WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President Engineering

October 13, 2010

ET 10-0028

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

Reference: 1) Letter ET 10-0014, dated April 13, 2010, from T. J. Garrett, WCNOC to USNRC

2) Letter dated August 18, 2010, from B. K. Singal, USNRC, to M. W. Sunseri, WCNOC, "Wolf Creek Generating Station - Request for Additional Information Regarding License Amendment Request to Revise Technical Specification Table 3.3.2, "Engineered Safety Feature Actuation System Instrumentation" (TAC NO. ME3762)"

Subject: Docket No. 50-482: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"

Gentlemen:

Reference 1 provided Wolf Creek Nuclear Operating Corporation's (WCNOC) application to revise Technical Specification (TS) Table 3.3.2-1, Function 8.a., of TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." Reference 2 provided a request for additional information related to the application. Attachment I provides a response to the request for additional information.

The response to the request for additional information does not expand the scope of the application as originally noticed, and does not impact the conclusions of the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the Federal Register (75 FR 33844).

A001
NRR

In accordance with 10 CFR 50.91, a copy of this submittal is being provided to the designated Kansas State official.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Richard D. Flannigan at (620) 364-4117.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachment: Response to Request for Additional Information

cc: E. E. Collins (NRC), w/a
T. A. Conley (KDHE), w/a
G. B. Miller (NRC), w/a
B. K. Singal (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, 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 ______________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 13th day of October, 2010.

Gayle Shepheard
Notary Public

GAYLE SHEPHEARD
Notary Public - State of Kansas
My Appt. Expires 7/24/2011

Expiration Date 7/24/2011

Response to Request for Additional Information

Reference 1 provided Wolf Creek Nuclear Operating Corporation's (WCNOC) application to revise Technical Specification (TS) Table 3.3.2-1, Function 8.a., of TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." Reference 2 provided a request for additional information related to the application. The specific NRC question is provided in italics.

1. *The Licensee Event Report (LER), 2009-009-01 (ADAMS Accession No. ML100890421), referenced within your LAR indicates that the implementation of the bypass to defeat the feedwater isolation on low Tavg coincident with P-4 interlock function is accomplished via plant procedure SYS SB-122, "EnablinglDisabling P-4/Lo-Tavg FWIS," through the use of the temporary installation of jumpers. Within your LAR and the referenced LER, it is indicated that the installation or removal of such jumpers is required when performing procedures GEN 00-006, "Hot Standby to Cold Shutdown," STS AE-201, "Feedwater Chemical Injection Inservice Valve Test," and GEN 00-002, "Cold Shutdown to Hot Standby."*

   Further, the referenced precedent in Section 4.2 of your LAR describes the installation of a new bypass switch at Callaway Plant, Unit 1, to accomplish the feedwater isolation function on P-4 prior to entering Mode 2.

   For each of the methods called for within the various plant procedures, which require the installation of jumpers to defeat the feedwater isolation signal on low Tavg coincident with P-4 interlock, please describe how plant operators are made aware that such protection features are being bypassed (or otherwise deliberately rendered inoperative), and indicate the approximate collective frequency (i.e., inclusive of all applicable operating or maintenance procedures) for which such bypasses are required to be implemented. Please describe how the installation of these jumpers addresses Institute of Electrical and Electronics Engineers (IEEE)-279-1971, "Criteria for Nuclear Power Plant Protection Systems," regarding indication of bypasses, access to means of bypasses, and independence between safety and non-safety functions, and NRC Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

Response: Bypassing the feedwater isolation function on low T_{avg} coincident with P-4 is typically only performed during refueling outages in MODE 3 (descending) and reinstated prior to entry into MODE 2 (ascending) to prevent an undesirable ESFAS actuation during plant cooldown and during outage testing. Procedures that require bypassing the feedwater isolation function on low T_{avg} coincident with P-4 indicate that the jumpers are to be installed per procedure SYS SB-122, "Enabling/Disabling P-4/Lo T_{avg} FWIS." Procedure SYS SB-122 provides the instructions for installation of jumpers for bypassing the feedwater isolation function on low T_{avg} coincident with P-4. Step 5.2 of the procedure requires notification to the Shift Manager. Procedure AP 21F-001, "Equipment Out-of-Service Control," provides a mechanism for tracking the status of inoperable or degraded systems, structures, or components. In accordance with this procedure, bypassing the feedwater isolation function on low T_{avg} coincident with P-4, would require the entering the out-of-service equipment into the Equipment Out-of-Service Log by the Control Room staff.

