UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 30, 2011

Mr. Matthew W. Sunseri
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: REVISE TABLE 3.3.2-1 OF TECHNICAL SPECIFICATION 3.3.2, "ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS) INSTRUMENTATION" (TAC NO. ME3762)

Dear Mr. Sunseri:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 194 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 13, 2010, as supplemented by letters dated October 13 and December 21, 2010, and January 18, 2011.

The amendment revises TS Table 3.3.2-1, Function 8.a (Reactor Trip, P-4) by adding footnote (m) to identify the enabled functions and the applicable modes for the Reactor Trip, P-4 interlock function.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Balwant K. Singal

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures:
1. Amendment No. 194 to NPF-42
2. Safety Evaluation

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 194
License No. NPF-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Wolf Creek Generating Station (the facility) Renewed Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated April 13, 2010, as supplemented by letters dated October 13 and December 21, 2010, and January 18, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-42 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 194, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: March 30, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 194

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Renewed Facility Operating License No. NPF-42 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are provided to maintain document completeness.

Renewed Facility Operating License

REMOVE	INSERT
4	4

Technical Specifications

REMOVE	INSERT
3.3-35	3.3-35

(5) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Operating Corporation is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 194, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Kansas Gas & Electric Company and Kansas City Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Environmental Qualification (Section 3.11, SSER #4, Section 3.11, SSER #5)*

Deleted per Amendment No. 141.

*The parenthetical notation following the title of many license conditions denotes the section of the supporting Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

Table 3.3.2-1 (page 5 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 35.5% of instrument span
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4[m]	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1979 psig

(a) The Allowable Value defines the Limiting Safety System Settings. See the Bases for the Trip Setpoints.

(m) The functions of the Reactor Trip, P-4 interlock required to meet the LCO are:
- Trips the main turbine – MODES 1 and 2
- Isolates MFW with coincident low T_{avg} – MODES 1 and 2
- Allows manual block of the automatic reactuation of SI after a manual reset of SI – MODES 1, 2, and 3
- Prevents opening of MFIVs if closed on SI or SG Water Level – High High – MODES 1, 2, and 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 194 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated April 13, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101100391), as supplemented by letters dated October 13 and December 21, 2010 and January 18, 2011 (ADAMS Accession Nos. ML102920142, ML103630045, and ML110250269, respectively), Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (TSs, Appendix A to Renewed Facility Operating License No. NPF-42) for the Wolf Creek Generating Station (WCGS).

The proposed amendment would revise TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." Specifically, the amendment would revise TS Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation," by adding a footnote to Function 8.a concerning the Reactor Trip P-4 ESFAS interlock. The new footnote (m) would specify the required functions and the applicable modes for the Reactor Trip, P-4 Interlock Function for WCGS.

The supplemental letters dated October 13 and December 21, 2010, and January 18, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 15, 2010 (75 FR 33844).

2.0 REGULATORY EVALUATION

The Commission's regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications." This regulation requires that the TSs include items in five categories. These categories include (1) safety limits, limiting safety system settings, and limiting control setting, (2) limiting conditions for operation (LCOs), (3) surveillance requirements, (4) design features, and (5) administrative controls.

The regulations in 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether an LCO is required to be included in TSs. These criteria are:

(A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

General Design Criterion (GDC) 10, "Reactor design," of Appendix A to 10 CFR Part 50 requires that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences [AOOs].

GDC 13, "Instrumentation and control," of Appendix A to 10 CFR Part 50 requires that appropriate controls be provided to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems within prescribed operating ranges.

GDC 15, "Reactor coolant system design," of Appendix A to 10 CFR Part 50 specifies that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

GDC 19, "Control room," of Appendix A to 10 CFR Part 50 requires, in part, that

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.

GDC 21, "Protection system reliability and testability," of Appendix A to 10 CFR Part 50 requires, in part, that

> The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

GDC 50, "Containment design basis," of Appendix A to 10 CFR Part 50 states, in part, that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 6.2.1.2, "Subcompartment Analysis," Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," and Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," dated March 2007, provides review guidance for the postulated effects of pipe breaks.

In 10 CFR 50.55a(h), "Protection and safety systems," the NRC requires compliance with Institute of Electrical and Electronics Engineers (IEEE) Standard 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet, dated January 30, 1995. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999 (WCGS's construction permit was issued by the NRC on May 17, 1977), the applicant/licensee may elect to comply with the requirements in IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." Both, IEEE Standard 603-1991 and IEEE Standard 279-1971, list requirements with regard to a bypassed and inoperable status indication for safety systems. In addition, Criterion XIV, "Inspection, Test, and Operating Status," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, requires that measures "be established for indicating the operating status of structures, systems, and components of the nuclear power plant..., such as by tagging valves and switches, to prevent inadvertent operation." The provisions of 10 CFR 50.34(f)(2)(v) also require an automatic indication of the bypassed and operable status of safety systems.

The applicable industrial standards contain the following guidance:

IEEE Standard 603-1991 defines a "safety function" as

> One of the processes or conditions (for example, emergency negative reactivity insertion, post-accident heat removal, emergency core cooling, post-accident radioactivity removal, and containment isolation) essential to maintain plant parameters within acceptable limits established for a design basis event.

IEEE Standard 609-1991 also defines a "division" as

> The designation applied to a given system or set of components that enable the establishment and maintenance of physical, electrical, and functional independence from other redundant sets of components.

IEEE Standard 279-1971 states, in part, that

> The nuclear power generating station protection system encompasses all electric and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating those signals associated with the protective function. These signals include those that actuate a reactor trip and that, in the event of a serious reactor accident, actuate engineered safety features, such as containment isolation, core spray, safety injection, pressure reduction, and air cleaning.

IEEE Standard 279-1971 defines "protective function" as

> A protective function is the sensing of one or more variables associated with a particular generating station condition, signal processing, and the initiation and completion of the protective action at values of the variables established in the design basis.

Clause 16 of IEEE Standard 338-1987, Section 5, "Design Requirements," states that

> Indication should be provided in the control room if a portion of the safety system is inoperable or bypassed. Systems that are frequently placed in a bypass or inoperative condition for the purposes of testing should have automatic indication.

Regulatory guidance for addressing these regulations is provided in NRC Regulatory Guide 1.47, Revision 1, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," February 2010 (ADAMS Accession No. ML092330064). This guidance provides for key criteria to be met for the design of such bypassed and inoperable status indication systems. Specifically, the RG 1.47, Revision 1, Regulatory Position C.1, states that,

> Administrative procedures should be supplemented by an indication system that automatically indicates, for each affected safety system or subsystem, the bypass or deliberately induced inoperability of a safety function and the systems actuated or controlled by the safety function. Provisions should also be made to allow the operations staff to confirm that a bypassed safety function has been properly returned to service.

The previous revision of this regulatory guide, RG 1.47, May 1973, which was in effect when the WCGS received its operating license in June of 1985, contained guidance for establishing which types of plant conditions should be depicted on such an indication system, since it is recognized that automatic indication of inoperability or a bypassed condition is not feasible for all the

possible means by which safety-related systems could be completely or partially rendered inoperative. RG 1.47, states that,

> A practical indicating system covering a wide range of commonly expected conditions ... could be designed if it included provisions for automatic indication of each bypass or deliberately induced inoperable condition that meets all three of the following guidelines:
>
> 1. The bypass or inoperable condition affects a system that is designed to perform automatically a function that is important to the safety of the public;
>
> 2. The bypass will be utilized by plant personnel or the inoperable condition can reasonably be expected to occur more frequently than once per year; and
>
> 3. The bypass or inoperable condition is expected to occur when the affected system is normally required to be operable.
>
> Such a design is considered practical because: (1) appropriate emphasis on testability early in the design process can reduce to a minimum the number of bypasses that are needed for frequent activities such as testing, and (2) activities such as modification, repair, and maintenance either are conducted infrequently or can be restricted to times when plant conditions do not require the affected system to be available.

