WOLF CREEK
NUCLEAR OPERATING CORPORATION

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March 8, 1996

WO 96-0036

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

Subject: Docket No. 50-482: Revision to Technical Specifications to Implement Cycle 9 Reactor Coolant System Thermal Design Flow Reduction

Gentlemen:

This letter transmits an exigent change to Facility Operating License No. NPF-42 for Wolf Creek Generating Station (WCGS). Wolf Creek Nuclear Operating Corporation (WCNOC) requests that this license amendment request be reviewed under the exigent circumstances provisions of 10 CFR 50.91(a)(6). This proposed change would revise Technical Specification Figure 2.1-1, "Reactor Core Safety Limit - Four Loops in Operation," Table 2.2-1, "Reactor Trip System Instrumentation Setpoints," and Table 3.2-1, "DNB Parameters." The requested changes are needed to allow operation of the WCGS with decreased indicated Reactor Coolant System (RCS) flow. A decrease in the predicted RCS flow is a consequence of the effect of the Cycle 9 core reload geometry on hot leg streaming and its subsequent influence on indicated RCS flow. The requested exigent change is required to allow WCGS to operate at full rated power following restart after the eighth refueling outage, should the indicated flow be below the current minimum measured flow.

Attachment I provides a detailed safety evaluation/analysis including a description of the proposed changes. Attachment II provides a No Significant Hazards Consideration Determination and Attachment III provides an Environmental Impact Determination. Marked-up pages indicating the specific changes to the technical specifications proposed by this request are provided in Attachment IV.

WCNOC is proposing to implement the proposed changes prior to the end of the current refueling outage, which is scheduled for completion on March 30, 1996.

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

If you have any questions concerning this matter, please contact me at (316) 364-8831, extension 4450, or Mr. Richard D. Flannigan, at extension 4500.

Very truly yours,

Otto L. Maynard

OLM/jra

Attachments
I - Safety Evaluation
II - No Significant Hazards Consideration Determination
III - Environmental Impact Determination
IV - Proposed Technical Specification Change

cc: G. W. Allen (KDHE), w/a
L. J. Callan (NRC), w/a
W. D. Johnson (NRC), w/a
J. F. Ringwald (NRC), w/a
J. C. Stone (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

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

By Otto L. Maynard
Otto L. Maynard
Vice President
Plant Operations

SUBSCRIBED and sworn to before me this 17TH day of MARCH, 1996.

Mary E. Gifford
Notary Public

MARY E. GIFFORD
Notary Public - State of Kansas
My Appt. Expires 12/09/1999

Expiration Date 12/09/1999

ATTACHMENT I

SAFETY EVALUATION

Introduction

This license amendment request proposes to revise the Wolf Creek Generating Station (WCGS) Technical Specifications to allow plant operation at 100% rated thermal power (RTP) with a 3.5% reduction in Thermal Design Flow (TDF) and an increase in the Low Pressurizer Pressure trip setpoint. This revision in thermal design flow represents a decrease in TDF from the current value of 374,400 gpm to 361,296 gpm. The corresponding vessel average temperature (T_{avg}) will remain at the current value of 586.5 °F, however, the decreased TDF will result in a slight decrease in the core inlet temperature (T_{in}) and an approximate 1 °F increase in the hot leg temperature (T_{hot}). The Low Pressurizer Pressure trip setpoint will be raised from the Safety Analysis Limit (SAL) of 1915 psig to 1940 psig to preclude the occurrence of Departure from Nucleate Boiling (DNB), ensuring that core thermal protection is provided for all conditions of operation.

The Limiting Condition for Operation (LCO) flow value listed in Technical Specification Table 3.2-1 is, by definition, Minimum Measured Flow. Minimum Measured Flow is defined as 102.5% of TDF. The proposed new TDF of 361,296 gmp corresponds to a new Minimum Measured Flow of 370,328 gpm, rounded up to 371,000 gpm to provide additional margin. Therefore, the Technical Specification LCO flow value listed in Table 3.2-1 will change from 384,000 gpm to 371,000 gpm.

The limiting USAR Chapter 15 event analyses have been evaluated assuming a 3.5% reduction in flow and assuming operating parameters consistent with a T_{avg} equal to 588.4°F. The results of these analyses show that the applicable acceptance criteria for each event continue to be met.

Background

The WCGS Cycle 9 reload design was performed utilizing Westinghouse's core design methodology as approved in Technical Specification Amendment No. 92. Based on this methodology a Low Leakage Loading Pattern (LLLP) was developed within the constraints of the cycle energy requirement and fuel available for reload in Cycle 9. The loading pattern was optimized such that it would minimize the number of feed (new fuel) assemblies to be purchased for Cycle 9 as well as reduce neutron fluence at the core periphery, ultimately maximizing the fuel economy for Cycle 9 operation.

During the Cycle 9 reload design process, the vendor reviewed the proposed Cycle 9 LLLP and raised concerns based on their experience. The concerns were based on the apparent tendency for large power gradients at the core periphery in LLLPs to influence hot leg streaming. An increase in hot leg streaming could result in a biased T_{hot} measurement such that the indicated T_{hot} would be greater than the actual bulk fluid temperature in the hot leg.

Because RCS flow is calculated based on a flow calorimetric which is dependent on the hot leg temperature measurement, an increase in hot leg streaming could therefore lead to a calculated RCS flow below the Technical Specification LCO. An RCS flow measurement below the LCO value would result in shut down of the WCGS unit and a considerable loss of revenue to Wolf Creek's owners. This

Technical Specification change is being prepared to ensure that the WCGS can continue to operate at 100% power in the event that hot leg streaming leads to conservative biasing of the RCS flow measurement below the current Technical Specification LCO value.

Evaluation

The proposed operating conditions presented in this document are justified based on the original safety analyses performed at the lower bound temperatures for the power rerate program in combination with the additional safety analyses performed specifically at the reduced flow conditions for Cycle 9 (henceforth referred to as "the previous Safety Analysis"). The proposed operating parameters for Cycle 9 are presented in Table 1. The safety analyses have been either performed or evaluated at the lower and upper bound conditions. By performing the analyses at these conditions, the proposed condition is assured to be bounded and plant operational changes may be made in the future without significant impact to the safety analyses.

The Technical Specification parameter changes necessary for reducing RCS flow are provided Attachment IV, and are summarized in Table 2.

The remainder of this evaluation details each specific evaluation performed to demonstrate the acceptability of the proposed Technical Specification changes.

Table 1: Proposed Operating Parameters

Parameter	Current Condition 0 °F T_{hot} Reduction	Lower Bound 15 °F T_{hot} Reduction	Proposed Condition 0 °F T_{hot} Reduction	Proposed Upper Bound 1.8 °F T_{hot} Increase
NSSS Power, MWt	3579	3579	3579	3579
Reactor Power, MWt	3565	3565	3565	3565
Thermal Design Flow				
Per loop, gpm	93600	93600	90324	90324
Total flow, gpm	374400	374400	361296	361296
Reactor Flow, Total, (Mlbm/hr)	139.8	142.9	135.1	134.7
Reactor Coolant Press, psia	2250	2250	2250	2250
Core Bypass, %	8.4	8.4	8.4	8.4
Fuel Design	17x17 V5H w/IFMs	17x17 V5H w/IFMs	17x17 V5H w/IFMs	17x17 V5H w/IFMs
Reactor Coolant Temperature, °F				
Core Outlet	623.3	608.5	624.5	626.2
Vessel Outlet	618.2	603.2	619.3	621.1
Core Average	591.1	575.1	591.3	593.2
Vessel Average	586.5	570.7	586.5	588.4
Vessel/Core Inlet	554.8	538.2	553.7	555.8
Steam Generator Outlet	554.5	538.0	553.4	555.5
Vessel ΔT (T_{hot}-T_{in})	63.4	65.0	65.6	65.3
Steam Generator				
Steam Temperature, F	536.3	519.4	535.5	537.6
Steam Pressure, psia	934	807	928	944
Steam Flow, total, Mlbm/hr	15.91	15.83	15.91	15.92
Feedwater Temp, F	446	446	446	446
Zero Load Temp, F	557	557	557	557
SG Tube Plugging, %	10	10	10	10

Table 2: Summary of Technical Specifications Changes

Technical Specification	Page	Description Of Change	Reason For Change
Figure 2.1-1	2-2	Revise Reactor Core Safety Limits to ensure protection from DNB.	Reduction in Technical Specification RCS flow.
Table 2.2-1	2-4	Revise Low Pressurizer Pressure reactor trip setpoint from 1915 psig to 1940 psig and the Allowable Value (AV) from 1906 psig to 1931 psig to ensure protection from DNB.	Reduction in Technical Specification RCS flow.
Table 3.2-1	3/4 2-16	Revise Reactor System Coolant Flowrate from 384,000 gpm to 371,000 gpm.	Ensure sufficient Technical Specification flow margin exists to support future cycles of operation.

