

Enclosure 2
Meeting Summary Handouts of the
March 18, 2010 ROP Public Meeting
DATED MARCH 30, 2010

Preliminary Statistics of EDG Baseline Data Review

Base Period:	1999 – 2001 (3 years)
Total Number of EDG Related Failures (Based on System Codes: DC, DE, EK)	383
INL Identified Failures	72
Start	31
Load	31
Run	10
CDE Failures (IsMspifailure = "Y")	27
Start	22
Load	3
Run	2
CDE PUMP Failures (Based on System Codes: DC, DE, EK)	12
EDG PUMP Related Failures (IsMspifailure = "Y")	2
FOTP Failures	4
EDG FOTP Related Failures (IsMspifailure = "Y")	0
EDG FOTP Related Failures (EDG secured)	1 or 2

Preliminary Statistics of EDG Baseline Data Review

CDE Failure	Type	Description	Impact	IsMsp Failure	Monitored
1425	Fuel Injector Pump	Drawstring from spray shield lodged between fuel pump metering rod and pump housing	Speed control problem (60hz to 61.1hz variation)	Y – Start Failure	Emergency Diesel Generator / Not Monitored
1463	Fuel Injection Pump	Failure of fuel supply line from engine header to the jerk pump suction	EDG tripped 23 hours and 12 minutes into the 24 hour run	Y – Run Failure	Not Monitored / Emergency Diesel Generator
2644	Fuel Oil Booster Pump	Failure to support unloaded operation due to loss of prime causing inadequate supply to injectors	Slow down and shutdown of EDG due to air inleakage at mechanical seal	Blank	Emergency Diesel Engine / Not Monitored
12544	Portable Diesel Oil Pump	Could not be started – gas tank empty, needle valve fully closed	None	Blank	Not Monitored
12918	Engine-driven coolant pump	Seal leak	EDG declared inoperable	Blank	Not Monitored
16039	Injector Pump	Fuel Leak	EDG shutdown	Blank	Not Monitored
20236	FOTP	Day Tank overflowed 80 - 100 gallons. Lamda power supply problem	Unknown	Blank	Not Monitored
20244	FOTP	Auto stop failed. Pump stopped before overflow	EDG declared inoperable – not clear if EDG was running	Blank	Not Monitored
20582	Fuel Pump	Relay that starts DC pump failed. Shaft Pump available	None	N	Not Monitored
22325	DFOTP	Rapid Cycling during activity to recirc the day tank – not repeatable	None	Blank	Not Monitored
22568	FOTP	Would not pump. Discharge RV leak by causing pump to recirc	Unknown	Blank	Not Monitored
27924	Jacket Coolant Pump	Seal leak	EDG tripped first 20 minutes of 24 hour run	Blank	Not Monitored / Emergency Diesel Generator

No.	PI	Topic	Status	Plant/ Co.
09-06	EP01	Offsite Call Simulation	Tentative Approval	DAEC
09-09	IE03	Unplanned Power Changes	Tentative Approval	Generic
09-10	EP02	Common EOF	ON Hold for EP Task Force	Generic
10-01	NA	Withdrawal of FAQs	Tentative Approval	Generic
10-02	IE04	USwC	Introduced	Generic
10-03	IE04	USWC	New	Wolf Creek

FAQ

Plant: Duane Arnold Energy Center
Date of Event: 6/24/09
Submittal Date: 7/21/09
Licensee Contact: Mike Davis, Bob Murrell
Tel/email: 319-851-7032/ michael.davis@nexteraenergy.com
319-851-7900/ robert.murrell@nexteraenergy.com
NRC Contact: Randy Baker Tel/email: 319-851-7210

Performance Indicator: **Drill and Exercise Performance**

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, Rev. 6 page 45, lines 43 – 46:

Performance statistics from operating shift simulator training evaluations may be included in this indicator only when the scope requires classification. Classification, PAR notifications and PARs may be included in this indicator if they are performed to the point of filling out the appropriate forms and demonstrating sufficient knowledge to perform the actual notification.

NEI 99-02, Rev. 6 page 46, lines 17 – 19:

Simulation of notification to offsite agencies is allowed. It is not expected that State/local agencies be available to support all drills conducted by licensees. The drill should reasonably simulate the contact and the participants should demonstrate their ability to use the equipment.

Event or circumstances requiring guidance interpretation:

In accordance with Duane Arnold Energy Center (DAEC) procedures for making offsite notifications of emergency events, the Shift Technical Advisor (Key Communicator) fills out the notification form, gains approval from the Shift Manager (Key Decision Maker/Emergency Director), and hands the form off to the Security Shift Supervisor (not filling an NRC Participation PI key position). The Security Shift Supervisor then contacts offsite authorities using a telephone system (one call notifies all county and state authorities).