USAR Section 7.5.2.2 specifies that the system of bypass indication is designed to satisfy the requirements of IEEE Standard 279-1971 and Regulatory Guide 1.47, Revision 0, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." Regulatory Guide 1.47, Regulatory Position C.3, indicates that automatic indication should be provided in the control room for each bypass or deliberately induced inoperable status that meets the following conditions:

a. Renders inoperable any redundant portion of the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the systems it actuates to perform their safety-related functions;

b. Is expected to occur more frequently than once per year; and

c. Is expected to occur when the affected system is normally required to be operable.

As discussed in Reference 1, Feedwater isolation on low Tavg coincident with P-4 is not credited in any USAR Chapter 15 analyses and therefore does not provide a protective function. Automatic indication of bypassing the feedwater isolation function on low T_{avg} coincident with P-4 is not required since the subject circuitry does not provide a required safety function and the frequency for bypassing the function is expected to occur less frequently than once per year.

2. *Technical Specification Task Force (TSTF)-444, Revision 1, "ESFAS [Engineered Safety Features Actuation System] Interlock P-4, P-11 & P-12 LCO [Limiting Condition for Operation] Actions and Surveillance Requirements Revisions," previously evaluated ESFAS interlock function "P-4" (Reactor Trip). The evaluation noted that even though the permissive "P-4" may not be directly credited in the safety analyses, some safety analyses may incorporate into their model selected permissive functions. The TSTF evaluation cited the turbine trip is assumed to occur on a reactor trip, typically in non-loss-of-coolant accident events. If the trip function is expected to occur and incorporated into the models, then the results may affect the limiting transient for certain events.*

   The licensee states in the LAR that none of the Updated Safety Analysis Report (USAR) Chapter 15 accident analyses credit the turbine trip from reactor trip (P-4 interlock function) for accident mitigation. However, one of the statements in the TS Bases the licensee proposes to delete states, "Only the turbine trip function is explicitly assumed since it is an immediate consequence of the reactor trip function."

   Please confirm that there is not a limiting transient in the safety analyses model that incorporates the P-4 permissive function by assuming a turbine trip.

Response: The proposed amendment deletes the TS requirement for the turbine trip on reactor trip function in MODE 3 based on the function not being credited in the accident analysis to mitigate the consequence of an accident. A review of the USAR accident analyses determined that there are limiting transients (non loss-of-coolant accident (LOCA) transients, resulting during operation in MODES 1 and 2) that use the P-4 function of providing a turbine trip after a reactor trip. The proposed change does not impact the functionality of P-4 in MODES 1 and 2. Other safety analyses limiting transients either do not rely solely on the P-4 turbine trip or do not model the turbine trip on reactor trip function as discussed below.

2.1 LOCA Long Term (LT)/Short Term (ST) Mass & Energy (M&E) (Containment Integrity)

The USAR Chapter 6.2 mass and energy release analyses for postulated loss-of-coolant accidents analyses assume that the turbine trips with the loss of offsite power (LOOP), which is assumed to occur with reactor trip. Modeling turbine trip at this time ensures that no additional heat removal from the reactor coolant system (RCS) primary side through the steam generators will occur, other than from the steam generator safety valves, and forces the break to be the primary means of heat removal for the RCS.

The intent of the long term LOCA M&E releases analysis is to maximize the mass and energy release available to containment through assuming the limiting initial conditions. None of these conditions, models or methodology used becomes more limiting by implementing the P-4 technical specification change. The long term LOCA M&E releases used for containment pressurization assume that fuel geometry is unaffected by the forces generated by the LOCA event. Implementation of the P-4 permissive function will have no effect on the analysis since the LOCA forces and fuel assembly criteria continue to be met. Therefore, the current design basis analysis of record documented in the USAR would remain valid.

The short-term LOCA-related mass and energy releases are used as input to the subcompartment analyses, which are performed to ensure that the walls of a subcompartment can maintain their structural integrity. These analyses are performed to ensure that the walls in the immediate proximity of the break location can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) that accompanies a high energy line pipe rupture (LOCA) within the region (subcompartment). Due to the short duration of the event, Emergency Core Cooling System (ECCS) does not actuate; as such, the effect from the P-4 interlock functions does not affect the LOCA short-term mass and energy release analyses. Therefore, the current design basis short-term LOCA mass and energy releases analysis documented in the USAR remains valid.

