WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President Engineering

December 21, 2010

ET 10-0038

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)"
3) Letter ET 10-0028, dated October 13, 2010, from T. J. Garrett, WCNOC, to USNRC
4) Letter dated November 24, 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 Second 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. Reference 3 provided WCNOC's response to the request for additional information. In Reference 4, the Nuclear Regulatory Commission (NRC) staff indicated that the staff has reviewed Reference 3 and determined that additional information is needed to complete the review. The Attachment provides a response to the request for additional information.

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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).

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 Second 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 21st day of December , 2010.

NOTARY PUBLIC OFFICIAL SEAL STATE OF KANSAS
RHONDA L. TIEMEYER
MY COMMISSION EXPIRES
January 11, 2014

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2014

Response to Second 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. Reference 3 provided WCNOC's response to the request for additional information. In Reference 4, the Nuclear Regulatory Commission (NRC) staff indicated that the staff has reviewed Reference 3 and determined that additional information is needed to complete the review. The specific NRC question is provided in italics.

1. *In its letter dated October 13, 2010, the licensee stated in response to RAI2 that "the potential RCS [reactor coolant system] cooldown caused by the turbine trip is bounded by the cooldown caused by the MODE 2 HZP SLB [Hot Zero Power Steamline Break] event." In the Final Safety Analysis Report (FSAR) chapter 15, there are five cooldown events:*

   a. *Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature (Updated FSAR Section 15.1.1)*

   b. *Feedwater System Malfunctions the Result in an Increase in Feedwater Flow (Section 15.1.2)*

   c. *Excessive Increase in Secondary Steam Flow (Section 15.1.3)*

   d. *Inadvertent Opening of a Steam Generator Atmospheric Relief or Safety Valve (Section 15.2.4)*

   e. *Steam System Pipe Failure (Section 15.1.5)*

   Please discuss the effects of the turbine trip and main feedwater isolation function on each of the above cooldown events and show that with the proposed TS deletion of the turbine trip and feedwater isolation functions of the P-4 interlock, the consequences from the above events initiating from Mode 3 through 6 are bounded by the FSAR Chapter 15 analysis for corresponding events initiating from Mode 1 or 2.

Response:

<u>Feedwater System Malfunctions That Result in a Decrease in Feedwater Temperature (Updated Safety Analysis Report (USAR) Section 15.1.1)</u>

The Feedwater Malfunction Temperature Reduction event is caused by a malfunction in the feedwater control system which impacts the feedwater heaters, a failure of a bypass valve that diverts flow around a portion of the feedwater heaters, or operator error. Any of these failures could cause inadequate heating of the feedwater entering the steam generators resulting in excessive heat removal from the Reactor Coolant System (RCS). This event is limiting at full power operation (MODE 1) since this condition leads to the greatest potential cooldown. The heating capacity of the feedwater heaters is primarily provided by extraction steam from the main turbine. As power decreases, the temperature of the extraction steam also decreases. Once the plant enters MODE 2, the feedwater heaters are no longer capable of providing heat from the main turbine to the feedwater. Therefore, a failure of the feedwater heaters in MODE 2 or below will not result in a cooldown event and the P-4 functions, including turbine trip and feedwater isolation, would not provide any mitigating effects. Thus, since this event is not credible in MODE 3 and below, it is not impacted by the deletion of the proposed P-4 functions in MODE 3 and the current MODE 1 analysis presented in USAR Section 15.1.1 remains bounding.

Feedwater System Malfunctions That Result in an Increase in Feedwater Flow (USAR Section 15.1.2)

The Feedwater Malfunction Flow Increase event is caused by a malfunction in the feedwater control system which impacts the Feedwater Control Valve (FCV), a failure of the FCV, or operator error. Any of these failures could cause an increase in feedwater flow resulting in excessive heat removal from the RCS. This event is analyzed at both full power (MODE 1) and zero power (MODE 2). The MODE 2 analysis bounds MODES 3 and below since the initial conditions in MODE 2 enable a more severe cooldown due to the failed FCV. Furthermore, the MODE 2 analysis does not model any protective features including the P-4 functions. Rather, the event proceeds unmitigated and the maximum reactivity inserted due to the cooldown is confirmed to be bounded by (i.e., less than) the reactivity insertion modeled in the Uncontrolled Rod Cluster Control Assembly (RCCA) from a Subcritical or Low Power Startup Condition event (USAR Section 15.4.1). Thus, a Feedwater Malfunction Flow Increase event occurring in MODE 3 and below is not impacted by the proposed deletion of the P-4 functions in MODE 3 and the current MODE 2 analysis presented in USAR Section 15.1.2 remains bounding.

Excessive Increase in Secondary Steam Flow (USAR Section 15.1.3)

The Excessive Load Increase event is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. This event could result from a steam dump control malfunction, turbine throttle valve control failure, or operator error. Typically, the RCS is designed to accommodate a 10% step load in the range of 15% to 100% of full power without causing the Reactor Protection System (RPS) to actuate. Increases in steam flow below 15% power or in excess of 10% of initial load are addressed by the accidental depressurization of the Main Steam System and steam line break (SLB) events (USAR Sections 15.1.4 and 15.1.5). Since the Condition III and IV SLB events are analyzed to Condition II criteria, it is acceptable to bound these increased steam flow scenarios by the steamline break events. As discussed in the response to Question 2 of letter ET 10-0028 (Reference 3), since the steam generators have integral flow restrictors, any steam flow path regardless of location, would at worst, have the same effect on the RCS as a break corresponding to the throat area of the restrictors. Thus, the MODE 2 SLB analysis bounds a potential MODE 3 Excessive Load Increase event. Based upon the discussions provided in the response to Question 2 of Reference 3, the MODE 2 SLB analysis bounds a MODE 3 SLB event and is not impacted by the proposed deletion of the P-4 functions in MODE 3. Furthermore, the MODE 2 SLB analysis does not model the P-4 functions including feedwater isolation and turbine trip. Therefore, it is concluded that an Excessive Load Increase event occurring in MODE 3 and below is not impacted by the proposed Technical Specification change and the MODE 1 Excessive Load Increase analysis presented in USAR 15.1.3 remains bounding.

Inadvertent Opening of a Steam Generator Atmospheric Relief or Safety Valve (USAR Section 15.1.4)

The most severe core conditions resulting from an accidental depressurization of the Main Steam System, aside from steam line ruptures, are associated with an inadvertent opening of a single turbine bypass, atmospheric relief, or safety valve. The event, also known as the credible SLB, is most limiting at zero power (MODE 2) since the RCS cooldown caused by the inadvertent Main Steam System depressurization results in an insertion of positive reactivity (assuming a negative moderator temperature coefficient) which could overcome the shutdown margin (when including an allowance for a stuck RCCA) and cause the reactor to return to power. The MODE 2 analysis bounds MODES 3 and below since the initial conditions in MODE 2 enable a more severe cooldown with a greater chance of a return to power. The inadvertent opening of a steam generator relief or safety valve event creates a depressurization of the secondary side with an effective opening size that is within the spectrum of break sizes analyzed by the hypothetical SLB event, and thus, is bounded by the event discussed in USAR Section 15.1.5. The Section 15.1.5 event is addressed in the response to Question 2 in Reference 3. Furthermore, the SLB event discussed in USAR Section 15.1.5 does not model the P-4 functions, including feedwater isolation and turbine trip. Thus, the accidental depressurization of the Main Steam System is not impacted by the proposed Technical Specification change and the analysis presented in USAR 15.1.4 remains bounding.