3.0 TECHNICAL EVALUATION

3.1 Background

A plant uses a protection system to initiate a unit shutdown or an engineered safety feature actuation in accordance with the plant design. The reactor trip P-4 interlock is one of the engineered safety feature actuation system (ESFAS) instrumentation functions. The P-4 interlock function initiates on a reactor trip breaker opening. In its letter dated April 13, 2010, the licensee identified the associated functions of the P-4 interlock currently listed in the TS Bases as:

- Trips the main turbine;
- Isolates MFW [main feedwater] with a coincident low T_{avg};
- Allows manual block of the automatic re-actuation of SI [safety injection] after a manual reset of SI; and
- Allows arming of the steam dump valves and transfers the steam dump from the load rejection T_{avg} controller to the plant trip controller; and
- Prevents opening of the MFW isolation valves if they were closed on SI or SG [steam generator] Water Level – High High.

The licensee further discusses that during MODE 3, plant personnel perform surveillance testing activities (e.g., rod drop testing, cross calibration of the wide and narrow range resistance temperature detectors, which cycles the reactor trip breakers). Reactor trip breaker opening in turn enables the reactor trip P-4 interlock, which actuates balance of plant system components. The licensee considers these actuations an unwarranted and unnecessary impact concurrent with startup operations. WCNOC proposes a change to the existing TS requirement in order to prevent unnecessary cycling of the main feedwater isolation valves (MFIVs), cycling of auxiliary feedwater (AFW) system components, and tripping of the main turbine.

The licensee's TS define the modes of operation as MODE 1 – Power Operation, MODE 2 – Startup, MODE 3 – Hot Standby [average reactor coolant system (RCS) temperature ≥ 350 degrees Fahrenheit (°F)], MODE 4 - Hot Shutdown [average RCS temperature < 350 °F and >200 °F), and MODE 5 – Cold Shutdown (average RCS temperature ≤ 200 °F). The licensee proposed to relocate the above discussed P-4 interlock functions with changes to TS Table 3.3.2-1 as the new added footnote (m) to Function 8.a.

New footnote (m) would state:

> The functions of the Reactor Trip, P-4 interlock required to meet the LCO are:
>
> - Trips the main turbine - MODES 1 and 2
> - Isolates MFW with coincident low T_{avg} - MODES 1 and 2
> - Allows manual block of the automatic reactuation of SI after a manual reset of SI - MODES 1, 2, and 3
> - Prevents opening of the MFIVs if closed on SI or SG Water Level - High High - MODES 1, 2, and 3.

The NRC staff concludes that the licensee's approach of relocating the P-4 functions from the TS Bases to TS Table 3.3.2-1 is acceptable.

3.2 Technical Specification Changes

Currently, TS Table 3.3.2-1, Function 8.a requires the ESFAS P-4 interlock be operable in MODES 1, 2, and 3. The current TS Table 3.3.2-1 does not delineate each individual function that the P-4 interlock performs. The licensee proposes to modify Table 3.3.2-1 to identify each individual function of the P-4 interlock, and identify the applicable mode for each function. In doing so, the licensee proposes the following changes to the existing specification:

- removal of the P-4 function of the turbine trip in MODE 3,
- removal of the P-4 function of the MFW isolation in MODE 3,
- removal of the P-4 function related to the steam dump function in MODES 1, 2 and 3.

The licensee requests the changes to the TS in order to avoid unnecessary cycling of the MFIVs, tripping of the main turbine generator, and cycling of the AFW components. The NRC staff evaluated each of the proposed changes separately.

3.3 NRC Staff Evaluation

3.3.1 Turbine Trip

In the WCGS Updated Safety Analysis Report (USAR) Section 7.2.1.1.1, "Functional Performance Requirements," the licensee describes the reactor trip system initiating a turbine trip signal whenever a reactor trip occurs in order to prevent insertion of positive reactivity, which would result from an over cooling of the RCS if steam flow continued to the main turbine. The main turbine is typically placed in service above 15 percent power. Therefore, the P-4 interlock would be required to be in service at that time to provide the turbine trip upon a reactor trip. However, in MODE 3, the main turbine is tripped and not in service. Therefore, MODE 3 would not be an applicable mode for the turbine trip function of the P-4 interlock. Hence, the only applicable modes for the turbine trip function of P-4 would be MODES 1 and 2. The licensee proposes to maintain the turbine trip function applicable for MODES 1 and 2, for the P-4 interlock, and to remove the applicability for MODE 3.

The licensee bounds design basis events while in MODE 3 by its analyses of the plant in MODES 1 and 2. The NRC staff performed an evaluation of the licensee's proposal to eliminate the requirement for the turbine trip function performed by the P-4 interlock while in MODE 3, by reviewing the individual accident analyses described in the WCGS USAR Chapter 15.0, "Accident Analysis," and the containment integrity analysis described in USAR Chapter 6.2, "Containment Systems." The results of the assessment are documented in Sections 3.3.6 and 3.3.7 of this safety evaluation (SE). Based on the results of the evaluations described in Sections 3.3.6 and 3.3.7 of this SE, the NRC staff concludes that the licensee's proposed TS change is acceptable to exclude MODE 3 applicability for the main turbine trip function for the P-4 interlock. The licensee will still maintain MODES 1 and 2 as applicable for the main turbine trip function as required by 10 CFR 50.36(c)(2)(ii).

In reviewing the criterion for inclusion into the TSs, per 10 CFR 50.36, the NRC staff agrees with the licensee's assessment that the turbine trip function does not serve any of the functions delineated under 10 CFR 50.36(c)(2)(ii) while in MODE 3. The NRC staff concludes that the turbine trip function will continue to provide its designed safety function under the proposed TS change during the modes of operation when it is required to perform a safety function. Therefore, the turbine trip function of P-4 is only applicable for Function 8.a of TS Table 3.3.2-1 during MODES 1 and 2.

3.3.2 MFW Isolation

The MFW system is designed to provide sufficient feedwater to the SGs to support normal operations. It is also designed to limit the amount of feedwater being supplied in certain conditions in order not to increase the severity of certain accidents.

One of the safety functions for the P-4 interlock is to isolate MFW in order to prevent an excessive RCS cooldown. When the reactor trip breakers are opened, the P-4 interlock will signal for MFW isolation when T_{avg} decreases below 564 °F. Thus, the MFW isolation function of the P-4 interlock provides protection against an excessive cooldown of the RCS. In order to satisfy the licensing basis accident analyses, automatic feedwater isolation must be provided on receipt of a SG Water Level - High High signal, whenever the MFW system is in service.

In MODES 1 and 2, the reactor is critical and MFW is supplying water to the SGs for heat removal. Therefore, MODES 1 and 2 are applicable modes for the P-4 interlock MFW isolation function. However, in MODE 3, the reactor is not critical. The RCS is kept hot and pressurized from heat from the reactor coolant pumps and pressurizer heaters. There may be some contribution from the fuel due to decay heat, but the amount is not significant during plant startup. Therefore, while in MODE 3 when the reactor trip breakers open, there is no change in heat production in the RCS.

In MODE 3, the MFW pumps are not in service. A separate motor-driven startup feedwater pump is normally used to provide feedwater during low power operations until there is adequate steam production to support operation of the MFW pumps. Since only the startup pump would be in service during MODE 3, only the effects of the startup pump would have to be considered in determining whether the P-4 interlock signal is a required safety function to isolate MFW flow path in order to avoid insertion of reactivity from a subsequent RCS cooldown.

Design basis events crediting the isolation of MFW are analyzed with the plant at hot zero power, full power (FP), and partial power conditions. The licensee bounds design basis events while in MODE 3 by its analyses of the plant in MODES 1 and 2. The NRC staff performed an evaluation of the licensee's proposal to eliminate the requirement for the MFW isolation function performed by the P-4 interlock while in MODE 3, by reviewing the individual accident analyses described in the WCGS USAR Chapter 15.0 and the containment integrity analysis described in USAR Chapter 6.2. The results of the assessment are documented in Sections 3.3.6 and 3.3.7 of this SE.

