Anthony J Vitale
Site Vice President

PNP 2014-063

June 17, 2014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Response to Request for Additional Information – License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

References: 1. ENO letter, PNP 2012-106, "License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors," dated December 12, 2012 (ADAMS Accession Number ML12348A455)

2. ENO letter, PNP 2013-013, "Response to Clarification Request — License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors," dated February 21, 2013 (ADAMS Accession Number ML13079A090)

3. NRC electronic mail of August 8, 2013, "Palisades - Requests for Additional Information Regarding Transition to the Fire Protection Program to NFPA Standard 805 (TAC No. MF0382)" (ADAMS Accession Number ML13220B131)

4. ENO letter, PNP 2013-075, "Response to Request for Additional Information – License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors", dated September 30, 2013 (ADAMS Accession Number ML13273A469)

5. ENO letter, PNP 2013-079, "Response to Request for Additional Information – License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors", dated October 24, 2013 (ADAMS Accession Number ML13298A044)

6. ENO letter, PNP 2013-083, "Response to Request for Additional Information – License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors", dated December 2, 2013 (ADAMS Accession Number ML13336A649)

7. NRC electronic mail of March 11, 2014, "Requests for Additional Information – Palisades – NFPA 805 Project LAR - MF0382" (ADAMS Accession Number ML14118A293)

8. ENO letter, PNP 2014-035, "Revised Response to Request for Additional Information – License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors", dated April 2, 2014

9. ENO letter, PNP 2014-050, "Response to Request for Additional Information – License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors", dated May 7, 2014

10. NRC electronic mail of May 21, 2014, "Requests for Additional Information – PRA - Palisades – NFPA 805 LAR - MF0382" (ADAMS Accession Number ML14142A104)

Dear Sir or Madam:

In Reference 1, Entergy Nuclear Operations, Inc. (ENO) submitted a license amendment request to adopt the NFPA 805 performance-based standard for fire protection for light water reactors. In Reference 2, ENO responded to a clarification request. In Reference 3, ENO received electronic mail Request for Additional Information (RAIs). In Reference 4, ENO submitted the 60-day RAI responses. In Reference 5, ENO submitted the revised 90-day RAI responses. In Reference 6, ENO submitted the 120-day RAI responses. In Reference 7, ENO received electronic mail RAIs on Fire Modeling. In Reference 8, ENO submitted the revised response to RAI SSA 07. In Reference 9, ENO submitted responses to the Fire Modeling RAIs. In Reference 10, ENO received electronic mail RAIs on Fire PRA. Per discussion with the NRC, the RAI response schedule for the RAIs in Reference 10 is as follows:

PRA RAIs due in 30 days (no later than June 20, 2014):

- PRA 01.e.01, PRA 01.f.01, PRA 01.h.01, PRA 01.h.02, PRA 01.k.01, PRA 01.mm.01, PRA 01.q.01, PRA 01.r.01, PRA 01.y.01, PRA 12.01, PRA 31

PRA RAIs due in 90 days (no later than August 19, 2014):

- PRA 01.j.01, PRA 01.l.01, PRA 17.b.01, PRA 20.01, PRA 23.01,
  PRA 23.a.01, PRA 23.c.01, PRA 28.a.01, PRA 30

In Attachment 1, ENO is providing 30-day responses to the RAIs noted above.

A copy of this response has been provided to the designated representative of the State of Michigan.

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 17, 2014.

Sincerely,

ajv/jpm

Attachment:

1. Response to Request for Additional Information Regarding License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors

    cc: Administrator, Region III, USNRC
             Project Manager, Palisades, USNRC
             Resident Inspector, Palisades, USNRC
             State of Michigan

ATTACHMENT 1
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING LICENSE AMENDMENT REQUEST TO ADOPT NFPA 805
PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION FOR
LIGHT WATER REACTORS

NRC REQUEST

PRA RAI 01.e.01

The response to PRA RAI 01.e, in the letter dated December 2, 2013, Agencywide Documents Access and Management System (ADAMS) Accession No. ML13336A649, stated that the primary coolant pump (PCP) seal failure model used the methodology presented in WCAP-15749-P, Revision 1, "Guidance for the Implementation of the Combustion Engineering Owners Group (CEOG) Model for Failure of Reactor Coolant Pump Seals Given Loss of Seal Cooling (Task 2083)", December 2008. This topical has not been endorsed by the NRC.