<u>Steam System Piping Failure (USAR Section 15.1.5)</u>

This event is addressed in the response to Question 2 in Reference 3. The information provided in Reference 3 was discussed during the teleconference between WCNOC and the NRC on December 6, 2010.

2. *To address the acceptability of the proposed TS deletion of the turbine trip and feedwater isolation in Mode 3, the licensee stated in its response to RAIs 2 and 3 that the turbine trip function is not required to obtain acceptable results for Chapter 15 analyses. Also, the response to RAI 4 stated that neither the turbine trip nor the feedwater isolation functions are required to obtain acceptable results within the non-loss-of-coolant accident (LOCA) Chapter 15 analyses. Please provide the bases to support the above statements in the responses to RAI 2 and 3 and RAI 4 for each of the events in FSAR Chapter 15, and show that none of the "events analyzed in Modes 1 and 2 would become more severe if the events were analyzed in Mode 3 (or below) assuming the proposed P-4 function are defeated."*

Response:

<u>USAR Section 15.1</u>

See the response to Question 1 above.

<u>Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow (USAR Section 15.2.1)</u>

As discussed in USAR Section 15.2.1, this event is not applicable to WCGS.

<u>Loss of External Electrical Load / Turbine Trip / Inadvertent Closure of Main Steam Isolation Valves / Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip (USAR Sections 15.2.2, 15.2.3, 15.2.4, and 15.2.5)</u>

The Turbine Trip event bounds the Loss of External Load, Inadvertent Closure of the Main Steam Isolation Valves, Loss of Condenser Vacuum, and other events resulting in a turbine trip due to the more rapid loss of steam flow during the Turbine Trip event. Since these events are initiated by a turbine trip with coincident feedwater isolation, the proposed deletion of the P-4 functions in MODE 3 does not adversely impact the events. Thus, the MODE 1 analysis presented in Section 15.2.3 of the USAR remains bounding.

Loss of Nonemergency AC Power to the Station Auxiliaries (Blackout) / Loss of Normal Feedwater Flow (USAR Sections 15.2.6 and 15.2.7)

The Loss of Normal Feedwater (LONF) event is defined as a complete loss of main feedwater flow while the reactor is operating at full power. The Loss of Offsite AC Power (LOOP) event is identical to the LONF event, except that a loss of offsite power is assumed to occur following a reactor trip. The immediate consequence of the events is a reduction in steam generator inventory which, if left unmitigated, will result in a reactor trip and Auxiliary Feedwater (AFW) actuation on a Low-Low Steam Generator Level signal. Following a reactor trip, the decay heat (and Reactor Coolant Pump (RCP) heat input for LONF) may exceed the heat removal capability of the secondary system. This will result in an increase in RCS pressure, temperature, and pressurizer water level and will continue until the AFW System re-establishes the secondary side heat sink. Thus, the primary acceptance criterion for the events is to prevent the generation of a more serious plant condition as demonstrated by ensuring that the AFW System is capable of re-establishing the secondary side heat sink prior to pressurizer overfill. Therefore, these events are limiting in MODE 1 due to the decay heat contribution immediately following a reactor trip. Although a turbine trip is received following reactor trip, delaying its actuation is a benefit to the analyses since the turbine would provide an additional heat removal path prior to AFW actuation. The additional heat energy removed through the turbine would cause a less severe RCS heatup and extend the time to pressurizer overfill. This reduces the challenge on the AFW System to re-establish the secondary side heat sink. However, 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. Therefore, the proposed deletion of the P-4 functions in MODE 3 would be a benefit to the LONF and LOOP events and thus, the current MODE 1 analyses presented in USAR Sections 15.2.6 and 15.2.7 remain bounding.

Feedwater System Pipe Break (USAR Section 15.2.8)

A Feedline Break (FLB) event is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to maintain shell-side fluid inventory in the steam generators. If the break occurs between the check valve and the steam generator, the break cannot be isolated and the steam generator inventory will be lost. Depending on the location of the AFW piping, this break could also preclude subsequent addition of AFW to the affected steam generator. This will reduce the capacity of the secondary side heat sink and result in an RCS heat up due to the reduction in decay heat removal. Thus, the FLB event is limiting in MODE 1 due to the decay heat contribution immediately following a reactor trip. The event is non-limiting in MODE 2 and below since the decay heat contribution is minimal. Although a turbine trip is received following a reactor trip, delaying its actuation is a benefit to the analysis since the turbine would provide an additional heat removal path prior to AFW actuation. The analysis does not explicitly model feedwater isolation since main feedwater is conservatively assumed to become unavailable at event initiation and the feedline check valves are credited to prevent reverse flow from the intact steam generators. The minimum AFW flow rates to the intact steam generators modeled in the analysis are based upon the conservative assumption that all of the flow from the motor-driven AFW pump aligned with the affected steam generator spills out of the break. The flow from the turbine-driven AFW pump is also adjusted based upon this assumption. Therefore, the proposed deletion of the P-4 functions in MODE 3 would either be a benefit to the FLB event or have no impact on the analysis and thus, the current MODE 1 analysis presented in USAR Section 15.2.8 remains bounding.

Partial / Complete Loss of Forced Reactor Coolant Flow (USAR Sections 15.3.1 and 15.3.2)

A Partial or Complete Loss of Flow (PLOF/CLOF) is caused by a mechanical or electrical failure in an RCP, an interruption in the power supplying one or more of the RCPs, or a reduction in RCP motor supply frequency. If the reactor is at power (MODE 1), the immediate effect of a loss of flow event is a rapid increase in coolant temperature which could result in Departure from Nucleate Boiling (DNB) with subsequent fuel damage. The loss of flow events result in an immediate reactor trip on Low RCS Flow or RCP Undervoltage. Due to the reduced flow through the core, the temperature continues to increase until the RCCAs are fully inserted into the core. This critical time frame of interest, in which a DNB violation may occur, is the first 5 or 6 seconds after event initiation and before any of the P-4 functions can impact the event. The loss of flow events are non-limiting in MODE 2 and below since the reactor is not at power and natural circulation adequately cools the RCS. Thus, the proposed deletion of the P-4 functions in MODE 3 does not impact the loss of flow events and the current MODE 1 analyses presented in USAR Sections 15.3.1 and 15.3.2 remain bounding.

Reactor Coolant Pump Shaft Seizure (Locked Rotor) / Shaft Break (USAR Sections 15.3.3 and 15.3.4)

The consequences of an instantaneous seizure of the rotor or break of the shaft of an RCP are very similar. The initial rate of reduction in coolant flow is greater for the Locked Rotor event; however, with a failed shaft, the impeller could conceivably be free to spin in the reverse direction such that the endpoint (steady-state) core flow would be reduced when compared to a locked rotor. The Locked Rotor/Shaft Break events are also very similar to the loss of flow events except they are more limiting due to the more rapid reduction in coolant flow. As with the loss of flow events, the critical time frame of interest is the first 5 or 6 seconds after event initiation and before any of the P-4 functions can impact the event. The Locked Rotor/Shaft Break events are non-limiting in MODE 2 and below since the reactor is not at power and natural circulation adequately cools the RCS. Thus, the proposed deletion of the P-4 functions in MODE 3 does not impact the Locked Rotor/Shaft Break events and the current MODE 1 analyses presented in USAR Sections 15.3.3 and 15.3.4 remain bounding.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition (USAR Section 15.4.1)

