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U.S. Nuclear Regulatory Commission
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ULNRC-04684

Gentlemen:

DOCKET NUMBER 50-483
UNION ELECTRIC COMPANY
CALLAWAY PLANT
REVISION TO TECHNICAL SPECIFICATION 3.3.1,
"REACTOR TRIP SYSTEM (RTS) INSTRUMENTATION"

References: 1. ULNRC-04638 dated April 16, 2002
2. NRC letter from Jack Donohew to Garry L. Randolph dated May 21, 2002

AmerenUE herewith transmits an application for amendment to Facility Operating License No. NPF-30 for the Callaway Plant.

This amendment application would add Surveillance Requirement (SR) 3.3.1.16 to Function 3 of Technical Specification (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 is the power range neutron flux - high positive rate reactor trip function (hereafter referred to as the positive flux rate trip (PFRT) function).

Upon learning from Westinghouse that the PFRT function was credited in the analyses of certain transients, on April 3, 2002 Callaway Plant applied the provisions of Surveillance Requirement (SR) 3.0.3 since the response time testing surveillance (SR 3.3.1.16) had not been met for the PFRT function. In accordance with the provisions of Callaway OL Amendment 147, we applied the revised SR 3.0.3 and complied with the requirements stipulated in the SR 3.0.3 Bases. We determined that it is not risk-significant to delay performance of the response time test SR 3.3.1.16 up to the 18-month SR Frequency. We are currently managing this risk impact and have entered this issue into the Corrective Action Program.

Rather than performing an online response time test on the PFRT function, with its attendant risks and potential for perturbing the plant, AmerenUE submitted a request in Reference 1 above to apply WCAP-14036-P-A, Revision 1, and allocate a response time for this trip function. Reference 1 provided the

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information upon which a determination could be reached regarding this application of WCAP-14036-P-A, Revision 1. NRC approved the application of the topical report to the PFRT function in Reference 2. The attached amendment application is submitted in accordance with the guidance of NRC Administrative Letter 98-10, fulfilling the commitment made in Reference 1.

The Callaway Plant Onsite Review Committee and the Nuclear Safety Review Board have reviewed this amendment application. Attachments 1 through 5 provide the Evaluation, Markup of Technical Specifications, Retyped Technical Specifications, Proposed Technical Specification (TS) Bases Changes, and Proposed FSAR Changes, respectively, in support of this amendment request. Attachments 4 and 5 are provided for information only. Final TS Bases and FSAR changes will be processed under our programs for document updates per TS 5.5.14, Technical Specifications Bases Control Program, and 10CFR50.71(e), respectively, at the time this amendment is implemented.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10CFR50.92. Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment. The amendment will be implemented within 60 days after NRC approval. In accordance with 10CFR50.91, a copy of this amendment application is being provided to the designated Missouri State official. If you have any questions on this amendment application, please contact us.

Very truly yours,

John D. Blosser
Manager-Regulatory Affairs

Attachments:

1 - Evaluation
2 - Markup of Technical Specifications
3 - Retyped Technical Specifications
4 - Proposed Technical Specification Bases Changes (for information only)
5 - Proposed FSAR Changes (for information only)

STATE OF MISSOURI)
) S S
CITY OF ST. LOUIS)

John D. Blosser, of lawful age, being first duly sworn upon oath says that he is Manager, Regulatory Affairs for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company 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 JBlosser
John D. Blosser
Manager, Regulatory Affairs

SUBSCRIBED and sworn to before me this 17th day of June, 2002.

Melissa Orr

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EVALUATION

1.0 DESCRIPTION

This amendment application would revise Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," by adding Surveillance Requirement (SR) 3.3.1.16 to Function 3 of TS Table 3.3.1-1.

2.0 PROPOSED CHANGE

This amendment application would add SR 3.3.1.16 to Function 3 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 is the power range neutron flux - high positive rate trip function (hereafter referred to as the positive flux rate trip (PFRT) function).