Describe whether the PCP seal failure is the same for both the compliant and the post-transition PRA models such that the impact of this model on the change in risk estimates is minimal. If the PCP seal model differs between the compliant and post-transition PRA models, or if the model has a substantive impact on the change in risk estimates, provide a summary of the method and the quantitative results that are used in the PRA.

ENO RESPONSE

The primary coolant pump seal failure model is based on the topical report generated by the owners group and endorsed by the NRC (WCAP-16175-P-A).

As part of a model update the revised topical report WCAP-15749-P, was reviewed for impact on the implementation of the seal model. WCAP-15749-P provides guidance on implementation of the seal model as developed per WCAP-16175-P-A. The review of WCAP-15749-P documented that no changes to the existing seal model were required and none were made.

Therefore, the existing seal model remains consistent with the consensus model as endorsed by the NRC as documented in WCAP-16175-P-A.

The seal model incorporated into the PRA model consists of two principal elements. The first element is development and incorporation of seal failure probabilities into the PRA model. The second element includes the plant specific elements with respect to maintaining seal cooling, instrument and control related to primary coolant pump operation and the human error probability for failure to trip the primary coolant pumps.

The seal failure probabilities were developed per and remain consistent with the criteria of WCAP-16175-P-A. The probability of seal failure based on the seal model is the same for both the compliant and post-transition plant. The probability of seal failure was not altered in the post transition plant results.

The probability of failure of support systems required for seal cooling and instrument and control necessary to trip the pumps is a plant specific input to the PRA model logic and is not governed by the consensus model. This element of the model is based on plant specific features with one exception. The human error probability for tripping the primary coolant pumps is based on the time available to accomplish the action defined by WCAP-16175-P-A.

Modification S2-5 (Provide Alternate Method of Tripping Primary Coolant Pumps during Fire Event) as described in Attachment S Table S-2 of the original PNP LAR is being implemented as part of transition to NFPA 805. This modification will provide an alternate capability to trip the primary coolant pumps from the control room. Implementation of the modification impacts the plant specific inputs to the seal model.

Therefore, the difference between the variant and post-transition plant in the PRA model with respect to primary coolant pump seals is in the instrument and control logic associated with pump operation. The variant plant represents the existing plant (no modification). The post transition plant model includes the alternative capability to trip the pumps from the control room. The post-transition plant is compliant with respect to the requirement to ensure primary coolant pumps can be tripped from the control room following a fire. Consequently there is no difference between the 'compliant' and 'post-transition' plant.

The modification reduces the risk associated with the existing pump control circuits which may preclude the ability to trip the pumps due to fire affects. Logic associated with the proposed modification is the only difference between the variant and post-transition plant with respect to the pump seal model.

A summary of the method and the quantitative results that are used in the PRA are not required because the difference in the seal model is:

- in the plant specific element of the model,
- related to a modification to improve plant capability,
- and NOT related to the probability that the seal will fail on loss of cooling

REFERENCES:

1. WCAP-16175-P-A (Formerly CE NPSD 1199 P, Revision 1), Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS Plants, March 2007.

2. WCAP-15749-P, Guidance for the Implementation of the CEOG Model for Failure of RCP Seals Given Loss of Seal Cooling (Task 2083), Revision 1, December 2008.

NRC REQUEST

PRA RAI 01.f.01

The response to PRA RAI 01.f in the letter dated December 2, 2013, ADAMS Accession No. ML13336A649 indicates that the circuit analysis of identified instrumentation for "dominant" operator actions has been completed and will be incorporated into the transition fire PRA risk results, which is to be provided in response to PRA RAI 30.

a. *Discuss what is meant by "dominant" relative to RG 1.200's, "An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities", definition of a significant basic event and whether these non-"dominant" actions are assumed to be failed in the fire PRA.*

b. *If not assumed to be failed, justify this treatment by discussing the risk significance of the credited non-dominant operator actions on the transition risk results.*

ENO RESPONSE

a. 'Dominant' operator actions in the context of the discussion provided in the original response to 01.f was related to a set of operator actions which would be required to be maintained as detailed human error probabilities (HEPs) to offset increases in core damage frequency (CDF) resulting from the assignment of screening or scoping HEPs to other human failure events (HFEs). In addition, the discussion does not mean that other (non 'dominant') actions did not already have instrumentation supporting the operator action included in the model. The discussion was only meant to convey that some actions in the dominant set did not have instrumentation available at that time.

b. It is not the case that all non 'dominant' operator actions are assumed to be failed in the fire PRA. The group of non-'dominant' operator actions includes two subsets comprised of HFEs assigned either scoping or screening values. HFEs assigned a screening value (1.0), are assumed failed in the fire PRA. Events assigned scoping values are analyzed in the same manner as the 'dominant' HFEs to the extent that instrumentation is included in the model, fire induced impacts are considered; access to the area where the action is to be completed is required, operator ability to complete the action is required and instrumentation availability impacts are considered. Scoping HFEs without supporting instrumentation included in the model or those for which the fire fails the instrumentation would be failed in the fire PRA. Revised risk results reflecting the implementation of the above process for incorporation of operator actions will be provided in response to RAI 30.