Based upon a satisfactory assessment of the impact on the accident analysis, the NRC staff concludes the licensee's proposed TS change is acceptable to exclude MODE 3 applicability for the MFW isolation function for the P-4 interlock. The licensee will still maintain MODES 1 and 2 as applicable for the MFW isolation function as required by 10 CFR 50.36(c)(2)(ii).

3.3.3 Steam Dump Control

In its letter dated October 13, 2010, the licensee delineates that upon a reactor trip, the reactor trip breakers open, enabling the P-4 interlock to arm the steam dump valves and transfer steam dump control from the load rejection T_{avg} controller to the plant trip controller, which in turn defeats the load rejection controller. In the WCGS USAR Chapter 7.7, "Control Systems Not Required for Safety," the licensee's evaluation of the steam dump system concluded that the steam dump control system is not required or credited in any plant safety analysis. In reviewing the criterion for inclusion into the TSs, the licensee believes the instrumentation utilized to initiate transfer to the plant trip steam dump controller does not serve any of the functions delineated under Criterion 3 of 10 CFR 50.36(c)(2)(ii). Therefore, the steam dump function of

P-4 does not warrant inclusion into the TSs, and can be excluded from Function 8.a of TS Table 3.3.2-1.

The NRC staff reviewed the licensee's proposal to exclude the transfer of the steam dumps to the plant trip controller as a function from being required in MODES 1, 2, and 3 as part of the P-4 interlock. The staff agrees that the transfer of steam dumps to the plant controller upon a reactor trip is not a safety function credited to mitigate a design basis event in the plant's licensing basis. Therefore, the staff concludes that the licensee's proposal to exclude the steam dump transfer function from TS Table 3.3.2-1, Function 8.a, based on not meeting the criteria outlined in 10 CFR 50.36 (c)(2)(ii), is acceptable.

3.3.4 Manual Block of SI

Currently, the TSs require the operability of the P-4 interlock for allowing the manual block of the automatic reactuation of SI after a manual reset of SI in MODES 1, 2, and 3. The licensee proposes to maintain the same requirement for operability in MODES 1, 2, and 3. Since there is no change proposed, besides identifying the function, no NRC staff evaluation is required.

3.3.5 Reopening of the Main Feedwater Isolation Valves

Currently, TS require operability of the P-4 interlock function to prevent opening of the MFIVs if they were closed on SI or SG Water Level – High High in MODES 1, 2, and 3. The licensee proposes to maintain the same requirement for operability in MODES 1, 2, and 3. Since there is no change proposed, besides identifying the function, no staff evaluation is required.

3.3.6 WCGS USAR Chapter 15.0 Accident Analysis Review

The NRC staff performed an evaluation of the proposed changes by reviewing the WCGS USAR Chapter 15.0 analyses. The review was based on the information provided by the licensee in its letters dated December 21, 2010, and January 18, 2011. The results of the evaluation are as follows:

3.3.6.1 Feedwater System Malfunctions

Feedwater system malfunctions can result in a decreased feedwater temperature or an increased feedwater flow. The events decrease the RCS temperature, which, in the presence of the negative moderator coefficient of reactivity, causes power to increase. The analyses of the feedwater malfunction initiated from MODES 1 and 2 are discussed in the WCGS USAR Section 15.1.1, "Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature," and Section 15.1.2, "Feedwater System Malfunctions that Result in an Increase in Feedwater Flow," for a decreased feedwater temperature event and an increased feedwater flow event, respectively.

A decreased feedwater temperature event is caused by a malfunction in the feedwater control system which affects the feedwater heaters, or a failure of a bypass valve that diverts flow around a portion of the feedwater heaters. The event is limiting at FP operation (MODE 1) since this condition will have maximum change in feedwater temperature, leading 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. When the plant is in MODE 2, the feedwater heaters are not 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. Since the decreased feedwater temperature event is not credible in MODE 3 and below, it is not affected by the removal of the proposed P-4 functions in MODE 3 from the TS and the current MODE 1 analysis presented in the WCGS USAR Section 15.1.1 remains bounding.

An increased feedwater flow event is caused by a malfunction in the feedwater control system, which affects the feedwater control valve (FCV), or a failure of the FCV. In the WCGS USAR Section 15.1.2, this event is analyzed at both FP (MODE 1) and zero power (MODE 2). The MODE 2 analysis bounds MODE 3 and below because the initial conditions in MODE 2 will cause a more severe cooldown resulting from the failed FCV. In addition, the MODE 2 analysis does not credit any protective features including the P-4 functions. Therefore, an increased feedwater flow event initiating from MODE 3 and below is not affected by the proposed removal of the P-4 functions in MODE 3 out of the TSs and the existing MODE 2 analysis presented in the WCGS USAR Section 15.1.2 remains bounding.

3.3.6.2 Excessive Load Increase

Excessive load increase event is caused by a steam dump control malfunction or turbine throttle valve control failure. This event decreases the RCS temperature, which causes an increased power in the presence of the negative moderator coefficient of reactivity. Typically, the RCS is designed to accommodate a 10 percent step load in the range of 15 percent to 100 percent of FP without actuating the reactor protection system. Increases in steam flow below 15 percent power or in excess of 10 percent of initial load are addressed by the steamline break (SLB) events (WCGS USAR Section 15.1.5, "Steam System Piping Failure"). In MODE 3, since the turbine is being turned on the turning gear, the steam demand is essentially zero with minimal steam being supplied to warm up in preparation to ente MODE 2. Therefore, the amount of the steam release from the SGs is based upon the flow areas of the turbine admission valve. Since the SGs have integral flow restrictors, any steam flow path, regardless of location, would have the same effect on the RCS as a break corresponding to the throat area of the restrictors, therefore, the MODE 2 SLB analysis bounds an excessive load increase event initiating from MODE 3. Since the licensee analyzed the SLB events to meet the specified acceptable fuel design limits that are acceptance criteria of AOOs, and the excessive load increase events are AOOs, the NRC staff concludes that the licensee's approach of using the SLB analysis to bound the excessive load increases event is acceptable. Also, the MODE 2 SLB analysis does not credit the P-4 functions including feedwater isolation and turbine trip. Therefore, an excessive load increase event occurring in MODE 3 and below is not affected by the proposed TS changes and the MODE 1 excessive load increase analysis presented in the WCGS USAR Section 15.1.3, "Excessive Increase in Secondary Steam Flow," remains bounding.

3.3.6.3 Inadvertent Opening of an SG Atmospheric Relief or Safety Valve

The inadvertent opening of an SG relief or safety valve event creates a depressurization of the SG secondary side with an effective opening size that is within the spectrum of break sizes considered in the analysis of an SLB event, and thus, is bounded by the event discussed in the WCGS USAR Section 15.1.5. Since the licensee analyzed the SLB events to meet the specified acceptable fuel design limits that are acceptance criteria of AOOs, and the inadvertent opening of an SG relief or safety valve events are AOOs, the NRC staff concludes that the licensee's approach of using the SLB analysis to bound the inadvertent opening of an SG relief or safety valve events is acceptable. In addition, the SLB event discussed in USAR Section 15.1.5 does not credit the P-4 functions, including feedwater isolation and turbine trip. Therefore, the accidental depressurization of the SG secondary side is not affected by the proposed TS changes and the analysis presented in the WCGS USAR Section 15.1.4, "Inadvertent Opening of a Steam Generator Atmospheric Relief or Safety Valve," remains bounding.

3.3.6.4 Steam System Break

The steam release from a SLB causes a decrease in the RCS temperature, and in the presence of a negative moderator temperature coefficient, the decreased RCS temperature results in a positive reactivity addition. If the resulting positive reactivity is greater than the negative reactivity from the inserted control worth and from the borated water from SI system, the core may return to criticality for a post-trip core. In the WCGS USAR Section 15.1.5, the licensee analyzed the current licensing basis SLB event at MODE 2 conditions with a zero parts per million (ppm) boron concentration to bound all lower modes of operation. Also, the MODE 2 SLB analysis does not credit the P-4 functions including feedwater isolation and turbine trip. Therefore, a SLB occurring in MODE 3 and below is not affected by the proposed TS changes and the MODE 2 SLB analysis presented in the WCGS USAR Section 15.1.5 remains bounding.

