WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President Engineering

February 7, 2006
ET 06-0002

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

Subject: Docket No. 50-482: Revision to Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation"

Gentlemen:

Pursuant to 10 CFR 50.90, Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Facility Operating License Number NPF-42 for the Wolf Creek Generating Station (WCGS). The proposed license amendment request (LAR) proposes to revise Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation."

The LAR proposes to add Surveillance Requirement (SR) 3.3.1.16 to Function 3.a. of TS Table 3.3.1-1. SR 3.3.1.16 requires that RTS RESPONSE TIMES be verified to be within limits every 18 months on a STAGGERED TEST BASIS. Function 3.a. is the power range neutron flux – high positive rate reactor trip function (hereafter referred to as the positive flux rate trip (PFRT) function). This change is being proposed based on a reanalysis of the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power event.

Attachments I through VI provide the Evaluation, Markup of Technical Specification Pages, Retyped Technical Specification Pages, Proposed TS Bases Changes (for information only), Proposed Updated Safety Analysis Report pages (for information only) and Summary of Regulatory Commitments, respectively, in support of this amendment request. Final TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specification Bases Control Program," at the time the amendment is implemented.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. The amendment application was reviewed by the WCNOC Plant Safety Review Committee.

A001

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

WCNOC requests approval of this proposed License Amendment by December 15, 2006. The changes proposed are not required to address an immediate safety concern. It is anticipated that the license amendment, as approved, will be effective upon issuance, to be implemented within 90 days from the date of issuance. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Very truly yours,

Terry J. Garrett

TJG/rlt

Attachments:
- I - Evaluation
- II - Markup of Technical Specification pages
- III - Retyped Technical Specification pages
- IV - Proposed TS Bases Changes (for information only)
- V - List of Commitments

cc: T. A. Conley (KDHE), w/a
J. N. Donohew (NRC), w/a
W. B. Jones (NRC), w/a
B. S. Mallett (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

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

By ____________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 7TH day of FEB, 2006.

Mary E. Gifford.
Notary Public

MARY E. GIFFORD
Notary Public - State of Kansas
My Appt. Expires 12/09/2007

Expiration Date 12/09/2007

EVALUATION

1.0 DESCRIPTION

This amendment application revises Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," by adding Surveillance Requirement (SR) 3.3.1.16 to Function 3.a., Power Range Neutron Flux Rate – High Positive Rate, of TS Table 3.3.1-1.

2.0 PROPOSED CHANGES

This amendment application adds SR 3.3.1.16 to Function 3.a. of TS Table 3.3.1-1. SR 3.3.1.16 requires that RTS RESPONSE TIMES be verified to be within limits every 18 months on a STAGGERED TEST BASIS. Function 3.a. is the Power Range Neutron Flux – High Positive Rate trip Function (hereafter referred to as the positive flux rate trip (PFRT) Function).

Proposed revisions to the TS Bases are also included in this application. The changes to the affected TS Bases pages will be incorporated in accordance with TS 5.5.14, "Technical Specifications (TS) Bases Control Program."

3.0 BACKGROUND

SR 3.3.1.16 requires a verification that RTS RESPONSE TIMES are within their limits every 18 months on a STAGGERED TEST BASIS, as defined in the TS. As discussed in the SR 3.3.1.16 Bases, Table B 3.3.1-2 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 analysis. The SR 3.3.1.16 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 3.3.1-2."

In August 2002, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 02-11, "Reactor Protection System Response Time Requirements," which notified licensees some protection functions (e.g., PFRT Function) may be credited for protection against anticipated transients or postulated accidents, but not explicitly credited for primary protection in the specific safety analysis cases presented in the Updated Safety Analysis Report (USAR). An evaluation of the NSAL determined that the PFRT is not explicitly credited for primary protection in an analysis of record for WCGS. Subsequent to the evaluation of the NSAL, for other purposes WCNOC has reanalyzed the Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power (RWAP) event using more conservative techniques. The reanalysis of this event results in the PFRT being credited in the safety analysis for primary protection.

4.0 TECHNICAL ANALYSIS

Transient Description

A continuous uncontrolled RCCA Withdrawal at Power event due to improper operator action or an instrument or control system malfunction will result in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation, there is a net increase in the reactor coolant temperature with a corresponding rise in Reactor Coolant System (RCS) pressure and pressurizer level. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in a Departure from Nucleate Boiling (DNB) and/or fuel centerline melting, RCS overpressurization, or pressurizer overfill.

Analysis Description

The uncontrolled RCCA Withdrawal at Power event (USAR Section 15.4.2) was evaluated to demonstrate that the reactivity and plant control systems are sufficient to prevent: (1) DNB and consequent fuel damage; (2) RCS overpressurization and consequent pressure boundary failure; and (3) pressurizer overfill and consequent progression of the accident sequence. The uncontrolled RCCA Withdrawal at Power event was reanalyzed using RETRAN-3D (Reference 3) in the RETRAN-02 mode. The use of RETRAN-3D in the RETRAN-02 mode has been found acceptable by the NRC for use as a licensing basis safety analysis code (Reference 4).