NRC REQUEST

PRA RAI 01.h.01

In the letter dated December 2, 2013, ADAMS Accession No. ML13336A649, the response to PRA RAI 01.h, subsection 3) "Justifications for Assumptions Identified as Non-Conservative in the licensee's analysis" describes that the treatment of location in the dependency analysis differs from the guidance in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines, Draft Report for Comment". NUREG-1921 guidance does not "negate the possibility of success of all subsequent actions" after failure of an action in the main control room as stated in the RAI response but does state that there would be high dependence between all actions. Simply stating that the approach is not realistic is not sufficient justification to deviate from the NUREG. It also appears that the timing decision branch of Figure 6-1 of NUREG-1921 is not utilized by the dependency analysis for sequential actions due to this deviation.

Provide a time and distance justification for each set of control room actions considered to be in different locations or conform to the accepted method. Identify the final approach used in the response to PRA RAI 30.

ENO RESPONSE

Palisades Nuclear Plant (PNP) will follow the NUREG-1921 guidance and treat all actions taken in the control room as taking place within a single (same) location. The impact of these changes will be reflected in the quantification results documented in response to PRA RAI 30.

NRC REQUEST

PRA RAI 01.h.02

The dependency analysis described in response to PRA RAI 01.h does not indicate that a minimum value was utilized for the joint probability of multiple human failure events (HFE) and the response. The statement, "e.g., for zero dependence, the conditional human error probabilities (HEP) is equal to the independent HEP" implies that joint HEPs may take on any value. Section 6.2 of NUREG 1921 addresses the need to consider a minimum ("floor") value for the joint probability of multiple HFEs. Each value less than the floor value should be individually justified.

Considering this guidance, describe and justify that joint HEP values that appear in fire PRA cutsets including any values less than the floor value. If a HEP floor for cutsets was not used consistent with NUREG-1921 (i.e., 1E-5 with justifications for lower values), provide updated risk results as part of the aggregate change-in-risk analysis requested in PRA RAI 30, which is consistent with NUREG-1921 guidance.

ENO RESPONSE

PNP will follow the guidance of NUREG-1921 and utilize a floor value of 1E-5 for all conditional joint HEPs. The impact of these changes will be reflected in the quantification results documented in response to PRA RAI 30.

NRC REQUEST

PRA RAI 01.k.01

The response to PRA RAI 01.k, in the letter dated December 2, 2013, ADAMS Accession No. ML13336A649, indicates that main control room (MCR) abandonment is only postulated for those fires resulting in a loss of MCR habitability; however, the response to PRA RAI 03, in the letter mentioned above, states that "the RAI Response Fire PRA Model will include additional scenarios that model MCR abandonment due to equipment damage, with control being transferred to other locations, such as the alternate shutdown panel".

If the intent is to credit MCR abandonment due to loss of control, provide a description of the method and its technical justification. Include an explanation of the supporting analysis, work performed, and process followed in the technical justification.

ENO RESPONSE

The response to PRA RAI 01.k was intended to indicate that control room abandonment due to loss of control or function is not explicitly modeled in the Fire PRA. That is, specific identification of those fire events which lead to loss of control or function is not part of the fire scenario development and initial quantification process. Only scenarios that result in control room abandonment due to loss of habitability are explicitly identified as control room abandonment scenarios.

However, the Fire PRA model does include credit for operator deployment for local actions (including local actions at the alternate shutdown panel) as potential success paths in the accident sequence development. Use of these alternate success paths is not limited to control room abandonment scenarios due to loss of habitability.

The response to PRA RAI 03 for FSS-B1-01 was intended to indicate that additional control room scenarios are being added to the RAI Response Fire PRA model. These additional scenarios also credit operator deployment for local actions including local actions at the alternate shutdown panel. The intent is not to explicitly identify and credit control room abandonment due to loss of control.

NRC REQUEST

PRA RAI 01.mm.01

The response to PRA RAI 01.mm, in the letter dated December 2, 2013, ADAMS Accession No. ML13336A649, indicates that key assumptions and sources of uncertainty were identified. Provide a table that describes these key assumptions and sources of uncertainty that assesses their impact on the NFPA 805 application.

