November 7, 1991

Docket No. 50-482

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Mr. Bart D. Withers
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

Dear Mr. Withers:

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. 80600)

The Commission has issued the enclosed Amendment No. 52 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications in response to your application dated June 11, 1991 as supplemented by letters dated August 30, 1991, and September 20, 1991.

The amendment revises Technical Specification Tables 2.2.1, 4.3.1 and associated bases to reflect the replacement of the existing resistant temperature detector bypass system with a thermowell system.

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,

Original Signed By

William D. Reckley, Project Manager
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 52 to NPF-42
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDIV-2/LA	PDIV-2/PM	PRPB	OGC	PDIV-2/D		
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DATE	10/23/91	10/23/91	10/24/91	11/7/91	11/6/91		

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 52
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) Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated June 11, 1991 and supplemented by letters dated August 30, 1991, and September 20, 1991, 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 Facility Operating License No. NPF-42 is hereby amended to read as follows:

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 52, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne C. Black, Director
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 7, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 52

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
2-4	2-4
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
3/4 3-9	3/4 3-9
3/4 3-12a	3/4 3-12a
B 2-5	B 2-5

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤112.3% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤28.3% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤35.3% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤10^5 cps	≤1.6×10^5 cps
7. Overtemperature ΔT	7.2	3.50	2.72	See Note 1	See Note 2
8. Overpower ΔT	5.5	1.83	0.17	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.7	0.71	2.49	≥1915 psig	≥1906 psig
10. Pressurizer Pressure-High	7.5	0.71	2.49	≤2385 psig	≤2400 psig
11. Pressurizer Water Level-High	8.0	2.18	1.96	≤92% of instrument span	≤93.9% of instrument span

*RTP = RATED THERMAL POWER
**Loop design flow = 93,750 gpm

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.10;
K_2	=	$0.0137/^{\circ}F$;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 = 28$ s, $\tau_5 = 4$ s;
T	=	Average temperature, $^{\circ}F$;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	588.5°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000671;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -27% and + 7%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -27%, the ΔT Trip Setpoint shall be automatically reduced by 1.57% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 0.85% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.6% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_4	=	1.08;
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature;
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation;
τ_7	=	Time constant utilized in the rate-lag compensator for T_{avg}, $\tau_7 = 10$ s;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00128/°F for T > T" and K_6 = 0 for $T \leq T''$;
T	=	Average temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F);
S	=	Laplace transform operator, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.7% of ΔT span.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4) M(3, 4) Q(4, 6) R(4, 5)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5, 12)	S/U(1),Q(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low	S	R	Q	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R	Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High	S	R	Q	N.A.	N.A.	1
12. Reactor Coolant Flow-Low	S	R	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-Low-Low	S	R	Q(15)	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
15. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
16. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2##
b. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(13) DELETED.

(14) DELETED.

(15) The MODES specified for these channels in Table 4.3-2 are more restrictive and, therefore, applicable.

(16) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(17) Local manual shunt trip prior to placing breaker in service.

(18) Automatic undervoltage trip.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT Reactor trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own Trip Setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 52 TO 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 June 11, 1991, as supplemented by letters dated August 30, 1991, and September 20, 1991, Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station. The proposed changes would revise Technical Specification Tables 2.2.1, 4.3.1, and associated Bases to reflect the replacement of the existing resistant temperature detector (RTD) bypass system with an RTD thermowell system. This design modification is intended to overcome drawbacks associated with the RTD bypass system such as potential reactor coolant leakage from system components and radiation exposure of personnel performing work in proximity to the RTD bypass piping. The engineering design and portions of the actual plant modification are being provided by Combustion Engineering (CE) which has performed similar modifications at the Salem and Callaway plants. The August 30, 1991, and September 20, 1991, submittals provided additional clarifying information and did not change the initial no significant hazards consideration determination.

2.0 BACKGROUND

2.1 Current Method

The current coolant temperature measurement system design measures the primary coolant loop temperature by diverting a portion of the reactor coolant into the bypass manifolds. The bypass manifolds utilize direct immersion RTDs to measure the reactor coolant system (RCS) hot and cold leg coolant temperatures. These measurements are input to the protection and control system logics which also calculate average and differential temperatures.

