Enclosure 2

Handouts discussed during the January 15, 2014 ROP WG Public Meeting

NEI 99-02 White Paper
"Initial Transient"

Two of the questions in NEI 99-02 used to determine if a BWR reactor trip was an Unplanned Scram with Complications include the undefined term "initial transient"; "Was pressure control unable to be established following initial transient?" and "Following initial transient did stabilization of reactor pressure/level and drywell pressure meet the entry conditions for EOPs?" The failure to define the term has resulted in confusion, with some licensees interpreting "initial transient" to be equivalent to "scram response".

The following definition is proposed to be added to NEI 99-02:

> *Initial Transient* is intended to envelope the immediate, expected changes to reactor parameters, such as pressure and level, which normally accompany BWR scrams due to the collapsing of voids in the core and the routine response of the main feedwater and turbine control systems. For example, at some BWRs the reflected pressure wave resulting from the rapid closure of turbine valves during a turbine trip may result in a pressure spike in the reactor vessel that causes one or more safety-relief valves (SRVs) to briefly lift. The intent is to allow a licensee to exclude the momentary operation of SRVs when answering "Was pressure control unable to be established?" The sustained or repeated operation of SRVs in response to turbine control bypass valve failures or Main Steam Isolation Valve (Group I) isolations are NOT a part of routine BWR scram responses and are therefore NOT considered to occur within the initial transient. Similarly, a reactor level decrease to Level 3 following a reactor trip due to the expected collapsing of voids in the core can be excluded when answering the question "Following initial transient, did stabilization of reactor pressure/level and drywell pressure meet the entry conditions for EOPs?" as long as the feedwater control system and at least one feedwater pump were operating as designed. "Initial transient" is different from "scram response", which bounds the time during which the performance indicator is active. The initial transient is a subset of the overall scram response time.

NRC Staff White Paper on
Performance Indicator Validity during
Extended Shutdown and Subsequent Startup

Purpose

This white paper proposes to incorporate guidance into the current revision of NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," and NRC IMCs for determining performance indicator (PI) validity for plants in extended shutdown conditions and the start-up of plants that were in extended shutdown conditions. The staff considers the term "valid" in this white paper to mean that the PI adequately provides enough data for performance assessment purposes.

Guidance in NEI 99-02 and NRC IMCs for determining PI validity would support the ROP objectives of being objective, understandable, and predictable, as well as the NRC objectives of being open and effective. Such guidance would provide a publicly available decision-making framework for determining PI validity during extended shutdowns and plant start-ups. This framework would result in predictable NRC actions and improved effectiveness in communicating PI results to stakeholders and in developing inspection plans for plants.

Background

NEI 99-02 currently provides guidance for determining the applicability or validity of some PIs under certain conditions. However, NEI 99-02 does not provide guidance for determining the validity for other PIs and plant conditions. Plant conditions that would need such determinations include an extended shutdown, which IMC 0608, "Performance Indicator Program," defines as a condition where the reactor has been shutdown for at least six months and the start-up of a plant from an extended shutdown.

Past and current examples that demonstrate the need for such guidance include the restart of the Browns Ferry Nuclear Plant, Unit 1, the extended shutdown of the Crystal River Nuclear Generating Plant and San Onofre Nuclear Generating Station, Units 2 and 3, and plants in the oversight process prescribed in IMC 0350, "Oversight of Reactor Facilities in a Shutdown Condition due to Significant Performance and/or Operational Concerns," and IMC 0351, "Implementation of The Reactor Oversight Process at Reactor Facilities in an Extended Shutdown Condition for Reasons Other Than Significant Performance Problems."

Discussion

NRC staff proposes modifying NEI 99-02 to add and clarify reporting guidance for extended shutdown conditions and subsequent start-ups from extended shutdowns. The staff proposes that NRC Inspection Manual Chapters be updated with guidance for determining PI validity. These proposals are described as follows for each PI.

<u>IE01: Unplanned Scrams per 7,000 Critical Hours</u>

Recommended Changes to IMC 0308, Attachment 1:

> This indicator measures the rate of unplanned scrams over the previous four quarters. The indicator value is the number of unplanned scrams while critical in the previous four quarters times the ratio of 7,000 hours to the total number of hours critical in the previous four quarters.

For plants entering an extended shutdown, if there are fewer than 2,400 critical hours in the previous four quarters the indicator value is displayed as “Not Applicable” (i.e., it is considered invalid) because rate indicators can produce misleadingly high values when the denominator is small. The data elements (unplanned scrams and critical hours) are still reported.

For plants starting up from an extended shutdown, the indicator becomes valid the quarter in which the total number of critical hours within the past four quarters, regardless of the plant operating status during those four quarters, reaches 2400. This is intended to be a clarification of the current methodology for considering PI validity upon start-up from an extended shutdown.