Attachments 2 and 3 provide the TS markups and the retyped TS for the Table 3.3.1-1 revision. Attachment 4 provides an information-only copy of the associated TS Bases changes, which include the following:

- Additional discussion of the events which credit the PFRT function;

- An unrelated correction to the nominal trip setpoint for the PFRT function (although this change reflects the current setpoint for the installed bistables and is not directly related to the amendment request, this setpoint supports, and is supported by, the safety analysis limit for rod ejection discussed in Section 4.0 below); and

- A clarification to the SR 3.3.1.16 Bases that notes the PFRT rate circuit time constant will continue to be measured during the Nuclear Instrumentation System (NIS) CHANNEL CALIBRATION performed per SR 3.3.1.11. This change reflects current practice and is consistent with Section 4.6 of WCAP-14036-P-A, Revision 1 (i.e., the rate circuit was not evaluated in Section 4.6 because the time constant is measured during the CHANNEL CALIBRATION).

Changes to the FSAR will also be required in conjunction with this amendment application. Attachment 5 provides an information-only copy of the associated changes to FSAR Chapters 7 and 15 which correspond to the analyses of the Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power (hereafter referred to as RWAP) and the Spectrum of RCCA Ejection Accidents (hereafter referred to as rod ejection). The PFRT, which provides protection for certain cases of these analyses, is discussed in FSAR Section 7.2. The analyses are discussed in FSAR Sections 15.4.2 and 15.4.8, with trip settings, time delays, and credited trip functions discussed in FSAR Tables 15.0-4 and 15.0-6. The change to FSAR Table 16.3-1 is required to add the 0.5 second response time limit that will establish the acceptance criteria against which SR 3.3.1.16 will be based for Function 3 in TS Table 3.3.1-1. Response times for the NIS circuitry and SSPS input relays will be allocated per the discussion in References 1 and 2 of Section 7.0.

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, FSAR Table 16.3-1 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 3.3.1.16 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 Reference 8."

Reference 8 is FSAR Table 16.3-1, which lists the associated response time as N.A. for item 3 (Power Range, Neutron Flux, High Positive Rate). The response time for the PFRT function has always been listed as "N.A.", in both the initial licensing basis TS Table 3.3-2 and currently in FSAR Table 16.3-1. As a result, the current Callaway Response Time Testing Program does not verify the response time for the PFRT function.

Prior to initial licensing of Callaway in 1984, Westinghouse confirmed that the PFRT function was not credited in any FSAR Chapter 15 analysis of record. Since Westinghouse had performed all the licensing basis FSAR Chapter 15 accident analyses, it was appropriate to list "N.A." for the response time of the PFRT function in the original licensing basis version of TS Table 3.3-2, which

was subsequently relocated to the FSAR as Table 16.3-1 in OL Amendment 104 (November 22, 1995) in accordance with NRC Generic Letter 93-08, "Relocation of Technical Specification Tables of Instrument Response Time Limits."

However, Westinghouse has recently confirmed to us that they have a generic evaluation in which credit is taken for the PFRT to provide protection against a RWAP accident that is analyzed with assumptions intended to maximize the primary pressure response. Therefore, the PFRT function in TS Table 3.3.1-1 should have SR 3.3.1.16 listed as a surveillance requirement.

This concern was first identified to AmerenUE in recent discussions with the staff at South Texas Project. Although the PFRT function is credited in the generic Westinghouse evaluation for establishing that the RWAP event is not limiting for the reactor coolant system (RCS) overpressure event, Westinghouse had not previously identified this as a Callaway analysis requirement based on their conclusion that the assumptions related to the PFRT function were conservatively bounding so as to not require response time testing. That conclusion was based on the fact that the 3-second response time assumed for the PFRT in the generic evaluation is significantly greater than the maximum delay time of 0.5 second typically assumed for the NIS trip functions in the safety analyses. In addition, Westinghouse believed that the normal functional testing of the PFRT instrumentation would assure that the 3-second response time assumption would be satisfied since no failure was conceivable that would extend the response time past 3 seconds without rendering the channel inoperable, consistent with the basic premise in Reference 3 of Section 7.0 for allocating a response time to a trip function. However, given the fact that the PFRT response time assumption forms the basis for preventing the RWAP from causing RCS overpressure, it is AmerenUE's position that SR 3.3.1.16 should be applied to the PFRT function.

As discussed in Section 4.0 below, the response time limit that will be specified in FSAR Table 16.3-1 for item 3 will be 0.5 second to accommodate the rod ejection transient, which is a faster reactivity addition event than the RCCA bank withdrawal transient.

