WOLF CREEK
NUCLEAR OPERATING CORPORATION

Jaime H. McCoy
Vice President Engineering

January 21, 2015

ET 15-0001

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

Reference: 1) Letter ET 13-0035, dated November 21, 2013, from J. P Broschak, WCNOC, to USNRC

2) Letter dated December 4, 2014, from C. F. Lyon, USNRC, to A. C. Heflin, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Re: License Amendment Request to Revise the Fire Protection Program Related To Alternative Shutdown Capability (TAC NO. MF3112)"

Subject: Docket No. 50-482: Response to Request for Additional Information Regarding License Amendment Request to Revise the Fire Protection Program Related to Alternative Shutdown Capability

Gentlemen:

Reference 1 provided the Wolf Creek Nuclear Operating Corporation (WCNOC) application to revise the approved fire protection program as described in the Updated Safety Analysis Report (USAR) to incorporate a revised alternate shutdown methodology. Reference 2 provided a request for additional information related to the application. The Attachment provides WCNOC's response to the request for additional information. On January 14, 2015, a teleconference was held regarding question SRXB-RAI-1 and the due date for submitting this response was extended to no later than January 29, 2015.

The additional information does not expand the scope of the application and does not impact the no significant hazards consideration determination presented in Reference 1.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," a copy of this submittal is being provided to the designated Kansas State official.

A006
NRR

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4156, or Mr. Steven R. Koenig at (620) 364-4041.

Sincerely,

Jaime H. McCoy

Jaime H. McCoy

JHM/rlt

Attachment

cc: T. A. Conley (KDHE), w/a
M. L. Dapas (NRC), w/a
C. F. Lyon (NRC), w/a
N. F. O'Keefe (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Jaime H. McCoy, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Jaime McCoy
Jaime H. McCoy
Vice President Engineering

SUBSCRIBED and sworn to before me this 21st day of January, 2015.

Gayle Shepheard
Notary Public

GAYLE SHEPHEARD
Notary Public - State of Kansas
My Appt. Expires 7/24/2015

Expiration Date 7/24/2015

Response to Request for Additional Information

Reference 1 provided the Wolf Creek Nuclear Operating Corporation (WCNOC) application to revise the approved fire protection program as described in the Updated Safety Analysis Report (USAR) to incorporate a revised alternate shutdown methodology. Reference 2 provided a request for additional information related to the application. The specific Nuclear Regulatory Commission (NRC) question is provided in italics.

SRXB-RAI-1

In Section 3.7.5 of the submittal, it states that maximum operator response times were, when possible, set as less than or equal to 80 percent of the time-sensitive action required time. This section also claims that the 80 percent threshold partially accounts for instrumentation uncertainties. The effects of time-sensitive human actions are important to determining the acceptability of the presented sequences. Please provide:

a. *A list of time-sensitive actions in Evaluation SA-08-006 which were identified by procedure AL 21-017 to require between 80 percent and 100 percent of the four time-sensitive action required time to complete.*

b. *A list of conservative assumptions purposefully included in evaluation SA-08-006 to account for uncertainties in plant response.*

In addition,

c. *Please explain if the 80 percent threshold also is meant to account for delays due to operator errors. If yes, explain how the 80 percent threshold was developed. The NRC staff normally accounts for operator error by postulating credible errors for specific tasks and adding the time margin required to recover from the worst case credible error. If a default value is to be used instead of a task-specific margin, the time margin should be 100 percent (i.e., the estimated time required should be doubled; see Appendix B of NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," October 2007 (ADAMS Accession No. ML073020676), for details). If the margin of 100 percent is used, the NRC staff would expect your threshold for further action to be set at 50 percent, not 80 percent.*

Response:

a. On January 14, 2015, a clarification teleconference was held with the NRC staff regarding question SRXB-RAI-1, Item a. The NRC staff indicated that the intent of this question was to obtain a list of only the time-sensitive actions from Evaluation SA-08-006 that exceed the 80% threshold discussed in procedure AI 21-017, "Timed Fire Protection Actions Validation." The following table identifies each action that exceeds the 80% threshold.