The Uncontrolled RCCA Bank Withdrawal from Subcritical (RWFS) event is defined as an uncontrolled addition of reactivity while in a subcritical condition (MODE 2) caused by withdrawal of one or more RCCA banks resulting in a power excursion. The RWFS event could result from a malfunction in the automatic rod control system or operator error. The event is terminated by a reactor trip on the Power Range Neutron Flux – Low trip Function. In MODES 3 and below, protection is provided by the Source Range Neutron Flux trip Function since the Power Range Neutron Flux – Low trip Function is not required to be OPERABLE. The MODE 2 analysis bounds MODES 3 and below since the Source Range Neutron Flux trip Function will trip the reactor before any appreciable power is generated. Furthermore, none of the P-4 functions are modeled in the RWFS analysis and thus, they are not required to mitigate the event. Thus, the proposed deletion of the P-4 functions in MODE 3 does not impact the RWFS event and the current MODE 2 analyses presented in USAR Section 15.4.1 remain bounding.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (USAR Section 15.4.2)

A continuous Uncontrolled RCCA Bank Withdrawal at Power (RWAP) event, due to a malfunction of the Rod Control System or operator error, will result in an increase in the core heat flux due to the positive reactivity insertion. The RWAP event is analyzed in MODE 1. The RWFS event (USAR Section 15.4.1) addresses an uncontrolled RCCA bank withdrawal for MODES 2 and below. Thus, the RWAP event is not impacted by the proposed deletion of the P-4 functions in MODE 3 and the analysis presented in USAR Section 15.4.2 remains bounding.

Rod Cluster Control Assembly Misoperation (USAR Section 15.4.3)

USAR Section 15.4.3 includes the following events:

a. One or more dropped RCCAs within the same group, or dropped RCCA bank (Dropped Rod)

b. Statically Misaligned RCCA

c. Withdrawal of a Single RCCA

An RCCA may become misaligned with respect to the other RCCAs of its bank due to either a malfunction of the Rod Control System or operator error. The resulting asymmetric power distribution increases the local peaking factors which could cause a DNB violation. The Dropped Rod and Statically Misaligned RCCA events are not credible in MODE 3 and below since all of the control and protection RCCA banks are fully inserted into the core. The Withdrawal of a Single RCCA analysis does not model any of the P-4 functions. The analysis is performed at the burnup step which has the highest peak $F\Delta H$. Power distributions are generated for all combinations of Bank D inserted less one Bank D RCCA out. This determines the Bank D RCCA which produces the highest peak assembly powers when stuck out of the core. Therefore, the proposed deletion of the P-4 functions in MODE 3 does not impact the RCCA Misoperation events and the current analyses presented in USAR Section 15.4.3 remain bounding.

Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (USAR Section 15.4.4)

The Startup of an Inactive RCP event is administratively precluded in MODES 1 and 2 as discussed in the WCGS Technical Specification Bases (Reference 5). In MODE 3 and below, the startup of an inactive RCP cannot result in a "cold water" criticality, even if the maximum difference in temperature exists between the steam generator and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required shutdown margin. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition. Thus, the proposed deletion of the P-4 functions in MODE 3 does not impact the Startup of an Inactive RCP event and the current analysis presented in USAR Section 15.4.4 remains bounding.

A Malfunction or Failure of the Flow Controller in a BWR Loop That Results in an Increased Reactor Coolant Flow Rate (USAR Section 15.4.5)

This event is not applicable to WCGS.

Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (USAR Section 15.4.6)

The Boron Dilution event is defined as an inadvertent dilution of the RCS boron concentration due to a Chemical and Volume Control System (CVCS) malfunction or operator error. The Boron Dilution event is analyzed in MODES 1 through 6; however, as discussed in the response to Question 2 in Reference 3, the event does not model the P-4 permissive, feedwater isolation, or turbine trip. Therefore, the proposed deletion of the P-4 functions in MODE 3 does not impact the Boron Dilution event and the current analyses presented in USAR Section 15.4.6 remain bounding.

Inadvertent Loading and Operation of a Fuel Assembly in Improper Position (USAR Section 15.4.7)

Fuel and core loading errors that can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod or full fuel assembly during manufacture with one or more pellets of the wrong enrichment, or loading one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods will lead to increased heat fluxes if the error results in placing the fuel in core positions designed for fuel of lesser enrichment. This event is analyzed under steady-state conditions at limiting time frames throughout core life. Therefore, neither reactor trip nor any of the P-4 functions are modeled in the analysis. Thus, the proposed deletion of the P-4 functions in MODE 3 does not impact the event and the analysis presented in USAR Section 15.4.7 remains bounding.

Spectrum of Rod Cluster Control Assembly Ejection Accidents (USAR Section 15.4.8)

The RCCA Ejection event is the result of a mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. The consequence of this event is a rapid positive reactivity insertion and system depressurization together with an adverse core power distribution, possibly leading to localized fuel rod damage. The event is analyzed at both full power (MODE 1) and zero power. The zero power event is modeled to bound operation in both MODES 2 and 3 since it assumes only two RCPs are operating at the time of the ejection. However, the proposed deletion of the P-4 functions in MODE 3 does not impact the RCCA Ejection event since the analysis does not model the P-4 functions. Therefore, the current analyses presented in USAR Section 15.4.8 remain bounding.

Inadvertent Operation of the Emergency Core Cooling System During Power Operation (USAR Section 15.5.1)

In Inadvertent Emergency Core Cooling System (ECCS) Actuation at Power event results in an increase in RCS inventory leading to the potential filling of the pressurizer. The event could be caused by operator error or a spurious electrical actuating signal. The event is limiting in MODE 1 due to the decay heat contribution immediately following reactor trip. Also, the time to pressurizer overfill is minimized in MODE 1 since the initial pressurizer water level is substantially higher than at no-load conditions which results in more initial pressurizer inventory. Although the ECCS actuation is the primary source for the increase in RCS volume, the analysis also models an immediate turbine trip on reactor trip which causes the RCS inventory to swell resulting in an additional increase in pressurizer level. Thus, a delay in the time of turbine trip would provide a small benefit to the event in that pressurizer overfill could be delayed. However, 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. The event is terminated by the operators terminating the ECCS flow. Therefore, the proposed deletion of the P-4 functions in MODE 3 does not impact the Inadvertent ECCS Actuation at Power event and the current analysis presented in USAR Section 15.5.1 remains bounding.

Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory (USAR Section 15.5.2)

CVCS Malfunctions that increase the RCS inventory are caused by operator error or the failure of the changing pump controller. The event is very similar to the Inadvertent ECCS Actuation at Power event (USAR Section 15.5.1), except that the flow rate is lower and thus, pressurizer overfill is delayed. The analysis models an immediate turbine trip on reactor trip which causes the RCS inventory to swell resulting in an additional increase in pressurizer level. Thus, a delay in the time of turbine trip would provide a small benefit to the event in that pressurizer overfill could be delayed. Consistent with the Inadvertent ECCS Actuation at Power event, operator action to terminate charging is required to mitigate the event. Therefore, the proposed deletion of the P-4 functions in MODE 3 does not impact CVCS Malfunctions that increase the RCS inventory and the current analysis presented in USAR Section 15.5.2 remains bounding.

A Number of BWR Transients (USAR Section 15.5.3)

These events are not applicable to WCGS.

Inadvertent Opening of a Pressurizer Safety or Relief Valve (USAR Section 15.6.1)

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer safety or relief valve. The event is limiting in MODE 1 since the flow rate out the failed valve will be maximized at full power operation. The event is terminated by operator action to either close the open valve or close an isolation valve in the affected path. However, the analysis presented in the USAR does not examine this part of the transient since it only addresses the short term phase of the event to the point of reactor trip. Although a turbine trip is modeled to occur on reactor trip, the turbine trip is not used to mitigate the event. Furthermore, if a direct turbine trip on reactor trip does not occur, the turbine will be tripped directly by one of the turbine protection signals due to the reduction in steam flow caused by the reactor trip. Thus, the P-4 functions do not provide any mitigating effects for this event. Therefore, the proposed deletion of the P-4 functions in MODE 3 does not impact the RCS Depressurization event and the current analysis presented in USAR Section 15.6.1 remains bounding.