2.2 Main Steam Line Break (MSLB) Inside Containment(IC)/Outside Containment (OC) M&E – Dose Steam Release (Containment Integrity)

The turbine trip coincident with reactor trip assumption is modeled in the steamline break mass and energy release analysis. However, it is modeled to conservatively maximize the steam release out the break (rather than to the turbine). As such, the turbine trip on reactor trip is not credited in the analysis as a mitigating function. It is a penalty, not a benefit with respect to main steamline break mass and energy releases. If the turbine is on or is not isolated, it might provide a diversionary path for some of the mass release through a break (steamline break mass/energy release analyses) or through relief or dump valves to the atmosphere (steam release for dose).

2.3 Transient Analysis

The Chapter 15 non-LOCA safety analyses performed in MODES 1 and 2 typically model the turbine trip on reactor trip function; however, the proposed Technical Specification change does not impact the functionality of P-4 in these Modes. The only Chapter 15 event that is explicitly analyzed in MODES 3 through 6 is the Boron Dilution event, which does not credit the P-4 permissive, feedwater isolation, or turbine trip. The Hot Zero Power Steamline Break (HZP SLB) event was also evaluated in MODE 3 and determined to be bounded by the current limiting HZP SLB scenario initiated in MODE 2 (see response to Question 4).

The main purpose of the turbine trip on reactor trip function is to prevent against an excessive RCS cooldown. In MODE 3, the turbine is in a tripped state with the turbine air-oil system depressurized. Since the turbine is no longer being driven by steam load, it is connected to the turning gear to prevent warping of the turbine shaft. In this state (MODE 3), the potential RCS cooldown that would result by not receiving a turbine trip on reactor trip is substantially less severe than the cooldown that could occur at full power (MODE 1). In MODE 3, since the turbine is being turned on the turning gear, the steam demand is essentially zero with minimal steam being supplied for warmup in preparation to enter MODE 2. Therefore, the cooldown is based upon the flow area of the turbine admission valve. Since the steam generators have integral flow restrictors, any steam flow path (e.g., pipe rupture, stuck open / inadvertently open steam system valve, etc.), regardless of location, would have the same effect on the RCS as a break corresponding to the throat area of the restrictors. Therefore, the potential RCS cooldown caused by the turbine is bounded by the cooldown caused by the MODE 2 HZP SLB event.

The function also provides turbine overspeed protection in response to the sudden loss of steam flow and reduction in turbine inlet pressure. Assuming the turbine trip signal was not received on a reactor trip, the turbine would still be tripped by one of the direct turbine trip signals which guard against overspeed. For most of the Chapter 15 events, a delayed turbine trip signal will have no impact on the analysis results. The only analyses that are sensitive to the timing of a turbine trip following reactor trip are overpressure events, in which a delayed turbine trip would be a benefit since the turbine would provide an additional release path until the steam flow was reduced. This additional release could lower the peak pressures obtained during the event. Therefore, it can be concluded that this function is not required to obtain acceptable results for the Chapter 15 analyses.

2.4 LOCA Analysis – Small Break and Large Break

The USAR Chapter 15 Appendix K small break and large break loss-of-coolant accident (SBLOCA and LBLOCA) do not credit the reactor trip P-4 interlock turbine trip function. The pending WCGS specific Best Estimate LBLOCA (ASTRUM) analysis methodology does not credit the reactor trip P-4 interlock turbine trip function.

2.5 LOCA Long Term Cooling

The post-LOCA methodology only takes into consideration the primary side of the steam supply system. No modeling is contained in the post-LOCA methodology pertaining to the turbine. Therefore, there is no impact on the post-LOCA methodology pertaining to the P-4 permissive function. Since there is no impact on the post-LOCA methodology due to the consideration of the P-4 permissive function, there would be no change in the conclusions of the current licensing basis post-LOCA results regardless of the plant operating mode when the LOCA occurs.

2.6 LOCA Forces

LOCA hydraulic forces are generated as inputs to the analyses of the RCS components. These analyses of the components are performed in order to comply with 10 CFR 50 Appendix A, General Design Criteria 4 – Environmental and Dynamic Effects Design Bases.

LOCA forces analyses are performed to support different system and component qualification. Reactor vessel LOCA forces analyses support qualification of the reactor vessel, reactor vessel supports, and reactor vessel internals, including fuel qualification. Reactor coolant loop LOCA forces analyses support qualification of the reactor coolant loop piping and associated piping supports. Steam generator LOCA forces analyses support qualification of the steam generators. These analyses are performed using different models with a focus on the component of interest.