Core Thermal Limit Protection

Protection of the core thermal limits is illustrated in the WCGS USAR Figure 15.0-1 and is provided in Figure 1 below. The figure demonstrates that the OTΔT and OPΔT protection system setpoints and the steam generator safety valves prevent violation of the vessel exit boiling limits, Departure from Nucleate Boiling Ratio (DNBR) correlation exit quality limit, and DNB limits for a range of pressurizer pressures which bound allowable operating space. This range is defined by the Low and High Pressurizer Pressure trip setpoints.

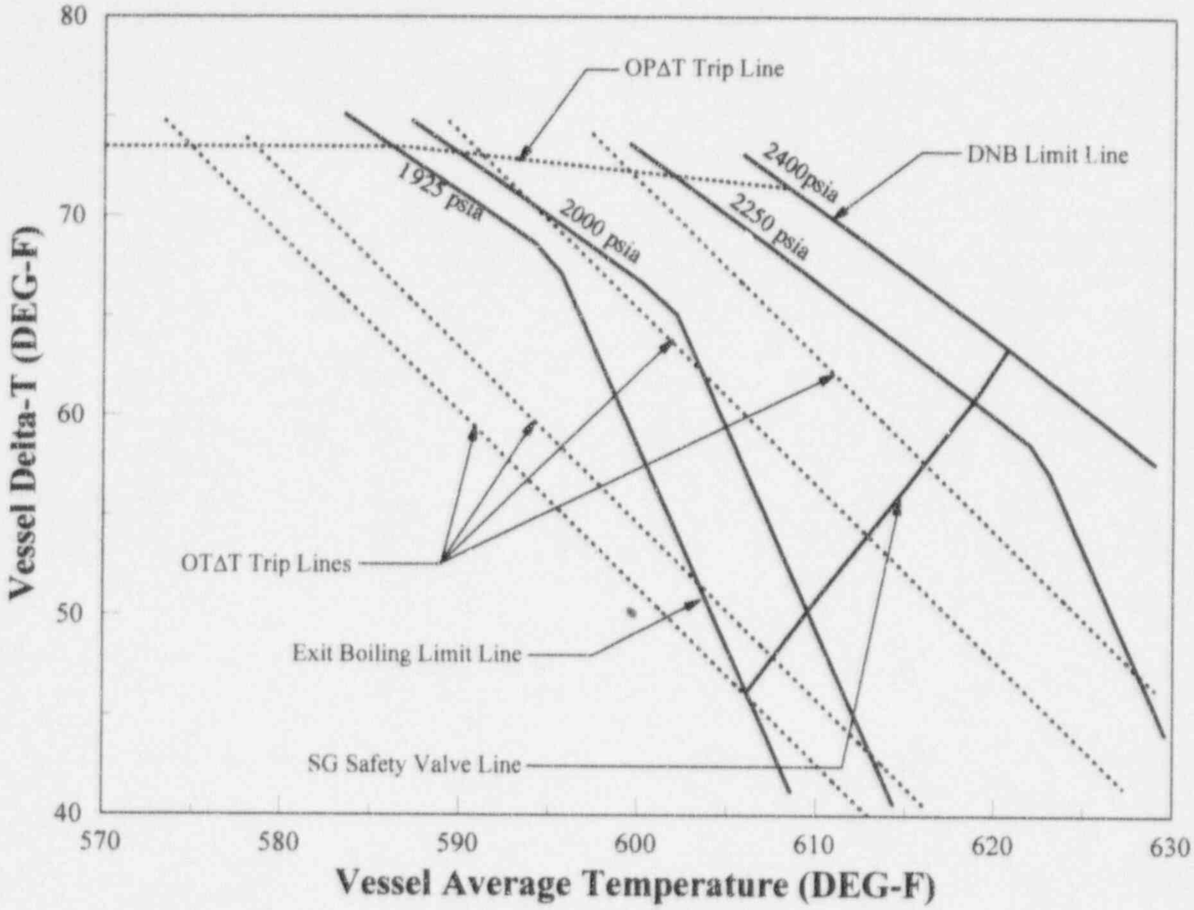

Figure 1 : Core Thermal Limit Protection

Changes in the primary side parameters associated with the reduction in thermal design flow affect the core thermal limit calculations. The reduction in flow penalizes the calculated limits, restricting the range of allowable operating space. Thus, the core thermal limits have been revised to account for the reduction in design flow.

The vessel exit boiling limit lines are a family of curves which represent the locus of points at which the vessel exit enthalpy is equal to the saturation enthalpy at the applicable system pressure. While the vessel exit boiling limits do not serve a direct function in core protection, it is necessary to limit the vessel exit enthalpy to a subcooled state, since many of the reactor protection system functions use vessel ΔT as indicator of core power. The inherent assumption in these protection system functions is that vessel ΔT is proportional to core power; a valid assumption as long as the coolant remains subcooled. Thus, to guarantee that vessel ΔT remains proportional to power, the core exit enthalpy is restricted such that exit boiling will not occur.

The core DNE limit lines are lines of constant DNBR below which there is a 95% probability at the 95% confidence level that DNB will not occur during normal operation, operational events, or as a result of conditions arising from any fault of moderate frequency. The reduction in thermal design flow results in

restrictions on allowable operating space. However, examination of the OTΔT and OPΔT trip functions indicate that these restrictions are bounded.

An additional effect of the reduction in thermal design flow is a change in the location of the steam generator safety valve line resulting from the changes in secondary side parameters. This line, which represents the locus of conditions under which the steam generator safety valves open, shifts such that a broader range of attainable operating conditions exist. Initial evaluation of the core thermal limit protection with the revised position of the steam generator safety valve limit line indicated violations in the Vessel exit boiling limits for low values of system pressure. Restricting the allowable pressure range at the low end produces margin to the corresponding pressure OTΔT trip setpoint conditions. The low end of the allowable system pressure range is defined by the Low Pressurizer Pressure trip setpoint. The system pressure range was therefore shortened by increasing this setpoint. An increase in the Low Pressurizer Pressure trip setpoint safety analysis limit from the current value of 1915 psig to 1940 psig ensures that the revised Vessel exit boiling limits are protected by the existing OTΔT trip setpoint safety analysis limit and the revised steam generator safety valve limit lines.

Thermal-Hydraulic Analysis

A complete description of the thermal-hydraulic methods used by WCNOC for Cycle 9 DNB evaluations is provided in WCNOC's June 14, 1995 submittal for Implementation of Advance Physics and Core Thermal-Hydraulic analysis methods (Letter ET 95-0051, from R. C. Hagan WCNOC to USNRC).

No changes to the thermal-hydraulic methods are required due to the reduction in thermal design flow. The limiting ANS Condition II event with respect to DNB, the Complete Loss of Flow event, as well as all other Updated Safety Analysis Report (USAR) Chapter 15 events, have been evaluated to insure that the DNB design basis continue to be met.

Event Analyses

Each USAR Chapter 15 event was evaluated to determine the impact of the reduction in thermal design flow. The events in which the margin to the acceptance criteria was decreased were reanalyzed to support the proposed flow reduction. Generally, the RCS heat-up events fall into this category as the reduction in RCS flow results in decreased heat removal capacity. Those USAR Chapter 15 events in which the reduction in thermal design flow would result in an increase in margin to the acceptance criteria were not reanalyzed. Generally, the RCS cool-down events fall into this category. Rather, evaluations of these events were performed using bounding core state parameters based on the previous Safety Analysis. Results of the analyses and evaluations performed for the reduction in thermal design flow indicate that all acceptance criteria for USAR Chapter 15 events continue to be met.