Recommended Changes to NEI 99-02, Revision 7:

Page 12, data example and corresponding change to the graph:

Unplanned Scrams per 7,000 Critical Hours *** indicates first quarter in which reactor is critical if starting up from an extended shutdown lasting longer than a year.**											
	2Q97*	**3Q97**	**4Q97**	**1Q98**	**2Q98**	**3Q98**	**4Q98**	**~~Prev. Qtr~~ 1Q99**	**2Q99**	**3Q99**	**4Q99**
# of Scrams critical in qtr	1	0	0	1	1	1	2	2	0	0	0
Total Scrams ~~over~~within 4 qtrs	1	0	0	2	2	3	5	6	5	4	2
# of Hrs Crit in qtr	1500	1000	2160	2136	2160	2136	2136	1751	0	0	0
Total Hrs Critical in 4 qtrs	1500	2500	4660	6796	7456	8592	8568	8183	6023	3707	1751
	2Q97	**3Q97**	**4Q97**	**1Q98**	**2Q98**	**3Q98**	**4Q98**	**~~Prev. Q~~ 1Q99**	**2Q99**	**3Q99**	**4Q99**
Indicator value	N/A	2.8	1.5	2.1	1.9	2.4	4.1	5.1	5.8	7.55	N/A

IE03: Unplanned Power Changes per 7000 Critical Hours

Recommended Changes to IMC 0308, Attachment 1:

This indicator measures the rate of unplanned power changes over the previous four quarters. The indicator value is the number of unplanned power changes in the previous four quarters times the ratio of 7,000 hours to the total number of hours critical in the previous four quarters.

For plants entering an extended shutdown, if there are fewer than 2,400 critical hours in the previous four quarters the indicator value is displayed as “Not Applicable” (i.e., it is considered invalid) because rate indicators can produce misleadingly high values when the denominator is small. The data elements (unplanned power changes and critical hours) are still reported.

For plants starting up from an extended shutdown, the indicator becomes valid the quarter in which the total number of critical hours within the past four quarters, regardless of the plant operating status during those four quarters, reaches 2400. This is intended to be a clarification of the current methodology for considering PI validity upon start-up from an extended shutdown.

Recommended Changes to NEI 99-02, Revision 7:

Page 18, data example and conforming change to the graph:

	2Q97*	3Q97	4Q97	1Q98	2Q98	3Q98	4Q98	~~Prev. Qtr~~ 1Q99	2Q99	3Q99	4Q99
# of Power Changes in previous qtr	1	0	0	1	2	2	1	3	0	0	0
Total Power Changes in previous 4 qtrs	1	1	1	2	3	5	6	8	6	4	3
# of Hrs Critical in qrtr	1500	1000	2160	2136	2160	2136	2136	1751	0	0	0
Total Hrs Critical in previous 4 qtrs	1500	2500	4660	6796	7456	8592	8568	8183	6023	3707	1751
	2Q97	**3Q97**	**4Q97**	**1Q98**	**2Q98**	**3Q98**	**4Q98**	~~**Prev. Q**~~ **1Q99**	**2Q99**	**3Q99**	**4Q99**
Indicator value	N/A	2.8	1.5	2.1	2.8	4.1	4.9	6.8	7.0	7.6	N/A
*** indicates first quarter in which the reactor is critical after an extended shutdown lasting longer than a year.**											

IE04: Unplanned Scrams with Complications (USwC)

NRC staff and industry discussed the practicality of graying out this indicator. Because the indicator has a 4-quarter look back period, new greater-than-green inspection findings that start within the USwC 4-quarter period, even though the plant has been shutdown and/or fuel removed, could potentially still aggregate with this PI if the PI was white. In addition, the indicator allows for counting scrams that could result even though a reactor was initially subcritical. To eliminate any potential unintended consequences of graying out the indicator, NRC and industry decided not to modify the indicator's validity status. As an alternative, NRC and industry agreed that licensees shall provide a comment with the PI data reporting to indicate that when a scram was not possible for the entire duration of a quarter so that an explanation is available on public NRC Web sites.

Recommended Changes to IMC 0308, Attachment 1:

> This indicator measures the number of unplanned scrams with complications while the reactor was critical during the past four quarters.
>
> For plants that are in extended shutdown conditions, NRC staff and industry discussed the practicality of graying out this indicator. Because the indicator has a 4-quarter look back period, new greater-than-green inspection findings that start within the USwC 4-quarter period, even though the plant has been shutdown and/or fuel removed, could potentially still aggregate with this PI if the PI was white. In addition, the indicator allows for counting scrams that could result even though a reactor was initially subcritical. To eliminate any potential unintended consequences of graying out the indicator, NRC and industry decided not to modify the indicator's validity status. As an alternative, NRC and industry agreed that licensees shall provide a comment with the PI data reporting to indicate that when a scram was not possible for the entire duration of a quarter so that an explanation is available on public NRC Web sites.

Recommended Changes to NEI 99-02, Revision 7:

Page 20, between current guidance at lines 31 and 33:

> The PI data elements continue to be reported for plants that are in extended shutdown conditions. Licensee shall provide a comment in the PI data if for the entire duration of a quarter, plant conditions did not allow for the possibility of a scram to occur. The

comment will acknowledge and describe these conditions (e.g., no fuel was in the reactor for the entire duration of the quarter).