Spectrum of BWR Steam System Piping Failures Outside of Containment (USAR Section 15.6.4)

These events are not applicable to WCGS.

A Number of BWR Transients (USAR Section 15.6.6)

This event is not applicable to WCGS.

Anticipated Transients Without Scram (USAR Section 15.8)

An Anticipated Transient Without Scram (ATWS) is an anticipated operational occurrence followed by failure of the reactor trip portion of the RPS. By definition, an ATWS is only postulated in MODE 1 operation since the reactor is in a tripped condition (i.e., all RCCAs are inserted into the core) in MODES 2 and below. Therefore, the proposed deletion of the P-4 functions in MODE 3 does not impact the ATWS event and the current analysis presented in USAR Section 15.6.1 remains bounding.

3. *The licensee's response to RAI 4 discussed an analysis of the steam line break (SLB) event initiating from Mode 3 below P-11 with no feedwater isolation. Please provide a description of the SLB analysis. The requested information should include the results of the analysis, a sequence of the events, a discussion of the methods and computer code used in the analysis, and a discussion of the compliance with the restrictions and limitations in the NRC safety evaluation report approving the methods and code. Please identify the key parameters considered and discuss the bases of the selection of the values (including measurement uncertainties and parameter fluctuation around the normal values) for the initial plant conditions that minimize the margin to acceptable limits.*

Response: The current licensing basis SLB event is conservatively analyzed at HZP with a 0 ppm boron concentration (corresponding to End of Life in a fuel cycle) to bound all lower modes of operation. The pressurizer pressure-low and steam line pressure-low signals are credited as the primary protection for the mitigation of the limiting SLB event. The limiting SLB is analyzed based on maximizing the RCS cooldown effect with a most negative moderator temperature coefficient. The RCS cooldown then results in a significant addition of positive reactivity to the core, and a return to criticality and power excursion even with the control rods inserted (less the most reactive rod stuck out of the core). The pressurizer pressure-low and steam line pressure-low signals actuate safety injection to provide borated ECCS flow to limit the power excursion and main steam line isolation to limit the secondary cooldown of the RCS.

However, a question had been raised in 2002 regarding a SLB during MODE 3 operation when automatic safety injection (SI) actuation on Low Pressurizer Pressure and/or Low Steam Line Pressure has been manually blocked and whether this is bounded by the licensing basis HZP SLB analysis presented in the USAR Section 15.1.5. Note: The blocking of the Low Pressurizer Pressure and/or Low Steam Line Pressure signals is permitted following receipt of the P-11 permissive to allow the plant to initiate RCS cool down without the initiation of SI. For this configuration, the main steam line isolation function is provided by the steam line pressure-negative rate-high signal. However, the only available SI signal for a SLB event would have to be generated by the containment pressure–high 1 signal. Since the SLB could occur outside containment, it is possible to have a SLB event below the P-11 interlock setpoint that does not generate a SI actuation of borated ECCS flow. In the absence of this protective action, whether or not the consequences of a SLB occurring below the P-11 setpoint would be bounded by the limiting scenario initiating from HZP conditions is questionable.

In response to this concern, an analysis was performed to simulate the consequences of a SLB occurring in MODE 3 below the P-11 setpoint. The RCS cooldown for a SLB event occurring in MODE 3 below the P-11 setpoint would be essentially identical to the limiting USAR case, since the RCS can be at or near the HZP, no load temperature of 557°F and still meet the RCS subcooling requirements. With the exception that an automatic SI actuation signal would not be available, because the SI on low pressurizer pressure or low steam line pressure may have been manually blocked. However, the results of the analysis has confirmed that the consequences of a postulated SLB occurring below the P-11 setpoint would be bounded by the current licensing basis analysis, if the steam line isolation provided by the Steam Line Pressure Negative Rate – High trip Function and the initial RCS boron concentration that is required to be maintained during startup or shutdown evolutions are credited.

The LOFTRAN code (Reference 9) is used to perform the core response analysis for a postulated SLB occurring in MODE 3 below the P-11 setpoint. LOFTRAN is a digital computer code, developed to simulate transient behavior in a multi-loop pressurized water reactor system. The program simulates neutron kinetics, thermal-hydraulic conditions, pressurizer, steam generators, RCPs, and control and protection system operation. Reference 9 and the USNRC Topical Report Evaluation (Reference 10) contained therein, document the appropriateness of the application of LOFTRAN for simulating the plant response for the SLB – core response event. Reference 10 provides a list of events that can be analyzed with LOFTRAN and includes the SLB transient. The application of LOFTRAN for this analysis is consistent with the information presented in Reference 10.

Consistent with the assumptions used for the SLB analysis assuming the HZP conditions, the following plant conditions and features are assumed to exist at the time of a main SLB accident occurring in MODE 3 below the P-11 setpoint:

a. The reactor is already tripped with the most reactive RCCA stuck in its fully withdrawn position at the time the SLB occurs.

b. The most negative reactivity coefficients (primarily moderator and Doppler), corresponding to the End of Life (EOL) in a fuel cycle, are assumed because they cause the highest power increases as the result of cooldown transients.

c. Pumped SI is not available since the automatic actuation of SI has been manually blocked when the RCS pressure reaches the P-11 setpoint (i.e., 1970 psig). Only the passive accumulators are available for injection of highly concentrated boric acid solution.

d. Feedwater isolation is not actuated since the automatic SI signal has been manually blocked when the RCS pressure reaches the P-11 setpoint.

e. Steamline isolation is provided by the Steam Line Pressure Negative Rate – High trip Function. Steam line isolation is assumed in all loops except the faulted steam line. Blowdown from the three intact steam generators is terminated upon receipt of the signal to isolate and valve closure. The main steam line isolation function is accomplished via the main steam isolation valves in each of the three unbroken steam lines.

f. Offsite power is assumed to be available. The major effect of the offsite power assumption is the operation of the reactor coolant pumps. If the RCPs continue to operate throughout the event, the forced reactor coolant flow results in higher primary-to-secondary heat transfer and thus conservatively enhances cooling effect.

g. The initial boron concentration in the core is assumed to be maintained at a level just meeting the minimum shutdown margin requirements. This assumption is conservative compared with the normal practice. According to Operating Procedure GEN 00-006, "Hot Standby to Cold Shutdown," prior to initiating RCS cooldown, sufficient boron concentration is established to maintain shutdown margin in 50°F increments. For instance, at a no load temperature of 557°F, procedures requires the operator to proactively determine whether the boron concentration is sufficient to maintain the shutdown margin at 500°F prior to cooling down to 500°F.