Non-LOCA Analyses

The USAR Chapter 15 non-LOCA events are categorized into the following sections

- 15.1 Increase in Heat Removal by the Secondary System
- 15.2 Decrease in Heat Removal by the Secondary System
- 15.3 Decrease in Reactor Coolant System Flowrate
- 15.4 Reactivity and Power Distribution Anomalies
- 15.5 Increase in Reactor Coolant System Inventory
- 15.6 Decrease in Reactor Coolant System Inventory
- 15.7 Radioactive Release from a Subsystem or Component

With the exception of USAR Section 15.7, each USAR section can be further classified into RCS heat-up events or RCS cool-down events, preparing a basis from which each event's sensitivity to the RCS flow reduction may be determined. The heat-up events are generally comprised of USAR Sections 15.2, 15.3, portions of 15.4, and 15.6. The cool-down events are comprised of USAR Sections 15.1, portions of 15.4, and 15.5. In general, the RCS flow reduction will impact the heat-up events which rely on the RCS to remove core heat to prevent DNB and RCS overpressurization, however, a short evaluation of each cooldown event will also be provided for completeness.

The USAR Chapter 15.6 events, i.e., Steam Generator Tube Rupture (SGTR), small break LOCA and large break LOCA, are discussed further below. In addition to the USAR events, the Steamline Break Coincident with Rod Cluster Control Assembly (RCCA) Withdrawal at Power event has been evaluated and is also discussed below.

Each event assumes initial conditions consistent with those listed in Table 1 and assumes the appropriate uncertainties and steady state errors as presented in Table 3. The DNB analyses are performed in accordance with Westinghouse's Revised Thermal Design Procedure (RTDP) using the WRB-2 correlation. The uncertainties and steady state errors assumed in the initial conditions for these analyses are treated statistically in the DNB analysis and are therefore initiated from nominal conditions in the event analyses. DNB analyses which fall outside the range of applicability of the RTDP methodology are analyzed utilizing the W-3 correlation and therefore are initiated from the same conditions as the remainder of the non-LOCA event analyses.

Table 3: Steady State Error and Uncertainty

Parameter	Standard	RTDP
Core Power	± 2%	Nominal
RCS Temperature	± 6.5 °F	+ 1.65 °F
RCS Pressure	± 30 psi	Nominal
RCS Flow	Thermal Design	Minimum Measured

In general, the low pressurizer pressure trip setpoint is chosen at a conservatively low value for the safety analyses. Increasing this reactor trip setpoint 25 psi from 1915 psig to 1940 psig would result in a net benefit to all analyses which assume its use. Therefore the current Safety Analysis Limit of 1885 psig will continue to be used in the WCGS event analyses. Based on this conclusion the low pressurizer pressure trip setpoint will not be explicitly addressed further in this evaluation.

A detailed evaluation of each of the USAR Chapter 15 non-LOCA events is provided below for the proposed reduction in TDF.

Feedwater Malfunction (USAR 15.1.2)

This USAR event is classified as an ANS Condition II event and is analyzed primarily to show that the DNB design basis is met. As stated in the USAR, feedwater system malfunctions that result in decreased feedwater temperature (USAR 15.1.1) are bounded by those which result in increased feedwater flow (USAR 15.1.2). The increase in feedwater flow event is considered for both the no-load and full power conditions.

The USAR full power event assumes automatic rod control and the reactor trips on a Low-Low Steam Generator Water Level signal following feedwater isolation due to high steam generator water level. The reduced primary side TDF will not change the time at which this signal is initiated. Core power matches the increased load demand up until the time of feedwater isolation on high-high steam generator level. After feedwater isolation, power drifts downward as the steam generator level decreases until the reactor is tripped on low steam generator level. Since T_{avg} and system pressure to which the plant would be controlled have not changed, there would be no significant change to these parameters at either the time of feedwater isolation or the time of reactor trip. The severity of the Feedwater Malfunction event is dependent on the secondary system being capable of over cooling the primary RCS due to an increase in feedwater flow, resulting in a power excursion in the presence of negative moderator temperature coefficient. A decrease in the RCS flowrate would result in less RCS cooling in the core and a less conservative event analysis. Therefore the results of the current analysis bound the reduced RCS flow case. The event has been evaluated for DNBR based on the current licensing basis analysis assuming a 3.5% RCS flow reduction in the DNB model. Results of the revised DNB analysis show that the event continues to meet the applicable acceptance criteria. Therefore, the event is shown to continue to meet all acceptance criteria for an ANS Condition II event and the results

presented in the USAR remain valid.

The USAR 15.1.2 Feedwater Malfunction event bounds the USAR 15.1.1 event and is therefore not analyzed in the WCGS USAR.

Excessive Increase in Secondary Steam Flow (USAR 15.1.3)

The Excessive Increase in Secondary Steam Flow event is modeled as a 10% step increase in steam flow. Because the increase in steam flow is small compared to both the increase in steam flow from inadvertently opening a steam generator relief or safety valve and a postulated steamline break, the Excessive Increase in Secondary Steam Flow event is bounded by both the Inadvertent Opening of a Steam Generator Relief or Safety Valve event and the Main Steamline Break event.

This ANS Condition II event is analyzed to show that the DNB design basis is met following a step load increase from rated power. The event is characterized by a rapid increase in steam flow that causes a power mismatch between reactor core power and steam generator load demand. Cases analyzed assume both manual and automatic rod control for minimum and maximum reactivity feedback conditions. In all cases equilibrium is reached corresponding to the increased load demand. The calculated minimum DNBR is significantly greater than the safety analysis limit. This event has been evaluated for DNBR based on the current licensing basis analysis assuming a 3.5% RCS flow reduction in the DNB model. Results of the revised DNB analysis show that the event continues to meet the applicable acceptance criteria. Therefore, the event is shown to continue to meet all acceptance criteria for an ANS Condition II event and the current USAR analysis remain valid for the Excessive Increase in Secondary Steam Flow event.

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

The inadvertent opening of a steam generator relief or safety valve is an ANS Condition II event which is analyzed to show that the DNB design basis is met. The energy removal from the RCS due to an Inadvertent Opening of a Steam Generator Relief or Safety Valve causes a reduction in primary coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, a positive reactivity insertion and increase in reactor power results from the uncontrolled steam release through the valve. The event's assumptions are chosen to maximize the resulting reactivity insertion. A reduction in TDF reduces primary to secondary heat transfer and thus the reactivity insertion due to the negative moderator temperature coefficient. This effect is, therefore, an event benefit.

The impact on the DNBR of the reduced TDF can be conservatively estimated by considering a 3.5% reduction in flow while maintaining the other DNBR statepoints. Results of the revised DNB analysis show that the event continues to meet the applicable acceptance criteria. Therefore, the event is shown to continue to meet all acceptance criteria for an ANS Condition II event and the current USAR analysis remain valid for the Inadvertent Opening of a Steam Generator Relief or Safety Valve event.

Main Steam Line Break (USAR 15.1.5)

The steam system piping failure is an ANS Condition IV event analyzed to show that the core remains intact and in place and that the radiation doses do not exceed the guidelines of 10CFR100. This is demonstrated by showing that the DNB design basis is met, even though DNB and possible clad perforation are not necessarily unacceptable for a Condition IV event.

The energy removal from the RCS due to a steamline rupture causes a reduction in primary coolant temperature and pressure. Assuming a sufficiently conservative negative moderator temperature coefficient produces a positive reactivity insertion and a limiting event response. The Steamline Break assumptions are chosen to maximize the resulting reactivity insertion. A reduction in TDF reduces primary to secondary heat transfer and thus the reactivity insertion due to the negative moderator temperature coefficient. This effect is, therefore, a event benefit. In the main steam line rupture event, the event is mitigated by borated safety injection flow initiated by a Low Steamline Pressure signal. The time of safety injection would not be adversely impacted by the reduction in TDF. This coupled with the reduction in reactivity insertion produces a DNBR benefit.