ENO RESPONSE

In the development of each Fire PRA report, a section was included that identified assumptions related to each of the associated Fire PRA tasks included in that specific notebook. For each of the identified assumptions, a qualitative assessment was documented regarding the potential quantitative impact as it applies to the base fire PRA model which serves as part of the characterization of the assumptions. In the PNP Fire PRA Quantification and Summary Notebook [1], these assumptions were reviewed to develop a table that identified sources of uncertainty by each NUREG/CR-6850 task and assessed the sensitivity of their impact on the NFPA 805 application. A modified version of this table is provided below. It has been updated to account for the status of the RAI Response Fire PRA model and updated to specifically identify the potential key assumptions associated with the sources of uncertainty.

FPRA UNCERTAINTY AND SENSITIVITY MATRIX

TASK NO.	TASK TITLE	TASK DESCRIPTION	POTENTIAL KEY ASSUMPTIONS AND SOURCES OF UNCERTAINTY	SENSITIVITY OF THE RESULTS TO THE SOURCE(S) OF UNCERTAINTY
1	**Plant Boundary Definition and Partitioning**	The fire PRA analysis boundary was determined, and the plant was partitioned into discrete physical analysis units (PAUs) based on the physical characteristics of the various areas.	This task posed a limited opportunity for the identification of potentially key assumptions and related sources of uncertainty beyond the credit taken for the physical presence of boundaries and partitions.	During scenario development, the zone of influence was not limited to the physical analysis unit boundary for most compartment scenarios. If the zone of influence included targets in adjacent fire areas/zones, these targets were also included, regardless of their fire area/zone location. In addition, a multi-compartment analysis further reduced uncertainty by addressing the potential impact of failure of partition elements on quantification.
2	**Fire PRA Component Selection**	The fire PRA components were selected by reviewing the components in the FPIE PRA model and the equipment included in the deterministic Nuclear Safety Capability Assessment (NSCA) analysis. The data were analyzed with respect to their suitability to be included in the fire PRA model. Additional considerations, including the potential effects of Multiple Spurious Operations (MSOs), were used to evaluate the need to include additional components.	This task posed perhaps the highest potential for error if not uncertainty. The mapping of basic events to components required not only the consideration of failure modes (active versus passive) but an understanding of the Appendix R/NSCA functions not previously considered risk significant in the FPIE model.	The potential for uncertainty was reduced as a result of multiple overlapping tasks including the MSO expert panel process combined with reviews of screening initiating events, screened containment penetrations, and screened ISLOCA scenarios. Additional internal reviews and the change evaluation process provided the opportunity to further reduce uncertainty in this task.

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3	**Fire PRA Cable Selection**	Cables were assigned to the components based on existing Fire Safe Shutdown cable evaluations and additional cable identification.. Tasks 2 and 3 were performed iteratively with the Plant Fire Induced Risk Model (Task 5).	No treatment of uncertainty is typically required for this task beyond the understanding of the cable selection approach for the various iterations of cable identification tasks. Additionally, PRA credited components for which cable routing information was not provided (credit by exclusion) represents a potential key assumption and source of uncertainty. Recognizing that the potential exists to improperly credit these components where their cables are located (non-conservative), it can be assumed that these components are failed unnecessarily (conservative).	The cable selection approach was based on the failure fault consequences identified for each cable relative to the operation of the associated component. These fault consequences were identified in the original Appendix R data. A seperate effort was performed to review this data in light of current practices to assure its fidelity. Since Palisades has undergone an extensive effort to identify cables for components beyond those addressed in Appendix R, uncertainty associated with unknown cable locations (UNL components) has been greatly reduced. In order to eliminate excessive conservatism, UNL components were credited by exclusion – either explicitly or based on assumed cable routing.

In any event, the assumed cable routing is identified as a potential key source of uncertainty. |
| 4 | **Qualitative Screening** | A small number of plant areas met all of the criteria necessary for qualitative screening. | Structures from the global analysis boundary, and ignition sources deemed to have no impact on the FPRA, were excluded from the quantification based on qualitative screening criteria. The only assumptions subject to uncertainty are the judgments regarding the potential for plant trip used as part of the screening process. | No structure with credited PRA components was excluded. This exclusion criterion is not subject to uncertainty. In the event that a structure which could lead to a plant trip was excluded incorrectly, its contribution to CDF would be small (with a CCDP commensurate with base risk) and would likely be more than offset by inclusion of the additional ignition sources and the subsequent reduction of other scenario frequencies. A similar argument can be made for ignition sources for which scenario refinement was deemed unnecessary. |