The assumptions and methods used for this analysis are consistent with earlier analyses performed for this event as described in USAR Section 15.4.2 to a great extent. Various cases were analyzed using a spectrum of rod withdrawal rates, initial reactor powers, and reactivity feedback conditions.

This event is classified as a Condition II fault of moderate frequency. As such, RCS pressure must be maintained within 110% of design pressure.

The primary protection function is condition dependent and provided by the power range neutron flux – high, pressurizer pressure – high, and overtemperature ΔT reactor trips. For a narrow range of RCS overpressure cases, it was found that the PFRT is additionally required to provide primary protection to prevent calculated peak RCS pressure from exceeding 110% of design pressure.

For this analysis, a rate setpoint of 9% RATED THERMAL POWER (RTP) (per second) with a time constant (lag) of 2.0 seconds and a 3.0 second trip delay were assumed for consistency with the Westinghouse Safety Analysis Standard (Reference 5) and for conservatism. These values bound the WCGS Trip Setpoint of ≤ 4% of RTP with a time constant of ≥ 2 seconds, listed in TS Bases Table B 3.3.1-1, as well as the Allowable Value of ≤ 6.3% RTP with a time constant of ≥ 2 seconds, as reflected in TS Table 3.3.1-1.

Peak RCS pressure attained in this analysis is 2733.97 psia for conditions of 10% RTP, beginning-of-life, and 41 pcm/sec reactivity insertion rate. This is lower than the acceptable pressure of 2748.2 psia, giving a margin of 14.23 psi.

Conclusion

This analysis shows that the acceptance criteria for the uncontrolled RCCA Withdrawal at Power event can be successfully met, with adequate protection for the primary and secondary system provided the Power Range Neutron Flux Rate – High Positive Rate trip is credited.

The Power Range Neutron Flux Rate – High Positive Rate trip Function is a sub-function of the Nuclear Instrumentation System, and is qualified as a primary protection function and accurately described in the TSs and USAR with the exception of the requirement for response time testing.

As a result, TS Table 3.3.1-1 should list SR 3.3.1.16 as a required Surveillance for the Power Range Neutron Flux Rate – High Positive Rate trip Function. This trip function has been administratively response time tested on an operating cycle basis, starting in Refueling Outage 13 with satisfactory results.

5.0 REGULATORY ANALYSIS

This section addresses the standards of 10 CFR 50.92 as well as the applicable regulatory requirements and acceptance criteria.

5.1 No Significant Hazards Consideration

Wolf Creek Nuclear Operating Corporation (WCNOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," Part 50.92(c), as discussed below:

(1) Does the proposed change 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 accident analysis since there are no hardware changes. The design of the Reactor Trip System (RTS) instrumentation, specifically the positive flux rate trip (PFRT) function, will be unaffected. The reactor protection system 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 additional surveillance requirements to assure safety related structures, systems, and components are verified to be consistent with the safety analysis and licensing basis. In this specific case, a response time verification requirement will be added to the PFRT Function.

The proposed changes will not modify any system interface. The proposed changes 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 changes will not alter any

assumptions or change any mitigation actions in the radiological consequence evaluations in the Updated Safety Analysis Report (USAR).

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

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

(2) Does the proposed change 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 requirements. The 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 these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes.

Therefore, the proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

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

Response: No

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation 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, DNBR limits, F_Q, $F\Delta H$, 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 safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis is changed. The imposition of

additional surveillance requirements increases the margin of safety by assuring that the affected safety analysis assumptions on equipment response time are verified on a periodic frequency. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Conclusion:

Based on the above evaluation, WCNOC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory guidance documents associated with 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.

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 RTS Function 3.a., 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.

There will be no changes to the RTS instrumentation design such that any of the regulatory requirements and guidance documents would come into question. This amendment application imposes additional surveillance requirements on RTS Function 3.a. consistent with the above requirements. The evaluations performed by WCNOC confirm that WCGS will continue to comply with all applicable regulatory requirements.

In conclusion, based on the considerations discussed above, (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) issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

WCNOC has determined that the proposed amendment would change requirements with the respect to the installation or use of a facility component located within the restricted area, as defined in A0 CFR 20, or would change an inspection or surveillance requirement. WCNOC has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

7.1 References

1. Westinghouse Nuclear Safety Advisory Letter (NSAL) 02-11, "Reactor Protection System Response Time Requirements," August 1, 2002.