MS05: Safety System Functional Failures

Recommended Changes to IMC 0308, Attachment 1:

This indicator monitors the number of events or conditions that prevented or could have prevented the fulfillment of the safety function of structures or systems in the previous four quarters. This indicator remains valid during an extended shutdown. The indicator should remain valid upon start-up from an extended shutdown with no break in the data.

NEI 99-02 does not provide explicit guidance for determining PI validity for extended shutdown conditions or for start-ups. The data example on page 30 of NEI 99-02 could be misinterpreted to mean that the indicator is not valid until four quarters have elapsed after a start-up.

Recommended Changes to NEI 99-02, Revision 7:

Page 30, beginning a new line at 36:

For plants that are in extended shutdown conditions and for subsequent start-ups, the PI data elements continue to be reported.

Page 31, data example and conforming change to the graph:

Quarter	2Q98*	3Q98	4Q98	1Q9~~8~~9	2Q9~~8~~9**	3Q9~~8~~9	4Q9~~8~~9	~~Prev. Q~~ 1Q00**	2Q00
SSFF in ~~the previous~~that qtr	1	3	2	1	1	2	0	1	0
	2Q98	3Q98	4Q98	1Q9~~8~~9	2Q9~~8~~9	3Q9~~8~~9	4Q9~~8~~9	~~Prev. Q~~ 1Q00	2Q00
Indicator: Number of SSFFs over 4 Qtrs	1	4	6	7	7	6	4	4	3
** In this example, the reactor was shut down in 2Q99 and restarted in 1Q00.									

MS06, MS07, MS08, MS09, MS10: Mitigating System Performance Index (MSPI)

These PIs monitor the performance of selected systems based on their ability to perform risk-significant functions. The MSPI is the sum of the changes in a simplified core damage frequency evaluation resulting from differences in unavailability and unreliability relative to industry standard baseline values. The MSPI is supplemented with system component performance limits. An unavailability index (UAI), unreliability index (URI), and a determination as to whether a system exceeded its component performance limits are reported data elements.

Recommended Changes to IMC 0308, Attachment 1:

MSPI is very reactive when critical hours diminish and is largely dependent on the specific plant's PRA model and MSPI inputs. Because of this sensitivity, the staff and industry agree that validity of MSPI during an extended shutdown and subsequent startup shall be determined on a case-by-case basis using the FAQ process. Once the licensee anticipates that a shutdown will enter an extended period (six months), a FAQ shall be submitted for the ROP Working Group to determine MSPI validity. The licensee shall submit an additional FAQ to establish MSPI validity upon subsequent startup.

For plants that are in extended shutdown conditions, the MSPI data elements continue to be reported. Once MSPI is determined to be invalid, the NRC will display the MSPI data on the NRC public Web-page as "Not Applicable" (i.e., it is considered an invalid indicator).

Recommended Changes to NEI 99-02, Revision 7:

Page 37, beginning a new line at 23:
Extended Shutdown

For plants that are in extended shutdown conditions, the MSPI data elements continue to be reported. Once the licensee anticipates that a shutdown will enter an extended period (six months), a FAQ shall be submitted for the ROP Working Group to determine MSPI validity. The licensee shall submit an additional FAQ to establish MSPI validity upon subsequent startup.

BI01: Reactor Coolant System (RCS) Specific Activity

Recommended Changes to IMC 0308, Attachment 1:

> This indicator monitors the maximum monthly RCS activity in accordance with Technical Specifications (TS) and is expressed as a percentage of the TS limit. The indicator is determined by multiplying 100 by the ratio of the maximum monthly value of calculated activity to the TS limit. If in the entire month, plant conditions do not require RCS activity to be calculated, as described by NEI 99-02, the data field is left blank for that month and the status "Final – N/A" is selected" (i.e., the PI is invalid for that month). This applies during extended shutdown conditions.

Recommended Changes to NEI 99-02:

Page 40, beginning a new line at 35:

> For plants that are in extended shutdown conditions and for subsequent start-ups, the PI data elements continue to be reported.

BI02: Reactor Coolant System Leakage

Recommended Changes to IMC 0308, Attachment 1:

> This indicator monitors the maximum monthly RCS leakage in accordance with Technical Specifications (TS) and is expressed as a percentage of the TS limit. The indicator is determined by multiplying 100 by the ratio of the maximum monthly value of identified (or total) leakage to the TS limit. If in the entire month, plant conditions, as described by NEI 99-02, do not require RCS leakage to be calculated, the data field is left blank for that month and the status "Final – N/A" is selected (i.e., the PI is invalid). This applies during extended shutdown conditions.

Recommended Changes to NEI 99-02, Revision 7:

Page 43, beginning a new line at 8:

> For plants that are in extended shutdown conditions and for subsequent start-ups, the PI data elements continue to be reported.

<u>EP01: Drill/Exercise Performance</u>

Recommended Changes to IMC 0308, Attachment 1:

> This indicator monitors timely and accurate licensee performance in emergency preparedness (EP) drills, exercises, and actual events when presented with opportunities for classification of emergencies, notification of offsite authorities, and development of protective action recommendations (PARs). The indicator is calculated as a ratio (expressed as a percent) of the number of timely and accurate classifications, notifications, and PARs during the previous eight quarters to the total number of opportunities to perform these actions during the previous eight quarters.
>
> This indicator remains valid during an extended shutdown because drills, exercises, or actual events will or could still occur. The indicator remains valid upon start-up from an extended shutdown with no interruption or reset in reporting.

