WOLF CREEK
NUCLEAR OPERATING CORPORATION

Richard A. Muench
Vice President Engineering
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DEC 7 2000

ET 00-0041

U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555

Subject: Docket No. 50-482: Revision to Technical Specification 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation"

Gentlemen:

Wolf Creek Nuclear Operating Corporation (WCNOC) herewith transmits an application for amendment to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS).

This amendment application would add Surveillance Requirement (SR) 3.3.2.10 to Functions 6.f and 6.h of Technical Specification (TS) Table 3.3.2-1. SR 3.3.2.10 requires that ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIMES be verified to be within limits every 18 months on a staggered test basis. Function 6.f covers the start of the turbine-driven auxiliary feedwater pump on a loss of offsite power. Function 6.h covers the auxiliary feedwater pump suction transfer from the condensate storage tank (CST) to the Essential Service Water (ESW) System on low suction pressure.

In accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That are Insufficient to Assure Plant Safety," administrative controls were implemented soon after the discovery of both of these issues involving insufficient Technical Specifications. In both cases, only a portion of the circuitry had not been response time tested. The response times of those portions of the affected functions were either determined from a review of plant computer data or were measured during supplemental testing.

As required by SR 3.3.2.10, the turbine-driven AFW pump spin-up time is measured every 18 months on a STAGGERED TEST BASIS. This test measures the time required for the turbine-driven AFW pump to start, including required valve strokes, and reach the required discharge pressure after slave relay K641 energizes when a steam generator water level low-low signal is initiated. The response time of the undervoltage relay that detects the loss of offsite power is measured in accordance with SR 3.3.5.4. The only portion of Function 6.f that was not previously response time tested is the BOP-ESFAS separation group 2 actuation circuitry. A review of plant computer data taken during Refueling Outage X ESFAS testing verified that the turbine-driven AFW pump start signal occurred in less than 2 seconds after the main feeder breaker opened upon a simulated loss of offsite power. After adding this value to the measured response times for the rest of the equipment required for Function 6.f, a total response time less

A001

than the required limit of 60 seconds was obtained. This function was response time tested prior to startup from Refueling Outage XI.

The portion of Function 6.h that was not previously response time tested extends from the pressure transmitters that detect low suction pressure from the CST through the BOP-ESFAS separation group 1 and 4 actuation bistables. The response time of this portion of Function 6.h was measured prior to startup from Refueling Outage XI. After adding this value to the measured response times for the rest of the equipment required for Function 6.h, a total response time less than the required limit of 60 seconds was obtained.

The WCNOC Plant Safety Review Committee and the Nuclear Safety Review Committee have reviewed this amendment application. Attachments I through VII provide the required affidavit, Description and Assessment, Markup of TS page, Markup of TS Bases pages, Retyped TS page, Proposed USAR Changes, and the List of Commitments, respectively, in support of this amendment request. Attachments IV and VI are provided for information only.

WCNOC requests approval of the proposed license amendment by May 1, 2001. The amendment will be implemented within 60 days of issuance of the license amendment.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared in connection with the issuance of this amendment.

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 should have any questions regarding this submittal, please contact me at (316) 364-4034, or Mr. Tony Harris at (316) 364-4038.

Very truly yours,

Richard A. Muench

RAM/rlr

Attachments:
- I - Affidavit
- II - Description and Assessment
- III - Markup of Technical Specification page
- IV - Markup of Technical Specification Bases pages (for information only)
- V - Retyped Technical Specification page
- VI - Proposed USAR Changes (for information only)
- VII - List of Commitments

cc: V. L. Cooper (KDHE), w/a
J. N. Donohew (NRC), w/a
W. D. Johnson (NRC), w/a
E. W. Merschoff (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Richard A. Muench, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering and Information Services 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 RAMuench
Richard A. Muench
Vice President Engineering
and Information Services

SUBSCRIBED and sworn to before me this 7th day of Dec., 2000.

NOTARY PUBLIC STATE OF KANSAS | RHONDA L. RODGERS My Appt. Exp. 5-11-2002

Rhonda L. Rodgers
Notary Public

Expiration Date May 11, 2002

ATTACHMENT II

DESCRIPTION AND ASSESSMENT

DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

1.1 This amendment application would revise Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," by adding Surveillance Requirement (SR) 3.3.2.10 to Functions 6.f and 6.h of TS Table 3.3.2-1.