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5	**Plant Fire Induced Risk Model**	The PNP fire PRA model was developed using applicable portions of the FPIE model. The model was expanded as necessary to include additional sequences associated with fire events. Cables were linked with basic events in the model and associated to plant locations allowing evaluation of fire-induced circuit failures on a per scenario basis.	The construction of the FPRA plant response model itself is a source of uncertainty. The same sources of uncertainty/sensitivity that are applicable to the base model are applicable to the FPRA. However, these are judged to be minor in the context of the overall Fire PRA model development process in the context of the NFPA 805 application. Some 9,000+ failure modes (random and fire) are included in the FPRA plant response model. This includes a highly detailed representation of potential failures (e.g., down to the contact pair level) and fully developed common cause failure modeling. Several thousand cables are mapped to the associated basic events. The bookkeeping challenge of managing this amount of data introduces potential error.	FPIE and FPRA peer reviews (including the F&O resolution process and the subsequent RAI resolution process), internal assessments, and the change evaluation process are useful in exercising the model and identifying weaknesses. In addition, the FPRA model changes are incorporated into the FPIE model. This assures that these sequences are exercised and reviewed continually – not just for fire PRA applications. The potential for managing this amount of data was addressed by employing different industry codes that were used to validate the quantified results. By employing different codes, problems with input are better captured as each code provides different reports, different diagnostic capabilities, etc. The detailed modeling employed in the Palisades analyses ensures better rigor, insights, and reduces errors, and reduces the epistemic uncertainty. Moreover, such detailed modeling results in conservative numerical results as failures are double counted; however, this increases the aleatory uncertainty. It is considered that the importance of reducing the epistemic uncertainty at the expense of increasing the aleatory uncertainty greatly benefits the development of additional risk insights.

FPRA UNCERTAINTY AND SENSITIVITY MATRIX

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6	**Fire Ignition Frequency**	A fire ignition frequency was estimated for each plant compartment based on fixed sources and transient factors. The frequencies were ultimately applied on a scenario basis. The approtionment of the fire frequency was done in accordance with NUREG/CR-6850 guidance and associated FAQs.	The frequency values from NUREG/CR-6850 and EPRI Report 1016735 include uncertainty to account for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates, based on limited detailed data. A potential key assumption is that the fire ignition frequency data is applicable and provides an accepted estimate of the fire frequency for PNP.	A Bayesian update process for PNP events after 2000 was applied to the generic frequencies taken from NUREG/CR-6850 and the EPRI 1016735 data. *The applicablilty of the ignition frequency data is identified as a potential key source of uncertainty.*
7	**Quantitative Screening**	An initial quantification of the fire PRA model was performed to identify the relative risk contribution of each physical analysis unit (PAU). No actual screening was performed as all PAUs were retained in the quantification. This step was used to identify compartments where detailed analyses would be appropriate.	Other than the conservative treatment asscoiated with retaining all scenarios, there is no uncertainty from this task on the FPRA results.	Quantitative screening was limited to refraining from further scenario refinement of those scenarios with a resulting CDF / LERF below the screening threshold. All of the results were retained in the cumulative CDF / LERF.
8	**Scoping Fire Modeling**	Scoping fire modeling is a coarse approach used to bound the fire effects of certain ignition sources. A more refined approach, generic modeling, was employed at PNP. A detailed analysis was performed for typical ignition sources based on their physical properties and prescribed heat release rates. This analysis yielded a guideline for the evaluation of fire damage effects for the various ignition sources. This enabled the development of a basic scenario for many sources that could be treated as bounding.	This task by itself does not contribute to uncertainty. However, the approach taken for this task included: 1) generic fire modeling treatments used in lieu of conservative scoping analysis techniques and 2) limited detailed fire modeling performed to refine the scenarios developed using the generic fire modeling solutions. The primary conservatism introduced by this task is associated with the heat release rates specified in NUREG/CR-6850.	The employment of generic fire modeling solutions did not introduce any significant conservatism. Detailed fire modeling was performed on those scenarios which otherwise would have been notable risk contributorsand applied where the reduction in conservatism was likely to have a measurable impact. The NUREG/CR-6850 heat release rates introduce significant conservatism given the limited fire test data available to define the heat release rates and the associated fire development timeline. However, alternative treatments are not currently accepted.

