Enclosure 2
Meeting Summary Handouts
of the May 26, 2010
ROP Public Meeting
Dated June 15, 2010

ROP Task Force Proposal on
Plant Modifications and Changes to the MSPI Basis Document

(Presented Orally at April 21, 2010 ROP Meeting with NRC)

The current guidance in NEI 99-02, Revision 6, is silent concerning the impact of Plant Modifications on the basis document. The current guidance starts on page 33 of NEI 99-02, Revision 6, and is titled "Documentation and Changes". There are only two types of changes discussed: (1) changes to PRA coefficients and (2) changes to non-PRA information.

It appears that another section is needed to deal with changes to the plant design. It would implement the following guidance:

Plant modifications should be evaluated for their impact on the MSPI basis document.

Modifications to the plant design that result in either (a) a change to segment or train boundaries, or (b) a change to the set of monitored components, either by addition or deletion shall be reflected in the basis document the quarter following the completion of implementation.

Modifications that result in a monitored component changing type, such as replacing an AOV with a MOV, will be reflected in the basis document the quarter following the completion of implementation.

Modifications of sub-components that are within the boundary of a monitored component, such as the replacement of the emergency AC voltage regulator with a different type, are not required to be reflected in the basis document until the next required basis document update.

A change to the basis document does <u>not</u> require a change to the PRA-of-record. The normal criteria used by a station for evaluating the need to update the PRA will be followed.

A case could be made that some types of changes to the plant design could be made without immediate reflection in the basis document, but each case would need to be evaluated and that evaluation documented anyway. It is more transparent to just change the basis document.

Considerations in Developing a new PI

A) Capable of being objectively measured

B) Allows for the establishment of a risk-informed threshold to guide NRC and licensee actions

C) Provides a reasonable sample of performance in the area being measured

D) Represents a valid indication of performance in the area being measured

E) Represents a verifiable (auditable) indication of performance in the area being measured

F) Encourages appropriate NRC and licensee actions

G) Provides sufficient time for the NRC and licensees to correct declining performance prior to posing undue risk to public health and safety

H) Adheres to the overall objectives of the ROP (i.e., risk-informed, objective, predictable, and understandable).

Proposed Traits for Evaluating Potential New Performance Indicators
Provided by NEI to NRC on 5/20/2010

A. Capable of being objectively measured
1. PIs should be "fact-based", not subjective. They must be "what happened," not "what might have happened."
2. PIs should be capable of being clearly, unambiguously defined. This impacts the ability to report the data and to avoid, where possible, FAQs.

B. Allows for the establishment of a risk-informed threshold to guide NRC and licensee actions
1. PIs should allow users to differentiate performance by risk level or outlier status.
2. PIs should remain "green" when variations in PI inputs are random and turn white only when they become non-random and provide an indication of declining performance.

C. Provides a reasonable sample of performance in the area being measured
1. PIs should be based on data that is easy to collect and report.
2. PIs should not conflict with the data INPO collects. Where possible, PIs with the same names used by NRC and INPO should not have different definitions.

D. Represents a valid indication of performance in the area being measured
1. PIs should measure <u>outcomes</u> or be clearly linked to outcomes, not measure intermediate conditions upstream of observable outcomes.
2. PIs should reflect behaviors and outcomes within the control of management.

E. Represents a verifiable indication of performance in the area being measured
1. PIs should be based on data that is auditable.
2. A single failure, or a single failure above baselines used in PIs (such as MSPI) should not cause a threshold to be crossed

F. Encourages appropriate NRC and licensee actions
1. PIs should permit defining performance bands that are consistent with the risk bands assigned to inspection findings (i.e., a white or yellow require a certain amount of additional inspection. The importance of crossing a threshold in a PI should be commensurate with the importance of crossing an inspection finding threshold).
2. PIs should be transparent and understandable.
3. PIs should provide <u>actionable</u> intelligence.
4. The PI should not encourage inappropriate actions on the part of the licensee.

G. Provides sufficient time to correct declining performance prior to posing undue risk to public health and safety
1. PIs should be predictive or leading indicators.
2. PIs should change slowly, not discontinuously, as performance changes, so management has time to act.
3. PIs should reflect current, not past performance (i.e., PIs should be timely. In practical terms, a one year PI best captures management performance. Longer periods may be necessary for PIs with very low failure rates. Three years is really the maximum appropriate.)