The currently installed bypass system is designed to account for streaming (non-uniform stratified temperatures) in the hot legs by having three hot leg sampling scoops, 120 degrees apart, protrude into each hot leg. Each scoop has five holes which sample the hot leg flow along the leading edge of the scoop. The flows from each hole mix to provide an average scoop temperature and then the flows from each scoop are combined to provide a representative average temperature for each hot leg. The cold leg temperature is measured in a similar manner except a single sample line is provided and a scoop does not protrude into the cold leg piping. The cold leg sampling method is less complicated due to the uniform temperature profiles in the pipe due to the mixing provided by the reactor coolant pumps. After the temperatures are measured by the direct immersion RTDs in the hot and cold leg manifolds, the samples are returned to the RCS via a return line which penetrates the crossover piping between the steam generator and reactor coolant pump.

The bypass system valves and socket-welded pipes act as crud traps which increase the radiation exposure to personnel during plant maintenance and surveillance activities being conducted on both the bypass system and other nearby systems or components. The bypass system also increases plant startup delays because of valve and flange leakage and potential RTD Swagelock fitting leakage and potential flow problems associated with the bypass system valves. Therefore, due to the maintenance and ALARA ("as low as is reasonably acheivable" radiation doses) concerns, the bypass system is being removed and the RTDs are being housed in thermowells which are mounted directly into the RCS hot and cold legs.

2.2 New Method

The proposed method for measuring hot and cold leg temperatures uses narrow-range, dual element, fast response RTDs manufactured by the Weed Company. One element of each RTD is used while the other is provided as an installed spare. The RTDs are placed in thermowells to allow replacement without draindown of the RCS. Three RTDs for each hot leg will be located within the existing sampling scoops. Outlet ports will be added to each scoop to direct the flow past the thermowell and the sensing element of the RTD located inside. The temperature measurement of the three RTDs in each hot leg are averaged electronically to arrive at a representative loop hot leg temperature. A single dual-element RTD is planned for a thermowell in each cold leg since compensation for temperature streaming is not required.

3.0 EVALUATION

The evaluation of the proposed replacement of the RTD bypass system with the thermowell system requires consideration of the potential impacts on measurement accuracies, system response time, system testing, system qualification (functional, seismic, and environmental), RTD failure detection ability, protection system setpoint determinations, safety analysis assumptions, and radiation exposure to plant personnel. Each of these areas are addressed in the evaluations provided below.

3.1 Measurement Accuracies

The new method of measuring each hot leg temperature with three RTDs installed in thermowells and obtaining an average by using electronic components has been analyzed to be slightly more accurate than the existing RTDs and the bypass system. The accuracy of the Weed RTDs has been verified by CE to be an improvement over the existing RTDs. This accuracy includes errors from hysteresis, repeatability, and drift over a 24-month period. The measurement uncertainty for each hot leg RTD is also reduced in importance due to the averaging of the three RTDs to calculate each hot leg temperature. The additon of electronic components to perform the averaging was accounted for in the uncertainty calculations. The accuracy determinations also accounted for temperature streaming and the limitations of the scoop and thermowell arrangement. A model test has been completed and calculations performed to ascertain that an accurate mixed mean temperature will be measured. The model test provided information for the selection of the proper location of the RTD sensor in the scoop for accurate measurement and the expected temperature bias. The licensee has committed to obtain confirmatory information on the mixed mean temperature accuracy. This will be done by comparing pre-installation and post-installation calorimetric data for the RTD measurements at Wolf Creek for matching operating conditions. As stated in the August 30, 1991 letter, the licensee will make this data available to the staff.

3.2 System Response Time

The overtemperature delta-T (OTDT) reactor trip function response time is the time lag from when the hot leg temperature reaches trip conditions at the scoop until the control rods start to drop into the core. As shown in Table 1, the OTDT response time for the proposed system has some gains and losses compared to the existing system, but the total response time of the proposed system is improved over the existing system (5.5 seconds vs. 6.5 seconds).