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9	**Detailed Circuit Failure Analysis**	Circuit failures were evaluated on a failure mode basis using the data provided in the original Appendix R analysis and additional cable data selection efforts. In many cases additional circuit reviews were necessary to determine the specific failure consequences of cables on individual equipment.	Uncertainty considerations are limited to errors in circuit failure analysis where a cable was deemed incapable of causing loss of a particular function credited in the FPRA. Similar to Task 2 (with the exception of the MSO process), this task has no associated uncertainty if performed correctly.	Circuit analysis was performed as part of the Fire Safe Shutdown / NSCA analysis and supplemental circuit selection efforts. Refinements in the application of the circuit analysis results to the fire PRA were performed on a case by case basis where the scenario risk quantification was large enough to warrant further analysis.
10	**Circuit Failure Mode Likelihood Analysis**	Circuit failures based off the failure mode were evaluated in Task 9. In some cases, additional circuit failure likelihood analysis was needed. If applicable, failure probabilities were applied to specific cable failure modes.	The uncertainty associated with the applied conditional failure probabilities posed competing considerations. On the one hand, a failure probability for spurious operation could be applied based solely on cable scope without consideration of less direct fire effects (e.g., a failure likelihood applied to the spurious operation of an MOV without consideration of the fire-induced generation of spurious signal to close or open the MOV). On the other hand, a failure probability for spurious operation could be applied despite the absence of cables capable of causing spurious operation in that location.	Circuit failure mode likelihood analysis was generally limited to those components where spurious operation could not be caused by the generation of a spurious signal. This approach limited the introduction of non-conservative uncertainties. Additional refinement to this approach was performed on risk significant scenarios. Given this treatment, the application of circuit failure probabilities is not considered to be a potential key source of uncertainty.

FPRA UNCERTAINTY AND SENSITIVITY MATRIX

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11	**Detailed Fire Modeling**	The application of detailed fire modeling was limited to the Main Control Room (MCR) abandonment scenario, and a few risk significant areas (e.g., in the 1C and 1D swithcgear rooms). The majority of the other scenarios were analyzed using the generic fire modeling treatments. This task also includes the devleopment of a multi-compartment analysis and structural steel analysis.	Utlimately, the treatment of these issues has evolved through the various RAIs and subsequent model refinements to reduce the number of potential key assumptions. The analysis methodology conservatism is primarily associated with conservatism in the heat release rates specified in NUREG/CR-6850. The primary potential key assumption and related source of uncertainty in this task is in the area of the time delay associated with cable damage that resulted in several different related RAIs.	Detailed fire modeling was performed only on those scenarios which otherwise would have been notable risk contributors and only where removal of conservatism in the generic fire modeling solution was likely to provide benefit either via a smaller zone of influence or to credit automatic suppression. Additional refinement of the fire scenarios was pursued using multi-point analysis of the heat release rates as opposed to the use of a bounding fire for most scenarios. Additional fire modeling was pursued in areas of high risk, notably the switchgear rooms. *The time delay associated with cable damage that was incorporated into the fire modeling is identified as a potential key source of uncertainty.*
12	**Post-Fire Human Reliability Analysis (HRA)**	The post-fire HRA was performed by developing a post-fire human error probability (HEP) for each credited action. For cases where detailed post-fire HEPs were not developed, screening or scoping values were used consistent with the guidance provided in NUREG-1921.	Human error probabilities represent a potentially large uncertainty for the FPRA given the importance of human actions in the base model. A potential key assumption is that the HRA methods utilized for PNP provide representative HEP values in the analysis commensurate with their importance.	Detailed fire HEP values have not been developed in all cases, and screening or scoping HEP values have been applied to some of the less risk significant HEPs. This approach should help reduce the impacts of uncertainty associated with this issue. *In any event, the human error probabilities used in the Fire PRA model are identifed as a potential key source of uncertainty.*
13	**Seismic Fire Interactions**	A qualitative seismic-fire review was performed and documented.	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	Seismic-fire interaction has no impact on fire risk quantification.

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14	**Fire Risk Quantification**	The fire PRA was quantified using the FRANC analysis tool. The quantitative results are summarized in the Fire PRA Quantification and Summary Notebook.	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. One source of uncertainty is the selection of the truncation limit.	Since the fire PRA solves for CCDP (prior to the application of frequency) at a truncation limit of 1.0E-09 for CDF and 1.0E-10 for LERF, there should not be a significant truncation contribution. These truncation limits are several orders of magnitude below the typical values calculated. Additionally, the final truncation values utilized in the integrated one-top model are compared to the PRA standard requirement of less than 5% change per decade of truncation and further discussed in the Fire PRA Quantification and Summary Notebook. As such, the truncation values utilized are not identified as a potential key source of uncertainty.
15	**Uncertainty and Sensitivity Analysis**	Uncertainty and Sensitivity are discussed in the Fire PRA Quantification and Summary Notebook.	This task does not introduce any new uncertainties but is intended to address how uncertainties may impact the fire risk.	N/A
16	**Fire PRA Documentation**	The FPRA is documented in a series of reports.	This task does not introduce any new uncertainties to the fire risk. Uncertainty considerations should be documented in a manner that facilitates FPRA applications, upgrades, and peer review.	The documentation task compiles the results of the other tasks. See specific technical tasks above for a discussion of their associated uncertainty and sensitivity.