H. Adheres to the overall objectives of the ROP (i.e., risk-informed, objective, predictable, and understandable)
1. Inspections cover areas that the PIs do not. If an additional PI is added, there must be a reduction in inspection hours in the area the PI covers.
2. PIs should provide a benefit to safety that is commensurate with the burden added by the new PI. That burden is on the licensee is collect and report the data and on NRC to inspect the accuracy of the data. In addition, the resources of both industry and NRC are added when additional inspection occurs. This must be balanced against the safety value added of a new PI.)
3. PIs should provide a basis for modulating the NRC's inspection effort in the target area of performance.

Open FAQs on NEI 99-02
Status Date: 5/24/10

No.	PI	Topic	Status	Plant/Co.	Point of Contact
09-10	EP02	Common EOF	On Hold for EP Task Force after 3/18 mtg. Discussed 4/21/10. *NRC NSIR meeting with Marty Hug, et.al., on this subject on 5/26.*	Generic	Tony Feltman, Marty Hug
10-02	IE04	USwC	Introduced at 3/18 mtg. Discussed 4/21/10.	Generic	PGN, Ken Heffner
10-03	IE04	USwC	Introduced at 3/18 mtg. Discussed 4/21/10. *Revised FAQ was e-mailed to NRC on 5/21/10.*	Wolf Creek	WCNOC, T. Damashek
10-04	MSPI	Missing CCF Value	Introduced at 4/21 mtg. *Editorial changes to be conveyed in 5/26 ROP meeting.*	BFN1	TVA, Rod Miller
10-05	IE04	USwC – LOFC EOP	Introduced at 4/21 mtg.	Generic	APS, Mark McGhee, Del Elkinton

FAQ 09-10

FAQ TEMPLATE

Plant: Plant Generic

Date of Event:	10/19/2009		
Submittal Date:	11/09/2009		
Licensee Contact:	Tony Feltman Martin Hug	Tel/email:	ahfeltman@tva.gov mth@nei.org
NRC Contact:		Tel/email:	

Performance Indicator: NEI 99-02 (rev. 6) 2.4 Emergency Preparedness Cornerstone Emergency Response Organization Drill Participation

Site-Specific FAQ (Appendix D)? No

FAQ requested to become effective when approved.

Question Section

NEI 99-02 Guidance needing interpretation (include page and line citation):

Page 50, Lines 3-8

Purpose

This indicator tracks the participation of ERO members assigned to fill Key Positions in performance enhancing experiences, and through linkage to the DEP indicator ensures that the risk significant aspects of classification, notification, and PAR development are evaluated and included in the PI process. This indicator measures the percentage of ERO members assigned to fill Key Positions who have participated recently in **performance-enhancing experiences** such as drills, exercises, or in an actual event.

FAQ 09-10

Page 50, Lines 10 - 13
Indicator Definition
The percentage of ERO members assigned to fill Key Positions that have participated in a drill, exercise, or actual event during the previous eight quarters, **as measured on the last calendar day of the quarter.**

Page 50, Lines 13 - 14
If an ERO member filling a Key Position has participated in more than one drill during the eight quarter evaluation period, the most recent participation should be used in the indicator statistics.

Page 52, Lines 20-22
If a person is assigned to more than one Key Position, it is expected that the person be counted in the denominator for each position and in the numerator only for drill participation that addresses each position. **Where the skill set is similar**, a single drill might be counted as participation in both positions.

Page 52, Lines 24-29
Assigning a single member to multiple Key Positions and then only counting the performance for one Key Position could mask the ability or proficiency of the remaining Key Positions. The concern is that an ERO member having multiple Key Positions may never have a performance enhancing experience for all of them, yet credit for participation will be given when any one of the multiple Key Positions is performed; particularly, if more than one ERO position is assigned to perform the same Key Position.

Page 52, Lines 31-41
ERO participation should be counted for each Key Position, **even when** multiple Key Positions are assigned to the same ERO member. In the case where a utility has assigned two or more Key Positions to a single ERO member, each Key Position must be counted in the denominator for that ERO member and credit given in the numerator when the ERO member performs each Key Position.

Similarly, ERO members **need not individually** perform an opportunity of classification, notification, or PAR development in order to receive ERO Drill Participation credit. The evaluation of the DEP opportunities is a crew evaluation for the entire Emergency Response Organization. ERO members may receive credit for the drill if their participation is a meaningful opportunity to gain proficiency in their ERO function.