4.0 TECHNICAL ANALYSIS

Pressure-maximizing RWAP Case

Based on the concern identified via discussions with South Texas Project staff, AmerenUE transmitted a request for NRC approval to apply WCAP-14036-P-A, Revision 1, to the PFRT function which was subsequently approved by NRC (see References 1 and 2 in Section 7.0). Although not discussed in Callaway FSAR Section 15.4.2, Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power (RWAP), Westinghouse has informed us that credit is taken for the PFRT to maintain RCS pressure below the TS 2.1.2 Safety Limit when they analyze the RWAP transient with assumptions intended to maximize primary pressure. A response time of 3.0 seconds is credited in the generic Westinghouse evaluation. The discussion currently contained in FSAR Section 15.4.2 involves the analysis of the RWAP transient from a departure from nucleate boiling (DNB) perspective. The assumptions used for this FSAR Section 15.4.2 transient analysis, listed in FSAR Table 15.0-2, and the use of the Improved Thermal Design Procedure (ITDP) and the WRB-2 DNBR correlation to perform the analysis are associated with a transient being evaluated against DNB acceptance criteria. DNB is more of a concern as primary pressure decreases. When the RWAP analysis assumptions are changed in order to maximize primary pressure, the PFRT must be credited as the primary reactor trip signal.

The overtemperature delta-T (OTΔT) trip function and the power range neutron flux - high setting trip function are the primary reactor trips credited in FSAR Section 15.4.2 to protect against DNB conditions. However, the discussions mentioned above indicate that the PFRT function, which is designed to initiate a reactor trip due to a large insertion of positive reactivity, was credited in a generic Westinghouse evaluation of a RWAP event that evaluates its acceptance criteria against the peak RCS pressure. This evaluation is not currently discussed in the FSAR. Since the RWAP event is more DNB limiting with a functioning pressurizer pressure control system, the RWAP event with a malfunctioning pressurizer pressure control system was not originally evaluated and was not considered limiting for RCS overpressure concerns at the time the Section 15.4 analyses were last updated in the Callaway FSAR.

The generic Westinghouse evaluation discussed above in Section 3.0 notes that a low power RWAP could result in the RCS pressure exceeding the 110% design limit (2750 psia) if only the typically credited FSAR Chapter 15 trip functions (high

pressurizer pressure, overtemperature delta-T, and power range neutron flux - high) are credited and assuming that the pressurizer pressure control system malfunctions. The heat generated by the nuclear fuel in response to the positive reactivity addition resulting from the postulated rod withdrawal would cause an increase in RCS pressure. The potential for RCS overpressure increases as the time between the reactivity insertion and a reactor trip increases due to the time lag associated with transfer of the increased heat generated in the core through the fuel and into the RCS coolant.

The magnitude of the RCS pressure increase resulting from the RWAP is a function of the reactivity insertion rate, the initial power level, and the amount of reactivity feedback. For small positive reactivity insertion rates, the nuclear power and RCS temperature increase relatively slowly and in equilibrium such that this thermal lag effect on RCS overpressure is not a concern. For large reactivity insertion rates at the end of core life conditions, there is a large reactivity feedback effect such that the nuclear power and RCS temperature still increase in relative equilibrium and RCS overpressure is not a concern. However, for large reactivity insertion rates at the beginning of core life and corresponding minimum reactivity feedback effects, the nuclear power increases much faster than the rate at which the energy can be transferred into the RCS. For fast RWAP transients which occur at a relatively high initial power level, the high flux trip is reached before the RCS heatup rate has significantly increased and RCS overpressure is still not a concern. However, for a RWAP event at low initial power levels, adequate time is available to transfer the heat generated as a result of the positive reactivity addition to the reactor coolant before any other reactor trip is actuated.