During licensed operator continuing training simulator evaluations, Security personnel are sometimes not available to participate. In these cases, the simulator instructor/evaluator role-plays as the Security Shift Supervisor. When this occurs, the instructor does not pick up the phone and simulate making a call to offsite authorities.

The NRC resident has challenged counting these as successful DEP opportunities because there is no demonstration of using the phone equipment.

NEI 99-02, Rev. 6 seems to differentiate the extent of demonstrating notification between operations simulator evaluations and drills. This is also discussed in a previous FAQ 202.

What extent of simulation is required to “demonstrate sufficient knowledge to perform actual notification”? Should “demonstration of their ability to use the equipment” be applied to operations simulator evaluations?

In the simulator evaluations in question, the simulator scenario was developed to have the instructor role-play as the Shift Security Supervisor and did not require any participant to demonstrate use of the phone if security personnel were not available. If these instances do not meet the intent for demonstrating sufficient knowledge of performing notifications and there were no errors made by the participants, should these opportunities be counted in the performance indicator as failures?

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

The NRC has concluded that the opportunities are failures due to not demonstrating the use of phone equipment.

Potentially relevant existing FAQ numbers

None

Response Section

Proposed Resolution of FAQ

During operator simulator training, personnel filling a non-key position for making a phone call to offsite agencies may not be available. In these instances where the Shift Manager (Emergency Director) and the Shift Communicator do not perform the notification phone call, it is acceptable to demonstrate the notification process up to the point of filling out the appropriate forms and providing the completed notification forms to a person role-playing as the phone-talker.

At a later time off sequence a phone talker will complete the process of using the telecommunications equipment.

Past opportunities performed by Licensees in a similar manner as the FAQ submitter will not require revision. Data will be collected using this new process going forward.

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

NEI 99-02, Rev. 6 page 45, lines 43 – 46:

Current wording is italicized, proposed additions are underlined.

Performance statistics from operating shift simulator training evaluations may be included in this indicator only when the scope requires classification. Classification, PAR notifications and PARs may be included in this indicator if they are performed to the point of filling out the appropriate forms and demonstrating sufficient knowledge to perform the actual notification. It is recognized that key control room positions may not perform the actual communication with offsite agencies as part of the notification process. Personnel filling non-key positions for contacting offsite agencies (phone-talker) may not be available during simulator training. Therefore, “demonstrating sufficient knowledge” during the simulator session includes demonstrating knowledge of the notification process and interface with persons (actual or evaluator role-playing) assigned to contact offsite agencies using equipment (phone-talker). If an evaluator role-plays the phone talker during the simulator session, a phone talker is required to complete the notification process out of sequence (e.g. notification form completed in the simulator is provided to a phone talker at a later time and the phone talker demonstrates use of the telephone equipment). Timeliness is determined by adding the time

FAQ 09-06

required to complete the notification form in the simulator to the time required by the phone talker to utilize the communications equipment out of sequence.

Plant: N/A
Date of Event: N/A
Submittal Date: October 15, 2009
Licensee Contact: Jeff Thomas, 704-382-3438, cjthomas@duke-energy.com
NRC Contact: John Thompson, 301 415-1011, john.thompson@nrc.gov

UNPLANNED POWER CHANGES PER 7,000 CRITICAL HOURS

Purpose

This indicator monitors the number of unplanned power changes (excluding scrams) that could have, under other plant conditions, challenged safety functions. It may provide leading indication of risk-significant events but is not itself risk-significant. The indicator measures the number of plant power changes for a typical year of operation at power.

Indicator Definition

The number of unplanned changes in reactor power greater than 20% of full-power, per 7,000 hours of critical operation excluding manual and automatic scrams.

Data Reporting Elements

The following data is reported for each reactor unit:

- the number of unplanned power changes, excluding scrams, during the previous quarter
- the number of hours of critical operation in the previous quarter

Calculation

The indicator is determined using the values reported for the previous 4 quarters as follows:

$$value = \frac{(total\ number\ of\ unplanned\ power\ changes\ over\ the\ previous\ 4\ qtrs)}{total\ number\ of\ hours\ critical\ during\ the\ previous\ 4\ qtrs} \times 7,000\ hrs$$

Definition of Terms

Unplanned ~~change-change~~ in reactor power, for the purposes of this indicator, is a change in reactor power that (1) ~~was-was~~ initiated less than 72 hours following the discovery of an off-normal condition that required or ~~resulted-resulted~~ in a power change of greater than 20% full power to resolve and (2) has not been excluded ~~from-from~~ counting per the guidance below. Unplanned changes in reactor power also include uncontrolled excursions of greater than 20% of full power that occur in response to changes in reactor or plant conditions and are not an expected part of a planned evolution or test.