LOCA forces are driven by the rarefaction wave that travels through the RCS at the RCS fluid's local speed of sound. Due to the speed at which the rarefaction wave travels through the system, the transient time is relatively short, on the order of 500 milliseconds to 1 second, which captures the dampening of the transient with peak forces typically occurring in the first 200 milliseconds. As a consequence of this relatively short transient time, Engineered Safety Feature Actuation System (ESFAS) instrumentation and the mitigating effects of equipment that may be actuated as a result of a LOCA are not modeled in the LOCA forces analyses. Therefore, the limiting LOCA forces transient does not incorporate the P-4 permissive function since a turbine trip is not modeled.

3. *Chapter 15 of the USAR documents the results of the analyses for the limiting cases of all the anticipated operational occurrences (AOOs) and accidents. The licensee indicates that the P-4 interlock functions are not credited in the USAR Chapter 15 analyses for justification of the TS removal. The licensee clarifies that the Chapter 15 cases, other than (1) the control rod withdrawal event from subcritical and (2) the boron dilution event, are initiated from Modes 1 and 2 plant conditions.*

   Although the initial plant conditions, such as power levels, reactor coolant system, and steam generator (SG) temperatures and pressures in Modes 3 through 6, may be less limiting than the Modes 1 and 2 conditions, the reactor trip and engineered safety feature (ESF) actuation functions may not be included in the TS for Modes 3 through 6 and, therefore, may not be credited in the safety analysis. The capabilities of the ESF may be reduced for accident mitigation for events initiating from Modes 3 through 6. Single failure considerations in the systems affected by the proposed TS changes may identify a worst single failure that is different from that assumed in the Chapter 15 analysis.

   Please provide a discussion or the results of an analysis to demonstrate that with the proposed blockage of the three P-4 interlock functions in Mode 3, the Chapter 15 analyses for all AOOs and accidents bounds the results of the corresponding events initiating from Modes 3 through 6, with a combination of the worst single failure consideration.

Response: A review of the USAR accident analyses determined that, as documented below, with the proposed modification to the TS requirements for the P-4 interlock functions in MODE 3 (deletion of the turbine trip and feedwater isolation functions), the Chapter 15 accident analyses remains valid and would bound the results of the corresponding events initiating from MODES 3 through 6 with a combination of the worst single failure consideration.

3.1 LOCA LT/ST M&E (Containment Integrity)

Relative to Chapter 6 LOCA mass and energy release methodology, the intention is to model the turbine trip at a time to isolate the steam generators as early as possible. This traps the energy from the secondary side for the post-reflood portion of the transient. The turbine trip is not credited as a mitigating feature in the Chapter 6 LOCA mass and energy release analysis. The turbine trip is modeled as a penalty, not a benefit. The LOCA M&E release analyses are performed considering MODE 1 operation, at power normal operating pressure and temperature, to maximize all stored and generated mass and energy release available. In addition, MODES 3 through 6 conditions are less limiting relative to mass and energy generation.

3.2 MSLB IC/OC M&E – Dose Steam Release (Containment Integrity)

As stated in the response to Question 2, the turbine trip is not credited as a mitigating feature in the steamline break mass and energy release or steam release for dose analyses. Therefore the proposed elimination of the turbine trip function does not adversely impact the analyses. This is true for all modes of operation.

The feedwater isolation on low T_{avg} coincident with reactor trip appears to be a normal method of isolating the main feedwater in MODE 3 since the T_{avg} setpoint is higher than the noload temperature. The elimination of this logic allows for the potential for the main feedline to be used as the pathway to deliver water to the steam generators. During a steamline break, there is a concern if this increases the mass of water added to the faulted steam generator, compared to the current situation with the feedwater isolation function in place. However, the pumped flowrate would be limited because the main feedwater pumps are not in service. Only the motor driven startup feedwater pump or motor driven auxiliary feedwater (AFW) pumps are in service, which have a relatively low capacity of 210 klbm/h and 600 klbm/h, respectively. Continued flow from the AFW System is accounted for within steamline break mass/energy release analyses. There is no concern if the flow is being delivered via the main feedwater piping or the AFW piping.