As a result of the above, the factors that result in a potential overpressure condition in the RCS are only present at low reactor power levels. Westinghouse identified that the limiting RWAP overpressure case occurred at an initial power level of 10% rated thermal power (RTP). This is the lowest power level at which the power range neutron high flux - low trip function could be blocked by permissive P-10. Westinghouse determined that by the time a reactor trip on high pressurizer pressure, OTΔT, or power range neutron flux - high would occur, the RCS heatup and volumetric expansion rate could exceed the relief capacity of the pressurizer safety valves. Westinghouse determined in their generic evaluation that crediting the PFRT function with a setpoint of 9% RTP and a time constant on the NIS rate circuit of 2 seconds, as well as a 3-second delay time on the rest of the PFRT function circuitry, would mitigate this event before the RCS pressure limit was exceeded. Note that any discussion in this submittal of "time constant" is referring to the NIS rate circuit in each of the four NIS power range channels which is surveilled under the SR 3.3.1.11 CHANNEL

CALIBRATION. Westinghouse concluded that the assumptions related to the PFRT function were conservatively bounding and did not require response time testing, therefore they did not notify AmerenUE. The 3-second response time assumed for the PFRT in the generic evaluation was significantly greater than maximum delay time of 0.5 second typically assumed for the NIS trip functions in the safety analyses.

Although not currently documented in the FSAR, the Westinghouse generic RWAP evaluation for the RCS overpressure case represents part of the Callaway design basis for establishing that the Chapter 15 events do not result in exceeding the 110% RCS design pressure limit. The Westinghouse evaluation provides assurance that the PFRT function is required for RWAP cases which initiate from reactor power levels between 10% RTP and 60% RTP. Above 60% RTP Westinghouse concluded that the power range neutron flux - high setting trip occurs quickly enough that RCS overpressure is not a concern and that crediting the PFRT function for RCS overpressure protection is not necessary.

On the above basis, AmerenUE has concluded that the generic evaluation performed by Westinghouse is applicable to Callaway. As a result, TS Table 3.3.1-1 should list SR 3.3.1.16 against the PFRT function. Note that the PFRT function with a safety analysis limit of 9% reactor power and a time constant $\geq$ 2 seconds bounds both the Callaway Nominal Trip Setpoint of 4.25% RTP with a time constant of $\geq$ 2.0 seconds, listed in TS Bases Table B 3.3.1-1, as well as the Allowable Value of $\leq$ 6.3% RTP with a time constant of $\geq$ 2.0 seconds, as reflected in TS Table 3.3.1-1.

Rod Ejection Accident

Westinghouse has also informed us that the PFRT is credited to mitigate the consequences of certain partial power, low rod worth, rod ejection events. Those events are not discussed in Callaway FSAR Section 15.4.8, Spectrum of RCCA Ejection Accidents. The FSAR discussion is limited to the worst case analyses, initiated from either hot zero power or from 102% RTP. The typical rod ejection event for which the PFRT would be credited is initiated from just above 10% RTP, when the P-10 permissive would allow a manual block of the power range neutron flux - low setting trip function. The transient phenomenology is very much like that discussed above for the RWAP event; however, the rod ejection transient occurs much more quickly. For these partial power rod ejection events, a safety analysis limit of 7.15% RTP with a time constant $\geq$ 2 seconds and with a response time of 0.5 second for the rest of the PFRT function circuitry is required.

Probabilistic Risk Assessment (PRA) Evaluation

There is no impact on the Callaway PRA since that study is concerned mainly with time-averaged equipment functionality, not specific response time assumptions. The functional performance of the RTS is demonstrated by the Surveillance Requirements of TS 3.3.1, including CHANNEL CHECKS, CHANNEL OPERATIONAL TESTS, and CHANNEL CALIBRATIONS. This amendment would also add SR 3.3.1.16 on response time verification, subject to the use of allocated response times as discussed in References 1 and 2 of Section 7.0. Acceptable equipment response time is inherent in the PRA assumption that the modeled equipment will perform its intended function (i.e., the PRA does not model equipment unavailability solely due to a slow response time). Adding SR 3.3.1.16 to Function 3 of TS Table 3.3.1-1 reconciles equipment performance against its intended function.

Summary/Conclusion

Adding SR 3.3.1.16 to Function 3 TS Table 3.3.1-1 will assure that accident analysis assumptions regarding the response time of equipment credited in accident mitigation are verified on a periodic basis. The discussions presented above assess the potential impact of these changes on the safety analyses that credit these trip functions. These assessments demonstrate that the change will not adversely affect the design basis, safety analyses, or the safe operation of the plant.

5.0 REGULATORY SAFETY ANALYSIS

This section addresses the standards of 10CFR50.92 as well as the applicable regulatory requirements and acceptance criteria.