The plant initial conditions and input parameters used in the analysis are based on the original licensing basis analysis for the HZP SLB accident. The values for the initial conditions and assumptions are presented and discussed as follows:

a. Power: 1% (used for the zero power runs for code stability)

b. Heat input from reactor coolant pumps: 14 MWt

c. RCS average temperature: 557°F (no load temperature)

d. RCS pressure: 1985 psia (based on P-11 setpoint of 1970 psig)

e. RCS design flow: 361,296 gpm (thermal design flow)

f. Pressurizer water volume: 516.98 ft^3 (corresponding to the programmed water level at no-load temperature, i.e., 27%)

g. Feedwater enthalpy: 50.0 Btu/lb (based on feedwater temperature corresponding to zero power)

h. Steam generator fluid mass: 164,500 lbs (corresponding to the nominal steam generator water level at zero power)

i. Core boron concentration: 386 ppm (corresponding to the required boron concentration that meets the minimum shutdown margin at no load temperature of 557°F at EOL)

j. Reactor trip: Reactor tripped at 0 seconds

Several sensitivity runs based on the input parameters mentioned above were made for analyzing the SLB core response. Highlights of the different cases used to determine the core response to a postulated SLB initiated from the HZP conditions or MODE 3 below the P-11 setpoint, with or without SI actuation and steam line isolation, are presented by the following:

a. The licensing basis SLB scenario, which initiates from the HZP conditions with a zero ppm initial boron concentration, remains bounded as far as the power excursion is concerned. The power excursion is calculated to reach a peak value of about 17%. This excursion of core power would eventually be reversed by isolating the steam line to reduce the overcooling effect and injecting highly borated water into the core from the ECCS.

b. If the reactor core is borated to a level just meeting the minimum shutdown margin requirement (for instance, about 386 ppm at 557°F, EOL conditions per the Cycle 13 design), the core would return to critical with a power excursion to a higher level (~21%), without crediting steam line isolation by the Steam Line Pressure Negative Rate – High trip Function. However, the power excursion would be reduced to about 13% if credit is taken for the Steam Line Pressure Negative Rate – High trip Function to isolate the steam line.

c. For a minor SLB, the core might still return to criticality, but the power excursion would be less severe. For instance, a 0.5 ft^2 partial double-ended steam line rupture, at an RCS temperature of 557°F with and an initial RCS boron concentration of 386 ppm (meeting the minimum shutdown margin requirement at EOL), would result in a core power excursion of about 9%, if the Steam Line Pressure Negative Rate – High trip Function is not credited. By taking credit for the Steam Line Pressure Negative Rate – High trip Function, the power excursion would only be about 7%.

Table 3-1 presents a comparison of the consequences of the case scenarios described above. Note: The reference scenario (Case 1) differs from the other cases because SI and the associated feedwater isolation are credited. The time sequence of event for Case 2 is provided in Table 3-2. Figures 3-1 through 3-10 show the transient results for the key plant parameters as a function of time for Case 2: full double-ended rupture with an initial boron concentration of 386 ppm, crediting the steam line isolation by the Steam Line Pressure Negative Rate – High trip Function.

Table 3-1 Comparison of the Consequences of a Steamline Break Scenario with Different Initial Conditions and Assumptions

Case	Break Size (ft^2)	Initial Conditions	SI/Feedwater Isolation	Steam Line Isolation	Peak Nuclear Power(%)@sec
1	Full DER(1)/4.2	HZP/ 0 ppm	YES	YES(2)	16.63 @96.5
2	Full DER(1)/4.2	Mode 3 /386 ppm	NO	YES(3)	12.57 @97.0
3	Full DER(1)/4.2	Mode 3 /386 ppm	NO	NO	20.51 @43.8
4	Partial DER(1)/0.2	Mode 3 /386 ppm	NO	YES(3)	6.97 @170.5
5	Partial DER(1)/0.2	Mode 3 /386 ppm	NO	NO	8.98 @130.0

(1) Double-ended rupture
(2) Steam line isolation by Steam Line Pressure Low trip Function.
(3) Steamline isolation by the Steam Line Pressure Negative Rate – High trip Function.

Table 3-2 Sequence of Events for Case 2

Event	Time (sec.)
Steam line ruptures	0.0
Negative steam pressure rate-high setpoint reached	0.032
Steam line isolation begins	2.032
Steam line isolation complete	7.032
Pressurizer emptied	22.0
Criticality attained	31.2
Peak core thermal power occurs	98.0