As shown in Table 1, the testable time delay Technical Specification (TS) limit is 6.0 seconds. The testable time delay (excludes transport delay and thermal lag) for the existing system is 4.5 seconds and 5.25 seconds for the proposed system. This makes the proposed testable system response time for the thermowell installed system slightly (0.75 seconds) longer than the existing system. However, the total time delay for the proposed system is compensated for by a reduction in the loop and scoop transient thermal lag response time, resulting in a lower first order lag for the proposed system versus the existing system (5.0 seconds vs. 6.0 seconds). The electronic delay is 0.5 seconds for each system and therefore the total time delay for the thermowell system is 5.5 seconds which is less than the 6.5 second delay associated with the bypass system. The response time used in the safety analysis is 8.0 seconds which provides ample margin between the safety analysis and the TS values.

3.3 System Testing

The response time of the thermowell RTDs will be tested in place by using a Loop Current Step Response (LCSR) test during each refueling outage interval in accordance with TS 4.3.1.2. The allocated response time for the RTDs includes a 10 percent error allowance for LCSR testing. The LCSR method of response time testing is recommended by the NRC as detailed in NUREG-0809, "Safety Evaluation Report Review of Resistance Temperature Detector Time Response Characteristics." The LCSR method of response time testing uses an external electrical current (20-40 ma) to heat the RTD element and the temperature transient in the element is recorded. From this transient, the response of the RTD to changes in external temperature is inferred. The Weed RTD is capable of being tested by the in-situ LCSR method and a continuous current of 20-40 ma will not damage the RTD.

3.4 System Qualification

The proposed system will use a 4-wire Weed model N9004 RTD, fast response dual element RTD/thermowell assembly with the required fittings and sealed flexible tubing necessary to mate up with the quick disconnect assembly, junction boxes, splices, field cables, and containment penetration module assemblies. It will be connected by field wiring to the existing 7300 process cabinets. All of the above components and systems have been qualified pursuant to the requirements of the applicable IEEE standards and 10 CFR 50.49.

3.5 RTD Failure Detection

A failed RTD would be detected by the loop delta-T versus auctioneered (high) delta-T alarm and/or the loop average temperature versus auctioneered (high) average temperature alarm. In addition, each channel is checked once per 12 hours as required by TS 4.3.1.1. When a failed RTD element is identified, the TS Action Statement would require the protection channel associated with the failed RTD to be placed in the trip condition. Since the OTDT function utilizes a 2-out-of-4 logic, the failed RTD would not prevent safe operation or a safe shutdown of the plant. The second element of each RTD is an installed spare which is connected to the master test cards in the 7300 process cabinets; therefore, this facilitates changing to the spare element as well as minimizing the time that one channel would be in a tripped condition.

3.6 Protection System Setpoints

Setpoint calculations were made for the OTDT and the overpower delta-T (OPDT) reactor trip functions using previously approved setpoint methodologies to account for the new hot leg RTDs and the added electronics. The setpoint calculations also accounted for the various process uncertainties such as temperature streaming in the hot legs that could not be totally compensated for by the physical arrangement of the RTD thermowells. Although the setpoints for the OTDT and OPDT functions did not change, the revised setpoint calculations did result in changes to the allowable values, sensor error and "Z" values provided in TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints."

In addition to the protection system setpoints, the proposed changes to the RCS temperature measurement system has the potential to affect the accuracy of the calorimetric measurements of reactor power and reactor coolant flowrate. The revised uncertainties associated with the proposed RTDs were incorporated into an evaluation of the power and flow uncertainties and it was determined that the existing allowances remain conservative for the RTD thermowell system.

3.7 Safety Analysis

The impact of the RTD bypass elimination on USAR Chapter 15 non-LOCA accidents was evaluated by the licensee. As previously mentioned, the RTD outputs are used by the OTDT and OPDT protection functions. The OTDT reactor trip function is the primary trip credited, while the OPDT reactor trip provides backup protection against excessive power increases. Since the temperature measurement response time and accuracy is not degraded by the replacement of the RTD bypass system with the thermowell system, the existing conclusions of the USAR transient analyses remain valid.

The replacement of the RTD bypass system has been found to not impact the uncertainties associated with RCS temperature measurement or the calorimetric measurements of reactor power or reactor coolant flowrate. The removal of the RTD bypass system and installation of the thermowell system will not affect the LOCA analyses inputs and hence, the results of the existing analyses remain unaffected. Therefore, the plant design changes due to the removal of the RTD bypass system are acceptable from a LOCA analysis standpoint without requiring any detailed reanalysis.