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

This amendment application would revise Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," by adding Surveillance Requirement (SR) 3.3.1.16 to Function 3 of TS Table 3.3.1-1. The proposed change does not involve a significant hazards consideration for Callaway Plant based on the three standards set forth in 10CFR50.92(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 design of the 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 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 FSAR.

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 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 amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

This amendment 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 change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Do the proposed changes 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 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, AmerenUE 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 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.

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.

10CFR50.55a(h) requires that the Callaway protection systems, including RTS Function 3, 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 above would come into question. This amendment application imposes additional surveillance

requirements on RTS Function 3 consistent with the above requirements. The evaluations performed by AmerenUE confirm that Callaway Plant 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

AmerenUE 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 10CFR20, or would change an inspection or surveillance requirement. AmerenUE has evaluated the proposed amendment and has determined that the amendment 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 amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22 (c)(9). Therefore, pursuant to 10CFR51.22 (b), an environmental assessment of the proposed amendment is not required.

7.0 REFERENCES

1. ULNRC-04638 dated April 16, 2002.

2. NRC letter from Jack Donohew to Garry L. Randolph dated May 21, 2002.

3. WCAP-14036-P-A, Revision 1, Elimination of Periodic Protection Channel Response Time Tests, October 1998.

ATTACHMENT TWO

MARKUP OF TECHNICAL SPECIFICATIONS

Table 3.3.1-1 (page 1 of 8)
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 - 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
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

(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.

ATTACHMENT THREE

RETYPED TECHNICAL SPECIFICATIONS

Table 3.3.1-1 (page 1 of 8)
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 - 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
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

(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.

ATTACHMENT FOUR

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES
(for information only)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

b. Power Range Neutron Flux - Low (continued)

The LCO requires all four of the Power Range Neutron Flux - Low channels to be OPERABLE (two-out-of-four trip logic). The Trip Setpoint is $\leq$ 25% RTP.

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

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 trip uses the same channels as discussed for Function 2 above.

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 (two-out-of-four trip logic). The Trip Setpoint is $\leq$ ~~4%~~ 4.25% RTP with a time constant $\geq$ 2 seconds.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

Power Range Neutron Flux - High Positive Rate (continued)

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. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup (automatic rod withdrawal is no longer available). This trip Function provides redundant protection to the Power Range Neutron Flux - Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function (one-out-of-two trip logic). The Trip Setpoint is ≤ 25% RTP.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2 above the P-6 setpoint, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

analysis. Response time verification acceptance criteria are included in Reference 8. 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 Reference 8. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor until loss of stationary gripper coil voltage (at which point the rods are free to fall).

The safety analyses include the sum of the following response time components:

(a) Process delay times (e.g., scoop transport delay and thermal lag associated with the narrow range RCS RTDs used in the OTΔT, OPΔT, and SG low-low Vessel ΔT (Power-1, Power-2) functions) which are not testable;

(b) Sensing circuitry delay time from the time the trip setpoint is reached at the sensor until a reactor trip is generated by the SSPS;

(c) Any intentional time delay set into the trip circuitry (e.g., undervoltage relay time delay, NLL cards (lag, lead/lag, rate/lag) and NPL cards (PROM logic cards for trip time delay) associated with the OTΔT, OPΔT, and SG low-low level Vessel ΔT (Power-1, Power-2) trip functions, and NLL cards (lead/lag) associated with the low pressurizer pressure reactor trip function) to add margin or prevent spurious trip signals;

(d) For the undervoltage RCP trip function, back EMF delay from the time of the loss of the bus voltage until the back EMF voltage generated by the bus loads has decayed;

(e) The time delay for the reactor trip breakers to open; and

(f) The time delay for the control rod drive stationary gripper coil voltage to decay and the RCCA grippers to mechanically release making the rods free to fall (i.e., gripper release time measured during the performance of SR 3.1.4.3).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time verification is performed with the time constants set at their nominal values. Time constants are verified during the performance of SR 3.3.1.10 and SR 3.3.1.11. The response time may be verified by a

(continued)

Table B 3.3.1-1
(Page 1 of 3)