The impact on the DNBR of the reduced TDF can be conservatively estimated by considering a 3.5% reduction in flow while maintaining the other DNBR statepoints. Results of the revised DNB analysis show that the event continues to meet the applicable acceptance criteria. Therefore, the event is shown to continue to meet all acceptance criteria for an ANS Condition II event and the current USAR analysis remain valid for the Main Steam Line Break event.

Loss of Electrical Load/Turbine Trip (USAR 15.2.3)

This ANS Condition II event is analyzed to show that the DNB design basis is met and that primary and secondary side system pressures do not exceed 110% of design values. Whether from loss of external electrical load or turbine trip, this event is characterized by an increase in core power which exceeds the secondary side power extraction. The result is a primary side heat up and RCS pressure increase. The analysis assumptions for this event are chosen to maximize primary side heat-up, therefore this event was reanalyzed to incorporate the reduced flow assumption.

The reanalysis of the Turbine Trip event was accomplished utilizing the RETRAN-02 computer code in accordance with WCGS's approved methodology. Results of the analysis show that the peak RCS pressure continues to be maintained below the pressure limit of 110% of design pressure on both the primary and secondary coolant systems. DNB analysis of the revised core statepoints shows that the minimum DNBR continues to be maintained above the design limit. Therefore the Turbine Trip event is shown to meet all applicable acceptance criteria for an ANS Condition II event and therefore there is no reduction in margin of safety.

The results of the revised Turbine Trip analysis will be incorporated in Revision 10 of the WCGS USAR.

Loss of Non-Emergency AC Power (USAR 15.2.6)

This ANS Condition II event is analyzed to demonstrate that adequate heat removal capability exists via natural circulation flow, aided by the Auxiliary Feedwater System, to remove core decay heat and stored energy following reactor trip. This is ensured by turn-around of the RCS heat-up event. This event is also analyzed to ensure that the DNB and RCS overpressure criteria are met. The Loss of Non-Emergency AC Power event is initiated by the termination of feedwater flow. Reactor trip occurs on Steam Generator Low-Low Level and is followed shortly by reactor coolant pump coastdown. After appropriate delays, auxiliary feedwater is delivered. Eventually, the auxiliary feedwater heat removal capability exceeds core decay heat generation.

To ensure the acceptance criteria for the Loss of Non-Emergency AC Power event continue to be met for the reduced RCS flow condition, this event was reanalyzed. The reanalysis of the Loss of Non-Emergency AC Power event was accomplished utilizing the RETRAN-02 computer code in accordance with WCGS's approved methodology. Results of the analysis show that the peak RCS pressure continues to be maintained below the pressure limit of 110% of design pressure on both the primary and secondary coolant systems. DNB analysis of the revised core statepoints shows that the minimum DNBR continues to be maintained above the design limit. The long term core cooling event also demonstrates that the Auxiliary Feedwater System can provide sufficient cooling to prevent this event from progressing to a worse condition event. Therefore the Loss of Non-Emergency AC Power event is shown to meet all applicable acceptance criteria for an ANS Condition II event and therefore it is demonstrated that the margin of safety is maintained.

The results of the revised Loss of Non-Emergency AC Power analysis will be incorporated in Revision 10 of the WCGS USAR.

Loss of Normal Feedwater (USAR 15.2.7)

This ANS Condition II event is analyzed to demonstrate that adequate heat removal capability exists via natural circulation flow, as aided by the Auxiliary Feedwater System, to remove core decay heat and stored energy following reactor trip. This is ensured by turn-around of the RCS heat-up event. This event is also analyzed to ensure that the DNB and RCS overpressure criteria are met. The Loss of Normal Feedwater event is initiated by the termination of feedwater flow. Reactor trip occurs on Steam Generator Low-Low Level and, after appropriate delays, auxiliary feedwater is delivered. Eventually, the auxiliary feedwater heat removal capability exceeds core decay heat generation.

To ensure the acceptance criteria for the Loss of Normal Feedwater event continue to be met for the reduced RCS flow condition, this event was reanalyzed. The reanalysis of the Loss of Normal Feedwater event was accomplished utilizing the RETRAN-02 computer code in accordance with WCGS's approved methodology. Results of the analysis show that the peak RCS pressure continues to be maintained below the pressure limit of 110% of design pressure on both the primary and secondary coolant systems. DNB analysis of the

revised core statepoints shows that the minimum DNBR continues to be maintained above the design limit. The long term core cooling event also demonstrates that the auxiliary feedwater system can provide sufficient cooling to prevent this event from progressing to a worse condition event. Therefore the Loss of Normal Feedwater event is shown to meet all applicable acceptance criteria for an ANS Condition II event and therefore it is demonstrated that the margin of safety is maintained.

The results of the revised Loss of Normal Feedwater analysis will be incorporated in Revision 10 of the WCGS USAR.

Feedwater Line Break (USAR 15.2.8)

This ANS Condition IV event is analyzed to show that adequate heat removal capability exists using the Auxiliary Feedwater System to remove core decay heat, stored energy and RCS pump heat following reactor trip. This is demonstrated by ensuring that the RCS heat-up is turned around prior to the time at which the hot leg inventory would become saturated.

The current USAR analysis results support the conclusion that the auxiliary feedwater is sufficient to remove decay heat by demonstrating that there is no hot leg boiling and the peak RCS pressure is within 110% of design. To ensure the acceptance criteria for the Feedwater Line Break event continue to be met for the reduced RCS flow condition, this event was reanalyzed. The reanalysis of the Feedwater Line Break event was accomplished utilizing the RETRAN-02 computer code in accordance with WCGS's approved methodology. Results of the analysis show that the peak RCS pressure continues to be maintained below the pressure limit of 110% of design pressure on both the primary and secondary coolant systems. DNB analysis of the revised core statepoints shows that the minimum DNBR continues to be maintained above the design limit. The analysis also demonstrates that the RCS coolant temperatures remain below the saturation temperature. Therefore the Feedwater Line Break event is shown to meet all applicable acceptance criteria for an ANS Condition II event and therefore it is demonstrated that the margin of safety is maintained.

The results of the revised Feedwater Line Break analysis will be incorporated in Revision 10 of the WCGS USAR.

Partial Loss of Forced Reactor Coolant Flow (USAR 15.3.1)

The Partial Loss of Forced Reactor Coolant Flow event is an ANS Condition II event and analyzed to demonstrate that the minimum DNBR acceptance criteria continues to be met. Due to the proposed reduction of RCS flow, the event was reanalyzed assuming a 3.5% reduction in RCS flow to account for any degradation of heat transfer due to the flow reduction. The analysis was performed utilizing the RETRAN-02 computer code following the WCGS approved methodology. The results of the analysis show that the Partial Loss of Forced Reactor Coolant Flow continues to meet the acceptance criteria for an ANS Condition II event therefore ensuring that the margin of safety is not reduced.

The results of the revised Partial Loss of Forced Reactor Coolant Flow analysis will be incorporated in Revision 10 of the WCGS USAR.

Complete Loss of Forced Reactor Coolant Flow (USAR 15.3.2)

The Complete Loss of Forced Reactor Coolant Flow event is an ANS Condition III event. The Complete Loss of Forced Reactor Coolant Flow is the most limiting of the loss of flow DNBR events. This event is also analyzed to ensure that RCS pressure is maintained below 110% of the design pressure. Due to the proposed reduction of RCS flow, the event was reanalyzed assuming a 3.5% reduction in RCS flow to account for any degradation of heat transfer due to the flow reduction. The analysis was performed utilizing the RETRAN-02 computer code following the WCGS approved methodology. The results of the analysis show that the Complete Loss of Forced Reactor Coolant Flow continues to meet the acceptance criteria for an ANS Condition II event therefore ensuring that the margin of safety is not reduced.

The results of the revised Complete Loss of Forced Reactor Coolant Flow analysis will be incorporated in Revision 10 of the WCGS USAR.

Reactor Coolant Pump Shaft Seizure (USAR 15.3.3)

The Reactor Coolant Pump Shaft Seizure event is classified as a Condition IV event. As such, compliance with the guidelines of 10CFR100 regarding the release of radioactive material must be demonstrated, although DNB is not precluded. For conservative clad temperature calculations, DNB is assumed to occur at the beginning of the event. The event is characterized by an instantaneous seizure of a reactor coolant pump rotor at 100% power. The flow reduction in the affected loop is so rapid that the time of reactor trip on low flow does not change due to the reduction in TDF. However, the TDF reduction may result in an increase in system pressure and fuel clad temperature. Therefore this event has been reanalyzed based on the proposed flow reduction. The analysis was performed utilizing the RETRAN-02 computer code following the WCGS approved methodology. The results of the analysis show that the Reactor Coolant Pump Shaft Seizure event continues to meet the acceptance criteria for an ANS Condition IV event therefore ensuring that the margin of safety is not reduced.