3.8 Radiation Exposure

A motivating factor for the replacement of the existing RTD bypass system with the thermowell system is to eliminate the potential leakage and crud traps which result in increased radiation exposures to personnel performing work on or near the RTD bypass system components. An exposure savings of approximately 60-90 man-rem per refueling outage has been projected as a result of this plant modification. This is equivalent to approximately 2000 man-rem dose savings over the remaining life of the plant, assuming a 40-year operating license. The exposure savings identified in the licensee's ALARA cost-benefit analysis is based upon the reduced radiation levels, reduced maintenance requirements for the RTD bypass system, increased accessibility inside the bioshield, elimination of inservice inspection weld examination interferences, exposure associated with implementing the modification, reduced forced outage potential, and increased unit reliability over the life of the plant.

The licensee has taken several steps to ensure that the radiation dose experienced by those workers performing the replacement of the RTD bypass system is maintained as low as is reasonably acheivable. Wolf Creek personnel observed and taped portions of the implementation of the plant modification to remove the RTD bypass system at the Callaway plant. Additionally, licensee personnel have participated in an RTD tooling/activity demonstration provided by the vendor. These activities have been factored into the ALARA techniques being considered for this modification which include:

a. Mock-up training for shielding installation and pipe demolition.

b. Design and fabrication of special pipe handling tools for cutting, removing and transferring pipe from inside the bio-shield.

c. Utilization of remote cameras to allow fire watches and health physics personnel to monitor modification activities from outside the bio-shield.

d. Testing of various pipe cutting techniques for determining the best method of minimizing debris, improving the timeliness of cuts and minimizing the number of blade changes.

4.0 EVALUATION OF TECHNICAL SPECIFICATIONS

As a result of the modifications associated with the removal of the RTD bypass system and the installation of the thermowell system, changes to the plant's Technical Specifications were proposed. The following Technical Specifications were examined:

Change 1: Table 2.2.1

Table 2.2.1 is revised to reflect the changes to the values of the maximum allowable deviation from the setpoint, the sensor error (S), and the combination of uncertainties included in the "Z" term for the OTDT and OPDT functions. These changes are a result of the reanalysis of the measurement uncertainties and the differences in the various uncertainty contributors between the existing bypass system and the proposed thermowell system. The table is also revised to remove a reference to the RTD manifold instrumentation. These revisions have been reviewed by the staff and determined to be acceptable.

Change 2: Table 4.3.1

Table 4.3.1 has been revised to delete Note 13 from the instrumentation surveillance requirements. This note requires that the RTD bypass loops flow rate be included in the surveillance performed each refueling outage. The note is not relevant after the removal of the bypass system and therefore the staff deems this change to be appropriate.

Change 3: Bases

This change to the Bases deletes a reference to the transit delay associated with the temperature measurements by the RTD bypass system. Since the modification involves the removal of the RTD bypass piping, the time delays associated with the temperature measurement have changed as detailed in Table 1 and a revision to the Bases is required.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to 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 (56 FR 37595). 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.

Attachment:
Table

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Date: November 7, 1991

Table 1

Overtemperature Delta-T Response Times

	Time Delays (seconds)		
	Existing	Proposed	Safety Analysis
I. First Order Lags			
a. Direct Immersion (Manifold) RTD	4.0*	n/a	
b. Combined RTD and Thermowell	n/a	4.75*	
c. Bypass Piping and Thermal Lag	2.0	n/a	
d. Scoop Transport and Thermal Lag	included in c	0.25	
SUBTOTAL- First Order Lags	6.0	5.0	6.0
II. Pure Time Delays			
a. Electronics	0.30	0.30	
b. SSPS	0.001	0.001	
c. Reactor Trip Breakers	0.167	0.167	
SUBTOTAL - Pure Delays	0.50**	0.50**	2.0
TOTAL - Testable Time Delays***	4.5	5.25	
TOTAL - Time Delays	6.5	5.5	8.0

* Includes 10% testing allowance for LCSR testing. Existing RTD response time makes use of margin available in OTDT analysis

** Delays total 0.468 seconds but were rounded to 0.50 seconds

*** Technical Specification limit is 6.0 seconds (excludes transport delays and thermal lags)