Procedure Step	Time Sensitive Action Description	Required Time (min)	Demonstrated Time (min) (Note 1)	% of Required Time
A2	Close Steam Generator A and C Atmospheric Relief Valves from the auxiliary shutdown panel	7	6.2	88.6
A7	Control Backup Group B Pressurizer Heaters	11.5	10.2	88.7
B9	Isolate Normal Charging	14	12.4	88.6
B13	Establish Charging Flow	28	23.8	85.0
C2	Isolate Pressurizer Pilot Operated Relief Valves (PORVs)	3	2.83	94.3

Note 1 – Demonstrated times include the time necessary to don flash suit (1 min 30 sec) and close manually operated valves.

b. As noted in Sections 3.7.1 and 3.7.5 of Attachment I of Reference 1, Evaluation SA-08-006 was performed using a best-estimate RETRAN 3D model with the model initialized for nominal plant conditions and setpoints. As such, the model was developed without purposefully including conservative assumptions.

c. The 80% threshold is not intended to account for operator errors. The acceptance criterion in procedure AI 21-017 is that the action be performed on or before the required time limit. Performance of the action on or before the required time ensures the actions are bounded by Evaluation SA-08-006. If the action is performed between 80% and 100% of the required time, consideration is given to performing additional validations using other performers and/or evaluate for a degrading trend. The 80% threshold is an internal administrative limit and is not intended as the pass/fail criteria.

At WCNOC, it is assumed that reactor trip and evacuation of the control room occurs when the fire starts. This establishes time = 0 for the purpose of the evaluation and procedure. This assumption was approved by the NRC in NUREG-0881, Supplement 5, "Safety Analysis Report related to the operation of Wolf Creek Generating Station, Unit No. 1."

Procedure OFN RP-017, "Control Room Evacuation," is predominately a non-symptom based procedure. When operators enter the procedure, they perform their tasks from start to finish generally without reliance on other operators or without the need to make key decisions that would delay the completion of the procedure. In some cases, an operator may be required to wait for another operator to complete a step. For example, an operator may have to wait for a pump suction valve to be opened before starting a pump. These factors have been considered in the layout of the procedure and are evaluated in drawing E-1F9915, "Design Basis Document for Procedure OFN RP-017, Control Room Evacuation."

Regulatory Guide 1.189, Revision 2, "Fire Protection for Nuclear Power Plants," Section 5.5, "Postfire Safe-Shutdown Procedures," states the following with respect to alternative post fire safe shutdown procedures:

> Procedures for effecting safe shutdown should reflect the results and conclusions of the safe-shutdown analysis. Implementation of the procedures should not

> further degrade plant safety functions. Time-critical operations for effecting safe shutdown identified in the safe-shutdown analysis and incorporated in postfire procedures should be validated.

WCNOC meets this guidance with procedures OFN RP-017 and AI 21-017.

For the time-sensitive actions that exceed the 80% threshold listed in the response to Item a, the available margin provides reasonable assurance that the required time would not be exceeded. For Step C2, the required time is 3 minutes whereas the demonstrated time is 2.83 minutes (10 second margin). This is the lowest margin of all the time-sensitive actions. Additional operators were timed and completed the action in 1.95 minutes (63 second margin), 2.62 minutes (23 second margin) and 1.77 minutes (74 second margin). Step C2 is an immediate action step, requiring operators to complete the step from memory before obtaining the procedure. All switches are clearly labeled and emergency lighting is provided in accordance with 10 CFR 50, Appendix R. The switches are located 2 floors below the control room in the non-RCA area. Based on all documented times being less than 3 minutes, there is reasonable assurance that operators can reliably perform Step C2 in 3 minutes or less.

The goal of the fire protection program at WCNOC is to provide "reasonable assurance" that a fire will not adversely impact the ability to achieve and maintain safe shutdown in the event of a fire. Section 3.4 of Reference 1 outlines the fire protection defense-in-depth methods employed in the control room. The fire protection defense-in-depth features makes a fire that causes evacuation of the control room extremely unlikely. However, in the event this does occur, procedure OFN RP-017 will guide the operators to a safe hot standby condition. Procedure OFN RP-017, coupled with drawing E-1F9915 and procedure AI 21-017, as well as successfully timed evolutions performed by NRC Inspectors during recent Triennial Fire Protection Inspections provides reasonable assurance that, in the unlikely event procedure OFN RP-017 needs to be used, operators will be able to bring the plant to a safe hot standby condition.

SRXB-RAI-2

Please provide an explanation of the intended purpose of Figures 3 and 5 of the submittal; as presented, the differences between the plots of the individual sequences are indistinguishable.

Response: The intent of Figures 3 and 5 in Section 3.7.1.1 of Attachment I to Reference 1 is to illustrate graphically the range of transient boundary conditions spanned by all scenarios. These boundary conditions include normalized power, core outlet pressure, and core input flow and enthalpy. These are characterized across the VIPRE-01 cases as follows.