The results of the revised Reactor Coolant Pump Shaft Seizure analysis will be incorporated in Revision 10 of the WCGS USAR.

Reactor Coolant Pump Shaft Break (USAR 15.3.4)

The Reactor Coolant Pump Shaft Break event is an ANS Condition II event and is bounded by the Reactor Coolant Pump Shaft Seizure event. This event has also been reanalyzed assuming a 3.5% RCS flow reduction. The results of the analysis show that the Reactor Coolant Pump Shaft Break event meets all ANS Condition II acceptance criteria and therefore the margin of safety is not reduced.

The results of the revised Reactor Coolant Pump Shaft Break analysis will be incorporated in Revision 10 of the WCGS USAR.

Uncontrolled RCCA Withdrawal from a Subcritical or Low Power Startup Condition (USAR 15.4.1)

The Uncontrolled RCCA Withdrawal from Subcritical event is an ANS Condition II event performed at zero power conditions. It is characterized by a rapid power increase. The power excursion is retarded by Doppler reactivity feedback and the event is terminated by a reactor trip on the Power Range High Neutron Flux Low setpoint. The principal acceptance criterion for this event is DNB. Other calculated event parameters include peak fuel clad average temperature and peak fuel average temperature. The reduced TDF would result in a reduction in the fuel-to-coolant heat transfer; the associated statepoint heat flux reduction would be a DNB benefit due to increased Doppler reactivity feedback. This event has been reanalyzed assuming a 3.5% reduction in RCS flow and the results have been confirmed to be less limiting than the current licensing basis analysis due to increased Doppler reactivity feedback afforded by hotter fuel temperatures due to the TDF reduction. Therefore the current analysis presented in the USAR will remain the licensing basis analysis.

Uncontrolled RCCA Withdrawal at Power (USAR 15.4.2)

This ANS Condition II event is analyzed to show that the DNB design basis is met. Various power levels and reactivity insertion rates for both minimum and maximum reactivity feedback are analyzed. The events are terminated by an OTΔT or High Neutron Flux reactor trip. The more rapid reactivity insertion cases trip on High Neutron Flux at 118% power regardless of flow. The slower reactivity insertion cases trip on OTΔT. Lower flow may cause earlier reactor trip in these cases unless conditions are such that the steam generator safety valve setpoint is reached. In the latter situation, the OTΔT trip may be delayed and event results may worsen. Therefore this event has been reanalyzed to confirm that the acceptance criteria will continue to be met.

The Uncontrolled RCCA Withdrawal at Power event has been reanalyzed utilizing the RETRAN-02 computer code following the approved WCGS event analysis methodology. The analysis assumes a 3.5% reduction in RCS flow. The results of the reanalysis show that the event continues to meet the DNB design basis for an ANS Condition II event. The analysis also demonstrates that the primary and secondary coolant system pressures are maintained below their respective pressure limits (110% of design pressure). Finally it is

demonstrated that the event does not result in pressurizer overfilling, thus demonstrating that the event meets all ANS Condition II acceptance criteria and that the margin of safety is therefore not reduced.

The results of the revised Uncontrolled RCCA Withdrawal at Power analysis will be incorporated in Revision 10 of the WCGS USAR.

Rod Cluster Control Assembly Misoperation (USAR 15.4.3)

RCCA Misoperation is categorized into four types of events. Three of these are classified as ANS Condition II events: dropped RCCA, dropped RCCA bank, and statically misaligned RCCA. The fourth, Single RCCA Withdrawal, is classified as an ANS Condition III event. The statepoint T_{avg} and pressure for these events are insensitive to small variations in reactor coolant flow. Heat flux, primarily a function of power, dropped rod worth, control bank worth and moderator feedback, also will not be significantly affected. On this basis, flow will be the only change to the DNBR statepoints. The ANS Condition II dropped rod statepoints have been re-evaluated for the 3.5% TDF reduction. Evaluation of DNBR assuming the reduced flow shows that the event continues to be maintained above the design limit and that the current licensing basis analysis presented in the USAR remains valid.

Similarly, the Condition III Single RCCA Withdrawal event has been evaluated assuming the core statepoints have remained constant with the exception of a 3.5% reduction in RCS flow. Evaluation of DNBR assuming the reduced flow shows that the event continues to be maintained above the design limit and the calculated percentage of fuel failure will not increase. Therefore the current licensing basis analysis presented in the USAR remains valid.

Startup of an Inactive Loop (USAR 15.4.4)

The Startup of an Inactive Loop event is an ANS Condition II event analyzed to demonstrate that the DNB design basis is met. The event is caused by the starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature. This causes a rapid core power increase due primarily to moderator reactivity feedback. The event's assumptions are chosen to maximize the resulting reactivity insertion. A reduction in TDF will reduce the flowrate to the core from the cooler idle loop. Since this event is driven by the cooling effect on the moderator temperature in the presence of a negative moderator temperature coefficient, the flow reduction will result in a slower event with a reduced power increase. This effect is, therefore, a event benefit.

The impact on the DNBR of the reduced TDF can be conservatively estimated by considering a 3.5% reduction in flow while maintaining the other DNBR statepoints from the current licensing basis analysis. Results of the DNB evaluation show that the event continues to meet the applicable acceptance criteria. Therefore, the event is shown to continue to meet all acceptance criteria for an ANS Condition II event and the current USAR analysis remains valid for the Startup of an Inactive Loop event.

Boron Dilution (USAR 15.4.6)

The Boron Dilution event in the USAR is analyzed for the six technical specification modes of operation, i.e., dilution during refueling, cold shutdown, hot shutdown, hot standby, start-up, and full power operation. The time to lose shutdown margin, which is a measure of the boron dilution event's severity, is a function of dilution flowrate, RCS active or mixing volume, boron worth, boron concentration, and shutdown margin. Therefore, the boron dilution event is insensitive to variations in thermal design flow and, therefore, the conclusions in the USAR remain valid.

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

This ANS Condition III event addresses the possibility and consequences of one or more fuel pellets having the wrong enrichment or the loading of a fuel assembly without the prescribed amount of burnable poisons. The USAR concludes that any significant perturbation from the intended core inventory would be detectable due to the resulting effects on power distribution. The reduced TDF does not affect the ability of the core instrumentation to detect unexpected power shapes. Therefore, the USAR conclusions remain valid.

RCCA Ejection Event (USAR 15.4.8)

The RCCA Ejection event is an ANS Condition IV event analyzed to demonstrate that the following acceptance criteria are met:

Peak Clad Average Temperature	< 2700°F
Fuel Melt	< 10%
Peak Fuel Pellet Enthalpy	< 200 cal/gm, irradiated
	< 225 cal/gm, fresh

Four cases are presented in the USAR. They are beginning of life (BOL) and end of life (EOL) hot full power (HFP) and hot zero power (HZP). The limiting HFP case assumed BOL conditions and the limiting HZP case assumed EOL conditions. The RCCA Ejection Event has been reanalyzed assuming a 3.5% reduction in RCS flow. The analysis was performed with the TWINKLE and FACTRAN computer codes following Westinghouse methodology. The results of the RCCA Ejection event show that the reduction in RCS flow results in a more limiting condition at BOL condition and a benefit at EOL conditions due to reactivity feedback effects. All cases continue to meet the acceptance criteria and therefore there is no reduction in the margin of safety.

The results of the revised BOL RCCA Ejection analysis will be incorporated in Revision 10 of the WCGS USAR.