In summary, defeating these functions does not create a concern for a more limiting condition in MODE 3 than has already been analyzed for MODE 1 and MODE 2. Elimination of these functions in MODE 3 also does not create any new single failure scenarios that need to be considered.

3.3 Transient Analysis

None of the non-LOCA Chapter 15 events analyzed in MODES 1 or 2 would become more severe if the event were analyzed in MODE 3 (or below) assuming the proposed P-4 functions are defeated in combination with the limiting single failure. As discussed in the response to Question 2, the turbine trip function is not required to obtain acceptable results for the Chapter 15 analyses. Analyses which require feedwater isolation to mitigate the event (e.g., steamline break and feedwater malfunction) do not receive the isolation signal via the P-4 functions; thus, it also is not required to obtain acceptable results for the analyses. The only Chapter 15 event that is explicitly analyzed in MODES 3 through 6 is the Boron Dilution event, which does not credit the P-4 permissive, feedwater isolation, or turbine trip. The HZP SLB event was also evaluated in MODE 3 and determined to be bounded by the current limiting HZP SLB scenario initiated in MODE 2 (see response to Question 4). Since none of the limiting single failures assumed in the Chapter 15 analyses cause the events to rely on these P-4 functions for protective action, deletion of the proposed P-4 functions remains acceptable with respect to single failure considerations.

3.4 LOCA Analysis – Small Break and Large Break

From the LOCA perspective, the main concern relative to blocking the two reactor trip P-4 interlock functions stems from the availability of sufficient secondary side heat sink capacity. Both LBLOCA and SBLOCA events are typically insensitive to small changes in secondary side heat removal characteristics due to the heat removal from break discharge.

Previous investigations of MODE 3 and 4 LOCA behavior examined both large and small break events. LBLOCA events are low probability events in MODE 1 and have been determined to be significantly lower in probability during MODES 3 and 4 operation. The risk of core damage is also greatly reduced during MODES 3 and 4 from a MODE 1 LBLOCA accident. This conclusion extends to MODES 5 and 6 as well, during which the residual heat removal (RHR) System is relied upon as the primary decay heat removal path, rather than the steam generators, reducing the impact of steam generator heat removal characteristics further still. As such, LBLOCAs are not a concern for operation in MODES 3 through 6 and the proposed blockage of the P-4 interlock functions in question does not impact this conclusion.

Assumptions for the small break LOCA MODE 3 and 4 specific examinations resulted in a turbine trip and feedwater isolation at transient initiation; both of which ensure that the break is the primary path for heat removal. An increase in the steam generator secondary side heat removal, through either increased feed or steaming, would have a positive or negligible impact on transient results. The two P-4 interlock functions in question have not been specifically modeled in the MODE 3 and 4 analysis work and the analyses have been found to comply with 10 CFR 50.46 acceptance criteria in a conservative manner and the MODE 1 analysis work continues to be representative for the plant. This conclusion can be extended to MODES 5 and 6 as well because the RHR System becomes the primary means of heat removal thus minimizing the impact of the steam generator secondary performance changes.

3.5 LOCA Long Term Cooling

The post-LOCA methodology only takes into consideration the primary side of the steam supply system. No modeling is contained in the post-LOCA methodology pertaining to the secondary side of the steam supply system or the balance of plant operation. Furthermore, WCAP-12476, Revision 1, "Evaluation of LOCA During Mode 3 and 4 Operations for Westinghouse NSSS," Section 5.0 states that, "ECCS and ECCS actuation system designs for Westinghouse NSSS are adequate for Mode 3 and Mode 4 operation." Therefore, there is no impact on the post-LOCA methodology pertaining to the P-4 permissive function. Since there is no impact on the post-LOCA methodology due to the consideration of the P-4 permissive function, there would be no change in the conclusions of the current licensing basis post-LOCA results regardless of the plant operating mode when the LOCA occurs.

3.6 LOCA Forces

LOCA forces are driven by the rarefaction wave that travels through the RCS at the RCS fluid's local speed of sound. Due to the speed at which the rarefaction wave travels through the system, the transient time is relatively short, on the order of 500 milliseconds to 1 second, which captures the dampening of the transient with peak forces typically occurring in the first 200 milliseconds. As a consequence of this relatively short transient time, Engineered Safety Feature Actuation System (ESFAS) instrumentation and the mitigating effects of equipment that may be actuated as a result of a LOCA are not modeled in the LOCA forces analyses. Therefore, the current methodology (only MODE 1 considered due to the limiting initial conditions) remains bounding for MODES 3 through 6 of operation.