2. WCGS Updated Safety Analysis Report, Section 15.4.2, Revision 18.

3. Electric Power Research Institute Topical Report, EPRI-7450(A), "RETRAN-3D: A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," July 2001.

4. NRC Safety Evaluation dated January 25, 2001, "Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems" (TAC NO. MA4311)."

5. Westinghouse Safety Analysis Standard No. 2, "Uncontrolled RCCA Bank Withdrawal at Power," Revision 6, March 2003.

7.2 Precedent

A similar change was approved for the Callaway Plant in Amendment No. 151 on September 3, 2002. However, AmerenUE allocates a response time for this trip function based on WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests." WCNOC has not implemented WCAP-14036-P-A, Revision 1, and would therefore perform the response time test.

ATTACHMENT II

MARKUP OF TECHNICAL SPECIFICATION PAGES

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	3(b), 4(b), 5(b)	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 112.3% RTP
b. Low	1(c),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 28.3% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 6.3% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤6.3% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(c), 2(d)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 35.3% RTP
5. Source Range Neutron Flux	2(e)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.6 E5 cps
	3(b), 4(b), 5(b)	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.6 E5 cps

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlock.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.

ATTACHMENT III

RETYPED TECHNICAL SPECIFICATION PAGES

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE [(a)]
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	$3^{(b)}, 4^{(b)}, 5^{(b)}$	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 112.3% RTP
b. Low	$1^{(c)}$,2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 28.3% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 6.3% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤6.3% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	$1^{(c)}, 2^{(d)}$	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 35.3% RTP
5. Source Range Neutron Flux	$2^{(e)}$	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.6 E5 cps
	$3^{(b)}, 4^{(b)}, 5^{(b)}$	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.6 E5 cps

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlock.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
6.	Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1 (Page 3.3-19)
7.	Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 2 (Page 3.3-20)
8.	Pressurizer Pressure					
	a. Low	1(g)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1930 psig
	b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2395 psig
9.	Pressurizer Water Level - High	1(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.9% of instrument span
10.	Reactor Coolant Flow - Low	1(g)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 88.9%(m)
11.	Not Used.					
12.	Undervoltage RCPs	1(g)	2/bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 10355 Vac

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.
(m) % of design flow - 90,324 gpm.

ATTACHMENT IV

PROPOSED TS BASES CHANGES
(for information only)

BASES

APPLICABLE SAFTY ANALYSES, LCO, and APPLICABILITY

b. Power Range Neutron Flux - Low (continued)

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Power Range Neutron Flux Rate

The Power Range Neutron Flux Rate trips use the same channels as discussed for Function 2 above.

a. Power Range Neutron Flux - High Positive Rate

The Power Range Neutron Flux - High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux - High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

This Function also provides protection for the rod withdrawal at power event.

The LCO requires all four of the Power Range Neutron Flux - High Positive Rate channels to be OPERABLE.

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux - High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions.

b. Power Range Neutron Flux - High Negative Rate

The Power Range Neutron Flux - High Negative Rate trip Function ensures that protection is provided for multiple rod drop accidents. At high power levels, a multiple rod drop accident could cause local flux peaking that would

TABLE B 3.3.1-2
(Page 1 of 2)

	FUNCTIONAL UNIT	RESPONSE TIME
1.	Manual Reactor Trip	N.A.
2.	Power Range Neutron Flux	
	a. High	≤ 0.5 second[1]
	b. Low	≤ 0.5 second[1]
3.	Power Range Neutron Flux	
	a. High Positive Rate	~~N.A.~~ ≤ 0.5 second[1]
	b. High Negative Rate	≤ 0.5 second[1]
4.	Intermediate Range Neutron Flux	N.A.
5.	Source Range Neutron Flux	N.A.
6.	Overtemperature ΔT	≤ 6.0 seconds[1]
7.	Overpower ΔT	≤ 6.0 seconds[1]
8.	Pressurizer Pressure	
	a. Low	≤ 2.0 seconds
	b. High	≤ 2.0 seconds
9.	Pressurizer Water Level - High	N.A.
10.	Reactor Coolant Flow - Low	
	a. Single Loop (Above P-8)	≤ 1.0 second
	b. Two Loops (Above P-7 and below P-8)	≤ 1.0 second
11.	Not Used	
12.	Undervoltage - Reactor Coolant Pumps	≤ 1.5 seconds
13.	Underfrequency - Reactor Coolant Pumps	≤ 0.6 second
14.	Steam Generator Water Level - Low-Low	≤ 2.0 seconds
15.	Not Used	

[1] Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

LIST OF COMMITMENTS

The following table identifies those actions committed to by 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. Kevin Moles at (620) 364-4126.

COMMITMENT	Due Date/Event
The proposed changes to the Technical Specification Bases and USAR will be implemented within 90 days of NRC approval.	Within 90 days of NRC approval