Based on the uncertainty and sensitivity review summarized above, potential "key" assumptions (i.e., those that could impact the NFPA 805 application) were identified to include: non-suppression probabilities associated with the cable damage time, human error probabilities, fire ignition bin frequencies (in addition to the sensitivity analysis required by the use of NUREG/CR-6850 Supplement 1 (EPRI) ignition frequencies for all bins), and assumed cable routings.

Sensitivity analysis are performed for each of the potential key sources of uncertainty identified above, and these sensitivity cases will be re-performed with the base RAI Response Fire PRA Model. The results of these sensitivity cases will be included in the updated revision to the Fire PRA Fire Risk Quantification and Summary Notebook for the RAI Response Fire PRA Model.

REFERENCES:

1. Palisades Nuclear Plant Fire Probabilistic Risk Assessment Fire Risk Quantification and Summary, ERIN Report 0247-07-0005.01, Revision 1, November 2012.

NRC REQUEST

PRA RAI 01.q.01

The response to PRA RAI 01.q, in the letter dated December 2, 2013, ADAMS Accession No. ML13336A649, states that the "time delay method" will replace the "damage accrual method" originally employed by the fire PRA. Note that in Section H.1.5.2 of NUREG/CR-6850, the failure times reported in Table H-8 assume steady-state fire exposure conditions and are therefore, not applicable for use in calculating exposure conditions that evolve over time. Provide a technical justification for how the "time delay method" accounts for pre-heating of targets that occurs at heat fluxes prior to reaching the peak heat flux for the fire being analyzed including those below the target damage threshold, and those not already taken into account by Table H-8.

Provide updated risk results as part of the aggregate change-in-risk analysis requested in PRA RAI 30 that appropriately account for pre-heating or that conservatively do not credit the time delay associated with the pre-heating period.

ENO RESPONSE

Consistent with industry precedent (References 1, 2), PNP will revise the Fire PRA RAI Response Model to use the 'damage accrual' method using elements of the Arrhenius methodology (Reference 3, 4). As such, technical justification of the 'time delay' method is not provided. The updated risk results will be included in the response to RAI 30.

Due to the revised approach of using the 'damage accrual' method, reference to the 'time delay' method in the previously submitted responses for RAI FM 01.p and RAI FM 02.b is superseded.

REFERENCES:

1. Turkey Point – NFPA 805 LAR RAI Responses 4-4-14

2. Turkey Point – NFPA 805 LAR RAIs 5-27-14 ML14132A081

3. User Need Request on the Acceptability of the Arrhenius Methodology for Environmental Qualification (EQ) for LOCA and POST-LOCA Environments, ML003701987, February 24, 2000

4. PLP-RPT-00057, Attachment PRA-RAI-01.q.01

NRC REQUEST

PRA RAI 01.r.01

The response to PRA RAI 01r, in the letter dated December 2, 2013, ADAMS Accession No. ML13336A649, states that a one-minute time delay will be applied for credited automatic detection systems.

a. *How is the probability of failure of automatic detection included in the PRA?*

b. *If the automatic detection fails, is manual detection then credited?*

c. *When manual detection is credited after automatic detection fails, is the 15 minute delay used?*

d. *If a logical scenario of detection failure, manual detection with 15 minute delay, and attempted manual suppression is not included in the PRA. Evaluate the impact on the results of not including this scenario or add it to the PRA.*

ENO RESPONSE

a. The fire PRA model is being updated to include the failure probability of automatic detection systems credited in the calculation of manual non-suppression probabilities (NSPs). As stated in the response to PRA RAI 01.r, no automatic detection systems were credited in support of the activation of automatic suppression systems as the automatic suppression systems are all wet-pipe systems. In order to account for the failure probability of automatic detection systems credited in support of manual suppression, two sets of manual non-suppression probabilities are being calculated for each applicable set of fire scenarios.;

   1) The first set is calculated assuming the automatic detection system fails and the corresponding manual detection time is used (e.g. 15 minutes).