In the MODE 2 SLB analysis, the licensee assumed that, in accordance with the requirement Items 1.d and 1.e in Table 3.3.2-1 of the licensee's TSs, the pressurizer pressure-low signal would actuate SI to provide borated emergency core cooling system (ECCS) flow to limit the power excursion and, the steamline pressure-low signal would actuate main steamline isolation to limit the secondary cooldown of the RCS.

However, note (b) of Items 1.d and 1.e in TS Table 3.3.2-1 permits the blocking of the low pressurizer pressure and/or steamline pressure-low signals following receipt of the P-11 permissive to allow the plant to initiate RCS cooldown without the initiation of SI. For this configuration, the main steamline isolation function is provided by the steamline pressure negative rate-high signal (Item 4.e. (2) in TS Table 3.3.2-1). 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. With no borated ECCS flow supplied to the core, a return-to-criticality and subsequent power increase in the core could result. The licensee performed an analysis for a SLB occurring in MODE 3 below the P-11 setpoint and confirmed that the MODE 3 SLB below P-11 with no feedwater isolation due

to SI being blocked would be bounded by the licensing basis MODE 2 SLB analysis presented in the WCGS USAR Section 15.1.5. The following discussion of the SLB analysis occurring in MODE 3 is based on the licensee's analysis described in its letter dated December 21, 2010.

The licensee performed the core response analysis for a postulated SLB occurring in MODE 3 below the P-11 setpoint using the NRC-approved code, LOFTRAN. The licensee used the following assumptions to maximize the cooldown effects resulting from the SLB: (1) the reactor is tripped and the most reactive rod cluster control assembly (RCCA) is stuck in its fully withdrawn position; (2) the most negative reactivity coefficients, corresponding to the end of life (EOL) in a fuel cycle, are assumed; and (3) offsite power is assumed to be available. To reflect plant conditions that exist at the time of a SLB is occurring in MODE 3 below the P-11 setpoint that does not generate a SI actuation of borated ECCS flow, the licensee used the following assumptions:

- The pumped SI is not available and only the accumulators are available for injection of the highly borated water.

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

- Steamline isolation is provided by the steamline pressure negative rate - high trip (in accordance with the requirement of Item 4.e. (2) in TS Table 3.3.2-1). Steamline isolation is assumed in all loops except the faulted steamline. Steam release from the three intact SGs is terminated upon receipt of the signal to isolate and valve closure.

- The initial boron concentration in the core is 386 ppm, which is corresponding to the required boron concentration that meets the minimum shutdown margin at no load temperature of 557 °F at EOL.

The licensee used the values of the plant's initial conditions in the analysis based on the original licensing basis analysis for the hot zero-power (MODE 2) SLB accident. The values of the initial plant conditions are specified in the licensee's letter dated December 21, 2010, for the key parameters including power level, reactor coolant pump heat output, RCS average temperature, RCS pressure, RCS flow, pressurizer water volume, feedwater enthalpy, SG fluid mass, and reactor trip occurring at initiation of the SLB.

With the above assumptions, the licensee performed a sensitivity study for SLB cases including:

(1) a MODE 3 SLB case with a double-ended steam line rupture, assuming no SI and no feedwater isolation while taking credit of the steam line pressure negative rate – high trip function to isolate steam line,

(2) a MODE 3 SLB case with a double-ended steam line rupture, assuming no SI, no feedwater isolation, and no steam line isolation,

(3) a MODE 3 SLB case with a steam line break (SLB) size of 0.2 ft^2, assuming no SI and no feedwater isolation while taking credit of the steam line pressure negative rate – high trip function to isolate steam line, and

(4) a MODE 3 SLB case with a SLB size of 0.2 ft^2 , assuming no SI, no feedwater isolation, and no steam line isolation.

The results showed that for the double-ended steamline rupture cases, the power excursion is calculated to reach a range of peak value of about 13 to 21 percent of the FP, and for the small SLB cases, the calculated peak power ranges from about 7 to 9 percent of the FP.

The results also showed that the peak power (21 percent of the FP) for Case 2 is greater than that (13 percent of the FP) of Case 1. Since Item 4.e.(2) in TS Table 3.3.2-1 allows the licensee to credit steamline isolation on a signal from the steamline pressure negative rate - high trip, Case 1 above is considered as the credible limiting MODE 3 SLB case. A comparison of Case 1 with the MODE 2 SLB of the WCGS USAR Section 15.1.5 shows that Case 1 would be bounded by the licensing basis MODE 2 SLB analysis, which shows the peak power of about 17 percent of the FP.

Based on the above discussion, the NRC staff concluded that the analyses of the MODE 3 SLB events are acceptable, since (1) the NRC-approved code is used, (2) the assumptions used are conservative, resulting in a higher peak power, and (3) the peak power for the credible limiting MODE 3 SLB (Case 1) is bounded by the MODE 2 SLB analysis in the WCGS USAR Section 15.1.5.

3.3.6.5 Decrease in Heat Removal by the Secondary System

The consequences of a decrease in heat removal by the secondary system are discussed in the WCGS USAR Section 15.2, "Decrease in Heat Removal by the Secondary System." The category of the events includes:

(1) Loss of external electrical load (Section 15.2.2),

(2) Turbine trip (Section 15.2.3),

(3) Inadvertent closure of main steam isolation valves (Section 15.2.4),

(4) Loss of condenser vacuum and other events resulting in turbine trip (Section 15.2.5),

(5) Loss of offsite power (LOOP) (Section 15.2.6, "Loss of Nonemergency AC [Alternating Current] Power to the Station Auxiliaries"),

(6) Loss of normal feedwater (LONF) (Section 15.2.7), and

(7) Feedwater system pipe break (Section 15.2.8).

These events are characterized by rapid reductions in heat removal capability of the SGs. The loss of heat removal capability results in a rapid rise in the SGs' secondary system pressure and

temperature, and a subsequent increase in the RCS pressure and temperature. Reactor trip and actuation of primary and secondary safety valves mitigate the effects of the primary-to-secondary power mismatch during these events. The severity of these events is increased as the primary-to-secondary power mismatch is increased. The occurrence of the event at FP (MODE 1) produces a higher and more rapid power than that at lower power or operations below MODE 2 because of a higher initial power and a higher decay heat level. Therefore, the worst cases are the events initiating from MODE 1 conditions.

For the above events (1) through (4), the turbine trip is the limiting event due to the more rapid loss of steam flow during this 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 affect the events. Thus, the MODE 1 analysis presented in the WCGS USAR Section 15.2.3, turbine trip, remains bounding.

The above event (5), a LOOP event, is identical to event (6), a LONF event, except that the LOOP event is assumed to occur following a reactor trip. The immediate consequence of the events is a reduction in SG inventory which, will result in a reactor trip and AFW actuation on a low-low SG level signal. Following a reactor trip, the decay heat (and reactor coolant pump 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. The acceptance criterion for the events is to prevent the RCS overpressurization and pressurizer overfill from occuring. Although the WCGS USAR assumes a turbine trip following reactor trip, delaying its actuation is a benefit to the results of the analyses since the turbine would provide an additional heat removal path. The additional energy removed through the turbine would cause a less severe RCS overpressurization and increase the time to pressurizer-overfill. Therefore, the proposed removal of the P-4 functions in MODE 3 out of the TSs 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.