Page 53, Lines 1-3
Participation may be as a participant, mentor, coach, evaluator, or controller, but not as an observer. Multiple assignees to a given Key Position could take credit for the same drill if their participation is a meaningful opportunity to gain proficiency.

FAQ 09-10

Event or circumstances requiring guidance interpretation:
The event/circumstance principally involves utilities with common EOFs where the functions of EOF Senior Manager, EOF Key Protective Measures and EOF Communicator are assigned to Key Positions that generically support multiple nuclear sites.

Utilities with a common EOF established to support multiple nuclear sites have made Key Position assignments to provide implementation of the three functions mentioned above and described in NEI 99-02 rev 6.

ERO members assigned to each function are grouped and monitored to ensure that each member receives a “meaningful opportunity to gain proficiency”. This membership is accounted for at the end of each quarter and entered into the ROP process.

Where common EOFs are established supporting multiple sites the EOF, ERO membership is trained, including involvement in a drill and exercise program to ensure that they are fully qualified to respond to each site served by that EOF when emergencies are declared.

To restate the issue another way, this membership represents each nuclear site served by the EOF operationally and functionally.

In general given this prescribed condition procedures, processes and protocols have been established that have generic application or in words the **skill set is similar** in application regardless of the nuclear site involved.

Where benchmarking has been conducted, a common approach to calculating Participation Credit for this EOF Key Position set is as follows;

Participation Credit is given for these “generic” key positions and counted (as specified in NEI 99-02) for all nuclear sites served by the EOF when a Key Position member is provided a meaningful opportunity to gain proficiency during any one nuclear site drill or exercise. This practice is not a new practice nor is this practice the result of a collaborative effort. This has been establish by each utility separately
.

DEP Credit is only provided to the nuclear site included in the drill or exercise additionally as invoked by NEI 99-02.

FAQ 09-10

If licensee and NRC resident/region do not agree on the facts and circumstances explain

NRC region does not agree with the generic participation credit approached and has specified that participation credit can ONLY be provided to the specific site involved in the drill or exercise.

Potentially relevant existing FAQ numbers

NA

<u>Response Section</u>

Proposed Resolution of FAQ

1) Revise NEI 99-02 to provide clarifying language to more effectively communicate counting participation credit for NEI 99-02 EOF positions when centralized Emergency Offsite Facilities are utilized.
2) The concept of a centralized Emergency Offsite Facility was being utilized prior to the issuance of NEI 99-02 at a minimum of three utilities. Tennessee Valley Authority, Exelon and the Salem-Hope Creek facility each had centralized Emergency Offsite Facilities. Additionally Exelon executed a pilot for NEI 99-02 where participation credit was counted for each plant served by the centralized Emergency Offsite Facility.

If appropriate, provide proposed rewording of guidance for inclusion in next revision.
[PARTICIPATION]

NEI 99-02 Revision 6, page 54

expected to be just a phone talker who is not tasked with filling out the form. There is no intent to track a large number of shift communicators or personnel who are just phone talkers.

Where an approved centralized Emergency Offsite Facility (EOF) serves multiple nuclear plant sites at a number of locations (fleet concept) participation may be counted for each of the nuclear sites served by the centralized EOF when;

- Key EOF Positions are functionally aligned as prescribed in NEI 99-02.
- Key EOF Positions support similar key skills and functions
  - When only site specific attributes (i.e., evacuation sections, EALs, etc.) differ but the key skills and functions to attain the attributes are similar then participation credit may be counted.
- All other NEI 99-02 criteria for participation are met.
- Specifically the following criteria shall be met to grant participation credit:
  - Dose assessment – same software used for all sites.
  - Field monitoring team tracking and control are the same if EOF directs teams. Radio systems are the same.
  - PAR process is the same.
  - Notification form and equipment the same.
  - There are advisors on each team in the EOF that are familiar with each plant so that the EOF Senior Manager and EOF Key Protective Measures ERO Member may be advised on EALs, site terrain and special weather condition attributes, plant operation (BWR and PWR experience) and radiation monitoring system characteristics.

[DRILL AND EXERCISE PERFORMANCE]

NEI 99-02 Revision 6, page 48

the exercise. Thus, a licensee may choose to not include a PAR beyond the 10-mile EPZ as a DEP PI statistic due to its ad hoc nature.