<u>Core Power</u> - All cases have a similar power response with an early reactor trip and the power decreasing quickly to decay heat values. Departure from nucleate boiling ratio (DNBR) tends to be low when the power is high, and high when the power is low.

<u>Core Outlet Pressure</u> - The pressure varies considerably amongst the cases. In some scenarios it increases to the PORV opening pressure, and in others, it decreases to approximately 1200 psia due to operator action. Treated in isolation, low pressure is typically conservative regarding DNBR. However, low pressure can also affect density gradients and improve natural circulation, which indirectly increases DNBR.

Core Inlet Flow - Core flow can be characterized by three phases. First, there is high flow with reactor coolant pumps (RCPs) running. Second, RCP trip causes a flow coast-down that spans approximately 200 seconds. And third, the system enters a natural circulation phase. The scenarios differ primarily regarding RCP trip timing, which occurs almost immediately, or at approximately 420 seconds (depending on the scenario). From a DNBR perspective, low natural circulation flow is conservative; however, low flow occurs only after reactor trip, and consequently the minimum DNBR does not accompany low flow for these scenarios.

Core Inlet Enthalpy - The core inlet enthalpy is generally between 550 and 570 Btu/lbm for the various scenarios. However, where cooldown via the steam generators is significant, the inlet enthalpy decreases as low as 500 Btu/lbm. Low inlet enthalpy implies higher DNBR.

SRXB-RAI-3

The discussion of the applicability of the Chexal-Lellouche correlation is difficult to follow in its current form. Please provide a clear explanation of the following:

a. *The parameter ranges in Table 3 of Attachment 1 extend outside the Chexal-Lellouche experimental database. Please explain what the plant conditions are in both the primary and secondary systems during which the Chexal-Lellouche correlation is applied to the RETRAN-30 calculation.*

b. *The licensee stated that the Chexal-Lellouche correlation could be used to determine when and if binding occurs in the steam generator tubes, but the evaluations in SA-08-006 do not predict significant steam accumulation in the steam generator tubes. Please clarify when and where the Chexal-Lellouche correlation is or is not used in each of the 24 scenarios.*

c. *The analysis of the submittal's sequences 1, 1 A, and 1 C show that voids develop in the core regions and then move to the steam generators, where the voids collapse in the first steam generator tube volume. The lowest void fraction in the steam generator tubes is reported as 0.0 in Table 3 of Attachment 1. Please explain whether the void fraction reported in Table 3 is a rounded value, or whether the voids collapse in the steam generators so quickly that no voids exist at the bottom of the steam generator.*

d. *Please explain the purpose of applying the Chexal-Lellouche to the primary system calculations, if no voids exist in the bottom of the steam generator tubes.*

Response:

a. The Chexal-Lellouche correlation is available throughout the model, but is actually applied only when and where two phase conditions are present. The first row of Table 3 shows a void fraction that is only zero; hence, conditions are single-phase liquid and the Chexal-Lellouche correlation is never applied within the steam generator tubes. This also implies that although the mass flux is out of range high, this is not relevant because conditions are single phase. The second row of Table 3 shows the void fraction reaching 1.0, which is above the data range. However, this implies that conditions are single-phase vapor, so the Chexal-Lellouche correlation is not applied at this point (it is applied for void fractions below 1.0). In summary, Table 3 presents a broad range of fluid conditions. Where these conditions are two phase, they fall within the correlation data range.

b. The Chexal-Lellouche correlation is referred to as a "void fraction correlation" or as a "slip correlation." Context can be given to either of these terms by visualizing an adiabatic two-phase up-flow experiment, with both vapor and liquid injected at the bottom of a transparent pipe, flowing upward, and exiting from the top. The experiment measures the flowing quality at the test inlet and exit, and these will be nearly identical (adiabatic). However, the static quality within the test rig will be different from the flowing quality. An extreme example: solely vapor can bubble up through the test rig that also contains water, if the water velocity is zero. In this case the flowing quality is 1.0, while the static quality may be 0.1. The static quality is the quality present within a volume, rather than passing through it. By various experimental means, it is relatively easy to measure static void fraction, rather than static quality, so the static void fraction is measured. This type of experiment is relatively common, and unfortunately it has become customary to shorten the term "static void fraction" to "void fraction," and to shorten "flowing quality" to "quality." In any event, given the static void fraction and the flowing quality, the velocity difference between phases is implied. In computer codes such as RETRAN, this velocity difference (slip velocity) is calculated at each junction when and where two-phase conditions are present. Under certain conditions, this velocity difference can lead to phase separation and the formation of vapor bubbles at high spots such as at the apex of steam generator tubes.

   In the RETRAN-3D scenarios, two-phase conditions are sometimes seen entering the tubes. At this location/junction the Chexal-Lellouche correlation is applied, and the velocity of each phase is calculated. The fluid then passes through the first tube volume, where all of the vapor is condensed by heat transfer. This implies that only liquid exists at the junction exiting the first volume, so the Chexal-Lellouche correlation is not applied at this junction. Subsequent downstream junctions also see only liquid (no boiling), so both (1) vapor is not present to collect at the steam generator apex, which is 12 nodes distant from the inlet, and (2) the Chexal-Lellouche correlation is not applied at these junctions. So although the Chexal-Lellouche correlation is capable of predicting the bubble formation, it is never put to use in this manner (for these scenarios).

c. Voids are present at the steam generator inlet junction. However, heat transfer is sufficient to condense the voids before they reach the subsequent junction. Hence, the void fraction is exactly zero within the steam generator. If the first volume was divided successively into smaller volumes, then eventually the first volume would contain voids. However, the voids would still disappear within the height of the current first volume.

d. The Chexal-Lellouche correlation is available regardless of fluid conditions, but is applied only when and where two phase conditions are present. The Chexal-Lellouche correlation is used to model the velocity difference between phases (when and where two phase conditions exist) including in the steam generator tubes and elsewhere.

SRXB-RAI-4

In Section 3.5 of the application, the licensee states, in part, that, "there is reasonable assurance that the fire will remain in the cabinet of origin and will not spread." This conclusion has important implications as to the acceptability of the selected sequences in the supporting analysis. If the fire were to spread beyond the cabinet of origin, the impact upon systems could be significantly different than that of a single cabinet fire with significant implications for the thermal-hydraulic reasons to the accident scenarios. Thus, validation of the submittal conclusion that a fire would not spread beyond the cabinet of origin is important. Please explain or illustrate whether or not the electrical cabinet separation requirements for adjacent cabinets, that resulted from the tests documented in NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets," April 1987 (ADAMS Accession No. ML060960351), are consistent with the actual main control room cabinets at the site (e.g., the separation between SB038 and SB037 cannot be compared to the requirements established in NUREG/CR-4527).

Response: Section 3.6.5 of Reference 1 includes a detailed discussion of the control room cabinet fire testing and its applicability to the Wolf Creek Generating Station (WCGS). The exact quote in Section 3.5 of Reference 1 is ".... there is reasonable assurance that the fire will remain in the cabinet of origin and will not spread to adjacent cabinets [emphasis added]." The important distinction here is that it is not implied that the fire will not spread, but it is implied that the fire would not spread to adjacent cabinets containing opposite train circuits. This is further elaborated in Section 3.6.5 of Reference 1 on page 20 of 64 of Attachment I, second paragraph, which states, in part "As shown in Attachments V and VI of this submittal, the configuration of the cabinets at WCGS is such that there is at least a one inch air gap and a double metal barrier between trains [emphasis added] of cabinets."

The example given in question SRXB-RAI-4 is that the separation between cabinets SB038 and SB037 cannot be compared to the requirements established in NUREG/CR-4527 "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part 1: Cabinet Effects Test." Per Sections 3.6.2, 3.6.3 and Attachment V of Reference 1, cabinets SB038 and SB037 contain Train A components and circuits. The assumption in Reference 1 is that both cabinets are consumed by the fire due to their close proximity. Physical separation exceeding 1-inch is provided between trains. This physical separation consists of a walkway between Train A cabinets SB037/SB038 and Train B cabinets SB041/SB042. Section 3.6 of Reference 1 provides a detailed analysis of the effects of a fire in one train of cabinets.

Separation between trains of cabinets and cables has been compared to the requirements of NUREG/CR-4527 in Section 3.6.5 of Reference 1. The 1-inch minimum air gap between adjacent cabinets in the test report is significantly less than the approximately 4 foot wide walkway between cabinets. Therefore, as stated in Reference 1, there is reasonable assurance that a credible fire will not affect both trains of equipment.

SRXB-RAI-5

Multiple spurious auctions are important in the progression of accident sequences. Please provide the following:

a. A discussion regarding how single or multiple spurious actuations have been considered in each of the accident scenarios, consistent with the requirements in Regulatory Guide 1.189, Revision 2, "Fire Protection for Nuclear Power Plants," October 2009 (ADAMS Accession No. ML092580550), Sections 5.4.1 and 5.4.4, which specify that such actuations should be considered after control has been transferred from the control room to the alternative or dedicated shutdown system and after control of the plant has been achieved.

b. A discussion regarding the consideration of spurious actuations that could defeat the alternate safe shutdown system (e.g., spurious actuations that could negate the successful isolation of the main feedwater system or the chemical injection system). This information will be used by the NRC staff to validate the input assumption used in the supporting analysis.

Response:

a. Multiple spurious actuations occurring after control of the plant has been achieved have been considered. Appendix 3, "Control Room Multiple Spurious Operation (MSO) Review) in drawing E-1F9915 provides an evaluation of multiple spurious operations based on the guidance in Appendix G of NEI 00-01, Revision 3, "Guidance for Post Fire Safe Shutdown Circuit Analysis." Drawing E-1F9915 Appendix 3 is provided in the enclosure to Reference 1.

   The actions taken in procedure OFN RP-017 line up Train B equipment needed to achieve hot standby and prevent spurious operation of equipment on either train that could affect safe shutdown. Many of the actions are precautionary in order to prevent spurious operation of valves and/or pumps. Other actions are taken to line up the Train B on-site power supply and place equipment in the proper lineup to achieve hot standby from outside the control room. Equipment whose spurious operation or mal-operation could adversely affect the ability to achieve hot standby from outside the control room is isolated from the effects of a control room fire, thereby preventing spurious operation after control is achieved.

b. As detailed in drawing E-1F9915, and Section 3.6 of Reference 1, spurious actuations that could defeat the alternate safe shutdown system following transfer of control to the alternative and dedicated shutdown system are not credible. Table 7.1 in drawing E-1F9915 provides a detailed step-by-step evaluation of procedure OFN RP-017 and shows the basis for each step and how the step contributes to the successful performance of procedure OFN RP-017. As discussed in the response to question SRXB-RAI-5a above, the actions taken in procedure OFN RP-017 are taken to either line up Train B equipment for hot standby or to prevent spurious operation that could affect safe shutdown. Following transfer of control to the alternative and dedicated shutdown system, spurious operation of equipment required for hot standby or whose spurious operation could affect safe shutdown is not credible because this equipment is isolated from the effects of a control room fire.

SRXB-RAI-6

Please explain how fuel thermal conductivity degradation is accounted for in the analysis.

Response: Fuel thermal conductivity degradation (TCD) is not accounted for in evaluation. To assess the effect of TCD, two scenarios from Evaluation SA-08-006 and Calculation WCNOC-CP-003, "VIPRE-01 MDNBR Analyses of Control Room Fire Scenarios," were re-evaluated in Calculation WCNOC-CP-004, "Control Room Fire Analysis for Power Shape and Thermal Conductivity Degradation RAIs." During steady reactor operation, the temperature difference between fuel surface and centerline is inversely proportional to conductivity. Hence, low (degraded) conductivity implies higher fuel temperature and higher fuel energy content. Following reactor trip, this implies greater energy deposition into the coolant as the reactor power coasts down. This delays feedwater isolation on low Tavg, which in turn produces higher steam generator water levels. In one re-evaluated scenario, the steam generator water level goes off-scale high for approximately one hour. As expected, TCD causes higher fuel temperatures; however, the elevated temperatures arise during steady-state operation, and are not specific to the alternative shutdown capability required by the fire protection program (the peak centerline temperature increases by only 1.6 °F after the scenarios begin). The effect on other important plant parameters is minimal.

Evaluation SA-08-006 establishes the following acceptance criteria.

1. The reactor core remains cooled and no core damage is anticipated. This is defined by a hot leg temperature below 630 °F and natural circulation being maintained.

2. No pressure vessel limits are exceeded.

3. The reactor reaches a stable steady-state condition representing safe shutdown.

These acceptance criteria are met for the evaluated scenarios.

SRXB-RAI-7

Please justify the core axial power shape that is used in the analysis.

Response: Various axial power shapes were considered in the minimum DNBR evaluation performed in Calculation WCNOC-CP-003. Results from the most conservative axial power shape (lowest minimum DNBR) are presented in Reference 1. Regarding overall plant response, Evaluation SA-08-006 considers a single middle-peaked zero axial offset power shape. A more top-peaked power will tend to reduce natural circulation, and hence is conservative. To assess the effect of a top-peaked axial power, two scenarios from Evaluation SA-08-006 were re-evaluated in Calculation WCNOC-CP-004. The results show that the axial power shape has very little effect on important plant parameters.

Evaluation SA-08-006 establishes the following acceptance criteria.