	FUNCTION	NOMINAL TRIP SETPOINT [a]
1.	Manual Reactor Trip	N.A.
2.	Power Range Neutron Flux	
	a. High	≤ 109% RTP
	b. Low	≤ 25% RTP
3.	Power Range Neutron Flux Rate - High Positive Rate	≤ ~~4%~~ 4.25% RTP with time constant ≥ 2 sec.
4.	Intermediate Range Neutron Flux	≤ 25% RTP
5.	Source Range Neutron Flux	≤ 1.0E5 CPS
6.	Overtemperature ΔT	See Table 3.3.1-1 Note 1.
7.	Overpower ΔT	See Table 3.3.1-1 Note 2.
8.	Pressurizer Pressure	
	a. Low	≥ 1885 psig
	b. High	≤ 2385 psig
9.	Pressurizer Water Level - High	≤ 92% of instrument span
10.	Reactor Coolant Flow - Low	≥ 90% of loop minimum measured flow (MMF=95,660 gpm)

(continued)

[a] The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value. This also applies to the Overtemperature ΔT and Overpower ΔT K values per Reference 14.

ATTACHMENT FIVE

PROPOSED FSAR CHANGES
(for information only)

certain rod withdrawal at power events and

channel testing during plant shutdown or prior to startup. This bypass action is annunciated on the control board.

4. Power range high positive neutron flux rate trip

partial power, low rod worth,

This circuit trips the reactor when a sudden abnormal increase in nuclear power occurs in two out of the four power range channels. This trip provides ~~DNB~~ protection against certain rod ejection accidents (see ~~Chapter 15.0).~~ Sections 15.4.2 and 15.4.8).
Figure 7.2-1 (Sheet 3) shows the logic for all of the nuclear overpower and rate trips.

b. Core thermal overpower trips

The specific trip functions generated are as follows:

1. Overtemperature ΔT trip

This trip protects the core against DNB and trips the reactor on coincidence, as listed in Table 7.2-1, with one set of temperature measurements per loop. The setpoint for this trip is continuously calculated by analog circuitry for each loop by solving the following equation (See Technical Specification Table 3.3.1-1 Note 1 for further details):

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left[\frac{1}{1+\tau_3 s}\right] \le \Delta T_o\left\{K_1 - K_2\left[\frac{(1+\tau_4 s)}{(1+\tau_5 s)}T_{avg}\left(\frac{1}{1+\tau_6 s}\right) - T^\circ_{avg}\right] + K_3(P-2235) - f_1(\Delta I)\right\}$$

Where:

ΔT = measured RCS ΔT, °F

ΔT_o = indicated ΔT at rated thermal power, °F

T_{avg} = measured RCS average temperature, °F

TABLE 7.2-4 (Sheet 2)

	Trip[a]	Accident[b]	Technical Specification[c]
		Excessive Increase in Secondary Steam Flow (15.1.3)	
		Inadvertent Opening of a Steam Generator Relief or Safety Valve (15.1.4)	
		Steam System Piping Failure (15.1.5)	
		Spectrum of Rod Cluster Control Assembly Ejection Accidents (15.4.8)	
3.	Intermediate range high neutron flux trip	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition (15.4.1)	See Note d, 3.3.1, Table 3.3.1-1, Function 4
4.	Source range high neutron flux trip	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical or Low Power Startup Condition (15.4.1)	See Note d, 3.3.1, Table 3.3.1-1, Function 5
		Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (Mode 2, 15.4.6)	
5.	Power range high positive neutron flux rate trip	Spectrum of Rod Cluster Control Assembly Ejection Accidents (15.4.8) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)	3.3.1, Table 3.3.1-1, Function 3
6.	Deleted	-	-

TABLE 15.0-4 (Sheet 1)

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delays (Seconds)
Power range high neutron flux, high setting	118% RTP	0.5
Power range high neutron flux, low setting	35% RTP	0.5
Power range neutron flux, high positive rate	7.15% RTP with a time constant ≥ 2 sec.***	0.5
Overtemperature ΔT	Variable see Figure 15.0-1; $k_1 = 1.29$	8.0*
Overpower ΔT	Variable see Figure 15.0-1; $k_4 = 1.165^{(a)}$	8.0*
High pressurizer pressure	2,420[b] psig	1.0
Low pressurizer pressure, reactor trip	1,845 psig	2.0
High neutron flux, P-8**	84% RTP	0.5
Low reactor coolant flow (from loop flow detectors)	87% loop minimum measured flow	1.0
Undervoltage trip	68% nominal (9384 Vac)	1.5

* Total time delay (including scoop transport delay and thermal lag effect, combined RTD/thermowell time response, and trip circuit channel electronics delay; 2 seconds of pure time delays (i.e. electronics, SSPS, reactor trip breaker time delays) and 6 seconds of first order lags) from the time the temperature difference in the coolant loop exceeds the trip setpoint until the rods are free to fall.