NEI 99-02 does not provide explicit guidance for PI reporting for extended shutdown conditions or for start-ups. The data example on page 49 of NEI 99-02 could imply that the indicator is not valid until eight quarters have elapsed after a start-up.

Recommended Changes to NEI 99-02, Revision 7:

Page 50, line 1:

> For plants that are in extended shutdown conditions and for subsequent start-ups, the PI data elements continue to be reported.

<u>EP02: Emergency Response Organization (ERO) Drill Participation</u>

Recommended Changes to IMC 0308, Attachment 1:

> This indicator monitors the participation of ERO members assigned to fill key positions in EP performance-enhancing experiences. The indicator is calculated as a ratio (expressed as a percent) of the number of ERO members assigned to key positions that have participated in drills, exercises, or actual events during the previous eight quarters to the total number of key positions assigned to ERO members.
>
> This indicator remains valid during an extended shutdown because drills, exercises, or actual events will or could still occur. The indicator remains valid upon start-up from an extended shutdown with no interruption or reset in reporting.

NEI 99-02 does not provide explicit guidance for PI reporting for extended shutdown conditions or for start-ups.

Page 56, line 39:

> For plants that are in extended shutdown conditions and for subsequent start-ups, the PI data elements continue to be reported.

<u>EP03: Alert and Notification System Reliability</u>

Recommended Changes to IMC 0308, Attachment 1:

> This indicator monitors the reliability of the offsite ANS and is a percentage of the sirens that are capable of performing their safety function. The indicator is calculated as the ratio (expressed as a percentage) of the number of successful siren-tests in the previous four quarters to the total number of siren-tests in the previous four quarters.
>
> This indicator remains valid during an extended shutdown because the potential need for alerting the public still exists during extended shutdowns. The indicator remains valid upon start-up from an extended shutdown with no interruption or reset in reporting.

Recommended Changes to NEI 99-02, Revision 7:

Page 61, line 25:

> For plants that are in extended shutdown conditions and for subsequent start-ups, the PI data elements continue to be reported.

OR01: Occupational Exposure Control Effectiveness

Recommended Changes to IMC 0308, Attachment 1:

> This indicator sums the number of occurrences for each of the following three data elements over the previous four quarters at the site.
>
> - The number of TS high radiation area occurrences during the previous quarter
> - The number of very high radiation area occurrences during the previous quarter
> - The number of unintended exposure occurrences during the previous quarter
>
> This indicator does not depend on the operational status of the plant (e.g., critical hours) and is valid during extended shutdowns and subsequent start-ups. For start-ups after extended shutdowns, a total of four quarters after start-up would not need to elapse in order for the data to be valid; data can be valid prior to completing four quarters after start-up.

Recommended Changes to NEI 99-02, Revision 7:

Page 67, line 19:

> For plants that are in extended shutdown conditions and for subsequent start-ups, the PI data elements continue to be reported.

PR01: REST/ODCM Radiological Effluent Occurrence

Recommended Changes to IMC 0308, Attachment 1:

> This indicator calculates the number of RETS/ODCM radiological effluent occurrences (dose rates from liquid and gaseous effluents that exceed rates listed in NEI 99-02) per site in the previous four quarters. This indicator is independent of the operational status of the plant (e.g., critical hours) and is valid during extended shutdowns and subsequent start-ups. For start-ups after extended shutdowns and for new plant start-ups, a total of

four quarters after start-up would not need to elapse in order for the data to be valid; data can be valid prior to completing four quarters after start-up.

Recommended Changes to NEI 99-02, Revision 7:

Page 70, line 18:

For plants that are in extended shutdown conditions and for subsequent start-ups, the PI data elements continue to be reported.

PP01: Protected Area (PA) Security Equipment Performance Index

Recommended Changes to IMC 0308, Attachment 1 (or 6):

This indicator monitors the availability of security equipment. The PI value is the sum of two indices divided by two. The two indices are the number of compensatory hours (the hours a guard needs to be posted because of the unavailability of security equipment) in the previous four quarters divided by the product of a normalization factor and 8760 hours. This indicator is independent of the operating mode of the plant and is intended to be valid during extended shutdowns and subsequent start-ups. For start-ups after extended shutdowns and for new plant start-ups, a total of four quarters after start-up would not need to elapse in order for the data to be valid; data can be valid prior to completing four quarters after start-up.

Recommended Changes to NEI 99-02, Revision 7:

Page 76, line 19:

For plants that are in extended shutdown conditions and for subsequent start-ups, the PI data elements continue to be reported.

Whitepaper on the Definition of "Concurrent" Failures

Introduction

The resolution of Reaction Oversight Process FAQ 12-04 required development of a revision to a footnote in the section on Occupational Exposure Control Efffectiveness (OR01). In Revision 7 of NEI 99-02, the crucial footnote is number 14 on page 66 of the line-in/line-out version, which reads as follows:

> "Concurrent" means that the nonconformances occur as a result of the same cause and in a common timeframe."