Event (7), 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 SGs. If the break occurs between the check valve and the SG, the break cannot be isolated and the SG inventory will be lost. Depending on the location of the AFW piping, this break could also preclude subsequent addition of AFW to the affected SG. 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. The FLB event is limiting in MODE 1 because of the high decay heat level immediately following a reactor trip. Although the analysis assumes a turbine trip following reactor trip, delaying its actuation is a benefit to the analysis since the turbine would provide an additional heat removal path. The WCGS USAR analysis does not model feedwater isolation since MFW is conservatively assumed to become unavailable at event initiation and the feedwater line check valves are credited to prevent reverse flow from the intact SGs. The minimum AFW flow rates to the intact SGs 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 SG spills out of the break. The flow from the turbine-driven AFW pump is also adjusted based upon this assumption. Therefore, the proposed removal of the P-4 functions in MODE 3 out of the TSs would either be a benefit to the FLB event or have no effect on the analysis and

thus, the current MODE 1 analysis presented in the WCGS USAR Section 15.2.8 remains bounding.

3.3.6.6 Decrease in Reactor Coolant Flow

The consequences of a decrease in RCS flow events are discussed in the WCGS USAR Section 15.3, "Decrease in Reactor Coolant System Flow Rate." The applicable events are:

(1) Partial Loss of Forced Reactor Coolant Flow (Section 15.3.1),

(2) Complete Loss of Forced Reactor Coolant Flow (Section 15.3.2),

(3) Reactor Coolant Pump Shaft Seizure (Locked Rotor) (Section 15.3.3); and

(4) Reactor Coolant Pump Shaft Break (Section 15.3.4).

For these events, a loss of RCS flow can reduce heat removal from the primary-to-secondary system and cause a heat-up in the RCS. The heat-up results in an increase in the RCS pressure and a decrease in the departure from nucleate boiling ratios (DNBRs). The occurrence of the events at MODE 1 produces a higher and more rapid heat-up than at lower power or operations below MODE 2. Therefore, the events initiating from MODE 1 are the limiting, resulting in a maximum peak RCS pressure and a minimum DNBR. The critical time frame of interest, in which the maximum peak RCS pressure and the minimum DNBR occur, is the first 5 or 6 seconds after event initiation and before any of the P-4 functions can affect the event. Thus, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the decrease in reactor coolant flow events and the current MODE 1 analyses presented in the WCGS USAR Sections 15.3.1 through 15.3.4 remain bounding.

3.3.6.7 Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical or Low Power Startup Condition

The uncontrolled RCCA bank withdrawal from subcritical (RWFS) event causes power to increase, which results in a decrease in DNBR. The analysis of this event for a subcritical condition (MODE 2) is discussed in the WCGS USAR Section 15.4.1, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition." The event is terminated by a reactor trip on the power range neutron flux - low trip function. In MODE 3 and below, protection is provided by the source range neutron flux trip function as the power range neutron flux - low trip function is not required to be operable. The MODE 2 analysis bounds MODE 3 and below because the source range neutron flux trip function will trip the reactor before any appreciable power is generated. Because none of the P-4 functions are modeled in the RWFS analysis, they are not required to mitigate the event. Therefore, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the RWFS event and the current MODE 2 analyses presented in the WCGS USAR Section 15.4.1 remain bounding.

3.3.6.8 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

As discussed in the WCGS USAR Section 15.4.2, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power," the continuous uncontrolled RCCA bank withdrawal at power

(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 affected by the proposed removal of the P-4 functions in MODE 3 out of the TSs and the analysis presented in the WCGS USAR Section 15.4.2 remains bounding.

3.3.6.9 Rod Cluster Control Assembly Misoperation

The following events are considered as RCCA misalignment events:

(1) One or more dropped RCCAs within the same group or dropped RCCA bank (dropped rod);

(2) Statically misaligned RCCA; and

(3) Withdrawal of a single RCCA.

These events results in core radial power distribution perturbations, which result in a decrease in the calculated DNBRs. The above events (1) dropped rod, and (2) statically misaligned RCCA, are not credible in MODE 3 and below because all of the control and protection RCCA banks are fully inserted into the core. For event (3), withdrawal of a single RCCA, the analysis discussed in the WCGS USAR Section 15.4.3, "Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)," does not model any of the P-4 functions. Thus, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the RCCA misalignment events and the current analyses presented in the WCGS USAR Section 15.4.3 remain bounding.

3.3.6.10 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

Starting of an idle reactor coolant pump (RCP) without bringing the inactive loop hot-leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which, in the presence of a negative moderator temperature coefficient of reactivity, would cause a power increase and a DNBR decrease. The analysis of the startup of an inactive RCP event is presented in the WCGS USAR Section 15.4.4, "Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature." The analysis uses initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure resulting in minimum initial margin to departure from nucleate boiling. These values are consistent with maximum steady state power level allowed with three loops in operation. The high initial power gives the highest temperature difference between the core inlet temperature and the inactive loop hot-leg temperature. In MODE 3 and below, the licensee estimated in its letter dated December 21, 2010, that for the worst case with the maximum difference in temperature existing between the SG 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. This ensures that the startup of an idle RCP in MODE 3 cannot produce a return-to-power from the MODE 3 condition, assuring no fuel failure to occur. Therefore, the proposed removal of the P-4 functions in MODE 3 from the TSs does not affect the startup of an inactive RCP event and the current analysis presented in the WCGS USAR Section 15.4.4 remains bounding.

3.3.6.11 Chemical and Volume Control System (CVCS) Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant

CVCS malfunctions result in decrease in boron concentration in the reactor coolant. The resulting boron dilution event is analyzed for MODES 1 through 6. Since the analyses do not credit the P-4 functions including feedwater isolation and turbine trip, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the results of the analyses. Therefore, the current analyses presented in the WCGS USAR Section 15.4.6, "Chemical and Volume Control System Malfunction that Results In a Decrease in the Boron Concentration in the Reactor Coolant," remain bounding.

3.3.6.12 Inadvertent Loading and Operation of a Fuel Assembly in Improper Position

Fuel loading errors may result in a core power-shape exceeding its design values. The core power-shape changes may result in a decrease in the calculated DNBRs. This event is analyzed under steady-state conditions at limiting time frames throughout core life without taking credit of reactor trip and any of the P-4 functions. Thus, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the event and the analysis presented in the WCGS USAR Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in Improper Position," remains bounding.

3.3.6.13 Spectrum of Rod Cluster Control Assembly Ejection Accidents

The RCCA ejection event is the result of a mechanical failure of control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. The event will result in 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 FP (MODE 1) and zero-power (MODE 2). 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. In addition, the P-4 functions are not credited in the analysis. Therefore, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the RCCA ejection event and the current analyses presented in the WCGS USAR Section 15.4.8, "Spectrum of Rod Cluster Control Assembly Ejection Accidents," remain bounding.

3.3.6.14 Inadvertent Operation of the Emergency Core Cooling System during Power Operation

An inadvertent ECCS actuation at power event results in an increase in RCS inventory, leading to the potential filling of the pressurizer. The event is limiting in MODE 1 because of the higher decay heat level immediately following reactor trip. In the USAR analysis, an immediate turbine trip on reactor trip is assumed. The turbine trip reduces the primary-to-secondary heat removal, resulting in an increase in RCS temperature and causing the RCS inventory to swell, which, in turn, results 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. The event is terminated by the operators securing the ECCS flow. Therefore, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the inadvertent ECCS actuation at

power event and the current analysis presented in the WCGS USAR Section 15.5.1, "Inadvertent Operation of the Emergency Core Cooling System During Power Operation," remains bounding.

3.3.6.15 CVCS Malfunction That Increases Reactor Coolant Inventory

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, 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. 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 adversely affect CVCS malfunctions that increase the RCS inventory and the current analysis presented in the WCGS USAR Section 15.5.2, "Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," remains bounding.