1.2 Updated Safety Analysis Report (USAR) Section

Changes to the USAR will also be required after this amendment application is approved. Changes to USAR Chapter 15 are required to reflect auxiliary feedwater (AFW) assumptions related to the analysis of the small break loss of coolant accident (LOCA).

2.0 DESCRIPTION

This amendment application would add SR 3.3.2.10 to Functions 6.f and 6.h of TS Table 3.3.2-1. SR 3.3.2.10 requires that ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIMES be verified to be within limits every 18 months on a STAGGERED TEST BASIS. Function 6.f covers the start of the turbine-driven AFW pump on a loss of offsite power. Function 6.h covers the AFW pump suction transfer from the condensate storage tank (CST) to the Essential Service Water (ESW) System on low suction pressure.

Attachment III provides the TS markups.

3.0 BACKGROUND

SR 3.3.2.10 requires a verification that ESF RESPONSE TIMES are within their limits every 18 months on a STAGGERED TEST BASIS, as defined in the TS. As discussed in the SR 3.3.2.10 Bases, TS Bases Table B 3.3.2-2 establishes the acceptance criteria time limits for the response time tests. These limits are less than or equal to the maximum values assumed in the accident analyses. However, the SR 3.3.2.10 Bases also states: "No credit was taken in the safety analyses for those channels with response times listed as N.A. No response time testing requirements apply where N.A. is listed in Table B 3.3.2-2." Table B 3.3.2-2 lists the associated response times as N.A. for items 10 (Loss-of-Offsite Power - Start Turbine Driven Auxiliary Feedwater Pump) and 12 (Auxiliary Feedwater Pump Suction Pressure-Low - Transfer to Essential Service Water). As a result, the current Wolf Creek Generating Station (WCGS) Response Time Testing Program does not verify the response times for the start of the turbine-driven AFW pump on a loss of offsite power signal or the switchover of the AFW pump suction from the CST to the ESW System.

Prior to initial licensing of WCGS in 1985, Westinghouse confirmed that neither the start of the turbine-driven AFW pump on a loss of offsite power signal nor the switchover of the AFW pump suction from the CST to the ESW System was assumed in any Westinghouse analyses. Since Westinghouse had performed all the licensing basis Final Safety Analysis Report (FSAR) Chapter 15 accident analyses, it was appropriate to list "N.A." for the response time of these functions in the original licensing basis version of TS Table 3.3-5, "Engineered Safety Features

Response Times." This Table was subsequently relocated to the USAR as Table 16.3-10 in License Amendment No. 93 (December 12, 1995) in accordance with NRC Generic Letter 93-08, "Relocation of Technical Specification Tables of Instrument Response Time Limits" and to TS Bases Table B 3.3.2-2 with the implementation of Amendment No. 123 (March 31, 1999).

Small Break LOCA Analysis

The original licensing basis WCGS small break LOCA analysis was performed by Westinghouse utilizing the WFLASH Evaluation Model. That Evaluation Model did not credit the turbine-driven AFW pump; only the motor-driven AFW pumps were modeled.

The current WCGS small break LOCA analysis of record was performed by Westinghouse utilizing the NOTRUMP Evaluation Model. This analysis was last performed in 1992 in support of the power rerate at WCGS. In the NOTRUMP Evaluation Model the turbine-driven AFW pump and one motor-driven AFW pump is modeled. The current WCGS small break LOCA analysis models AFW delivery and credits its actuation on a loss of offsite power signal. A delay of not more than 60 seconds has been modeled in the analysis to account for the time between the AFW actuation signal and the time at which the turbine-driven AFW pump is at full flow. The response time of less than or equal to 60 seconds accounts for the delays associated with the delivery of flow from the turbine-driven AFW pump for mitigation of a small break LOCA. A loss of power, assumed to occur coincident with the reactor trip signal, is sensed and processed by the circuitry for loss of power diesel generator start covered by TS LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation." The 4.16 kV ESF bus undervoltage relays provide inputs to the load shedder and emergency load sequencer (LSELS) cabinets, which in turn provide a signal to BOP-ESFAS cabinets via LSELS output relays. The BOP-ESFAS separation group 2 signal opens the steam supply valves to start the turbine-driven AFW pump. The response time includes the signal actuation delays as well as valve stroke and turbine-driven AFW pump spin-up times.

In 1992, neither Westinghouse nor Wolf Creek Nuclear Operating Corporation (WCNOC) recognized the impact on TS Table 3.3-5 (currently TS Bases Table B 3.3.2-2), item 10, as a result of the modeling of the turbine-driven AFW pump flow in the revised WCGS small break LOCA analysis.