Proposed Change to NEI 99-02

As a result of numerous discussions on the meaning and original intent of "concurrent nonconformances" during the resolution of FAQ 12-04, the following is the proposed new text of Footnote 14:

> "Concurrent" means that the nonconformances occur as a result of the same cause and in a common timeframe. Failing to take reasonable action that would end a nonconformance when new information (i.e., a survey indicates, or a knowledgeable individual finds evidence of, unidentified or unexpected radiological conditions) is presented, is itself a new and separate cause for the subsequent (or continued) Tech. Spec. nonconformance, and would not be concurrent with the original Technical Specification High Radiation Area Occurrence."

NRC Staff Proposal for Concurrent Failures

> "Concurrent" means that the nonconformances occur as a result of the same cause and in a common timeframe. Failing to take an action that would have reasonably ended a nonconformance is itself a new and separate cause for the subsequent (or continued) Technical Specification nonconformance and would not be concurrent with the original Technical Specification High Radiation Area Occurrence. Actions that would reasonably end a nonconformance include performing a plant procedure (e.g., a radiation survey, or a verification that Locked High Radiation Area is locked) that would have identified the plant condition, or responding in a timely manner to new information (e.g., the results of a radiation survey, or evidence of the nonconforming radiological condition that is identified by a knowledgeable individual) that indicates the nonconformance.

APPENDIX M

TECHNICAL BASIS FOR THE SIGNIFICANCE DETERMINATION PROCESS (SDP) USING QUALITATIVE CRITERIA

1.0 OBJECTIVE

The objective of this appendix to Inspection Manual Chapter (IMC) 0308, Attachment 3, "Technical Basis for the Significance Determination Process," is to provide a technical basis for using qualitative criteria in determining the safety significance of an inspection finding.

2.0 BACKGROUND

During the early implementation of the Reactor Oversight Process (ROP), the SDP received a significant amount of critical feedback. At the outset there was a need for more SDP tools for staff use, some of the tools available needed more refinement and benchmarking, and the overall process failed to meet timeliness expectations. As a result of these initial challenges, an SDP improvement initiative was developed by the staff of the Inspection Programs Branch (IIPB), which at the time was the lead organization for implementation of the ROP. Additionally, in the late summer of 2002, the Executive Director for Operations (EDO) directed the formation of a NRC task group to perform an independent and objective review of the SDP. This review was prompted, in part, by issues described in a Differing Professional Opinion (DPO) Panel Response dated June 28, 2002, (ML021830090) and an Office of the Inspector General (OIG) Audit Report dated August 21, 2002 (ML023080280). On December 13, 2002, the SDP task group finished its report and provided several recommendations, many of which were consistent with the SDP improvement initiatives developed by IIPB. Some common recommendations involved the use of uncertainty in the SDP, the need to improve clarity of risk-informed decision-making guidance, and the importance of making timely regulatory decisions. These common recommendations suggested that an alternative process (i.e., a new SDP tool) should be developed to estimate the safety significance of inspection findings that are difficult to estimate using quantitative risk analyses and evaluations. Although previous inspection program guidance required NRC management review for findings that could not be evaluated by the SDP, a focus group led by IIPB was created to develop a new SDP tool, which eventually became IMC 0609, Appendix M, "The Significance Determination Process Using Qualitative Criteria," and was initially issued on December 22, 2006.

3.0 TECHNICAL BASIS OF THE METHODOLOGY - OVERVIEW

The technical basis for using qualitative criteria to estimate the safety significance of an inspection finding involves balancing two competing objectives; accounting for uncertainty and making timely regulatory decisions. All probabilistic evaluations have an inherent level of uncertainty associated with their quantitative outcomes. However, the amount of uncertainty can vary depending on how well the risk impact of the finding can be modeled using the state-of-the-art tools (e.g., Standardized Plant Analysis Risk (SPAR) models, SDP appendices). Findings that have a high level of uncertainty with their quantitative results can have drastically different outcomes which are very sensitive to assumption made in the risk analysis. For example, if an initiating event frequency has a large uncertainty band and the mitigation capability to address this initiating event is expected to be unsuccessful, then a very small change in the point estimate of the initiating event frequency will have a significant change in the overall outcome. In these situations it can be challenging for the staff to make a risk-informed decision in a timely manner.

3.1 UNCERTAINTY

There are two types of uncertainty that need to be addressed when using probabilistic risk assessment (PRA) insights to make a risk-informed decision: aleatory and epistemic. Aleatory uncertainty is associated with events or phenomena being modeled that are characterized as occurring in a random or stochastic manner. Epistemic uncertainty is associated with the risk analyst's confidence in the predictions of the PRA model itself and reflects the analyst's assessment of how well the PRA model represents the actual system being modeled. Epistemic uncertainty is also referred to as state-of-knowledge uncertainty. Appendix M accounts only for epistemic uncertainty; aleatory uncertainty is built into the structure of the PRA model itself. It is useful to identify three classes of epistemic uncertainty that are addressed in and impact the results of PRAs: parameter uncertainty, model uncertainty, and completeness uncertainty.