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Clarifying Notes

The value of 7,000 hours is used because it represents one year of reactor operation at about an 80% availability factor.

If there are fewer than 2,400 critical hours in the previous four quarters the indicator value is displayed as N/A 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.

The 72 hour period between discovery of an off-normal condition ~~and the corresponding change in power level of greater than 20% of full power to resolve and the corresponding change~~ in power level is based on the typical time to ~~assess prepare for a planned power change. It includes time to assess~~ the plant condition, and prepare, review, and approve the necessary work orders, procedures, and necessary safety reviews, to effect a repair. The key element to be used in determining whether a power change should be counted as part of this indicator is the 72-hour period ~~and not the extent of the planning that is performed~~ between ~~the discovery of the condition and initiation of the power change.~~

~~recognizing the possible need for a change in power level of greater than 20% and completion of the power change. The licensee should have objective evidence to demonstrate when the possible need for the downpower was recognized such as logs documenting actions required by Technical Specifications, troubleshooting plans, meeting minutes, corrective action program entries, or similar type documentation.~~

Given the above, it is incumbent upon licensees to provide objective evidence that identifies when the off-normal condition was discovered and when the power change of more than 20% was initiated. Such objective evidence may include logs, troubleshooting plans, meeting minutes, corrective action program documents, or similar type documentation.

Examples of occurrences that would be counted against this indicator include:

- Power reductions that exceed 20% of full power and are not part of a planned and documented evolution or test. Such power changes may include those conducted in response to equipment failures or personnel errors or those conducted to perform maintenance.
- Runbacks and power oscillations greater than 20 % of full power. A power oscillation that results in an unplanned power decrease of greater than 20% followed by an unplanned power increase of 20% should be counted as two separate PI events, unless the power restoration is implemented using approved procedures. For example, an operator mistakenly opens a breaker causing a recirculation flow decrease and a decrease in power of greater than 20%. The operator, hearing an alarm, suspects it was caused by his action and closes the breaker resulting in a power increase of greater than 20%. Both transients would count since they were the result of two separate errors (or unplanned/non-proceduralized action).
- Unplanned downpowers of greater than 20% of full power for ALARA reasons

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Examples of occurrences that are not counted include the following:

- Planned power reductions (anticipated and contingency) that exceed 20% of full power and are initiated in response to an off-normal condition discovered at least 72 hours before initiation of the power change.
- Unanticipated equipment problems that are encountered and repaired during a planned power reduction greater than 20% that alone could have required a power reduction of 20% or more to repair.
- Apparent power changes that are determined to be caused by instrument problems.
- If conditions arise that would normally require unit shutdown, and an NOED is granted that allows continued operation before power is reduced greater than 20%, an unplanned power change is not reported because no actual change in power greater than 20% of full power

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occurred. However, a comment should be made that the NRC had granted an NOED during the quarter, which, if not granted, may have resulted in an unplanned power change.

- Anticipatory power reductions intended to reduce the impact of external events such as hurricanes or range fires threatening offsite power transmission lines, and power changes requested by the steam load dispatches.
- Power changes to make rod pattern adjustments
- Power changes directed by the load dispatcher under normal operating conditions due to load demand, for economic reasons, for grid stability, or for nuclear plant safety concerns.

Anticipated power changes greater than 20% in response to expected environmental problems (such as accumulation of marine debris, biological contaminants, or frazil icing) which are proceduralized but cannot be predicted greater than 72 hours in advance may not need to be counted unless they are reactive to the sudden discovery of off-normal conditions. However, unique environmental conditions which have not been previously experienced and could not have been anticipated and mitigated by procedure or plant modification, may not count, even if they are reactive. The licensee is expected to take reasonable steps to prevent intrusion of marine or other biological growth from causing power reductions. Intrusion events that can be anticipated as part of a maintenance activity or as part of a predictable cyclic behavior would normally be counted unless the down power was planned 72 hours in advance. The circumstances of each situation are different and should be identified to the NRC in a FAQ so that a determination can be made concerning whether the power change should be counted.

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Licensees should use the power indication that is used to control the plant to determine if a change of greater than 20% of full power has occurred.

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If a condition is identified that is slowly degrading and the licensee prepares plans to reduce power when the condition reaches a predefined limit, and 72 hours have elapsed since the condition was first identified, the power change does not count. If however, the condition suddenly degrades beyond the predefined limits and requires rapid response, this situation would count. If the licensee has previously identified a slowly degraded off-normal condition but has not prepared plans recognizing the potential need to reduce power when the condition reaches predefined limits, then a sudden degradation of that condition requiring rapid response would constitute a new off-normal condition and therefore, a new time of discovery.

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Off-normal conditions that begin with one or more power reductions and end with an unplanned reactor trip are counted in the unplanned reactor scram indicator only. However, if the cause of the downpower(s) and the scram are different, an unplanned power change and an unplanned scram must both be counted. For example, an unplanned power reduction is made to take the turbine generator off line while remaining critical to repair a component. However, when the generator is taken off line, vacuum drops rapidly due to a separate problem and a scram occurs. In this case, both an unplanned power change and an unplanned scram would be counted. If an off-normal condition occurs above 20% power, and the plant is shutdown by a planned reactor trip using normal operating procedures, only an unplanned power change is counted.

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.

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.

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 established 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.

Comment [MTH1]: NRC does not believe 99-02 invokes this process so I do not want to draw attention to this point.

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

Response Section

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

1 *expected to be just a phone talker who is not tasked with filling out the form. There is no intent*

2 *to track a large number of shift communicators or personnel who are just phone talkers.*

3

4 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.

5

[DRILL AND EXERCISE PERFORMANCE]

NEI 99-02 Revision 6, page 48

1 *the exercise. Thus, a licensee may choose to not include a PAR beyond the 10-mile EPZ as a*

2 *DEP PI statistic due to its ad hoc nature.*

3

4 *If a licensee discovers after the fact (greater than 15 minutes) that an event or condition had*

5 *existed which exceeded an EAL, but no emergency had been declared and the EAL is no longer*

6 *exceeded at the time of discovery, the following applies:*

7

- *If the indication of the event was not available to the operator, the event should not be*

8 *evaluated for PI purposes.*

9

- *If the indication of the event was available to the operator but not recognized, it should be*

10 *considered an unsuccessful classification opportunity.*

- 11 • *In either case described above, notification should be performed in accordance with*
- 12 NUREG-1022 and not be evaluated as a notification opportunity.
- 13
- 14 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 Sanfilippo Tel/email: 301-415-3951/nathan.sanfillipo@nrc.gov

Performance Indicator:

NA

Site-Specific FAQ (Appendix D)? No

FAQ requested to become effective when approved

Question Section

Existing Guidance on Page E-3 beginning at line 16

Withdrawal of FAQs

A licensee may withdraw a FAQ after it has been accepted by the joint ROP Working Group. Withdrawals must occur during an ROP Working Group monthly (approximately) meeting. However, the ROP Working Group should further discuss and decide if a guidance issue exists in NEI 99-02 that requires additional clarification. If additional clarification is needed then the original FAQ should be revised to become a generic FAQ.

Event or circumstances requiring guidance interpretation

The staff has expressed concern that when a licensee withdraws an FAQ, the efforts that they expend during the discussions preceding the withdrawal of the FAQ are not captured.

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

NA

Response Section

Proposed Resolution of FAQ

Recommended Change

Withdrawal of FAQs

A licensee may withdraw a FAQ after it has been accepted by the joint ROP Working Group. Withdrawals must occur during an ROP Working Group meeting. However, the ROP Working Group

FAQ 10-01

should further discuss and decide if a guidance issue exists in NEI 99-02 that requires additional clarification. If additional clarification is needed then the original FAQ should be revised to become a generic FAQ. In many cases, there are lessons learned from the resources expended by the ROP Working Group that should be captured. In those cases, the FAQ will be entered in the FAQ log as a generic FAQ. If there is disagreement between the staff and industry, both positions should be articulated in the FAQ. These withdrawn FAQs should be considered as historical and are not considered to be part of NEI 99-02. Although they do not establish precedence, they do offer insights into perspectives of both industry and NRC staff and, as such, can inform future decisions to submit an FAQ.

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

See proposed resolution

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 Sanfillipo 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

FAQ 10-02

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

Response Section

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: January , 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 then 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. Both Steam Driven Main Feedwater Pumps tripped as designed on the feedwater isolation signal and steam generator water levels were restored and maintained by auxiliary feedwater flow per design. Several days later after repairs to the MFRV controller were complete, during preparations for restart and return of the plant to power, both Steam Driven Main Feedwater Pumps required maintenance assistance to return them to service. The event was reported in the monthly performance indicator IE04 as an Unplanned Scram with out Complications.