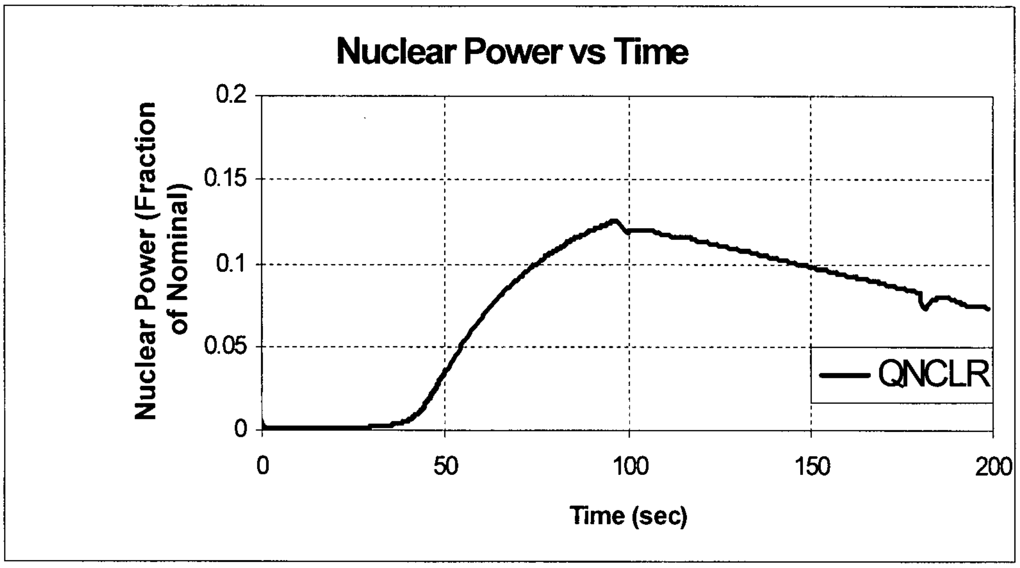

Figure 3-1: Nuclear Power versus Time

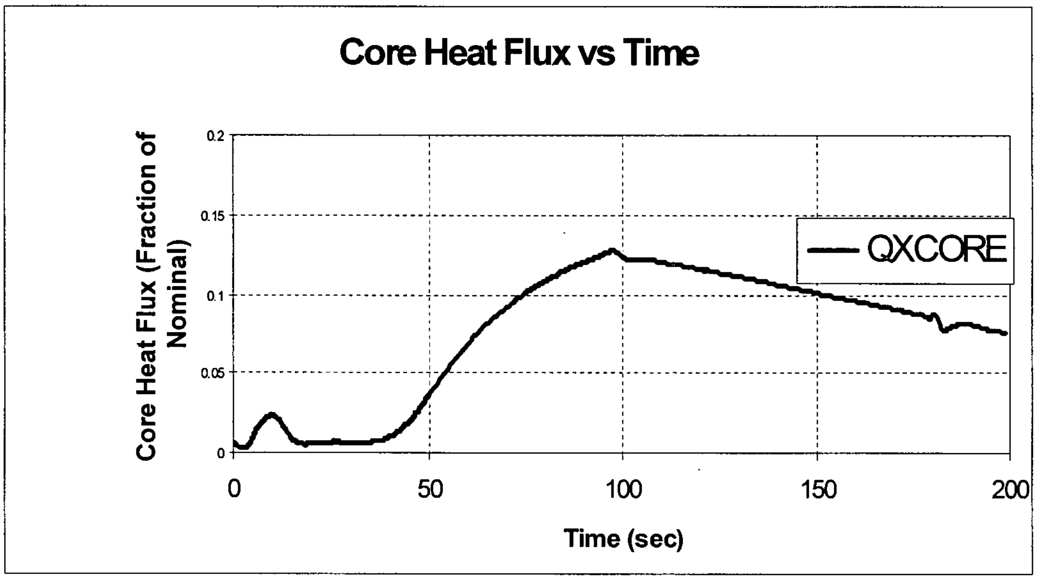

Figure 3-2: Core Heat Flux versus Time

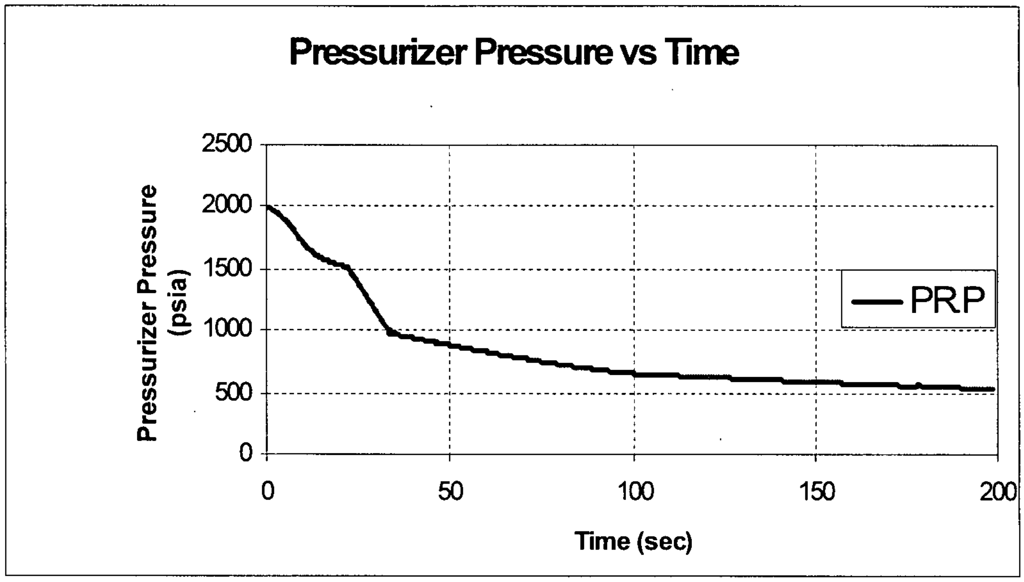

Figure 3-3 Pressurizer Pressure versus Time

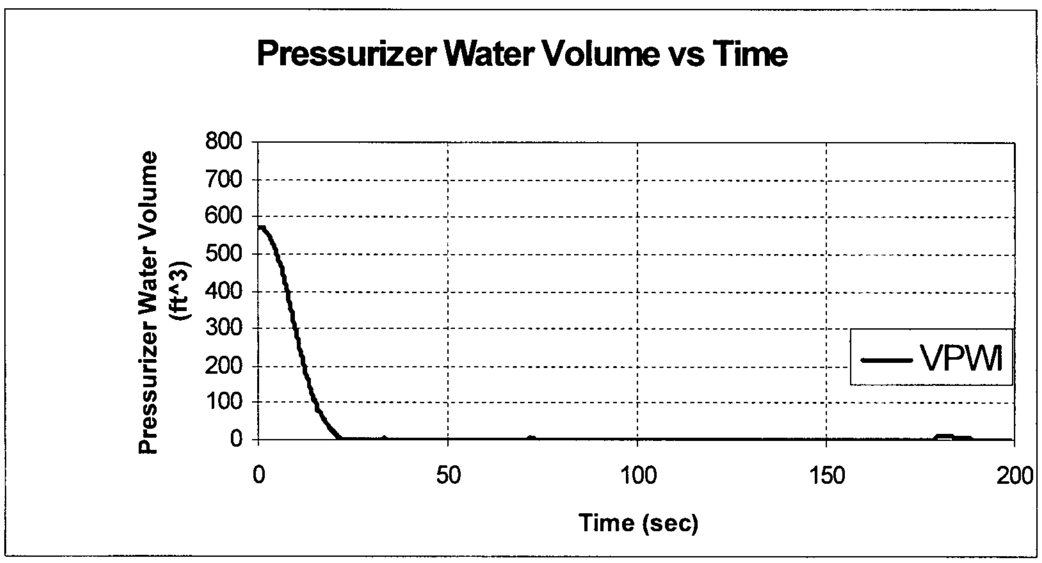

Figure 3-4 Pressurizer Water Volume versus Time

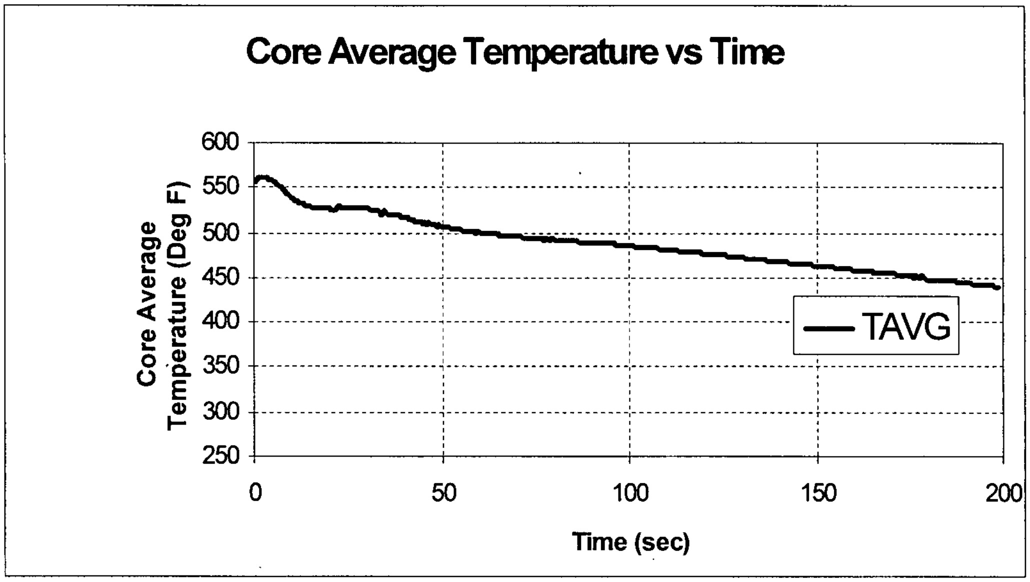

Figure 3-5 Core Average Temperature versus Time

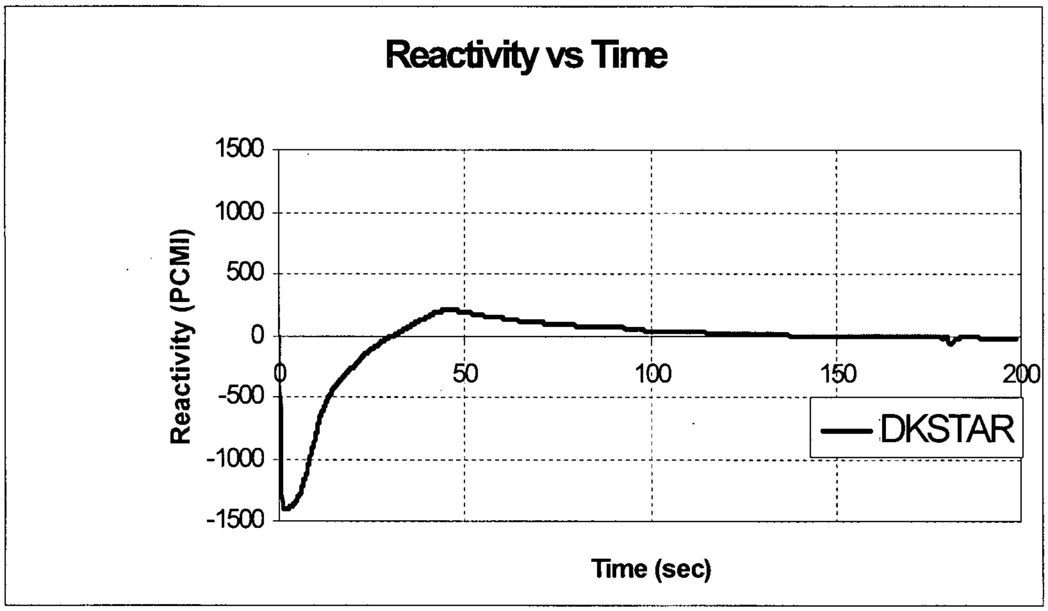

Figure 3-6 Reactivity versus Time

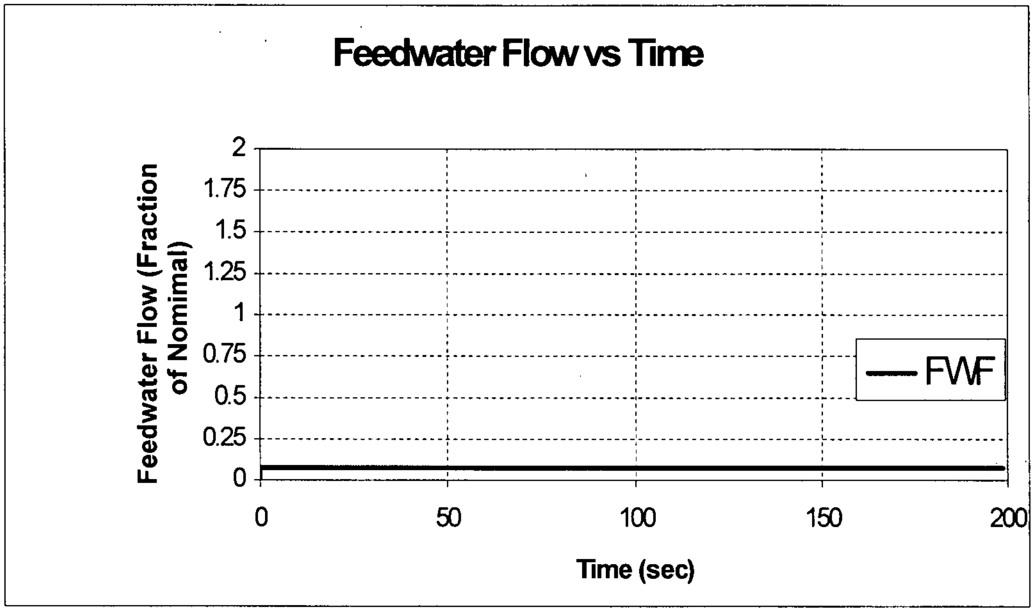

Figure 3-7 Feedwater Flow versus Time

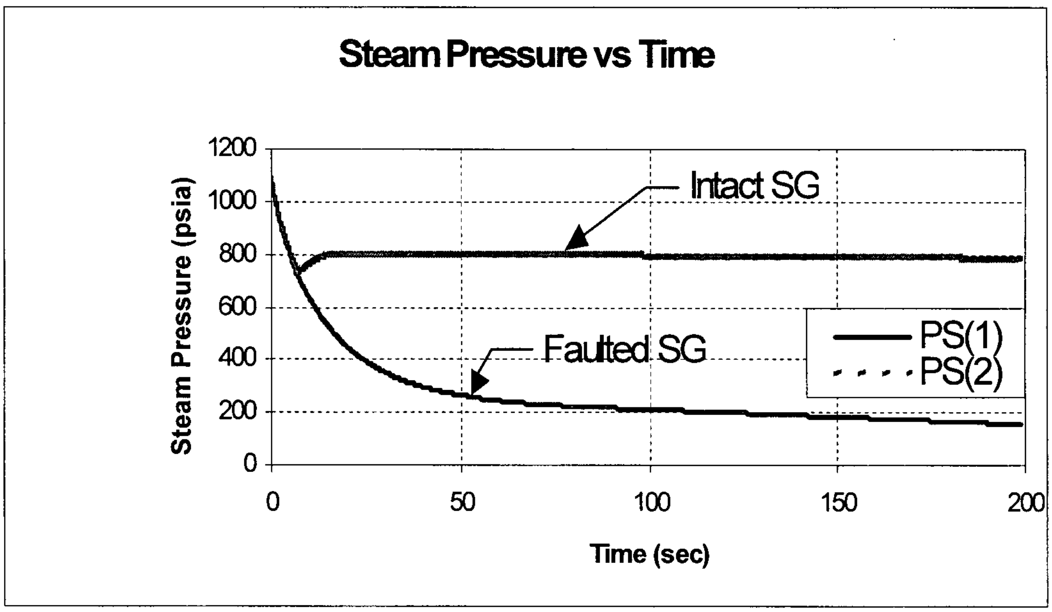

Figure 3-8 Steam Pressure versus Time

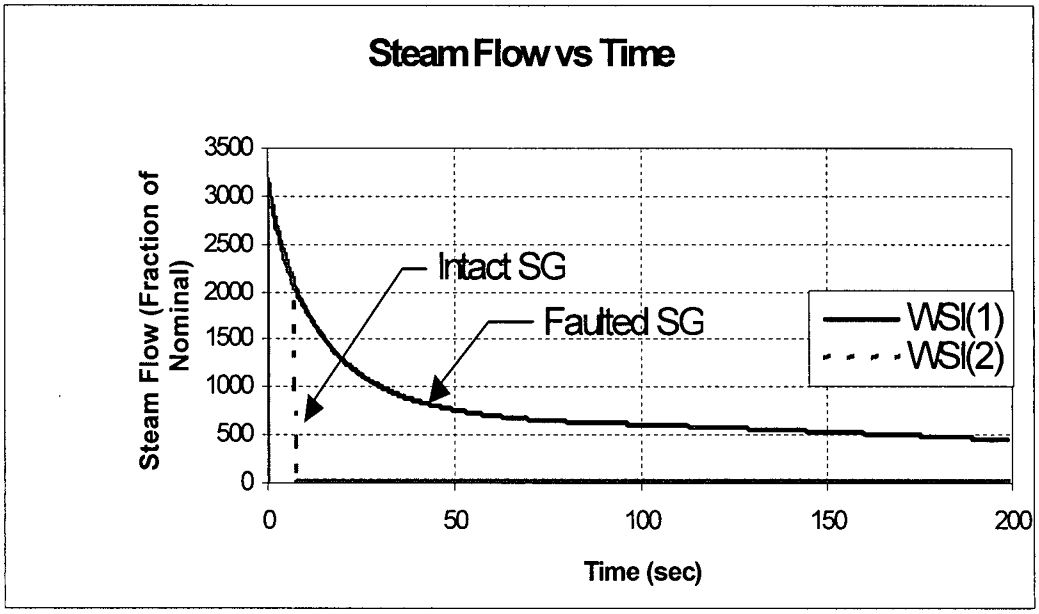

Figure 3-9 Steam Flow versus Time

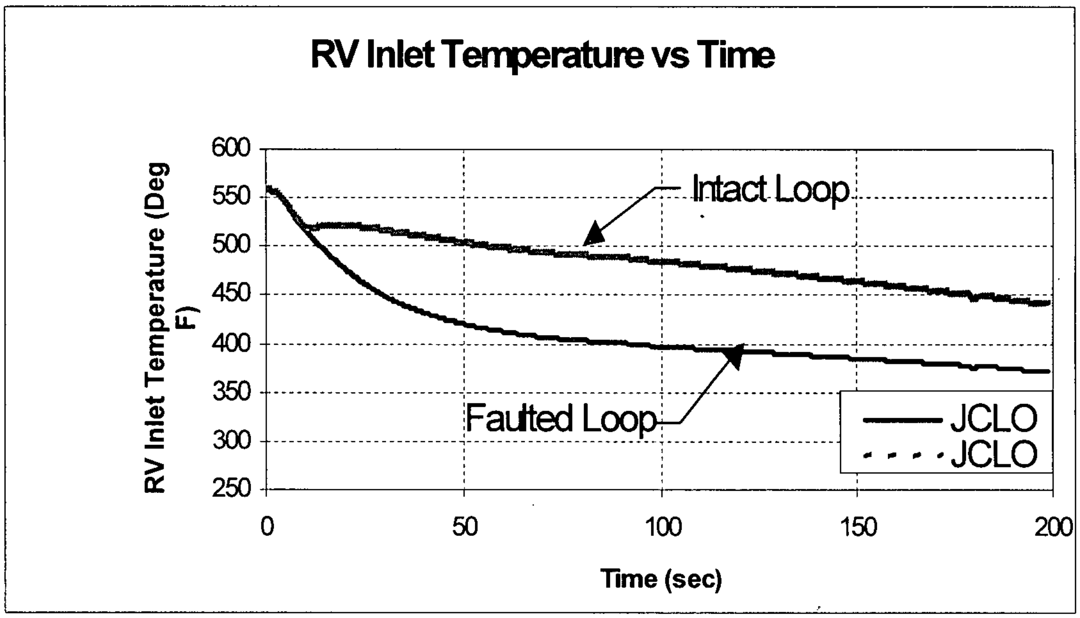

Figure 3-10 Core Inlet Temperature versus Time

Based on the above sensitivity study results, the following is concluded:

1. The combined effect of the negative reactivity associated with the initial RCS boration requirement to meet the minimum shutdown margin and the steam line isolation provided by the Steam Line Pressure Negative Rate – High trip Function to limit the steam blowdown is more than sufficient to limit the core power excursion following a return to criticality. As a result, the consequences of a postulated SLB event occurring in MODE 3 below P-11 with SI being blocked would be bounded by the current limiting SLB scenario initiating from HZP conditions with a 0 ppm boron concentration.

2. For a smaller steam line break (< 0.2 ft^2), the steam line depressurization rate may not be large enough to activate the Steam Line Pressure Negative Rate – High trip Function. However, the cooling effect associated with the slower steam blowdown would be less severe. Consequently, the core may still have a chance to return to criticality, but the subsequent power excursion would be limited by the negative reactivity associated with the initial RCS boron concentration. This conclusion is inferred by observing the transient results of the Cases 4 and 5 scenarios.

It should be noted that the analysis for a postulated SLB in MODE 3 below the P-11 setpoint assumes that the most reactive RCCA is stuck out of the core in accordance with 10 CFR Part 50 Appendix A (General Design Criteria). Plant procedures require that the RCS be borated to account for this condition. However, it is unlikely that the most reactive RCCA, if any, would actually be stuck out of the core. Taking credit for this additional negative reactivity makes a SLB scenario in MODE 3 significantly less severe with respect to applicable acceptance criteria.