3.3.6.16 Inadvertent Opening of a Pressurizer Safety or Relief Valve

An accidental depressurization of the RCS event could occur as a result of an inadvertent opening of a pressurizer safety or relief valve. This event is analyzed in the WCGS USAR Section 15.6.1, "Decrease in Reactor Coolant Inventory," with the plant initially at MODE 1 conditions. During the transient, the RCS pressure decreases rapidly. The event initiating from MODE 1 conditions will result in a decrease in DNBRs because of the RCS pressure decrease. For Modes 2 and below, violation of DNB safety limit is not of concern because of low decay heat levels. Therefore, the results of the MODE 1 event analyzed in the USAR bounds that of the event initiating from MODES 2 and below. The event is terminated by operator action by either closing the open valve or closing an isolation valve in the affected path. Although a turbine trip is modeled to occur on reactor trip, the turbine trip is not used to mitigate the event. Thus, the P-4 functions do not provide any mitigating effects for this event. Therefore, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the RCS depressurization event and the current analysis presented in the WCGS USAR Section 15.6.1 remains bounding.

3.3.6.17 Break in Instrument Line or Other Lines from the Reactor Coolant Pressure Boundary that Penetrate Containment

As indicated by the licensee in its letter dated January 18, 2011, a rupture of the letdown line outside of the containment was previously identified as the limiting case regarding radioactivity release during normal operation. The event is limiting in MODE 1 since the break flow will be maximized at FP operation. The analysis in the WCGS USAR Section 15.6.2, "Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment," assumes that the event is terminated by operator action by closing an isolation valve in the affected path within 30 minutes. Based on the calculated maximum leakage rate of 141 gallons per minute (gpm), the analysis shows that the event does not exceed a small fraction of the dose limits of 10 CFR Part 100, "Reactor Site Criteria." The analysis does not

model system transient response, including reactor trip, or any of the P-4 functions. Therefore, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the analysis of this event and the current analysis presented in the WCGS USAR Section 15.6.2 remains bounding.

3.3.6.18 SG Tube Rupture (SGTR)

The steam generator tube rupture (SGTR) event is analyzed to show its consequences of radiological releases are within the applicable dose limits. The radiological releases are calculated based on the mass releases from the SGTR transient analysis and site-specific meteorological parameters. The results of the radiological release analysis will be affected by the calculated steam or water released from the affected SG. Water releases have a significantly greater concentration of radioactive material when compared with that of steam releases and would result in worse radiological releases. As indicated by the licensee in its letter dated January 18, 2011, the WCGS USAR analysis determines the thermal and hydraulic response for the limiting case with respect to SG overfill and water release through SG safety valves, with a consequential failure of the safety valve following water release. The analysis credits operator actions to mitigate the event consequences. The major operator actions include identification and isolation of the affected SG, cooldown and depressurization of the RCS, and termination of SI. These actions are designed to equalize the RCS and affected SG pressures, and thus to terminate the primary to secondary leakage. Although a turbine trip is modeled to occur on reactor trip, it is not used to mitigate the event. Delay or deleting the actuation of turbine would be benefit to the analysis since the turbine would provide an additional heat removal path. The additional energy removal through the turbine would enhance the RCS depressurization and consequentially reduce the primary- to-secondary leakage. Thus, the P-4 function of turbine trip does not provide mitigating effects for this event. Therefore, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the analysis of this event and the current analysis presented in the WCGS USAR Section 15.6.3, "Steam Generator Tube Rupture (SGTR)," remains bounding.

3.3.6.19 LOCA Analysis – Small Break and Large Break

The WCGS USAR Section 15.6.5, "Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," discusses the analyses of the small and large break loss-of-coolant (LOCA) accident. Since the analyses do not credit the reactor trip P-4 interlock functions, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the LOCA accident and the current analysis presented in the WCGS USAR Section 15.6.5 remains bounding.

3.3.6.20 Anticipated Transients Without Scram

An anticipated transient without scram (ATWS) is an anticipated operational occurrence followed by failure of the reactor trip portion of the reactor protection system. 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 MODE 2 and below. Therefore, the proposed removal of the P-4 functions in MODE 3 out of the TSs does not affect the ATWS event and the current analysis

presented in the WCGS USAR Section 15.8, "Anticipated Transients Without Scram," remains bounding.

3.3.7 Containment Integrity

The licensee performed an evaluation of the WCGS USAR Chapter 15.0 accident analysis in order to ensure safe operation of the plant would be maintained with the proposed changes and results of the NRC evaluation are documented in Section 3.3.6 of this SE. In its letter dated October 13, 2010, in response to the NRC staff's request for additional information (RAI) dated August 18, 2010 (ADAMS Accession No. ML102180064), the licensee confirmed that for the WCGS USAR Chapter 6.0, "Engineered Safety Features," safety analyses limiting transients discussed below, either do not rely solely on the P-4 turbine trip or do not model the turbine trip on reactor trip function. The licensee concluded that the WCGS USAR Chapter 6.0 accident analyses remains valid and would bound the results of the corresponding events initiating in MODES 3 and 4 with a combination of the worst single failure consideration, as discussed below.

3.3.7.1 Long-Term (LT)/Short-Term (ST) Loss-of-Coolant Accident (LOCA) Mass & Energy (M&E) (Containment Integrity)

In its model for the LOCA M&E release analyses, the licensee assumed that the turbine trips with the LOOP, which is assumed to occur with reactor trip. By modeling the turbine trip at this time, the licensee ensures that no additional heat removal from the RCS primary side through the SGs will occur, other than from the SG safety valves, and forces the break to be the primary means of heat removal for the RCS, which is conservative.

The intent of the LT LOCA M&E release analysis is to maximize the mass and energy release available to containment through assuming the limiting initial conditions. In its letter dated October 13, 2010, in response to the NRC staff's RAI dated August 18, 2010, the licensee stated that none of the initial conditions, models, or methodology used for the LT LOCA M&E releases analysis become more limiting by implementing the P-4 technical specification change. Based on the above, the NRC staff agrees with the licensee's determination that the current design basis LT LOCA M&E release analysis documented in the WCGS USAR remains valid.

The ST LOCA M&E 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. In its letter dated October 13, 2010, in response to the NRC staff's RAI, the licensee explained that these analyses are performed to ensure that the walls in the immediate proximity of the break location can maintain their structural integrity during the rupture (LOCA) within the region (subcompartment). Due to the short duration of the event, ECCS does not actuate during the time of interest, so the effect from the P-4 interlock functions does not affect the ST LOCA M&E release analyses. Based on the discussion above, the NRC staff agrees with the licensee's determination that the current design basis ST LOCA M&E release analysis documented in the WCGS USAR remains valid.

In the licensee's LOCA M&E release methodology, the turbine trip is modeled at a time to isolate the SGs as early as possible, which traps the energy from the secondary side for the

post-reflood portion of the transient. The LOCA M&E release analyses are performed considering MODE 1 operation, at power normal operating pressure, and temperature, in order to maximize all stored and generated M&E release available. Since the turbine trip is not credited as a mitigating feature in the WCGS USAR Chapter 6.0 LOCA M&E release analyses, the NRC staff has determined that during MODES 3 and 4, conditions are less limiting relative to M&E generation.

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

In its model for the SLB M&E analysis, the licensee assumed a turbine trip coincident with reactor trip. As indicated in the WCGS USAR, the licensee's model for this accident conservatively models the steam release out the break, rather than to the turbine. If the turbine is on or is not isolated, it might provide a diversionary path for some of the mass release through a break or through relief or dump valves to the atmosphere. As such, the turbine trip on reactor trip is not credited in the analysis as a mitigating function.

As discussed above, the turbine trip was found to not give any credit as a mitigating feature in the SLB M&E release or steam release for dose analysis. In its letter dated October 13, 2010, in response to the NRC staff's RAI dated August 18, 2010, the licensee indicated the proposed elimination of the turbine trip function does not adversely impact the analyses for all modes of operation, and further stated:

> 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 [kilopounds mass per hour] 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.

The licensee reviewed its USAR Chapter 6.0 MSLB M&E analysis to ensure the results are bounded in MODES 3 and 4 with a combination of the worst single failure consideration. Based on the licensee's discussion above, the NRC staff concluded that defeating the turbine trip and feedwater isolation functions does not create a concern for a more limiting condition in MODE 3 than has already been analyzed for MODES 1 and 2, and further does not create any new single failure scenarios that need to be considered.