** The P-8 permissive is interlocked with reactor coolant flow such that low flow in any loop will cause a reactor trip when above P-8.

*** See Sections 15.4.2.1 and 15.4.8 (listed value is for RCCA ejection).

(a) The use of this safety analysis K_4 value is supported by cycle-specific evaluations that confirm that applicable DNBR and fuel centerline temperature limits are not exceeded for power levels as high as 118.52% RTP.

(b) 2410 psig used in setpoint calculations; 2420 psig used in transient analyses. Margin of 10 psi exists in the setpoint calculation.

TABLE 15.0-6 (Sheet 3)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment/Alarms	ESF Equipment
	Reactor coolant pump shaft seizure (locked rotor)	Low flow (1), high pressurizer pressure, manual	---	Pressurizer safety valves, steam generator safety valves	---
15.4	Reactivity and power distribution anomalies				
	Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition	Source range high flux, intermediate range high flux, power range high flux (low setpoint) (1), manual	---	---	---
	Uncontrolled rod cluster control assembly bank withdrawal at power	Power range high flux (high setpoint) (1), overtemperature ΔT (1), high positive flux rate (1), high pressurizer pressure, manual, OPΔT, high pressurizer water level	---	Pressurizer safety valves, steam generator safety valves	---
	Rod cluster control assembly mis-operation (single RCCA withdrawal)	Overtemperature ΔT (1), manual	---	---	---
	Startup of an inactive reactor coolant loop at an incorrect temperature	Power range high flux with low flow above P-8 (1), manual	---	Low-Low rod insertion limit annunciators for emergency boration, SDM verification, restoration of control bank insertion	---

TABLE 15.0-6 (Sheet 4)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment/Alarms	ESF Equipment
	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	Source range high flux (1-Mode 2), power range high flux, overtemperature ΔT (1-Mode 1)	---	BDMS-activated CCP suction valve swapover from VCT to RWST in Modes 3-5, at least one RCP in operation in Modes 3-5, (see Section 15.4.6.1 for complete list of status and annunciator indications)	---
	Spectrum of rod cluster control assembly (RCCA) ejection accidents	Power range high flux (high and low setpoints) (1), high positive flux rate (1), manual	SIS on low pressurizer pressure	---	Safety injection system
15.5	Increase in reactor coolant inventory				
	Inadvertent operation of the ECCS during power operation	Low pressurizer pressure (1-DNB Case), manual, safety injection trip (1,5-Pressurizer Filling Case)	---	---	Safety injection system (transient initiator)
	CVCS malfunction that increases reactor coolant inventory (3)	High pressure water level (4)	---	High pressurizer pressure and water level annunciators, pressurizer PORVs	---
15.6	Decrease in reactor coolant inventory				
	Inadvertent opening of a pressurizer safety or relief valve	Low pressurizer pressure, overtemperature ΔT (1), manual	---	---	---
	Steam generator tube failure	Low pressurizer pressure (1), SIS, manual, OTΔT (4)	SIS on low pressurizer pressure (1), AFAS and FWIS on SIS (1)	Essential service water system, component cooling water system, steam generator safety and/or relief valves, steam line isolation valves, feedwater isolation valves	Emergency core cooling system, auxiliary feedwater system, emergency power system

The reactivity insertion rate used is equal to that calculated for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region. Figure 15.4-1 shows the average nuclear power transient.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 15.4-2. The beneficial effect of the inherent thermal lag in the fuel is shown by a peak heat flux much less than the full power nominal value. There is a large margin to DNB during the transient, since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 15.4-3 shows the response of the hot spot average fuel and cladding temperature. The hot spot average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR at all times remains above the limit values.

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR greater than the safety analysis limit values. Thus, no fuel or clad damage is predicted as a result of DNB.

15.4.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad, the reactor protection system is designed to terminate any such transient before the DNBR falls below the safety analysis limit values. INSERT 15.4-6

This event is classified as an ANS Condition II incident (an incident of moderate frequency), as defined in Section 15.0.1.