Parameter uncertainty recognizes that the value of such parameters as initiating event frequencies, component failure probabilities or failure rates, and human error probabilities cannot be known with precision. PRAs are capable of addressing parameter uncertainty explicitly; however, the estimated mean value and spread of the uncertainty distribution can vary depending on the availability, quality, and source of data, the type of parameter that is being estimated, and other factors. Model uncertainty recognizes that the relationship between the real plant and its mathematical representation may differ. Model uncertainties that underlie the development of the PRA model are typically handled by making assumptions that then become part of the definition of the PRA model. When there are multiple assumptions that are equally plausible, sensitivity analyses may be conducted using different assumptions to assess their impact on the overall results. A common and significant example of model uncertainty is the determination of degraded conditions and exposure time. Often it is difficult to pinpoint the exact period of time a component was in a failed state and whether or not the component was capable of performing its intended function (i.e., the exact physics of failure). Completeness uncertainty, which can be regarded as a type of model uncertainty, recognizes that the model may not represent every aspect of the as-built as-operated plant, either because it may relate to an unknown dynamic or because accurate models do not exist for some systems or phenomena. The incompleteness of the model includes those aspects the analyst is aware are missing from the model and those that are not known given the current state-of-knowledge. Completeness uncertainties cannot be addressed analytically since, by definition, they stem from risk contributors that are missing from the model.

3.2 TIMELINESS

Timeliness is one of the key objectives of the ROP. The safety significance of inspection findings (i.e., SDP outcomes) are direct inputs into the ROP action matrix. When these inputs are of White, Yellow, or Red significance they have the potential to result in a

Comment [RRL1]: Need to balance realistic assessment versus timeliness supplemental inspection and other actions by both the regulator and licensee depending on the number, significance, and applicable cornerstone(s) of the finding(s). Prompt licensee and NRC staff response to identified findings ensures timely corrective actions to address the cause and to prevent recurrence.

4.0 TECHNICAL BASIS OF THE BOUNDING ASSESSMENT AND DECISION ATTRIBUTES

The results from the bounding evaluation, as practical, and decision attributes are used to provide technical staff and management with a framework to document qualitative information to support the safety significance of inspection findings. The bounding evaluation can vary in size and complexity depending on the nature of the situation. In cases where there are tools available to provide quantitative estimates, but there are large uncertainties associated with the estimated parameters, the bounding evaluation can become quite comprehensive and require a significant amount of resources. In complex systems it can be challenging to determine which assumptions lead to conservative results. Sometimes assumptions that appear to maximize a certain result or outcome could reflect a local maximum instead of a global maximum. In other cases where the available tools are not capable of providing a robust quantitative basis, a simple quantitative approach supplemented with qualitative inputs, as appropriate, might provide a reasonable bounding assessment. When the available tools are unable to provide any quantitative estimate, a completely qualitative approach is an acceptable method. Once the bounding assessment has been established, as practical, the decision attributes are reviewed for their applicability to the finding. If applicable, each decision attribute should have a basis, quantitative and/or qualitative, to justify its use as an input to the decision-making framework. After all the applicable decision attributes have been established with an appropriate basis, the bounding assessment and decision attributes should be evaluated as a whole to arrive at a risk-informed decision.

4.1 BOUNDING EVALUATION

To the extent possible, given the circumstances of the finding, quantitative tools should be used to frame the risk impact of the finding. A quantitative bounding evaluation may provide an upper and/or lower limit (i.e., worse case and/or best case analysis) to reduce the range of potential outcomes. If a quantitative bounding evaluation is not possible, then an appropriate qualitative bounding evaluation can be used to establish an upper limit.

4.2 DECISION ATTRIBUTES

4.2.1 Defense in depth – The defense-in-depth philosophy has traditionally been applied in reactor design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. It has been and continues to be an effective way to account for uncertainties in equipment and human performance and, in particular, to account for unknown and unforeseen failure mechanisms or phenomena, which (because they are unknown or unforeseen) are not reflected in either the PRA or traditional engineering analyses (Ref 1). The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy (Ref 3).

Comment [RRL2]: This by itself is a problem. SDP should be realistic, not bounding

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Comment [RRL3]: Saying bounding is an upper limit is one thing. In practice, it appears that the upper limit is typically used as the final result.

Defense-in-depth consists of a number of elements, and consistency with the defense-in-depth philosophy is maintained if the following occurs (Ref 1):

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- Over-reliance on programmatic activities as compensatory measures is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).
- Defenses against potential common-cause failures are preserved, and the potential for the introduction of new common-cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the plant's design criteria is maintained.

In addition, the introduction to the general design criteria in 10 CFR 50, Appendix A asserts that nuclear power plants consider (1) the need to design against single failures of passive components (as defined in 10 CFR 50, Appendix A) and (2) redundancy and diversity requirements for fluid systems (Ref 1).

4.2.2 Safety Margin – The impact of a finding is typically less if sufficient safety margins are maintained. The following are true when considering if safety margins are sufficient:

- Codes and standards or their alternatives approved for use by the NRC are met.
- Safety analysis acceptance criteria are met and provide sufficient margin to account for analysis and data uncertainty (Ref 1).

4.2.3 Extent of condition – If a finding is not isolated to a specific occurrence, condition, or event, its safety significance is typically greater. When a finding is capable of affecting multiple structures, systems, and components (SSCs), the number of degraded conditions has the potential to be greater than a case in which a finding is isolated to a specific SSC.

4.2.4 Degree of degraded condition (or programmatic weakness) – The magnitude and detailed circumstances of the degraded condition (or programmatic weakness) have a direct effect on the safety significance of the finding. As stated in IMC 0308, Attachment 3 "Technical Basis for the SDP," the finding (i.e., more than minor performance deficiency) is the proximate cause of the degraded condition or programmatic weakness. Logically, the more a condition is degraded or program is weakened, the more safety significant the finding.

4.2.5 Exposure time – Generally, the longer a finding is left uncorrected the more opportunities the finding has to manifest itself (i.e., act as the proximate cause of a degraded condition or programmatic weakness). As such, the longer the exposure

Comment [RRL4]: This should be fleshed out more

Comment [RRL5]: This should also be commensurate with the expected frequency of challenges

Comment [RRL6]: Once again, commensurate with the expected frequency of challenges.

Comment [RRL7]: Some credit should be allowed for codes and standards that have industry consensus endorsement, even if not yet approved by the NRC.

Comment [RRL8]: This needs to be bounded, similar to what is in the RASP handbook

Comment [RRL9]: This may not be a strong link, depending on what systems are impacted.

time the more safety significant the finding.

4.2.6 Recovery actions – Even if the extent of condition, degree of the degraded condition (or programmatic weakness), and exposure time increased the safety significance of a finding, crediting established recovery actions or mitigation strategies should be appropriately considered to determine the overall significance of the finding.

Comment [RRL10]: This should be defined. Some credit should also be allowed for knowledge based decisions.

4.2.7 Additional Qualitative Circumstances for Management Consideration – Depending on the situation, the previous six attributes may not capture all of the qualitative attributes needs to comprehensively describe the details of the finding. Therefore, the ability to add additional qualitative circumstances, as appropriate, needs to be part of this decision making process. Any additional qualitative circumstances for management consideration should have a clear and reasonable nexus to the safety significance of the finding.

4.3 INTEGRATED RISK-FORMED DECISION MAKING PROCESS BASED ON THE BOUNDING EVALUATION AND DECISION ATTRIBUTES

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After the bounding evaluation and decision attributes are established, the final step of the process is to evaluate all the inputs affecting the safety significance of the finding and make an integrated risk-informed decision. Overall, these decision-making inputs help in building an overall picture of the safety significance of the finding. Even though the different inputs (i.e., pieces of evidence) used to describe the safety significance of the finding may not be combined in a formal way, the integrated risk-informed decision needs to clearly document the synergistic effect of the inputs as a whole. The basis for the integrated risk-informed decision is a function of the confidence the NRC staff has in the combined effect the bounding evaluation and decision attributes have on the safety significance of the finding (Ref 1).

5.0 REFERENCES

1. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Revision 2, May 2011.

2. SDP Task Group Report December 13, 2002 (ML023470613)

3. 60 FR 42622, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy
Statement," *Federal Register,* Volume 60, Number 158, p. 42622, Washington, DC, August 16, 1995.

4. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Volume 1, March 2009.

5. The Office of Nuclear Reactor Regulation Office Instruction, LIC-504, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," Revision 3, April, 2010.

6. Inspection Manual Chapter 0609 “The Significance Determination Process”

7. Inspection Manual Chapter 0308, Attachment 3, “Technical Basis for the Significance Determination Process”

END

Attachment 1 – Revision History for IMC 0308, Attachment 3, Appendix M

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Training Required and Completion Date	Comment and Feedback Resolution Accession Number
N/A	MLXXXXXXX XX/XX/XX	Initial Issue.	N/A	MLXXXXXXX

Whitepaper Regarding Consequences to Safety System Functional Failures Performance Indicator

Issue

Evidence is beginning to accumulate that changes to Section 3.2.7 of NUREG-1022 made in Revision 3 are leading to an increase in reports of Safety System Functional Failures (SSFFs) for events in which there is not a true loss of safety function, but momentary conditions in which Technical Specification operability criteria are not satisfied. Per the Reactor Oversight Process, the step increase in reporting of these momentary conditions could inflate the counts of SSFFs and trigger a regulatory response not warranted by the facts.

Discussion

The Purpose section of NEI 99-02 for the MS05 indicator states that, "This indicator monitors events or conditions that prevented, or could have prevented, the fulfillment of the safety function of structures or systems that are needed to:

a. Shut down the reactor and maintain it in a safe shutdown condition;

b. Remove residual heat;

c. Control the release of radioactive material; or

d. Mitigate the consequences of an accident."