4. *As stated in Attachment 1 to the LAR, the licensee proposes to relocate the reactor trip P-4 interlock for (1) the feedwater isolation on low Tavg function and (2) the turbine trip function out of the TSs for Mode 3. The licensee indicates that none of the USAR Chapter 15 accident analyses credit the above-cited reactor trip (P-4 interlock functions) for accident mitigation.*

The proposed TS removal would allow the use of (1) the main feedwater to control SG levels and (2) the steam flow from the SGs to warm the main turbine during various surveillance testing and maintenance activities that involve opening the reactor trip breakers. The LAR does not specify a particular allowable time limit for these activities. A Chapter 15 licensing basis accident (LBA) may occur while these plant activities are ongoing.

Please provide a discussion or the results of analyses to show that with the proposed TS deletion of the turbine trip and feedwater isolation functions, the Chapter 15 analyses for applicable LBAs remain valid and bounds the results of the corresponding accidents initiating from Modes 3 through 6.

Response: The proposed amendment deletes the TS requirement for the turbine trip on reactor trip function and isolation of main feedwater coincident with low T_{avg} function in MODE 3 based on the functions not being credited in the accident analyses to mitigate the consequences of an accident. Since these P-4 functions are not credited there would not be a required time limt in MODES 3 through 6 in which these P-4 functions should be functional. A review of the USAR accident analyses determined that, as documented below, with the proposed modification to the P-4 interlock functions in MODE 3 (deletion of the turbine trip and feedwater isolation functions), the Chapter 15 accident analyses remains valid and would bound the results of the corresponding events initiating from MODES 3 through 6.

4.1 LOCA LT/ST M&E (Containment Integrity)

This is essentially the same as Question 3. Refer to response 3.1 above.

4.2 MSLB IC/OC M&E – Dose Steam Release (Containment Integrity)

This is essentially the same as Question 3. Refer to response 3.2 above.

4.3 Transient Analysis

None of the non-LOCA Chapter 15 events analyzed in MODES 1 or 2 would become more severe if the event were analyzed in MODE 3 (or below) assuming the proposed P-4 functions are defeated since neither of the functions are required to obtain acceptable results within the analyses. The only Chapter 15 event that is explicitly analyzed in MODES 3 through 6 is the Boron Dilution event, which does not credit the P-4 permissive, feedwater isolation, or turbine trip. Thus, the analysis is not impacted by the proposed changed.

The HZP SLB event was also evaluated in MODE 3 and determined to be bounded by the current limiting HZP SLB scenario initiated in MODE 2. During startup or shutdown evolutions, the operator manually blocks safety injection on low pressurizer pressure or low steamline pressure and steamline isolation on low steamline pressure when pressurizer pressure is less than the P-11 setpoint. Thus, it is possible to have a steamline break event below the P-11 interlock setpoint that does not generate a safety injection actuation of borated ECCS flow. With no borated ECCS flow supplied to the core, a return to criticality and subsequent power excursion in the core would result.

However, analysis has confirmed that the combined effect of the negative reactivity associated with the initial RCS boration requirement to meet the shutdown margin and the steamline isolation provided by the steamline high pressure negative rate trip function to limit the steam blowdown would be more than sufficient to limit the core power excursion following a return to criticality. The analysis confirmed that the consequences of a postulated steamline break event occurring in MODE 3 below P-11 with no feedwater isolation due to safety injection being blocked would be bounded by the limiting steamline break scenario initiated from the MODE 2 HZP conditions with a 0 ppm boron concentration.

4.4 LOCA Analysis – Small Break and Large Break

This is essentially the same as Question 3. Refer to response 3.4 above.

4.5 LOCA Long Term Cooling

This is essentially the same as Question 3. Refer to response 3.5 above.

4.6 LOCA Forces

This is essentially the same as Question 3. Refer to response 3.6 above.

References:

1. WCNOC Letter ET 10-0014, "Application to Revise Technical Specification 3.3.2, "Engineered Safety Feature Actuation System Instrumentation," Table 3.3.2-1," dated April 13, 2010.

2. Letter from B. K. Singal, USNRC, to M. W. Sunseri, WCNOC, "Wolf Creek Generating Station - Request for Additional Information Regarding License Amendment Request to Revise Technical Specification Table 3.3.2, "Engineered Safety Feature Actuation System Instrumentation" (TAC NO. ME3762)," August 18, 2010.