Wolf Creek Nuclear Operating Corporation (WCNOC) believes 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, 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 NEI 99-02 Rev 5, page H-4, Paragraph H 1.5, lines 36-39 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.”

On a normal scram from power, WCGS expects to receive a feedwater isolation signal on low Tav_g coincident with P-4 and a LoLo SG 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 feedwater isolation on low Tav_g 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. The normal response to a scram from power Emergency Response procedures do not include reset of feedwater isolation signal for low Tav_g coincident with P-4, or restart of the Steam Driven Main Feedwater Pumps. Once conditions stabilize and water levels are recovered in the SGs, the Normal Shutdown to Hot Standby procedure entered from the scram procedure provides 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.

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 reactivation 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]

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 WCNOG 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.

NRC Senior Resident Inspector Position:

SRI Position Summary

The SRI disagrees with Wolf Creek and feels that April 28 trip should have been reported as a scram with complications per NEI 99-02, revision 5, pages 19, H-4, and H-5. The three main feedwater pumps could not be started without maintenance. Any of the three main feedwater pumps can be procedurally started in Mode 3. The P4/FWIS can be cleared with the pushbutton per procedures. This is consistent with page H-5, lines 20-23. The justification for the SRI's position is below:

Use of Main Feedwater Pumps with a Subcritical Reactor

The SRI agrees that none of the three pumps in the main feedwater system could be started without maintenance. Although one main feedwater valve to steam generator B could not be opened after the trip, the three remaining steam generators could be fed and would have met the minimum requirements for a heat sink. The SRI disagrees that main feedwater cannot be used with a subcritical reactor. Inspectors reviewed this practice and found that Wolf Creek can and has rolled one steam driven main feedwater pump in Mode 3, specifically on November 9, 2006. GEN 00-005, "Minimum Load to Hot Standby," allows and provides steps to use any of the three main feed pumps including the two steam driven main feedwater pumps in Mode 3. This includes response to a reactor trip. Also, Wolf Creek has a motor driven non-safety related pump in the main feedwater system. Any one of the three pumps are capable of satisfying the Emergency Operating Procedures' required flow rate of 270,000 lbm/hr. This meets the NEI 99-02 page 19 guidance to use normal or off-normal procedures to supply main feedwater and satisfy the heat sink criteria. This also meets NEI 99-02 page H-5, lines 7 to 9 to supply main feedwater and satisfy the heat sink criteria.

P4 Coincident with Lo Tave Feedwater Isolation Signal (FWIS)

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 Lo Tave. 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 Lo Tave. 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 this was consistent with NEI 99-02 page 19 and pages H-4 through H-5. Plant design and procedures allow clearing of the P4/FWIS. 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. The SRI feels that because the P4/FWIS logic can be satisfied to feed with main feed water, the April 28, 2009 reactor trip should count towards the PI.

Wolf Creek has counted scrams with complications in the past. A 'Yes' to the Main Feedwater block for Scrams with Complications due to the April 28, 2009 trip would be consistent with Wolf Creek's March 2008 reactor trip in which all condensate pumps were lost. Main feedwater was unavailable and 'Yes' was answered to the flow chart block for the March 2008 reactor trip. There is no discussion of the FWIS in Wolf Creek's NRC PI procedure.

Use of Procedure EMG FR-H1, “Response to Loss of Secondary Heat Sink”

The SRI agrees with Wolf Creek’s position that use of EMG FR-H1 would count towards the PI because of entry into another EMG per NEI 99-02 section H 1.6 (“Was the scram response procedure unable to be completed without entering another [EMG]?”). This would be a ‘yes’ in the flow chart. 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 including the steam driven main feedwater pumps in Mode 3. 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. Consistent with page 19 of NEI 99-02, Revision 5 and page H-4, lines 24 to 29, the PI monitors whether main feedwater could be used to feed the steam generators if necessary in emergency operating procedures. Therefore, the April 28, 2009 reactor trip should count as a ‘hit’ in the Scrams with Complications performance indicator.

The clearing of the Lo Lo Steam Generator signal is outside of this FAQ’s requested paragraph H1.5 clarification and if performed under EMG FR-H1, the scram would count as a PI hit anyway.

Potentially Relevant Existing FAQ Numbers:

None

RESPONSE:

Proposed Resolution of FAQ:

This event should not count against the Unplanned Scrams w/Complications PI.