Inadvertent Actuation of the ECCS During Power Operation (USAR 15.5.1)

Inadvertent Actuation of the Emergency Core Cooling System (ECCS) During Power Operation is an ANS Condition II event analyzed to demonstrate that spurious ECCS operation without immediate reactor trip presents no hazard to the integrity of the RCS. The event is characterized by decreasing nuclear power, decreasing core water temperature, predominately decreasing pressurizer pressure and water level, and increasing DNBR. Once the reactor has tripped on low pressurizer pressure, the continued addition of ECCS flow in addition to coolant swelling due to decay heat and degraded primary-to-secondary heat transfer could lead to a pressurizer overfill condition. Since the proposed reduction of RCS flow could lead to decreased heat transfer rates, the amount of coolant swelling could increase due to increased RCS temperatures following the reactor trip. Therefore this event has been reanalyzed to ensure the acceptance criteria continue to be met. The results of the revised analysis show that the pressurizer does not reach a water solid condition throughout the event, confirming that the event will not lead to a more serious condition. Analyses were also performed to confirm that DNBR increases through the event and that the RCS pressure is maintained below the RCS pressure limit of 110% of design pressure. Based on the results of the analyses, all acceptance criteria for an ANS Condition II event continue to be met, therefore, maintaining the margin of safety.

The results of the revised Inadvertent ECCS Actuation analysis will be incorporated in Revision 10 of the WCGS USAR.

CVCS Malfunction That Increases Reactor Coolant Inventory (USAR 15.5.2)

Chemical and Volume Control System (CVCS) Malfunctions That Increase Reactor Coolant Inventory are ANS Condition II events which are analyzed in the USAR to demonstrate that the operator has sufficient time to mitigate pressurizer filling. This event is characterized by increasing pressurizer level and pressure and constant boron concentration. All of the USAR cases exhibit relatively constant core power and RCS average temperature and increasing RCS pressure. No automatic reactor trip is assumed to be actuated. The USAR analysis results show that none of the operating conditions during the event approach the core thermal limits. Calculated operator action times are documented in USAR Table 15.1-1.

The CVCS Malfunction That Increases Reactor Coolant Inventory is very similar to the USAR 15.5.1 event except that the CVCS cannot provide the same magnitude of charging flow as can the ECCS. Therefore the CVCS Malfunction Resulting in an Increase in RCS Inventory event is bounded by the USAR 15.5.1 event and the conclusions presented in the USAR remain valid for USAR Section 15.5.2.

LOCA and LOCA Related Analyses

Accidental Depressurization of the Reactor Coolant System (USAR 15.6.1)

This ANS Condition II event is analyzed to show that the DNB design basis is met. The USAR event postulates the RCS depressurization to result from the inadvertent opening of a pressurizer safety valve. The event is characterized by decreasing pressure, relatively constant power and core average coolant temperature, and a decreasing DNBR up until the time of reactor trip on OTΔT. The assumption of automatic rod control maintains T_{avg} at a fairly constant value and heat flux remains constant throughout the event. Due to the fact that automatic rod control provides a fairly constant reactor power and temperature throughout the event, the RCS flow reduction is not expected to perturb the DNB statepoints with the exception of RCS flow. Therefore the DNBR for this event was evaluated considering a 3.5% reduction in RCS flow while maintaining the other DNBR statepoints from the current licensing basis analysis. Results of the DNB evaluation show that the event continues to meet the applicable acceptance criteria. Therefore, the event is shown to continue to meet all acceptance criteria for an ANS Condition II event and the current USAR analysis remains valid for the Evental Depressurization of the Reactor Coolant System event.

Steam Generator Tube Rupture (USAR 15.6.3)

The Steam Generator Tube Rupture (SGTR) event is classified as an ANS Condition IV event. The event is characterized by decreasing primary side pressure, increasing charging pump flow, relatively constant power and a decreasing DNBR up until the time of reactor trip on OTΔT. On the secondary side, there is a indication of steam flow/feedwater flow mismatch before the trip as feedwater flow to the ruptured steam generator is reduced due to additional break flow being supplied to that loop.

Due to the nature of the SGTR event that allows a direct path of radioactivity releases to the atmosphere, the radiological doses following the SGTR event are the main concern. The major factors that affect the radiological doses of an SGTR event are the amount of primary coolant transferred to the secondary side of the ruptured steam generator, the amount of fuel failure, and the amount of steam release from the ruptured steam generator to the atmosphere.

An evaluation was performed to determine the impact of a reduced TDF on the radiological doses following the SGTR event. The DNB following the SGTR event was also evaluated. The results of the evaluation indicated that during the time between the reactor trip and ruptured SG isolation, both the primary to secondary break flow and the steam released via the ruptured SG will slightly increase. However based on the conservatism included in the current SGTR radiological consequences analysis, the consequences associated with the increase in mass releases for this evaluation will be bounded by the results of the current analysis.

It has been determined that the DNBR of a SGTR event is bounded by the DNBR of an inadvertent opening of a Pressurizer Safety or Relief Valve. A DNB evaluation has been performed for the Inadvertent Opening of a Pressurizer

Safety or Relief Valve event with TDF reduction. The results of this evaluation indicate that the DNB limits will not be violated. It therefore can be concluded that the core DNB limits will not be violated for the SGTR event due to a 3.5% reduction in RCS flow and the current USAR analysis remains valid.

Loss-of-Coolant Events (LOCA) (USAR 15.6.5)

Large Break LOCA Analysis

The licensed Large Break LOCA Analysis was performed with the 1981 Evaluation Model with BART. Analyses were performed for a range of RCS operating temperatures from an upper bound T_{avg} of 588.4 °F to a lower bound T_{avg} of 570.7 °F in order to provide future operational flexibility. Calculations for both conditions were performed for the limiting Moody break discharge coefficient (Cd = 0.4). The limiting peak cladding temperature (PCT) of 1916 °F was calculated assuming the limiting Moody break discharge coefficient at the lower bound T_{avg} under minimum safeguards assumptions. The lower bound T_{avg} therefore bounded the results of the case assuming the upper bound T_{avg} which resulted in a PCT of 1829 °F. Since the operating T_{avg} as a result of the proposed RCS flow reduction will remain within the T_{avg} window, the Large Break LOCA results presented in the USAR remain valid.

Small Break LOCA Analysis

The licensed Small Break LOCA analysis for WCGS was performed using the Westinghouse NOTRUMP Evaluation Model, for a range of RCS operating temperatures between an upper bound T_{avg} of 588.4 °F and a lower bound T_{avg} of 570.7°F. Analyses were performed for a spectrum of cold leg breaks ranging from an equivalent diameter of 2 inches to 6 inches. The PCT of 1510 °F was calculated for a 3-inch equivalent diameter cold leg break initiating from the upper bound T_{avg} conditions. The effects of the RCS flow reduction on the event characteristics are discussed below.

For a Small Break LOCA analysis, reactor trip occurs when the low pressurize pressure reactor trip setpoint is reached. As primary coolant inventory spills out the cold leg, the RCS depressurizes linearly until the reactor trip setpoint is reached. For equal size breaks, break flow is dependent on the delta-P across the break and the density of the water in the cold leg. The proposed reduction in TDF will not affect primary pressure thus the delta-P across the break will remain the same for this evaluation. However the TDF reduction will effect the temperature in the cold leg and thus the density of the water in the cold leg.

For the proposed reduction in TDF, the temperature in the cold leg was calculated to be less than 1.1 °F lower than the limiting high T_{avg} case analyzed at the current TDF. Thus the density of the cold leg with the reduced TDF would be slightly greater than the case analyzed. Because the differences in densities of water in the cold leg between the two cases is very small (less than 0.2%), there would be no significant difference in RCS depressurization rate or reactor trip time.

After reactor trip there would be no significant difference in the thermal hydraulic response between the original analysis and the reduced TDF case. Additionally, for a Small Break LOCA, a minor perturbation in initial operation conditions should not have an impact on calculated peak clad temperature because of the event sequence. The peak clad temperature of a Small Break LOCA occurs after loop seal clearing. The coolant inventory and core mixture level after loop seal clearing are strongly dependent on steam generator operation conditions and loop seal clearing oscillations during the event. These factors are not affected by the initial RCS flow rate.