1. The reactor core remains cooled and no core damage is anticipated. This is defined by a hot leg temperature below 630 °F and natural circulation being maintained.

2. No pressure vessel limits are exceeded.

3. The reactor reaches a stable steady-state condition representing safe shutdown.

These acceptance criteria are met for the evaluated scenarios.

SRXB-RAI-8

Please explain if loss of feedwater without offsite power may cause the pressurizer to overfill.

Response: Evaluation SA-08-006, Revision 3, analyzes 24 possible control room fire scenarios. Ten (10) scenarios analyze a loss-of-offsite power coincident with other failures. In these scenarios, feedwater is modeled by allowing the automatic feed water isolation signal to operate and stop feedwater flow on reactor trip with low Tavg which occurs within 15 seconds following reactor trip. Auxiliary feedwater is lined up to one steam generator within 15 minutes following reactor trip. In the cases evaluated no pressurizer overfill was observed.

SRXB-RAI-9

Please list the operator actions that will be taken in the control room prior to evacuation due to a fire, and identify how these operator actions are incorporated into the analysis used in support of this request.

Response: The only operator actions taken in the control room prior to evacuation are tripping the reactor and depressing the main steam isolation valve (MSIV) "all-close" switches. As per the WCNOC licensing basis, reactor trip is the only credited operator action taken in the control room prior to evacuation. Follow-up actions are taken outside the control room in accordance with procedure OFN RP-017 to ensure the MSIVs are closed. The analysis assumes reactor trip occurs at time = 0 and the MSIVs are closed at time = 3 minutes.

FP-RAI-01

Table 7.1 within E-1F9915, "Design Control Document For OFN RP-017, Control Room Evacuation," provides a detailed evaluation for each step in OFN RP-017. For the required actions from outside the control room, OFN RP-017 does not specify whether the operator manual actions have been evaluated for feasibility and reliability, e.g., per NUREG-1852 "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire (NUREG-1852)," October 2007 (ADAMS Accession No. ML073020676).

Provide a discussion of basis for the feasibility and reliability of operator manual actions performed outside of the control room.

Response: A formal evaluation of the operator manual actions specified in procedure OFN RP-017 has not been performed. As noted below, WCNOC's review of the regulatory guidance documents indicates that operator manual actions used as part of a Section III.G.3 area strategy, are not required to be evaluated for feasibility and reliability. However, as per the specific RAI question, provided below is a discussion of the feasibility and reliability of the operator manual actions taken in procedure OFN RP-017.

The WCNOC Fire Protection licensing basis is discussed in Section 3.3 of Reference 1. Appendix 9.5E of the USAR provides a comparison between the requirements of Section III of Appendix R and the WCNOC position. WCNOC had no exceptions to the requirements of Section III.G.3 and Section III.L of Appendix R regarding alternative/dedicated shutdown. The control room (Fire Area C-27) is the only Section III.G.3 compliance area at WCGS.

10 CFR 50, Appendix R, Section III.L.3, states the following:

> The shutdown capability for specific fire areas may be unique for each such area, or it may be one unique combination of systems for all such areas. In either case, the alternative shutdown capability shall be independent of the specific fire area(s) and shall accommodate postfire conditions where offsite power is available and where offsite power is not available for 72 hours. Procedures shall be in effect to implement this capability.

Enclosure 2 to Generic Letter 86-10, "Implementation of Fire Protection Requirements," provides several questions and answers related to Appendix R. Question 5.2.4 states the following:

> Do any NRC Staff guidance documents exist relative to the extent, form, nature, etc. of Appendix R post-fire operating procedures?

The response to this question is, in part, as follows:

> No. Other than the criteria of Section III.L, no specific post-fire shutdown procedure guidance has been developed. ...

WCNOC complies with the alternative/dedicated shutdown procedure requirement of Appendix R, Section III.L by providing post-fire safe shutdown operating procedures for hot standby and cold shutdown in the event of a control room fire. WCNOC has no regulatory commitments to perform feasibility and reliability evaluations for operator manual actions associated with alternative/dedicated shutdown. WCNOC performs periodic operator timing of the time sensitive manual actions per procedure AI 21-017 to ensure operators continue to meet the required response times documented in drawing E-1F9915 and evaluated in Evaluation SA-08-006.