Additional guidance from Lines 40 to 44 on page 29 of NEI 99-02 states that, "Unless otherwise specified in this guideline, guidance contained in the latest revision to NUREG-1022, "Event Report Guidelines, 10CFR 50.72 and 50.73," that is applicable to reporting under 10 CFR 50.73(a)(2)(v), should be used to assess reportability for this performance indicator. Questions regarding interpretation of NUREG-1022 should not be referred to the FAQ process." Therefore, this whitepaper is submitted to propose an exemption from the guidance for reporting specific data elements under the criteria for this Performance Indicator and not interpretation of the reporting requirements described in NUREG-1022.

Revision 3 to NUREG 1022 became effective on July 1, 2013. Federal Register Notice 78FR9744, Event Reporting Guidelines, NUREG-1022, Revision 3, Notice of Availability, states: "The comments also indicated that the changes, if implemented, will have the effect of requiring licensees to report events or conditions as a 'loss of safety function' where no function is lost since a system may be declared inoperable and still be capable of providing the function relied upon in the plant's safety analysis. Upon further review, the NRC disagrees and the position found in the draft Revision 3 to NUREG–1022 is retained in the final version. For systems within scope, the inadvertent TS inoperability of a system in the mode of applicability constitutes an event or condition for which there is no longer a reasonable expectation that equipment can fulfill its safety function. Therefore, such events or conditions are reportable."

Based on the above guidance, the scope of items previously reported under the loss of safety function criteria has expanded to include some items for which there is no loss of safety function, but that may result in momentary conditions where operability criteria from the Technical Specifications are not satisfied. A specific area of concern has been identified involving inadvertent momentary losses of secondary containment integrity due to the failure to have one secondary containment access door in each access opening closed. This situation can occur due to personnel error or the need to make minor adjustments to equipment. In these instances, secondary containment is considered inoperable due to failure to meet the surveillance requirement. In cases where these events are of short duration (i.e., less than two minutes), it can be readily demonstrated that the secondary containment function can still be satisfied as discussed below.

As discussed in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," the coolant activity phase of a loss of coolant accident begins with a postulated pipe rupture and ends when the first fuel rod has been estimated to fail. For BWRs, the coolant activity phase starts at the initiation of the accident and continues for the first two minutes following the accident. During this phase, the only activity released to the containment atmosphere is that associated with very small amounts of radioactivity dissolved in the coolant itself. The amount of radioactivity in the coolant itself is limited by Standard Technical Specifications 3.4.7, "RCS Specific Activity." The gap activity release phase begins when fuel cladding failure commences. This phase involves the release of that radioactivity that has collected in the gap between the fuel pellet and cladding. This process releases to containment a few percent of the total inventory of the more volatile radionuclides, particularly noble gases, iodine, and cesium. During this phase, the bulk of the fission products continue to be retained in the fuel itself.

As discussed in NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," which is based upon NUREG-1465, the gap release phase for BWRs starts two minutes following initiation of the accident, and continues until the early in-vessel release phase which starts 30 minutes following initiation of the accident.

The Bases for Standard Technical Specifications 3.6.4.1, "Secondary Containment," states that there are two principal accidents for which credit is taken for secondary containment operability. These are a loss of coolant accident (LOCA) and a fuel handling accident involving handling recently irradiated fuel.

Based on the information above, for BWR plants that have adopted an alternative source term in accordance with 10 CFR 50.67, "Accident source term," using the methodology described in NRC Regulatory Guide 1.183, no activity releases are assumed to occur for the first two minutes following initiation of the LOCA.

The above information supports a position that a momentary loss (i.e., less than two minutes) of secondary containment integrity due to the failure to have one secondary containment access door in each access opening closed does not constitute a loss of safety function of secondary containment, unless the momentary loss occurs when handling recently irradiated fuel.

Momentary losses of secondary containment integrity are, however, reportable to the NRC in accordance with 10 CFR 50.72(b)(3)(v) and 10 CFR 50.73(a)(2)(v) based on guidance in NUREG-1022, Revision 3, which states that "a SSC that has been declared inoperable is one in which the SSC capability is degraded to a point where it cannot perform with reasonable expectation or reliability." Secondary containment is required to be declared inoperable for momentary losses of secondary containment integrity as discussed below. Standard Technical Specifications Surveillance Requirement (SR) 3.6.4.1.3 requires verification that one secondary containment access door in each access opening is closed. In accordance with SR 3.0.1, the failure to meet SR 3.6.4.1.3 constitutes failure to meet LCO 3.6.4.1, "Secondary Containment," which requires secondary containment to be operable. Therefore, SR 3.0.1 would require secondary containment to be declared inoperable if SR 3.6.4.1.3 is not met (i.e., even momentarily).

Since July 2013, there has been a significant increase in reporting of safety system functional failures as a result the changes in the reporting guidance. Since these reports also require reporting under this performance indicator, the specified purpose of the indicator has been altered and will result in an inaccurate representation of safety risks associated with these events.

Although momentary losses of secondary containment integrity due to the failure to have one secondary containment access door in each access opening closed results in a reportable condition, there is no actual or potential loss of safety function as long as the access door is not in a degraded condition and is closed within two minutes. The changes to Section 3.2.7 of NUREG-1022 that were made in Revision 3 are driving an increase in PI MS05 and could trigger a regulatory response not warranted by the facts.