Non-LOCA Transient Analyses

Furthermore, every non-LOCA transient analysis in USAR Chapter 15 that credits AFW pump start within 60 seconds after the associated ESF actuation signal is generated (e.g., loss of non-emergency AC power, loss of normal feedwater, and feedwater system pipe break) must assume that the safety grade suction source, provided by the ESW System, is credited.

The preferred suction source for the AFW System is the CST, which is designed as a non-seismic Category I, non-nuclear safety related tank. When the AFW System is started, the CST provides a clean water source to feed the steam generators. In the event of a loss or depletion of the water supply in the CST, automatic switchover of the suction of the AFW pumps from the CST to the backup safety grade source of water from the ESW System is provided. USAR Section 10.4.9, states, in part:

> "The AFS, in conjunction with the condensate storage tank (nonsafety-related) or essential service water, provides feedwater to maintain sufficient steam generator level to ensure heat removal from the reactor coolant system in order to achieve a safe shutdown following a main feedwater line break, a main steam line break, or an abnormal plant situation requiring shutdown. The auxiliary feedwater system is capable of delivering full flow when required, after detection of any accident requiring auxiliary feedwater (refer to Chapter 15.0)."

The NRC Safety Evaluation Report (NUREG-0881 which refers to NUREG-0830) Section 9.2.6, states, in part:

> "The [condensate storage] system was evaluated and found to have no functions necessary for achieving safe shutdown conditions or for accident prevention or mitigation. Although the system normally serves as a water source for the safety-related auxiliary feedwater system because there is an automatic switchover to the essential service water system in the event of a loss of the condensate storage tank."

Based on the above, the implied safety analysis assumption is that an automatic switchover to the ESW System will occur for a loss of the CST. As such the Response Time Testing Program, that ensures the Chapter 15 analyses are met, must be revised. The response times included in TS Bases Table 3.3.2-2, items 10 and 12, must be revised from N.A. to 60 seconds.

4.0 TECHNICAL ANALYSIS

Design Basis Function

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The system has two motor-driven pumps and a turbine-driven pump. The AFW pumps normally take suction through a common suction line from the preferred source of water, the CST. A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water from the CST. Three pressure switches are located in the AFW pump suction line from the CST. A low pressure signal sensed by any two of the three switches coincident with an auxiliary feedwater actuation signal will automatically realign the AFW pump suctions to the safety related ESW System. The CST capacity allows the plant to remove decay heat from the primary system during a 4-hour station blackout event, however, the CST is not the safety related source of water to the AFW pumps. The CST is the preferred suction supply to the AFW pumps due to the quality of the water. However, the CST is a non-seismic structure that can not be credited in USAR Chapter 15 analyses. The safety related source of water to the AFW pumps is provided by the ESW System.

Loss of power to either ESF bus will start the turbine-driven AFW pump to ensure the steam generators contain sufficient inventory to serve as the heat sink for reactor decay heat and sensible heat removal. A loss of offsite power is indicated by the undervoltage relays detecting a loss of voltage on each ESF bus, as discussed in the Bases for TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," Function 6.f, "Auxiliary Feedwater - Loss of Offsite Power," and in TS 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation." Upon satisfying the two-out-of-four loss of voltage logic in the LSELS

cabinets, LSELS output relays feed the redundant separation groups 1 and 4 BOP-ESFAS cabinets. The separation groups 1 and 4 BOP-ESFAS cabinets perform several functions, but on a loss of power signal these cabinets perform only two functions, i.e., the steam generator blowdown and sample lines are isolated and electrically isolated outputs are provided to the separation group 2 BOP-ESFAS cabinet. The separation group 2 BOP-ESFAS cabinet is only associated with the turbine-driven AFW pump and is considered to be part of that end device, i.e., the turbine-driven AFW pump, in the Technical Specifications. On a loss of power signal, the turbine steam supply valves (ABHV0005 and ABHV0006) and the turbine trip and throttle valve (FCHV0312) are opened for the start of the turbine-driven AFW pump.

Impact on Nuclear Safety

Adding SR 3.3.2.10 to Functions 6.f and 6.h of TS Table 3.3.2-1 will ensure that accident analysis assumptions regarding the response time of equipment credited in accident mitigation are verified on a periodic basis.