The Abstract for NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," states the following:

> This report provides criteria and associated technical bases for evaluating the feasibility and reliability of postfire operator manual actions implemented in nuclear power plants. The U.S. Nuclear Regulatory Commission (NRC) developed this report as a reference guide for agency staff who evaluate the acceptability of manual actions, submitted by licensees as exemption requests from the requirements of Paragraph III.G.2 of Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to Title10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the *Code of Federal Regulations* (10 CFR Part 50), as a means of achieving and maintaining hot shutdown conditions during and after fire events. The staff may use this information in the review of *future* postfire operator manual actions to determine if the feasibility and reliability of the operator manual action were adequately evaluated.

Based on the Abstract, NUREG-1852 is intended to provide guidance for evaluating operator manual actions that are credited for compliance with Section III.G.2 of Appendix R, not Section III.G.3. Operator manual actions are not explicitly stated in Section III.G.2 as one of the approved methods of protecting safe shutdown capability where redundant trains of equipment are located in the same fire area. Manual actions, if used as part of a Section III.G.2 strategy, are required to be evaluated for feasibility and reliability and submitted to the NRC for an exemption/deviation as required by the plants' operating license.

SECY 03-0100, "Rulemaking Plan on Post-Fire Operator Manual Actions," dated June 17, 2003 was written to obtain Commission approval to proceed with a rulemaking related to operator manual actions. Throughout the SECY letter, reference is made to Appendix R, Section III.G.2, not Section III.G.3 or Section III.L. It is clear from reading the SECY letter that the concern being addressed by the proposed rulemaking was the use of operator manual actions to achieve compliance with Section III.G.2 separation requirements. Use of operator manual actions for Section III.G.3 areas has always been allowed by Appendix R, Section III.L.3.

Subsequent to the issuance of SECY 03-0100, the NRC withdrew the proposed rulemaking plan and, on June 30, 2006 issued Regulatory Issue Summary (RIS) 2006-10, "Regulatory Expectations with Appendix R Paragraph III.G.2 Operator Manual Actions." As indicated by the title, RIS 2006-10 specifically applies to operator manual actions credited for III.G.2 fire areas. NUREG-1852 was developed to assist the NRC Staff and licensees in evaluating the feasibility and reliability of operator manual actions credited for Section III.G.2 areas per the guidance in RIS 2006-10.

Each criterion in NUREG-1852 is discussed below with respect to applicability to control room fire.

Analysis Showing Adequate Time Available to Perform the Actions (To Address Feasibility)

Drawing E-1F9915 identifies the maximum time allowed to complete time-sensitive manual actions following a control room fire. The times listed are in part, based on a thermal hydraulic analysis documented in Evaluation SA-08-006, Revision 3. The baseline times used to develop Evaluation SA-08-006 were based on actual operator response times with margin added to ensure the actions can feasibly be completed within the time limitations of Evaluation SA-08-006.

Analysis Showing Adequate Time Available to Ensure Reliability

From NUREG-1852, for a feasible action to be performed reliably, it should be shown that there is adequate time available to account for uncertainties not only in estimates of the time available, but also in estimates of how long it takes to diagnose and execute the operator manual actions (e.g., as based, at least in part, on a plant demonstration of the action under nonfire conditions). Procedure OFN RP-017, once entered, requires minimal if any diagnosis. Operators perform actions from start to finish without the need to diagnose plant conditions. The operator at the Auxiliary Shutdown Panel (ASP) monitors plant conditions. Time-sensitive actions performed by operators outside the ASP room require minimal communication with the operator at the ASP.

Environmental Factors

From NUREG-1852, environmental factors are those factors that could negatively impact the ability to perform the manual actions, including radiation, lighting, temperature, humidity (caused, for instance, by water from sprinkler operation), smoke, toxic gases, and noise. The only actions performed in the control room prior to evacuation are reactor trip and isolation of the MSIVs, if permitted by the fire. Remaining manual actions performed per procedure OFN RP-017 are taken outside the fire affected area. Smoke and products of combustion are not expected to be a factor when completing the actions. Manual actions are taken in areas where radiation levels are low. Emergency lighting in accordance with 10 CFR 50, Appendix R, is provided where manual actions are taken and access and egress thereto. Therefore, environmental conditions are not expected to play a significant role in the reliability of completing manual actions.

Equipment Functionality and Accessibility

From NUREG-1852, this criterion addresses the need to ensure that the equipment that is necessary to enable implementation of an operator manual action to achieve and maintain postfire hot shutdown is accessible, available, and not damaged or otherwise adversely affected by the fire and its effects (such as heat, smoke, water, combustible products, spurious actuation). As discussed in drawing E-1F9915, equipment required to achieve and maintain safe shutdown following a control room fire is isolated from the effects of a fire. Equipment operated per procedure OFN RP-017 is accessible as demonstrated by the periodic performance of procedure AI 21-017.