3.3.8 Manual Bypass of P-4 Function to Avoid MFW Isolation and Main Turbine Trip

As stated in Section 3.2 of this SE, the licensee proposes to remove the P-4 function of the MFW isolation by installing temporary jumpers and the main turbine trip function by lifting leads when needed while in MODE 3 of the plant operation. The background information for the proposed change is described below:

On August 19, 2009, WCGS experienced a turbine trip and resulting reactor trip due to a momentary LOOP to the onsite buses. On August 22, 2009, while still in MODE 3 during recovery from this LOOP, the plant operators defeated the feedwater isolation signal (FWIS) on low T_{avg} coincident with P-4 function using plant procedure SYS SB-122, "Enabling/Disabling P-4/Lo T_{avg} FWIS." This procedure was performed to restore MFW flow through the MFIVs (MFIVs) to supply water to the SGs during troubleshooting of the rod drive motor generator sets. The plant TS Bases provide for the bypassing of some ESFAS interlocks to allow flexibility in unit operations. As stated in the WCGS TS Bases page B 3.3.2-30,

> These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

The NRC Senior Resident Inspector questioned the defeating of the feedwater isolation on low T_{avg} coincident with P-4 function while in MODE 3, because this defeating of the isolation signal appeared to be contrary to the requirements of the plant TSs, specifically TS Table 3.3.2-1, Function 8a. Subsequently, the NRC Integrated Inspection Report 05000482/2009004 for WCGS dated November 10, 2009 (ADAMS Accession No. ML093140803), identified a Green non-cited violation of LCO 3.0.3 in which both trains of TS Table 3.3.2-1, Function 8.a were defeated in accordance with a plant procedure.

Other normal operating conditions periodically occur at WCGS during which testing of plant safety features is required while the plant is in MODE 3, and automatic feedwater isolation is not required to control the RCS temperature. If such automatic isolations do occur during such testing, it results in unnecessary cycling of the MFIVs and the AFW system, which adversely impacts startup and shutdown evolutions. The results of the evaluations by the licensee concluded that there was no impact to the current WCGS USAR analyses or reduction in the analyzed margin of safety if the feedwater isolation function were to be defeated in MODE 3, while all other P-4 signals remained functional. The results of the NRC staff's evaluation for its impact of the WCGS USAR Chapter 15.0 analyses and Chapter 6.2 Containment Integrity are discussed in Sections 3.3 6 and 3.3.7 of this SE.

To prevent future potential non-compliances of this type with the plant TSs, the licensee is proposing to modify the plant TSs to prevent unnecessary cycling of the MFIVs and the AFW system and unnecessary trips of the main turbine during turbine warm-up that adversely impacts startups and shutdown evolutions. The licensee is proposing to add a new footnote (m) to Function 8.a. of TS Table 3.3.2-1 to identify the enabled functions and the applicable MODES for that function. The new footnote states:

(m) The functions of the Reactor Trip, P-4 interlock required to meet the LCO are:

- Trips the main turbine - MODES 1 and 2 [originally all functions were lumped together under MODES 1, 2, and 3]
- Isolates MFW with coincident low T_{avg} - MODES 1 and 2 [originally MODES 1, 2, and 3]
- Allows manual block of the automatic re-actuation of SI after a manual reset of SI - MODES 1, 2, and 3 [Same as original]
- Prevents opening of the MFIVs if closed on SI or SG Water Level - High High - MODES 1, 2, and 3 [Same as original]

To accomplish the bypass of the feedwater isolation function when needed in MODE 3 conditions, the WCGS operating staff plans to utilize an administrative procedure SYS SB-122, "Enabling/Disabling P-4/Lo T_{avg} FWIS," that requires the use of temporary jumpers to physically disable the feedwater isolation valve automatic isolation function. The licensee also states that installation or removal of such jumpers is required when performing plant 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." In November 2002, plant procedure SYS AC-120, "Main Turbine Generator Startup," was revised and included changes that allowed leads to be lifted at the turbine control panel that defeated the Reactor Trip, P-4 interlock for the turbine trip function when the plant is in MODE 3. The change was made to allow warming up of the main turbine during various plant activities that involve opening of the reactor trip breakers. The procedure change required that the trip function be reinstated prior to entry into MODE 2. The licensee has proposed to change the TSs to reflect this change.

The NRC staff notes that generally, a plant's administrative procedures require that a licensed operator give permission before the initiation of any activity that would or could affect a safety system. The decision to grant such permission should be based on knowledge of the operating status of the safety systems, the extent to which the activity will affect those systems, and whether that effect is permissible within the provisions of the license. Experience at operating plants, however, suggests that when the measures used to indicate inoperable status consist solely of administrative procedures, the operator is not always fully aware of the ramifications of each bypassed or inoperable component. An automatic indication display of any bypass or inoperability in a safety system supplements administrative procedures and aids the operator. As indicated above, Clause 16 of IEEE Standard 338-1987, Section 5 states that,

> Indication should be provided in the control room if a portion of the safety system is inoperable or bypassed. Systems that are frequently placed in bypass or

inoperative condition for the purposes of testing should have automatic indication.

The NRC staff notes that the temporary installation of jumpers to defeat a plant safety feature or interlock is not as risk-averse of a method as, for example, would be the use of a permanent control switch for ensuring that undesired safety system actuations do not take place. First, the temporary installation of such jumpers may cause a spurious undesired actuation or a short circuit and subsequent fuse failure if a slip by the technician is made during the installation process. Second, the fact that a temporary jumper is installed (especially if circumstances arise during the test process that require such jumpers to be installed for longer than necessary) does not provide the control room unit operating personnel with a continuous reminder as to the status of the desired safety feature being in an overridden condition. Finally, there is always the risk that such jumpers may be accidentally left in place longer than needed to accomplish the MFIV initiation signal bypass longer than necessary.

Consequently, the NRC staff requested the licensee to provide an explanation regarding how and under what circumstances jumpers will be used to accomplish the bypass function, and what indications there were to remind control room personnel that the required isolation signal was in a bypassed condition. Specifically, the licensee was requested to describe, 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 T_{avg} coincident with P-4 interlock, 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. In addition, the licensee was asked to describe how the installation of these jumpers addresses IEEE Standard 279-1971, "Criteria for Nuclear Power Plant Protection Systems" (part of the licensing basis for WCGS), regarding indication of bypasses, access to means of bypasses, and independence between safety and non-safety functions, and NRC RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

In its letter dated October 13, 2010, the licensee stated that during the conditions which are present when the jumpers are to be installed, "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 NRC RG 1.47 is not warranted.

The licensee also indicated that the guidance of NRC RG 1.47 provides for three criteria to be satisfied before permanent indication of such bypasses are recommended:

1. The bypass or inoperable condition affects a system that is designed to perform automatically a function that is important to the safety of the public;
2. The bypass will be utilized by plant personnel or the inoperable condition can reasonably be expected to occur more frequently than once per year; and

3. The bypass or inoperable condition is expected to occur when the affected system is normally required to be operable.

The licensee also indicated that implementation of the isolation signal bypass is performed so seldom that these criteria are not met, and therefore bypassed and inoperable status indication is still not warranted.

The NRC staff noted that there were other testing functions that were performed during startups or shutdowns during which require the licensee's performance of administrative procedure SYS SB-122, "Enabling/Disabling P-4/Lo Tavg FWIS." In an RAI letter dated November 24, 2010 (ADAMS Accession No. ML103260174), the NRC staff requested the licensee to 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.

In its letter dated December 21, 2010, the licensee stated that once the proposed TS change is approved, the only modes of Applicability requiring the P-4 interlock coincident with low T_{avg} to perform a safety function affecting the operability of the feedwater isolation valves would be MODES 1 and 2, and that installing jumpers into the affected circuitry would occur when the isolation of MFW coincident with low T_{avg} function and the turbine trip on reactor trip function of the Reactor Trip, P-4 interlock is <u>not</u> required to be OPERABLE. Therefore, the conditions 1) and 3) satisfying the need for a permanent bypassed and inoperable status indication as described above would not be met, and hence, the minimum conditions recommended in the guidance documents as to when a permanent bypassed and inoperable status indication should be provided for this interlock bypass would not be met.

The NRC staff also notes that when the licensee received its operating license in June of 1984, the version of NRC RG 1.47 in effect had been published in May 1973. Condition 2) described above is indicative of refueling outages originally occurring on a 12-month frequency (which was the common refueling cycle practice at that time). Condition 2) was intended to be a recommendation for a bypass indication to 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). Presently, the licensee conducts refueling outages on an 18-month frequency and, therefore, expects that the normal installation of these jumpers would occur on a frequency of once per 18 months.

The NRC staff notes, however, that it is possible for unexpected plant outages to occur between sesquiannual (once every 18 months) refueling outages. Such outages would require the installation of these jumpers during the subsequent start-up each time an unexpected outage occurred. Such unplanned outages could reasonably be expected to occur more than once per year; hence, the described condition 2) of the NRC RG 1.47 is still pertinent. However, the staff also notes that the licensee's procedure for implementing the jumpers cannot be exited unless the jumpers are first removed by one technician, and that this removal must be confirmed by independent personnel. Further, the procedure requires subsequent approval of its completion

by shift operations management personnel. Therefore, it is highly unlikely that these jumpers could accidentally be left in place without the operating staff's knowledge, and such removal confirmation within the plant procedure serves to compensate for the need to account for the occasional unplanned outage which could occur more than once per year when evaluating condition 2).

The licensee has made provisions for ensuring that the reactor interlock functions which ensure that bypassable functions for the feedwater isolation function will be in operation under the conditions assumed in the safety analyses (i.e., prior to entering into MODE 2 and/or MODE 1). Therefore, based on its review of the licensee's submittal and responses to the NRC staff's RAIs, the NRC staff concludes that there is reasonable assurance that the requirement of GDC 13, regarding the need for appropriate controls to be provided for maintaining variables and systems so that the containment and its associated systems can be operated within prescribed operating ranges, has been met. The licensee's procedure for confirming that the jumpers for bypassing the P-4 interlock coincident with low T_{avg} have been removed ensures that the control room operators are aware of their removal prior to entering into MODE 2 and MODE 1. Therefore, the NRC staff concludes that there is reasonable assurance that GDC 19 has been met, which requires that the nuclear plant can be operated safely from the control room under normal operating conditions. The NRC staff concludes that the licensee has proposed an acceptable alternative for meeting the requirements of Criterion XIV, "Inspection, Test, and Operating Status," as given in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, as well as the provisions of 10 CFR 50.34(f)(2)(v), which also require an automatic indication of the bypassed and operable status of safety systems. The Appendix B criterion requires that measures be established for indicating the operating status of structures, systems, and components of the nuclear power plant, such as by tagging valves and switches, to prevent inadvertent operation. The alternative proposed by the licensee is to administratively control the bypass of the feedwater isolation signal reactor trip interlock using formal plant procedures, requiring independent confirmation that the temporary jumpers have been removed. Therefore, there is reasonable assurance that GDC 21 has been met, regarding the need for the protection system to be designed for high functional reliability and inservice testability. Further, at the specified modes of operation when the proposed jumpers are to be installed, there is no safety requirement for the isolation signal to be present.

The NRC staff also reviewed the removal of the P-4 function of the main turbine trip in MODE 3 using the requirements of GDCs 13, 19, and 21 and guidance provided by IEEE 603-1991, IEEE 279-1971, and RG 1.47. Since the reasons and justification for removal of the P-4 function of the MFW isolation in MODE 3 are the same for removal of the P-4 function of the main turbine trip in MODE 3 based on the analysis described above for the main feedwater isolation function, the NRC staff concluded that the proposed TS change for the turbine trip function is acceptable.

In its technical evaluation, the NRC staff considered how the guidance provided in IEEE-338, IEEE-279, and NRC RG 1.47 has been achieved, and concludes that the licensee's proposed alternative method for addressing this guidance is in conformance with the general design criteria requirements. These considerations include the licensee's proposed method for accomplishing the bypass, the lack of a safety requirement for the isolation signal to be present

under the proposed modes when the signal is to be bypassed, and the planned frequency of use of the bypass procedure being less than once per year. Therefore, the NRC staff concludes that the licensee's proposed method for implementing this TS change meets the regulations in 10 CFR Part 50, Appendix A, GDCs 13, 19, and 21, and will be accomplished within the guidelines established within NRC guidance described in NRC RG 1.47, and industry standards IEEE 279-1971 and IEEE 338-1987.

Based on the discussion in sections 3.3.1 through 3.3.8 the NRC staff also concluded that the proposed TS change meets the requirements of 10 CFR Part 50, Appendix A, GDCs 10, 15, and 50.

3.3.9 Compliance with the 10 CFR 50.36(c)(2)(ii) Requirements

The proposed footnote (m) to Function 8.a in TS Table 3.3.2-1 includes removing out of the TSs the following functions (referred as the "subject functions"): (1) the P-4 function of transfer of the steam dump system to the plant trip controller in MODES 1, 2, and 3, (2) the P-4 turbine trip function in MODE 3, and (3) the P-4 feedwater isolation function in MODE 3. Based on the above, the NRC staff reviewed the "subject functions" against the 10 CFR 50.36(c)(2)(ii) criteria specified as follows.

- *Criterion 1:* The "subject functions" are not used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary.

- *Criterion 2:* The "subject functions" are not a process variable, design feature, or operating restriction that was an initial condition of a design basis accident or transient analysis.

- *Criterion 3:* As discussed in Sections 3.3.6 and 3.3.7 of this report, no credit is taken for the "subject functions" in the analysis of transients and accidents for WCGS. The "subject functions" are not considered as part of primary success path related to the integrity of a fission product barrier. Therefore, the "subject functions" are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- *Criterion 4:* The "subject functions" are not significant to public health and safety in that no credit is taken for the "subject functions" for consequence mitigation in applicable design basis accident or transient analysis. Therefore, the "subject functions" in applicable Modes are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Based on the above, the NRC staff concludes that the existing LCO and related surveillance requirement associated with the "subject functions" do not satisfy any of the criteria in 10 CFR 50.36(c)(2)(ii). Therefore, the proposed removal of the "subject functions" out of the TSs does not violate the 10 CFR 50.36(c)(2)(ii) requirements and is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on June 15, 2010 (75 FR 33844). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: March 30, 2011

March 30, 2011

Mr. Matthew W. Sunseri
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: REVISE TABLE 3.3.2-1 OF TECHNICAL SPECIFICATION 3.3.2, "ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS) INSTRUMENTATION" (TAC NO. ME3762)

Dear Mr. Sunseri:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 194 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 13, 2010, as supplemented by letters dated October 13 and December 21, 2010, and January 18, 2011.

The amendment revises TS Table 3.3.2-1, Function 8.a (Reactor Trip, P-4) by adding footnote (m) to identify the enabled functions and the applicable modes for the Reactor Trip, P-4 interlock function.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,
/RA/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures:
1. Amendment No. 194 to NPF-42
2. Safety Evaluation

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NAME	BSingal	JBurkhardt	REliott	GCasto	AUlses	RDennig
DATE	3/1/11	3/1/11	3/3/11 (W/comments)	3/2/11	3/2/11	3/4/11

OFFICE	NRR/DSS/SNPB/BC	NRR/DE/EICB/BC	OGC NLO	NRR/LPL4/BC	NRR/LPL4/PM
NAME	AMendiola	GWilson	MWright	MMarkley	BSingal
DATE	3/8/11	3/3/11 (w/comments)	3/11/11	3/30/11	3/30/11

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