Probabilistic Safety Assessment (PSA) Evaluation

There is no impact on the WCGS PSA since that study is concerned mainly with equipment functionality, not response time. The functional performance of the AFW and ESW Systems is demonstrated by the Surveillance Requirements of TS 3.7.5, "Auxiliary Feedwater (AFW) System," and TS 3.7.8, "Essential Service Water (ESW) System." Functional capabilities of the instrumentation is demonstrated by the Surveillance Requirements of TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," in Table 3.3.2-1 for Functions 6.f and 6.h as well as the Surveillance Requirements of TS 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation," which feeds a signal to the BOP-ESFAS by relay actuation for starting the turbine-driven AFW pump. Equipment response time is inherent in the PSA assumption that the equipment will perform its intended function. Adding SR 3.3.2.10 to Functions 6.f and 6.h of TS Table 3.3.2-1 will ensure the equipment performs its intended function.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

WCNOC has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation will be unaffected. These protection systems will continue to function in a manner

consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The proposed change imposes more stringent surveillance testing requirements to ensure safety related structures, systems, and components are tested in a manner consistent with the safety analysis and licensing basis.

The proposed change will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the USAR.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   There are no hardware changes nor are there any changes in the method by which any safety related plant system performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No performance requirements will be affected; however, the proposed change does impose additional surveillance testing requirements. These additional requirements are consistent with assumptions made in the safety analysis and licensing basis.

   No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. There will be no adverse effect or challenges imposed on any safety related system as a result of this change.

   This change does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

   Therefore, the proposed changes do not create a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak

cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

The imposition of more stringent surveillance testing requirements increases the margin of safety by ensuring that the affected safety analysis assumptions on equipment response time are verified on a periodic frequency.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above evaluations, WCNOC concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

5.2 Regulatory Safety Analysis

Applicable Regulatory Requirements/Criteria

The regulatory bases and guidance documents associated with the systems discussed in this amendment application include:

GDC-13 requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

GDC-20 requires that the protection system(s) shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC-21 requires that the protection system(s) shall be designed for high functional reliability and testability.

GDC-22 through GDC-25 and GDC-29 require various design attributes for the protection system(s), including independence, safe failure modes, separation from control systems, requirements for reactivity control malfunctions, and protection against anticipated operational occurrences.

GDC-34 requires that a system to remove residual heat shall be provided. When the plant is above the residual heat removal cut-in conditions, the secondary side of the plant including the AFW System removes residual heat.

GDC-44 requires a cooling water system(s) be provided to transfer heat to an ultimate heat sink. GDC-45 and GDC-46 require the cooling water system(s) be designed to permit inspection and testing. These criteria apply to the ESW System at WCGS.

Regulatory Guide 1.22 discusses an acceptable method of satisfying GDC-20 and GDC-21 regarding the periodic testing of protection system actuation functions. These periodic tests should duplicate, as closely as practicable, the performance that is required of the actuation devices in the event of an accident.

10 CFR 50.55a(h) requires that the WCGS protection systems, including the ESFAS Functions 6.f and 6.h, meet IEEE 279-1971. Sections 4.9 - 4.11 of IEEE 279-1971 discuss testing provisions for protection systems. Regulatory Guide 1.118, Revision 2, discusses acceptable methods for testing protection systems, including Section 6.3.4 of IEEE 338-1977 for response time testing.

Analysis

There have been no changes to the ESFAS instrumentation or AFW and ESW System design such that any of the above regulatory requirements and criteria would not be met. This amendment application imposes additional surveillance testing requirements on the ESFAS Functions 6.f and 6.h consistent with the above requirements.

Conclusion

The evaluation performed by WCNOC concludes that WCGS continues to comply with the above regulatory requirements.

6.0 ENVIRONMENTAL EVALUATION

WCNOC has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. WCNOC has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Therefore, pursuant to 10 CFR 51.22 (b), an environmental assessment of the proposed change is not required.