Available Indications

As stated earlier, most of the steps in procedure OFN RP-017 are performed without reliance on diagnostic indication. The operator at the ASP uses diagnostic indication to maintain the plant within required parameters and to bring the plant to cold shutdown. Indication used by the operator at the ASP is isolated from the control room and can be relied on during a control room fire.

Communications

Most of the operator actions taken following a control room fire are performed without the need for communication. Where necessary, portable radios are utilized for communication. The radio equipment room is located in a separate fire area and is unaffected by a control room fire. A fire in the control room will not affect communication.

Portable Equipment

Where portable equipment such as fuse pullers and special tools are required, the equipment is located in the emergency lockers and is retrieved by the applicable operator as one of the initial steps in the procedure. Equipment lockers are inventoried on a periodic basis to ensure equipment is available and lockers are sealed with breakaway locks to prevent unauthorized entry.

Personal Protection Equipment (PPE)

Most of the actions in procedure OFN RP-017 require only the usual PPE. Special PPE, where required, is located in designated emergency equipment lockers. Where actions require special PPE, the time to don the PPE has been considered.

Procedures and Training

Procedure OFN RP-017 covers hot standby from outside the control room following a control room fire. Procedure OFN RP-017A, "Hot Standby to Cold Shutdown From Outside the Control Room Due to a Fire," covers cold shutdown from outside the control room following a fire.

Staffing

Staffing levels are such that an adequate number of operators are available to perform OFN RP-017 actions. Personnel utilized for OFN RP-017 actions are independent of the minimum fire brigade staffing levels and have no other collateral duties that would preclude or delay the performance of procedure OFN RP-017 actions.

Demonstrations

Demonstrations are performed on a periodic basis per procedure AI 21-017. Time-sensitive actions are required to be completed on or prior to the required time. Where time to perform the action exceeds 80% of the required time, actions are taken to determine the reason and adjustments made as necessary. This could include additional training or reordering the procedure steps to increase the time margin.

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Drawing E-1F9915, Section 2.2, "Assumptions," item 2.2.3 states:

> *Prior to transfer of control to the Auxiliary Shutdown System only a single spurious actuation is assumed to occur at a time, except in the case of two redundant valves in a high/low pressure interface line. All potential spurious actuations are mitigated/prevented using OFN RP-017 but timing is based on the spurious actuations occurring one at a time, or two at a time in the case of high/low pressure interface lines.*

Regulatory Guide 1.189 "Fire Protection for Nuclear Power Plants," Revision 2, Section 5.4.4 states in part:

> *After control of the plant is achieved by the alternative or dedicated shutdown system, single or multiple spurious actuations that could occur in the fire-affected area should be considered, in accordance with the plant's approved FPP.*

Please justify the reasoning for assuming that after control of the plant is achieved from the alternative location that the timing of spurious actuations should be based on the spurious actuations occurring one at a time for non-high/low pressure interface lines.

Response: The fire protection program assumes that the timing of spurious actuations is based on single or multiple spurious operations occurring after control of the plant is achieved from the alternative location. Prior to transfer of control to the alternative and dedicated shutdown system, only a single spurious operation is assumed. This is consistent with RG 1.189, Section 5.4.4.

To eliminate confusion, WCNOC proposes to revise drawing E-1F9915 Assumption 2.2.3 as follows:

> Before transfer of control is achieved by the alternative and dedicated shutdown system only a single spurious actuation is assumed to occur, except in the case of two redundant valves in a high/low pressure interface line. All potential spurious actuations are mitigated or prevented using procedure OFN RP-017 but timing is based on one spurious actuation occurring prior to transfer of control to the alternative and dedicated shutdown system, or two spurious actuations in the case of high/low pressure interface lines.

A detailed analysis of potential multiple spurious operations that could occur following a control room fire and following transfer of control to the alternative and dedicated shutdown system is provided in Appendix 3 of drawing E-1F9915, included in the enclosure to Reference 1.

References:

1. WCNOC letter ET 13-0035, "License Amendment Request (LAR) for Revision to the Wolf Creek Generating Station Fire Protection Program Related to Alternative Shutdown Capability," November 21, 2013. ADAMS Accession No. ML13331A728.

2. Letter from C. F. Lyon, USNRC, to A. C. Heflin, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Re: License Amendment Request to Revise the Fire Protection Program Related To Alternative Shutdown Capability (TAC NO. MF3112)," December 4, 2014. ADAMS Accession No. ML14323A574.