The peak hot rod cladding and fuel temperatures are dependent upon the initial fuel rod temperatures that are reinitialized at core uncovery. From the discussion above, the thermal hydraulic conditions at core uncovery will not be significantly different for the reduced TDF case. The only parameter in LOCTA that is not reinitialized at core uncovery is the radial gap between the fuel rod and the fuel. However, calculations have been performed that show that the radial gap between the fuel and the fuel rod is not sensitive to changes in TDF. Consequently, no difference in the thermal hydraulic conditions at core uncovery means there will be no difference in peak clad temperature.

Based on these discussions and the fact that the resulting vessel/core inlet temperature will remain within the analyzed T_{in} window and the operating T_{avg} remains unchanged, the proposed reduction in TDF will have no significant effect on the small break LOCA analysis results. Thus, the reduction in TDF will have no effect on the WCGS small break margin to the PCT limit of 2200 °F and the conclusions presented in the USAR remain valid.

Post-LOCA Long-Term Core Cooling

The reduction in TDF has no impact on long term core cooling following a large break LOCA since this is controlled by the ECCS, the core reactivity, and the total mass of primary coolant and soluble boron that are collected in the containment sump boron concentration. Since the average temperature of the primary loop remains unchanged due to reduction in TDF, the change in primary coolant mass is expected to be negligible.

Hot Leg Switchover to Prevent Potential Boron Precipitation

Post-LOCA hot leg recirculation switchover time is determined for inclusion in emergency operating procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level and the RCS, Refueling Water Storage Tank (RWST), and accumulator water volumes and boron concentrations. For WCGS, the hot leg recirculation switchover time after a design basis Large Break LOCA is conservatively calculated to be 10 hours based on a maximum RWST boron concentration of 2700 ppm. There is significant margin in the calculated switchover time since the maximum Technical Specification value on RWST boron concentration is 2500 ppm. Since the reduction in thermal design flow does not affect power level or the maximum boron concentrations or volumes assumed for the RCS, RWST, and

accumulators, the reduced TDF would not affect the post-LOCA hot leg switchover time.

Radiological Consequences

The effect of a 3.5% TDF reduction would produce only small effects in the primary and secondary side design operating parameters. The primary system operating parameters remain very close to those presented in the USAR except for the reactor coolant flow, therefore, no impact on radiological consequences will occur due to the primary system effects. For the secondary system operating parameters, there is a small decrease in the steam pressure (6 psia) and steam temperature (0.8°F) due to the reduction of steam generator heat transfer capability. In addition, the steam flow remains the same as the current design operating condition. Since the changes are small and the radiological consequences of the events in the USAR are relatively insensitive to the steam pressure and temperature, no significant changes in the current USAR radiological consequences will result. Therefore, the conclusions in the USAR remain valid.

Rod Ejection Mass and Energy Release for Dose Calculation

For this calculation, a small break LOCA is performed with a break in the upper head the size of a control rod drive shaft in order to determine the primary coolant mass released to the containment through the break and the steam released from the steam generator safety valves. This information is then used to calculate the radiological consequences of a rod ejection event. With a reduction in thermal design flow, minor changes could be expected in rod ejection mass and energy releases, primarily as a result of the slight decrease in vessel inlet temperature. Based on previous sensitivity studies, a 1°F decrease in vessel inlet temperature would result in a very small effect on the primary and secondary mass releases. Thus, a reduction of TDF would have no significant impact on the calculated doses for the postulated rod ejection event. Therefore, there is reasonable assurance that the radiological consequences of a postulated rod ejection event will be maintained well within the exposure guidelines as set forth in 10 CFR Part 100 and the current USAR analysis remains valid.

Blowdown Reactor Vessel and Loop Forces

The blowdown vessel and loop forcing functions used as input to the design basis analysis for the effects of a postulated LOCA on the integrity of the reactor vessel internals and loops are documented in USAR Sections 3.6.2 and 3.9.2. A sensitivity study has shown that the primary influence of thermal design flow reduction on the LOCA hydraulic forcing functions is the change in the plant's operating temperatures. With a reduction in TDF, minor changes could be expected in LOCA reactor vessel forces, primarily as a result of a slight decrease in cold leg temperature. A review of the LOCA hydraulic forces analysis performed for the WCGS Power Rerate Program indicates that the forces were calculated for branch line breaks, in consideration of primary loop Leak Before Break methodology. The calculation was conservatively based on a cold leg temperature of 538°F, corresponding to a 15°F T_{hot} reduction.

Since the cold leg temperature for the proposed TDF reduction is still higher than the limiting temperature used in the record of analysis, the LOCA hydraulic forces calculated for the WCGS rerating program continue to bound the plant configuration with the reduction in thermal design flowrate.

Mass and Energy Releases

The following sections address mass and energy release analyses.

LOCA Mass and Energy Release Evaluation (USAR 6.2 and 6.3)

The mass and energy releases used in the containment analyses are described in the USAR sections 6.2.1.2 and 6.2.1.3. These sections consider the mass and energy releases following a design basis pipe break event for containment subcompartment and containment integrity analyses. The impact of the proposed reduction in TDF on these analyses is discussed as follows.

Containment subcompartment analyses of the pressure events in the reactor cavity, steam generator compartment, and pressurizer compartment are performed to demonstrate adequacy of containment internal structures and attachments when subjected to dynamic localized pressurization effects that could occur during the very early time period (i.e. less than 3.0 seconds) following a design basis pipe break event. Subsequent to the postulated rupture, the pressure builds up at a faster rate in the subcompartments than in the overall containment, thus imposing differential pressure across the walls of the subcompartment structures.

The short term mass and energy releases from the design basis breaks for these subcompartments are strongly affected by the initial temperature conditions of the fluid at the break locations. The releases are linked directly to the critical mass flux, which increases with decreasing temperatures. Since the primary effect of a TDF reduction is the impact upon the reactor coolant temperature, a review was performed pertaining to the background information utilized for the current USAR Section 6.2 containment analyses. The results of the review identify that the operating reactor coolant temperatures associated with the proposed reduction in TDF will be bounded by the RCS temperature assumptions used for the WCGS Power Rerate Program. Therefore, the short term LOCA mass and energy releases of record continue to be valid.

For the long term mass and energy release calculations, operating temperatures for the highest average coolant temperature were selected as the bounding conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. A review of the analysis of record indicated that core inlet temperature of 561.55 °F was assumed for the revised mass and energy releases used in the containment integrity evaluation for the WCGS Power Rerate Program. With the proposed reduction in TDF, the inlet core temperature is calculated to be 553.7 °F. Therefore, the calculated long term mass and energy releases will be enveloped by the revised mass and energy releases calculated at rerated conditions.

Secondary Pipe Rupture Mass and Energy Release (Inside and Outside Containment)

The objective of the USAR analysis is to maximize the release of high energy fluid to the containment environment during a steamline rupture. A reduction in TDF reduces primary to secondary heat transfer and the reactivity insertion in the presence of a negative moderator temperature coefficient. Since the assumptions are chosen to maximize the primary side cooldown effect, any reduction in heat transfer capability would tend to reduce the energy release. As such, the energy releases presented in the USAR bound the analysis results for a reduced TDF. It is concluded, therefore, that a reduction in TDF has no detrimental effect on the current USAR analysis results.

Analyses were performed to address the NRC concerns (described in IE Information Notice 84-90) on the effect of superheated steam releases on the environmental qualification of equipment located outside containment. The results of these analyses were documented in WCAP-10961, "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," and were used in plant-specific equipment qualification evaluations. The WCAP studies are performed to maximize the release of high energy fluid as a result of steamline ruptures occurring outside the reactor containment structure. Core reactivity coefficients are chosen to conservatively maximize the reactivity feedback effects of the cooldown resulting from the steamline break. A reduction in TDF which reduces primary to secondary heat transfer, and the reactivity insertion in the presence of a negative moderator temperature coefficient, will improve analysis results. It is concluded, therefore, that the calculated mass and energy release data that were used in the previous evaluations of environmental qualification of equipment remain bounding for a reduced TDF.