ATTACHMENT III

MARKUP OF TECHNICAL SPECIFICATION PAGE

Table 3.3.2-1 (page 4 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
6. Auxiliary Feedwater					
a. Manual Initiation	1,2,3	1 per pump	O	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c. Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)	1,2,3	2 trains	N	SR 3.3.2.3	NA
d. SG Water Level Low - Low	1,2,3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 22.3% of Narrow Range Instrument Span
e. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
f. Loss of Offsite Power	1,2,3	2 trains	P	SR 3.3.2.7 SR 3.3.2.10	NA
g. Trip of all Main Feedwater Pumps	1	2 per pump	J	SR 3.3.2.8	NA
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	M	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.12 SR 3.3.2.10	≥ 20.53 psia

(continued)

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

ATTACHMENT IV

MARKUP OF TECHNICAL SPECIFICATION BASES PAGES

Table B 3.3.2-2
(Page 2 of 3)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure - Low	
a. Safety Injection (ECCS)	≤ 39(3)/27(4)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation	≤ 2(5)
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	≤ 60(1)
6) Containment Cooling	≤ 60(1)
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	≤ 14(6)
9) Turbine Trip	N.A.
b. Steam Line Isolation	≤ 2(5)
5. Containment Pressure - High-3	
a. Containment Spray	≤ 32(1)/20(2)
b. Phase "B" Isolation	≤ 31.5
6. Containment Pressure - High-2	
Steam Line Isolation	≤ 2(5)
7. Steam Line Pressure - Negative Rate-High	
Steam Line Isolation	≤ 2(5)
8. Steam Generator Water Level - High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 7
9. Steam Generator Water Level - Low-Low	
a. Start Motor Driven Auxiliary Feedwater Pumps	≤ 60
b. Start Turbine Driven Auxiliary Feedwater Pumps	≤ 60
10. Loss-of-Offsite Power	
Start Turbine Driven Auxiliary Feedwater Pumps	~~N.A.~~ ≤ 60
11. Trip of All Main Feedwater Pumps	
Start Motor Driven Auxiliary Feedwater Pumps	N.A.

Table B 3.3.2-2
(Page 3 of 3)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Auxiliary Feedwater Pump Suction Pressure-Low Transfer to Essential Service Water	~~N.A.~~ ≤ 60 (1)
13. RWST Level-Low-Low Coincident with Safety Injection Automatic Switchover to Containment Sump	≤ 60

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting delay not included. Offsite power available.

(3) Diesel generator starting and sequence loading delay included. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(5) Does not include valve closure time.

(6) Includes time for diesel to reach full speed.

(7) Diesel generator starting and sequence loading delays included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. Response time assumes only opening of RWST valves.

ATTACHMENT V

RETYPED TECHNICAL SPECIFICATION PAGE

Table 3.3.2-1 (page 3 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE [a]
4. Steam Line Isolation (continued)					
d. Steam Line Pressure					
(1) Low	1,2[i], 3[b][i]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 571 psig[c]
(2) Negative Rate - High	3[g][i]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 125[h] psi
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	1,2[j]	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.14	NA
b. SG Water Level -High High (P-14)	1,2[j]	4 per SG	I	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 79.7% of Narrow Range Instrument Span
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(b) Above the P-11 (Pressurizer Pressure) Interlock and below P-11 unless the Function is blocked.
(c) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
(g) Below the P-11 (Pressurizer Pressure) Interlock; however, may be blocked below P-11 when safety injection on low steam line pressure is not blocked.
(h) Time constant utilized in the rate/lag controller is ≥ 50 seconds.
(i) Except when all MSIVs are closed.
(j) Except when all MFIVs are closed.

ATTACHMENT VI

PROPOSED USAR CHANGES

WOLF CREEK

TABLE 15.0-6 (Sheet 4)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.6	Decrease in reactor coolant inventory	-	-	-	-
	Inadvertent opening of a pressurizer, safety or relief valve	Low pressurizer pressure, overtemperature ΔT, manual	Low pressurizer pressure, manual	-	Safety injection system
	Steam generator tube rupture	Low pressurizer pressure, SIS, manual, overtemperaturer ΔT	Low pressurizer pressure manual	Essential service water system, component cooling water system, steam generator shell side fluid operating system, steam generator safety and/or atmospheric relief valves, steam line isolation valves	Emergency core cooling system, auxiliary feed-water system, emergency power system containment isolation
	Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Pressurizer low pressure, SIS, manual	Hi-1 containment pressure, low pressurizer pressure Hi-3 containment pressure, manual AFAS on loss of offsite power	Essential service water system, component cooling water system, steam generator safety and/or atmospheric relief valves	Emergency core cooling system, auxiliary feedwater system, containment heat removal system, emergency power system containment isolation

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Tony Harris, Manager Regulatory Affairs at Wolf Creek Generating Station, (316) 364-4038.

COMMITMENT	Due Date/Event
The amendment will be implemented within 60 days of issuance of the license amendment.	Within 60 days of issuance of the license amendment