If a licensee discovers after the fact (greater than 15 minutes) that an event or condition had existed which exceeded an EAL, but no emergency had been declared and the EAL is no longer exceeded at the time of discovery, the following applies:

- *If the indication of the event was not available to the operator, the event should not be evaluated for PI purposes.*
- *If the indication of the event was available to the operator but not recognized, it should be considered an unsuccessful classification opportunity.*

- *In either case described above, notification should be performed in accordance with* NUREG-1022 and not be evaluated as a notification opportunity.

Where an approved centralized Emergency Offsite Facility (EOF) serves multiple nuclear plants sites at a number of locations (fleet concept) DEP for any drill or exercise may be only counted for the participating nuclear sites served by the centralized EOF and principally involved in actual or simulated emergency event.

FAQ TEMPLATE

Plant: _Generic ___________________
Date of Event: __NA_____________________
Submittal Date: __January 21, 2010 __________
Licensee Contact: __Ken Heffner_________ Tel/email: _919-270-5611/kmh@nei.org
NRC Contact: __Nathan Sanfiilipo_____ Tel/email: _301-415-3951/nathan.sanfillipo@nrc.gov

Performance Indicator:
IE04 Unplanned Scrams with Complications

Site-Specific FAQ (Appendix D)? No

FAQ requested to become effective when approved

Question Section

NEI 99-02 Guidance needing interpretation (include page and line citation):

NEI 99-02 Revision 6, Page 20 lines 22 to 46, page 22 lines 35-45, and page 23 lines 1-10 discuss whether or not Main Feedwater was available following an unplanned scram.

Event or circumstances requiring guidance interpretation:

When FAQ # 467 was approved, the response section stated that the guidance in NEI 99-02 should be reviewed to see if it needs to be revised based on circumstances that might require the availability of feedwater beyond 30 minutes and whether consideration of the scram response time window remains an appropriate marker for judging a complication to recovery from an unplanned scram.

The purpose of this FAQ is to define what constitutes scram“ response” as opposed to scram “recovery.”

If licensee and NRC resident/region do not agree on the facts and circumstances explain

In FAQ #467, the plant’s recommendation was to change the guidance in two locations:

1. If operating prior to the scram, did Main Feedwater cease to operate and was it unable to be restarted during the reactor scram response? The consideration for this question is whether Main Feedwater could be used to feed the reactor vessel if necessary. When considering the availability of Main Feedwater, it should be able to be restarted within the first 30 minutes following the scram.

The Senior Resident’s response was that this guidance change would not capture those events that are of higher safety significance because main feed is not available, even if it was not required to be used, and 30 minutes is a completely arbitrary number.

2. Operations should be able to start a Main Feedwater pump and start feeding the reactor vessel with the Main Feedwater System within 30 minutes of the initial scram transient. During startup conditions where Main Feedwater was not placed in service prior to the scram, the question would not be considered, and should be skipped.

   This Senior Resident's response to this proposed change was that even if the main feed steam supply is temporarily isolated, the PI should capture those events where main feed couldn't be restored in a relatively short time. "It might be different if the equipment was designed such that restoration was not possible

Potentially relevant existing FAQ numbers
467

<u>Response Section</u>

Proposed Resolution of FAQ

The first 30 minutes after the scram is considered scram response and Main Feedwater must be available in the event that it could be needed. After 30 minutes is considered scram recovery.

If appropriate, provide proposed rewording of guidance for inclusion in next revision.

Plant: Wolf Creek Generating Station (WCGS)
Date of Event: April 28, 2009
Submittal Date: March 18, 2010
Revised: May 20, 2010
Licensee Contact: Terry Damashek Telephone: 620-364-8831, ext #8012
Email: tedamas@wcnoc.com
NRC Contact: Christopher Long Telephone: 620-364-8653
Email: chris.long@nrc.gov

Performance Indicator: IE04, Unplanned Scrams with Complications
Site-Specific FAQ (Appendix D)? No
FAQ requested to become effective when approved.

QUESTION

NEI 99-02 Guidance needing interpretation:

Page 19, "Was Main Feedwater unavailable or not recoverable using approved plant procedures following the scram." Attachment H, Page H-4, Lines 36 through 39, "Some other designs have interlocks in place to prevent feeding the steam generators with main Feedwater unless reactor coolant temperature is greater than the no-load average temperature. These plants should also answer this question as "No" and move on."