IE-79-22, Control and Protection Interaction (SLB/RWAP)

The Steamline Break at Power coincident with Rod Withdrawal is analyzed in order to demonstrate compliance with IE-79-22 (Control and Protection Interaction) and ANS Condition II acceptance criteria. The key analysis result is minimum DNBR. The postulated event scenario includes the failure of the rod control system as the result of the environment created by a steamline rupture. The analysis assumes the control rod withdrawal to occur at the initiation of the event. The steamline break causes an increased heat removal and subsequent decrease in primary pressure concurrent with an increase in reactor power and heat flux due to the rod withdrawal. The power and heat flux increase result in reactor trip on the OPΔT setpoint. This event has been reanalyzed following Westinghouse methodology. The analysis assumes a 3.5% reduction in RCS flow. The results of the reanalysis show that the event continues to meet the DNB design basis for an ANS Condition II event. The analysis also demonstrates that the primary and secondary coolant system pressures are maintained below their respective pressure limits (110% of design pressure). Therefore it is demonstrated that the event meets all ANS Condition II acceptance criteria and that the margin of safety is not reduced.

Events of a Different Type

Neither the proposed reduction in thermal design flow nor the increase in the pressurizer pressure low reactor trip setpoint will create the possibility of an event of a different type than previously evaluated in the USAR. The events assumed to occur at the current thermal design flow will be identical to those for the new thermal design flow.

Impact on Plant Equipment and Balance of Plant

Based on the conclusions reached in the Mass and Energy Releases discussion above, as well as the proposed conditions presented in Table 1, the proposed Technical Specification changes are bounded by the current conditions with respect to system dynamic loading, environmental equipment qualification, and rejection of heat to the Ultimate Heat Sink. These analyses are bounded by the current analyses due to the conclusion that the mass and energy releases will not be impacted by the proposed change. This conclusion is also based on the fact that the current operating conditions bound the proposed operating conditions with respect to the secondary system operating parameters.

Conclusions

Non-LOCA and LOCA safety analyses and evaluations confirm the acceptability of a 3.5% reduction in TDF and a 25 psi increase in the Low Pressurizer Pressure Trip setpoint. The USAR LOCA analyses have been evaluated or previously analyzed for conditions which bound the proposed reduction in TDF. The justification for non-LOCA events is based on event specific safety evaluations, maintenance of the existing DNB core thermal and axial offset limits, and an increase in the Low Pressurizer Pressure trip setpoint Safety Analysis Limit. It is confirmed that all analyses continue to, at a minimum, meet their respective acceptance criteria, demonstrating the acceptability of the proposed reduction in thermal design flow.

All LOCA and non-LOCA safety analysis conclusions remain valid for the proposed changes. In the cases where specific analyses were performed to justify the proposed change, the revised analysis results will be incorporated into Revision 10 of the WCGS USAR.

Therefore, based on the evaluation presented in the previous sections, it is concluded that the proposed technical specification changes for the reduced TDF and increased Low Pressurizer Pressure Trip setpoint may be implemented at the Wolf Creek Generating Station without decreasing the margin of safety and without increasing the risk to the health and safety of the public.

ATTACHMENT II

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

No Significant Hazards Consideration Determination

This license amendment request proposes to revise the Wolf Creek Generating Station (WCGS) Technical Specifications to allow plant operation at 100% rated thermal power (RTP) with a 3.5% reduction in Thermal Design Flow (TDF) and an increase in the Low Pressurizer Pressure Trip setpoint. This revision in thermal design flow represents a decrease in TDF from the current value of 374,400 gpm to 361,296 gpm. The corresponding vessel average temperature (T_{avg}) will remain at the current value of 586.5 °F, however, the decreased TDF will result in a slight decrease in the core inlet temperature (T_{in}) and an approximate 1 °F increase in the hot leg temperature (T_{hot}). The Low Pressurizer Pressure Trip setpoint will be raised from the Safety Analysis Limit (SAL) of 1915 psig to 1940 psig to preclude the occurrence of Departure from Nucleate Boiling (DNB), ensuring that core thermal protection is provided for all conditions of operation.

Standard I - Involve a Significant Increase in the Probability or Consequences of an Event Previously Evaluated

The probability of occurrence and the consequences of an event evaluated previously in the Updated Safety Analysis Report (USAR) are not increased due to the proposed technical specification changes. The technical specification changes being requested are to reflect revised core design parameters affected by the Cycle 9 core reload geometry, and instrumentation setpoint changes needed to ensure accurate measurement of reactor thermal power in order to allow the unit to operate at rated thermal power during Cycle 9. Each USAR Chapter 15 event was evaluated to determine the impact of the reduction in thermal design flow. The events in which the margin to the acceptance criteria was decreased were reanalyzed to support the 3.5% flow reduction. Generally, the RCS heat-up events fall into this category as the reduction in RCS flow results in decreased heat removal capacity. Evaluations of these events were performed using bounding core state parameters based on the previous Safety Analysis submitted in support of the WCGS Power Rerate Program, approved in WCGS Technical Specification Amendment 69. Results of the analyses and evaluations performed for the reduction in thermal design flow for Cycle 9 indicate that all acceptance criteria for USAR Chapter 15 events continue to be met.

Standard II - Create the Possibility of a New or Different Kind of Event from any Previously Evaluated

The requested changes do not create the possibility of a new or different kind of event or malfunction from any previously evaluated. The proposed changes do not change the method and manner of plant operation, nor is any new equipment being installed. Neither the proposed reduction in thermal design flow nor the increase in the Low Pressurizer Pressure Trip setpoint will create the possibility of an event of a different type than previously evaluated in the USAR.

The proposed Technical Specification changes are bounded by the current conditions with respect to system dynamic loading, environmental equipment qualification, and rejection of heat to the Ultimate Heat Sink. These analyses are bounded by the current analyses due to the conclusion that the mass and energy releases will not be impacted by the proposed change. This conclusion is also based on the fact that the current operating conditions bound the proposed operating conditions with respect to the secondary system operating parameters.

Standard III - Involve a Significant Reduction in the Margin of Safety

In general, the Low Pressurizer Pressure Trip setpoint is chosen at a conservatively low value (1885 psig) for the safety analyses. The safety margin (to prevent DNB) is provided by setting the Technical Specification limit for the Low Pressurizer Pressure Trip setpoint at its current value of 1915 psig. Increasing this reactor trip setpoint 25 psi (from 1915 psig to 1940 psig) would result in a net benefit to all analyses which assume its use, as well as offsetting a potential reduction in the margin of safety for this parameter, caused by the reduction in TDF. Therefore, the current Safety Analysis Limit of 1885 psig will continue to be used in the WCGS event analyses.

The proposed changes do not change the plant configuration in a way that introduces a new potential hazard to the plant and do not involve a significant reduction in the margin of safety. The analyses and evaluations discussed in the safety evaluation (Attachment I) demonstrate that all applicable design criteria continue to be met for the changes. Therefore, it is concluded that the margin of safety, as described in the bases to any technical specification, is not reduced.

Based on the above discussions, it has been determined that the requested technical specification changes do not involve a significant increase in the probability or consequences of an event or other adverse condition over previous evaluations; or create the possibility of a new or different kind of event or condition over previous evaluations; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration.

ATTACHMENT III

ENVIRONMENTAL IMPACT DETERMINATION

Environmental Impact Determination

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as specified below:

(i) the amendment involves no significant hazards consideration

As demonstrated in Attachment II, the proposed amendment request does not involve any significant hazards consideration.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite

The effect of a 3.5% TDF reduction at WCGS would produce only small effects in the primary and secondary side design operating parameters. The primary system operating parameters remain very close to those presented in the USAR except for the reactor coolant flow, therefore, no impact on radiological consequences will occur due to the primary system effects. For the secondary system operating parameters, there is a small decrease in the steam pressure (6 psia) and steam temperature (0.8°F) due to the reduction of steam generator heat transfer capability. In addition, the steam flow remains the same as the current design operating condition. Since the changes are small and the radiological consequences of the events in the USAR are relatively insensitive to the steam pressure and temperature, no significant changes in the current USAR radiological consequences will result.

(iii) there is no significant increase in individual or cumulative occupation radiation exposure

The proposed amendment request would not adversely affect the operation of the reactor, and would not affect any system that would affect occupational radiation exposure. The proposed changes do not create additional exposure to personnel nor affect levels of radiation present. The proposed changes will not result in any increase in individual or cumulative occupational radiation exposure.

Based on the above, it is concluded that there will be no impact on the environment resulting from the proposed amendment request, and that the proposed request meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to requiring a specific environmental assessment by the Commission.