4. *In Section 2.4 of its letter dated October 13, 2010, the licensee stated that the reactor trip P-4 interlock turbine trip function is also not credited for Appendix K small-break LOCA (SBLOCA) analyses. This could result in an additional increase in RCS depressurization during the SBLOCA, which is non-conservative. Besides, the LOCA blowdown load forces are much less for the limiting small breaks, so not tripping the turbine for SBLOCA appears to be inappropriate. Secondly, the licensee's statement in Section 3.4 of its letter dated October 13, 2010, that "SBLOCA events are typically insensitive to small changes in secondary side heat removal" is not always true. For plants with core uncover and high peak centerline temperatures (PCTs) for small breaks in the range 0.05 to 0.1 ft^2, small changes in secondary heat removal can have an appreciable impact on PCT because lower RCS pressures of 25 to 50 psia over several hundred seconds during the initial portion of a limiting SBLOCA can reduce PCT substantially, as a result of the additional high-pressure safety injection flow injected during this period due to this lower RCS pressure. Please show the impact of failure of turbine trip on the limiting SBLOCA and compare the results with the case where the turbine is tripped.*

Response: The WCGS Appendix K Small Break LOCA (SBLOCA) is analyzed with the approved Westinghouse Small Break LOCA Evaluation Model with NOTRUMP (NOTRUMP-EM) (References 6, 7, and 8). The standard application of the approved NOTRUMP-EM assumes a turbine trip coincident with the reactor trip. As such, the additional increase in RCS depressurization during the SBLOCA (and related sensitivity to small changes in secondary side heat removal described in this request for additional information), which could result from not tripping the turbine, do not occur. Furthermore, the impact due to the LOCA blowdown forces, as discussed in the request for additional information, is not a concern since the turbine is tripped for the analyzed SBLOCA. It is noted that the turbine trip is a conservative assumption modeled in the standard application of the approved NOTRUMP-EM; it is not modeled as a function of the P-4 interlock.

5. *In its response to RAI 1, the licensee indicated that "the subject circuitry does not provide a required safety function" and that the feedwater isolation signal bypass is not expected to be implemented more than once per year; therefore, required bypassed and inoperable status indication per Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," is not warranted.*

   Please confirm that inclusive of all the purposes identified for bypassing the feedwater isolation function during times when the feedwater system is expected to be operating, (e.g., when performing procedures GEN 00-006, "Hot Standby to Cold Shutdown," STS AE-201, "Feedwater Chemical Injection Inservice Valve Test," GEN 00-002, "Cold Shutdown to Hot Standby, and any others, if needed,) that jumpers for bypassing the feedwater isolation would not be installed more than once per year.