   2) The second set is calculated assuming the automatic detection is successful and the corresponding time to detection is used (e.g. 1 minute).

These two sets of NSPs are pro-rated by the automatic detection system success/failure rates. The first set of NSPs are multiplied by the automatic detection system failure probability (e.g. 0.05) and the second set of NSPs are multiplied by the complement of the failure probability (e.g. 0.95). The pro-rated NSPs from each set are summed and applied to the appropriate fire scenarios.

b. Yes, manual detection is credited if automatic detection fails as discussed in the response to part a) above.

c. Yes, as discussed in the response to PRA RAI 01.r, the application of a 15 minute manual detection time is applied when appropriate. If manual detection is not considered credible, manual suppression will not be credited when the automatic detection system is assumed to fail or is nonexistent.

d. As discussed in the response to part a) above, the fire PRA model is being updated so that the NSPs applied to fire scenarios crediting automatic detection also take into account the failure probabilities of these automatic detection systems, and the resulting impact on the detection times. An evaluation of the impact of not including these scenarios is therefore not required.

NRC REQUEST

PRA RAI 01.y.01

The response to PRA RAI 01.y, in the letter dated December 2, 2013, ADAMS Accession No. ML13336A649, appears to indicate that the barrier failure probability is defined by "the most limiting barrier (e.g., non-rated barrier, door, damper, or wall)" and not the sum of the types of barriers present.

Demonstrate that the impact on the results is not significant or provide updated risk results as part of the aggregate change-in-risk analysis requested in PRA RAI 30, summing the barrier failure probabilities for each type of barrier present per NUREG/CR-6850.

ENO RESPONSE

In response to this RAI, the multi-compartment barrier failure probability is being updated to sum the barrier failure probabilities for each type of barrier present per NUREG/CR-6850. The risk results provided with the response to PRA RAI 30 will reflect this change.

NRC REQUEST

PRA RAI 12.01

The ASME PRA standard calls for a focused scope peer review for PRA upgrades, where PRA upgrade is defined in the standard as:

> *"The incorporation into a PRA model of a new methodology or significant changes in scope or capability that impacts the significant accident sequences or the significant accident progression sequences."*

The response to RAI 12 states, "the detailed HEP methodology was reviewed by the peer review and has not been changed. As such, a focused scope review of the HEP analysis is also not warranted." The response to RAI 23.e states, "the use of NUREG-1921 methods for screening, scoping and detailed HEP values constitutes data and methods not included in the fire PRA peer review. However, these data and methods are considered acceptable for use."

a. *Clarify these conflicting statements considering that using data and methods acceptable for use is unrelated to the need for a peer review.*

b. *Describe the method that will be used to ensure that any PRA upgrade will be peer reviewed.*

ENO RESPONSE

a. The response to PRA RAI 12 should be clarified as:

> "the detailed HEP methodology was reviewed by the peer review and has not been changed. As such, a focused scope review of the detailed HEP methodology is also not warranted."

The response to PRA RAI 23.e should be clarified as:

> "the use of NUREG-1921 methods for scoping HEP values constitutes a method not included in the fire PRA peer review. Therefore, the new methods are considered to require a focused scope peer review."

A focused scope peer review on the use of NUREG-1921 scoping methods will be performed consistent with ASME/ANS RA-Sa-2009. Any findings and their resolution will be described in the response to PRA RAI 30.

b. ENO PRA configuration control procedure EN-DC-151 ensures that any PRA upgrades receive appropriate peer reviews.

REFERENCES:

1. NUREG-1921, "Fire Human Reliability Analysis Guidelines", Final Report, EPRI 1023001, EPRI/NRC-RES, July 2012.

2. ASME/ANS RA-Sa–2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", ASME/ANS RA-S Committee and ASME, February 2009.

3. EN-DC-151, Revision 5, "PSA Maintenance and Update", Nuclear Management Manual, November 2013.

NRC REQUEST

PRA RAI 31

The responses to several PRA RAIs (e.g., 01.g, 01.cc, and 03) are contingent on the development of a new "all-inclusive" fire response procedure. Describe if there is an Implementation Item in table S-3 that addresses the development and implementation of this procedure. If not, describe the method that will be used to ensure development of the procedure.

ENO RESPONSE

The completion of a new 'all-inclusive' procedure is an implementation action. Implementation item 1, in Table S-3 of the PNP NFPA 805 LAR, Attachment S, addresses the development and implementation of the new "all-inclusive" fire response procedure. Completion of this implementation item is controlled via the PNP Commitment Tracking Process, specifically under LO-LAR-2013-00052.