Event or Circumstances requiring guidance interpretation:

On April 28, 2009, WCGS experienced a reactor trip (scram)/turbine trip due to 'B' Steam Generator (SG) lolo water level caused by a main feedwater regulating valve (MFRV) controller failure. All equipment functioned as required. Steam generator water level control during and immediately after the scram was not an issue and the plant responded as expected. As designed, both Steam Driven Main Feedwater Pumps tripped on the feedwater isolation signal and steam generator water levels were restored and maintained by auxiliary feedwater flow. RCS temperature stabilized below 560°F and remained there. All required systems for a non-complicated scram functioned as required. Normal plant trip procedures were used and then normal plant recovery procedures were entered. Both the plant design and the approved EOPs do not allow for restart of the main feedwater pumps during a normal plant trip for WCGS.

Prior to the trip, the Main Feedwater Pumps were operating normally, and subsequently tripped per design on the expected Feedwater isolation signal. At the time of the trip, there was no indication that the main feedwater pumps would not have functioned. Several days later, during preparations for restart and return of the plant to power, both Steam Driven Main Feedwater Pumps and the Startup Feed pump required maintenance assistance to return them to service. The event was reported in the monthly performance indicator IE01 as an Unplanned Scram per 7000 Hours.

On a normal scram from power, WCGS expects to receive a feedwater isolation signal on low Tavg coincident with P-4 and a LoLo SG level Feedwater Isolation signal. If main feedwater does not isolate following a scram, manual isolation of feedwater is directed in the scram response procedures. The logic for main feedwater isolation on low Tavg coincident with P-4 can be reset any time after the signal is received, however the SG LoLo water level isolation signal cannot be cleared until the SG LoLo water level condition is cleared. This prevents feeding with the main feedwater pumps and adding positive reactivity via cooling of the moderator. Emergency Operating scram response procedures do not include reset of the feedwater isolation signal for low Tavg coincident with P-4, or restart of the Main Feedwater Pumps. After Emergency Operating procedures are exited, Normal Operating procedures are entered. The Normal Operating procedures provide the Operator options to restart the Steam Driven Main Feedwater Pumps, or the Startup Feedwater pump, or continue to maintain SG water level using the Auxiliary Feedwater Pumps.

Plant start up procedures do not place the Steam Driven Feedwater Pumps in service until after the reactor is restarted and producing power above the point of adding heat. This is due to the high steam demand needed for motive force.

The following information is from the WCGS Technical Specification Bases and describes the functions of the ESFAS interlock -Reactor Trip/P-4 (which include feedwater isolation coincident with P-4):

- *Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4*

  The P-4 interlock is enabled when a reactor trip breaker (RTB) and its associated bypass breaker is open. Manual reset of SI following a 60 second time delay, in conjunction with P-4, generates an automatic SI block. This Function allows operators to take manual control of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the RTBs have been manually closed.

  The functions of the P-4 interlock are:

  - Trips the main turbine;
  - *Isolates MFW with coincident low Tave;* [emphasis added]
  - Allows manual block of the automatic reactuation of SI after a manual reset of SI; and
  - Allows arming of the steam dump valves and transfers the steam dump from the load rejection Tave controller to the plant trip controller; and
  - Prevents opening of the MFW isolation valves if they were closed on SI or SG Water Level – High High.

  Each of the above Functions is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in core power. To avoid such a situation, the noted

> *Functions have been interlocked with P-4 as part of the design of the unit control and protection system.* [emphasis added]

Based on the emphasized information above, normal main feedwater is not required and unavailability does not impact normal scram recovery actions. A review of the Updated Safety Analysis Report showed that the Main Feedwater Pumps are not credited in the safety analysis for Wolf Creek Generating Station.

Wolf Creek Nuclear Operating Corporation's (WCNOC) position is that current plant design, which includes an Engineered Safety Features Actuation System (ESFAS) interlock (Reactor Trip, P-4) to prevent feeding the SGs with the Main Feedwater System when Tavg is < 564°F (no-load Tavg is 557 °F) and the reactor tripped, along with normal scram response procedures that do not permit reset of this signal, would result in answering "No" to the question "Was Main Feedwater unavailable or not recoverable using approved plant procedures following the scram?" WCNOC's position is based on the following guidance contained in NEI 99-02:

- NEI 99-02, Page 17, describes the purpose of Unplanned Scrams with Complications Indicator as follows: "This indicator monitors that subset of unplanned automatic and manual scrams that require additional operator actions beyond that of a normal scram. Such events or conditions have the potential to present additional challenges to the plant operations staff and therefore, may be more risk-significant than uncomplicated scrams." As described above, the condition of the Main Feedwater Pumps (tripped) does not require additional operator actions in response to a scram. The normal scram response procedures do not reset the P-4/Lo TAVG signal, and do not recover the Main Feedwater Pumps.
- NEI 99-02, Page 19, describes criteria for answering the question "Was Main Feedwater unavailable or not recoverable using approved plant procedures following the scram?". This section states the following: "If design features or procedural prohibitions prevent restarting Main Feedwater this question should be answered as 'No'." As described earlier, plant design (P-4 interlock) prevents restarting Feedwater and the scram response procedures to not permit resetting of the Feedwater Isolation signal for Low Tavg coincident with P-4.
- NEI 99-02, page H-4, Section H 1.5, second paragraph, which states: "Some other designs have interlocks in place to prevent feeding the steam generators with main Feedwater unless reactor coolant temperature is greater than the no-load average temperature. These plants should also answer this question as 'No' and move on." As described above, the P-4 interlock coincident with Lo Tavg isolates Main Feedwater and prevents feeding Steam Generators any time the reactor trip breakers are open and Tavg is below 564 °F.

If Auxiliary Feedwater cannot maintain adequate decay heat removal for any reason, guidance is provided in emergency response procedure EMG FR H-1, "Response to Loss of Secondary Heat Sink," to restore the Main Feedwater System on a loss-of-all-feedwater flow to the steam generators. It gives directions to defeat isolation signals by installing four to six jumpers per SG behind the main control boards. Utilization of this pathway would result in a scram with Complications because WCNOC would have to answer 'Yes' to the next question, "Was the scram response procedure unable to be completed without entering another EOP?" found on page 20, lines 2 & 3 and Figure 2.

In summary, this performance indicator was developed to track scrams where operators were required to perform actions outside of those expected for a normal scram. The importance of Main Feedwater as a mitigating system varies by plant design, and in WCNOCs case, Main Feedwater is not required for response to normal uncomplicated scrams. Availability of a component or system <u>when not required</u> should not be considered a factor for this indicator. While WCNOC was not satisfied with the performance of the Main feedwater pumps in this instance, their performance is monitored through Maintenance Rule indicators that are separate from the indicator discussed in this FAQ.

Although WCNOC reported an earlier SCRAM as complicated with similar circumstances, this should not be set as precedence. This was reported without a detailed review of the NEI 99-02 guidance contained in Attachment H.

<u>NRC Senior Resident Inspector Position:</u>

SRI Position Summary

The SRI disagrees with Wolf Creek and feels that the April 28 trip should have been reported as a scram with complications. On April 28, 2009, Wolf Creek did not have the ability to restore and use main feedwater in normal or emergency operating procedures because all three main feedwater pumps required maintenance, and not because of isolation signals. Any of the three main feedwater pumps can be procedurally started in Mode 3. The FWIS, including P4+Tavg <564F and lo lo S/G level, can be cleared with the pushbuttons or jumper wires per normal or emergency operating procedures. Page H-4, lines 27 to 29 state that the PI measures the **ability** [emphasis added] to implement emergency procedures on loss of auxiliary feedwater. Actual implementation of other emergency procedure is monitored elsewhere. This approach is also consistent with page H-5, lines 20-23, which provide for clearing of isolation signals in order to use main feedwater.

SRI Basis

The SRI believes that although there is a Feedwater Isolation Signal (FWIS, P4 interlock), the April 28, 2009 scram should still count towards the Scrams with Complications PI. Wolf Creek procedure GEN 00-005, "Minimum Load to Hot Standby," revision 62 directs reactor operators to depress the FWIS reset pushbuttons and check that the P4 FWIS annunciator is clear. Main feedwater valves can then be opened even if reactor trip breakers are open, coincident with reactor coolant system temperature below 564F. The control room pushbutton circuitry has a retentive memory device and the valves will remain open until the reactor trip breakers are cycled or the RCS goes above and below 564F. If this happens a second time, the reset button can be depressed again and main feedwater can be re-established. This interlock does not prevent feeding the steam generators with main feedwater because of normal (GEN 5) and off-normal

(EMG FR-H1) plant procedures and the reset pushbutton. The SRI felt page H-5, lines 20 to 23 state that a FWIS does not constitute a loss of main feedwater as long as it can be cleared and feedwater restarted. Procedure EMG FR-H1 also provides instructions for reactor operators to clear the P4+564F and lo lo steam generator level signals with jumper wires. The FWIS handswitch could also be used for P4. The flow path was viable.

The SRI agrees with Wolf Creek's position that actual use of EMG FR-H1 would count towards the PI because of entry into another EMG per NEI 99-02 section H 1.6. The plant trip on April 28, 2009, did not require entry into procedure EMG FR-H1.

Procedure EMG FR-H1 allows and provides steps to use any of the three main feed pumps. However, if procedure EMG FR-H1 was used on April 28, 2009, the main feedwater portion of the procedure would not have been successful because all three main feedwater pumps required maintenance (speed switch, servo valve, and a circuit breaker). Consistent with page 19 of NEI 99-02, Revision 6 and page H-4, lines 24 to 29, the PI monitors the ability of main feedwater to be used to feed the steam generators if necessary in emergency operating procedures. On April 28, 2009, Wolf Creek did not have the ability to restore and use main feedwater in normal or emergency operating procedures because all three main feedwater pumps needed maintenance, and not because of isolation signals.

Wolf Creek does not appear to be a design that applies to page H-4, lines 36 to 38. The P4 FWIS occurs with Tave at 564F which is above no load Tave of 557F cited on page H-4. A Tave of 564F corresponds to a reactor power of approximately 11%. The Wolf Creek total plant setpoint document defines low Tave as 553F (P-12) and lo lo Tave as 550F (Turbine loading stop). If auxiliary feedwater actually failed, and EMG FR-H1 was used, then the RCS is likely to be at 557F or above. RCS temperature is likely not to be a concern prohibiting initial use of main feedwater until the plant is cooled below 564F and the signal would have to be reset again.

Wolf Creek did count the March 2008 scram as complicated. There is no discussion of the main feedwater in Wolf Creek's NRC PI procedure.

Expected reactor trip parameters should not be used as a reason to exclude main feedwater availability from this performance indicator. But, if the NEI/NRC ROP Working Group determines that Wolf Creek is correct, then the Appendix H should be rewritten to explicitly exclude Westinghouse units from the main feedwater availability portion of this performance indicator.

<u>Potentially Relevant Existing FAQ Numbers:</u>
None

RESPONSE:

Proposed Resolution of FAQ:
This event should not count against the Unplanned Scrams w/Complications PI.

FAQ 10-04

Plant: **Browns Ferry Nuclear Plant, Unit 1 (BFN 1)**

Date of Event: _6/1/2007_____________________
Submittal Date: _4-21-2010____________________
Licensee Contact: _Rod Cook________________ Tel/email: _(423) 751-2834____
NRC Contact: ________________________ Tel/email: ________________

Performance Indicator: MS06 - MS10

Site-Specific FAQ (Appendix D)? Yes ~~or No~~

FAQ requested to become effective when approved or ____________

Question Section

NEI 99-02 Guidance needing interpretation (include page and line citation):

Add BFN 1 to Table 7 of Appendix F, Generic CCF Adjustment Values. The values for BFN 1 are the same as those presented for BFN 2 and BFN 3 since all BFN plants are of the same design.

Event or circumstances requiring guidance interpretation:

Return of BFN 1 to operating status during summer of 2007

If licensee and NRC resident/region do not agree on the facts and circumstances explain

NA

Potentially relevant existing FAQ numbers

NA

Response Section

Proposed Resolution of FAQ

Add BFN 1 to Table 7 of Appendix F with plant-specific Generic CCF Adjustment Values.

If appropriate, provide proposed rewording of guidance for inclusion in next revision.

The following is proposed to be added to Appendix F, Table 7:

	EDG	**MDP Running or Alternating⁺**	**MDP Standby**	**MDP Standby**	**TDP ****	**MDP Standby**
Browns Ferry 1	1.25	1	1	1	1	3

Figure E-1

FAQ 10-05
IE04 USwC FAQ

Plant:	**Palo Verde Nuclear Generating Station**		
Date of Event:	**December 3, 2009**		
Submittal Date:	April 14, 2010		
Licensee Contact:	Del Elkinton	Tele/email:	623-393-5656 Delbert.Elkinton@aps.com
NRC Contact	Ryan Treadway	Tele/email:	623-393-3737 Ryan.Treadway@nrc.gov

Performance Indicator: IE04 – Unplanned Scrams With Complications

Site-Specific FAQ (Appendix D)? Yes

FAQ requested to become effective when approved.

QUESTION SECTION

NEI 99-02 Guidance needing interpretation (include page and line citation):

IE04 page 21 Lines 2 -10:

> "Was the scram response procedure unable to be completed without entering another EOP?"

Appendix H2.3 PWR Case Study 3, page H-14 Line 9 through H-17 line 23:

> This case study discusses a PWR event with loss of forced circulation and entry into natural circulation that was answered "NO" for question six regarding entry into EOPs.

The IE04 guidance currently excludes counting loss of forced circulation (LOFC) under the Westinghouse ES01 Emergency Operating Procedure (EOP) scheme, but requires counting the same scenario under the Combustion Engineering CEN-152 EOP scheme. The proposed resolution would add an Appendix D FAQ to also exclude counting LOFC events under the Combustion Engineering CEN-152 EOP scheme.

The Westinghouse exclusion is based on normal scram recovery and restoration of forced circulation being addressed within the single Westinghouse ES01 EOP. Transition to another EOP is not required. For the same LOFC event, the CEN-152 EOP scheme organizes the response into two EOPs, the normal scram and LOFC.

The administrative arrangement of Westinghouse ES01 for a LOFC without a cooldown using natural circulation provides no safety benefit over the arrangement of CEN-152.

Without any other complications, an LOFC event does not require counting as an unplanned scram with complications in the ES01 scheme and it should not count in the CEN-152 scheme.

Event or circumstances requiring guidance interpretation:

On December 3, 2009, Palo Verde Unit 3 experienced a loss of containment instrument air that resulted in an eventual loss of normal reactor coolant pump (RCP) seal bleed-off flow. This caused the seal bleed-off relief valve to lift to send bleed-off to the reactor drain tank (RDT). To prevent overfill of the RDT and a breach of the RDT rupture disk, control room staff elected to scram the reactor and secure all four RCPs. After completing the standard post-trip actions (SPTAs), the plant remained in mode 3 via natural circulation until forced circulation was restored after instrument air was restored in containment. A cooldown using natural circulation was NOT initiated. The safety functions were met. All rods fully inserted, the turbine tripped automatically upon scramming the reactor, class and non-class AC busses remained energized, no safety injection occurred, and main feedwater remained in service or available throughout the event. During the event, charging remained available through the pressurizer auxiliary spray line. Letdown and the ability to pump down the RDT were lost because the respective air-operated containment isolation valves shut upon loss of instrument air pressure. These losses were addressed by the use of abnormal operating procedures that do not require entry into another EOP. A contingency action from EOP standard appendices was used to manually align turbine gland seal steam. The RDT rupture disk remained intact, and the each of the RCPs' 3-stage seals operated per design without experiencing abnormal leak-off or heating.

To address the event after diagnosing the loss of instrument air inside containment, the control room staff entered the SPTA EOP. The RCPs were secured and the LOFC EOP was entered to control the plant using natural circulation until forced circulation was restored.

If licensee and NRC resident/region do not agree on the facts and circumstances explain
The NRC resident and Palo Verde are in agreement on the facts of the event and the content of NEI guidance. Both agree that after the reactor trip and manual shutdown of the RCPs, the station entered a second EOP (the LOFC EOP) to maintain heat removal via natural circulation until instrument air and forced circulation were restored.

The NRC resident and Palo Verde differ on whether the guidance provided in NEI 99-02 regarding the Westinghouse ES01 EOP scheme provides an adequate basis for a plant specific exemption that would permit a "No" answer for the question whether the scram procedure was able to be completed without entering another EOP. The NRC resident's contention is based on the purpose of the performance indicator, which is track performance related to "events or conditions that may have the potential to present additional challenges to the plant operations staff and therefore, may be more risk-significant than uncomplicated scrams" given the challenges the Operations staff faced during the December 3, 2009, Unit 3 loss of instrument air event.

Potentially relevant existing FAQ numbers
There are no relevant existing FAQs

<u>RESPONSE SECTION</u>

Proposed Resolution of FAQ
Enter a Combustion Engineering NSSS vendor specific FAQ into Appendix D of NEI 99-02 that would permit a "NO" answer in response to the question "Was the scram response procedure unable to be completed without entering another EOP?" for specific scram events that require entry into the Loss of Forced Circulation EOP. This exception would not apply to LOFC events that were initiated by a loss of offsite power or resulted in a plant cooldown using natural circulation.

To align the December 3, 2009, Palo Verde scram with the indicator as described in the IE04 guidance for Westinghouse design and EOPs, approval of this FAQ would allow the event to be counted only as an unplanned scram.

If appropriate, provide proposed rewording of guidance for inclusion in next revision.
Not applicable – Appendix D FAQ