Response: Regulatory Guide 1.47, Revision 0, 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.

With the approval of the proposed Technical Specification change, the MODE of Applicability would be MODES 1 and 2. Jumpering of the affected circuitry would occur when the isolation of main feedwater coincident with low T_{avg} function and the turbine trip on reactor trip function of the Reactor Trip, P-4 interlock is not required to be OPERABLE. Therefore, conditions a. and c. would indicate that indication for each bypass would not be required.

Revision 0 was published in May 1973 and Condition b. is indicative of refueling outages occurring on a 12 month frequency and that bypass indication be provided if the system was expected to be bypassed or placed in an inoperable status more frequently than every refueling outage (once per year). WCNOC conducts refueling outages on an 18 month frequency and therefore, expects installation of the jumpers would occur on a frequency of 18 months. However, there may be unexpected plant outages that could require the installation of the jumpers. Unexpected plant outages would not invalidate the expectation of occurrence discussed in the Regulatory Guide.

As such, confirmation in plant procedures that the jumpers for bypassing the feedwater isolation would not be installed more than once per year is not required.

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," 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.

3. WCNOC Letter ET 10-0028, "Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," October 13, 2010.

4. 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,)" November 24, 2010

5. Wolf Creek Technical Specifications and Technical Specification Bases, Revision 47.

6. WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Non-Proprietary), "NOTRUMP – A Nodal Transient Small Break And General Network Code," August 1985.

7. WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Non-Proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

8. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

9. WCAP-7907-P-A, "LOFTRAN Code Description," Burner, T.W.T., et al., April 1984.

10. Letter from C. O. Thomas, USNRC to E. P. Rahe, Westinghouse, "Acceptance for Referencing of Licensing Topical Reports: WCAP-7907(P)/(NP) "LOFTRAN Code Desctription"; WCAP-7909(P)/(NP), as Superseded by WCAP-8843(P)/WCAP-8844(NP), "MARVEL – A Digital Computer Code for Tansient Analysis of a Multiloop PWR System"," July 29, 1983.