INSERT 15.4-6

If the analysis assumptions are chosen to maximize primary pressure rather than DNB, the power range neutron flux, high positive rate trip is credited to preclude an overpressurization of the RCS.

The automatic features of the reactor protection system which prevent core damage following the postulated accident include the following:

a. Power range neutron flux instrumentation actuates a reactor trip if two out of four channels exceed an overpower setpoint.

b. A reactor trip is actuated if any two out of four ΔT channels exceed the overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressure to protect against DNB.

c. A reactor trip is actuated if any two out of four ΔT channels exceed the overpower ΔT setpoint. This setpoint is automatically varied with coolant temperature to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

d. A high pressurizer pressure reactor trip actuated from any two out of four pressure channels which are set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

e. A high pressurizer water level reactor trip actuated from any two out of three level channels when the reactor power is above 10 percent (Permissive P-7).

f. INSERT 15.4-7

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks that discontinue manual rod withdrawal:

a. High neutron flux (one out of four power range or one out of two intermediate range)

b. Overpower ΔT (two out of four)

c. Overtemperature ΔT (two out of four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of RCS conditions is described in Section 7.2. Figure 15.0-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety

INSERT 15.4-7

f. The power range neutron flux, high positive rate trip is credited if the rod withdrawal event assumptions are chosen to maximize the primary pressure. This trip function was credited in a generic evaluation of a rod withdrawal case from 10% RTP with minimum reactivity feedback. The results of the generic evaluation are not included here, nor are the evaluation's initial condition assumptions reflected in Table 15.0-2; however, this evaluation credited a positive flux rate trip setpoint of 9% RTP, with a time constant of 2 seconds and a delay time of 3 seconds.

during the transient is accentuated by the lead-lag compensation, causing the overtemperature ΔT trip setpoint to be reached later, with resulting lower minimum DNBR.

For transients initiated from higher power levels (for example, see Figure 15.4-10) the effect described in item d above, which results in the sharp peak in minimum DNBR at approximately 7 pcm/sec, does not occur since the steam generator safety valves are never actuated prior to trip.

Figures 15.4-10, 15.4-11, and 15.4-12 illustrate minimum DNBR's calculated for minimum and maximum reactivity feedback.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will still remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118 percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will remain below the fuel melting temperature.

Since DNB does not occur at any time during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident is shown on Table 15.4-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.2.3 Conclusions

The high neutron flux (high setting, low setting, and high positive rate) and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the safety analysis limit values, and the RCS is not overpressurized.

d. The latch mechanism housing and rod travel housing are each a single length of forged Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures which will be encountered.

A significant margin of strength in the elastic range, together with the large energy absorption capability in the plastic range, gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds which are subject to periodic inspections.

Nuclear Design

Even if a rupture of an RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated for by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of an RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger-than-normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if the RCCAs approach their insertion limit or if one RCCA deviates from its bank. Operating instructions require the operator to clear the condition or to calculate shutdown margin at the low level alarm. Emergency boration is required at the low-low alarm if there is inadequate shutdown margin, as discussed in Section 7.7.1.3.3.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 8. The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip (for certain partial power, low rod worth events). These protection functions are described in detail in Section 7.2.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of an RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings.

TABLE 16.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES(2)

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	~~N.A.~~ ≤ 0.5 second*
4. Deleted	
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	≤ 6.0 seconds*
8. Overpower ΔT	≤ 6.0 seconds*
9. Pressurizer Pressure-Low	≤ 2.0 seconds
10. Pressurizer Pressure-High	≤ 1.0 seconds
11. Pressurizer Water Level-High	N.A.
12. 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
13. Steam Generator Water Level Low-Low	
a. Steam Generator Water Level Low-Low (Adverse Containment Environment)	≤ 2.0 seconds(1)
b. Steam Generator Water Level Low-Low (Normal Containment Environment)	≤ 2.0 seconds(1)
c. Vessel ΔT (Power-1, Power-2)	≤ 6.0 seconds(1)
d. Containment Pressure - Environmental Allowance Modifier	≤ 2.0 seconds(1)
14. Undervoltage-Reactor Coolant Pumps	≤ 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds