			
Availability of Radionuclide Suppression Systems					0.13			
Fire Frequency						0.33		
Vulnerability to Fire Damage						0.67		
Internal Flooding Frequency							0.50	
Vulnerability to Internal Flooding Damage							0.50	
Seismic Frequency								0.33
Vulnerability to Seismic Damage								0.67
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

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Washington, D.C. 20555-0001

10. SUPPLEMENTARY NOTES

J. Wood, NRC Project Manager

11. ABSTRACT (200 words or less)

The U.S. Nuclear Regulatory Commission (NRC) is performing a full-scope, site Level 3 probabilistic risk assessment (PRA), using a four-loop PWR as the reference plant. During the development of the Level 3 PRA, specifically the low-power shutdown (LPSD) analysis, the need to prioritize the plant operating states, hazards, and outage types to include in the full-scope site Level 3 PRA was identified by the Level 3 PRA project team. This need was further magnified by the fact that realistic LPSD modeling of plant outages involves consideration of the range of types of outages, from planned refueling and maintenance outages to unscheduled maintenance outages, and the significant variation in the types of activities that are performed during these outages. This report describes the PIRT process developed and used to meet the project objectives and associated results of implementing the process. The objective was to identify the plant operating states (POSs) and plant outage types (POTs), rank them according to their importance to LPSD risk in the context of different hazards, and consider important influences/phenomena associated with an LPSD model in the ranking process.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
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