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August 21, 2003

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

Subject: Duke Energy Corporation
Catawba Nuclear Station, Unit 1
Docket Numbers 50-413
Proposed Technical Specification Amendment to Relax Bypass Test Times for the Reactor Protection (RPS) and Engineered Safety Feature Actuation System (ESFAS)

References:

1) Letter from Thomas H. Essig (NRC) to Louis F. Liberatori Jr. (Westinghouse), "Review of Westinghouse Owners Group Topical Reports WCAP 14333P and WCAP-14334NP, dated May 1995," Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times" dated July 15, 1998.
2) Letter from William D. Beckner (NRC) to Anthony Pietrangelo (Nuclear Energy Institute) dated April 2, 2003.

Pursuant to 10 CFR 50.90, Duke Energy Corporation is requesting an amendment to the Catawba Nuclear Station Unit 1 Facility Operating License and Technical Specifications (TS). This proposed amendment will allow relaxation of the allowed bypass test times for Limiting Condition for Operation (LCOs) 3.3.1, Reactor Trip System (RTS) Instrumentation and 3.3.2, Engineered Safety Feature System Actuation System (ESFAS) Instrumentation. The proposed change will revise the time an inoperable channel may be in bypass from 4 hours to 12 hours. This change will be effective until the upcoming Unit 1 refueling outage, 1EOC14.

A001

On August 7, 2003, on Unit 1, two of the three hot leg temperature RTDs failed during maintenance activities on the "A" reactor coolant system (RCS) loop. The associated functions were placed in the tripped condition as required by TS and within the required TS completion time. The TS would allow continued operation in this condition indefinitely. These RTDs are not able to be replaced at power due to their location and environmental conditions present during power operation. Currently, the next scheduled opportunity to be in a mode to allow replacement is during the upcoming Unit 1 refueling outage this fall. Catawba has reviewed scheduled TS surveillances that could be affected by this failure. In order to complete some of these surveillances, the inoperable channel would have to be bypassed.

The current TS provide Notes that allow bypassing an inoperable channel for up to 4 hours for surveillance testing of other channels. A review of the scheduled surveillances indicates that some of those surveillances cannot be completed within the 4 hour time frame. The specific surveillances are the power range nuclear instruments channel operational tests (COTs) and the delta-T protection channel III COT. Catawba is requesting a change to the Notes for remedial actions to change the time allowed to bypass the inoperable channel from 4 hours to 12 hours. The first of these TS surveillances due date with the 25% grace period applied is October 25, 2003. If Catawba does not obtain relief prior to that date, Unit 1 will be required to shutdown to comply with TS. Therefore, Duke requests NRC review and approval of this amendment in an expeditious manner.

These relaxations are based on those that have been generically approved in WCAP-14333-P-A, Revision 1, Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times." Reference 1 documents the NRC approval of WCAP-14333 for referencing in plant-specific submittals.

These changes are similar to those proposed by the Technical Specification Task Force (TSTF) in the proposed changes to NUREG 1431.

The proposed changes were documented in Standard Technical Specification Change Traveler TSTF-418 revision 2, "RPS and ESFAS Test Times and Completion Times (WCAP-14333)," that the NRC approved by means of Reference 2. Duke has reviewed TSTF-418, Revision 2 and concluded that it is applicable to Catawba Unit 1.

The attached justification supports these proposed changes.

The contents of this amendment request package are as follows:

Attachment 1 provides marked copies of the affected TS pages for Catawba Unit 1 showing the proposed changes.

Attachment 2 contains reprinted page of the affected TS page for Catawba. Attachment 3 provides a description of the proposed changes and technical justification. Pursuant to 10 CFR 50.92, Attachment 4 documents the determination that the amendment contains No Significant Hazards Considerations.

Pursuant to 10 CFR 51.22(c)(9), Attachment 5 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement. Attachment 6 provides the analysis of WCAP-14333 applicability to Catawba Unit 1. Attachment 7 contains the Tier 2 and 3 analysis required by the implementation guidelines of WCAP-14333.

Implementation of this amendment to the Catawba Facility Operating License and TS will not impact the Catawba Updated Final Safety Analysis Report (UFSAR). Duke has determined that the standard 30 day implementation period is acceptable for this proposed amendment. There are no regulatory commitments contained in this letter or its attachments.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, this proposed amendment has been previously reviewed and approved by the Catawba Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this proposed amendment is being sent to the appropriate State of South Carolina official.

Inquiries on this matter should be directed to R. D. Hart at (803) 831-3622.

Very truly yours,

Dhiaa M. Jamil

RDH/s

Attachments

Dhiaa M. Jamil affirms that he the person who subscribed his name to the foregoing statement, and that all statements and matters set forth herein are true and correct to the best of his knowledge.

Dhiaa M. Jamil, Site Vice President

Subscribed and sworn to me: 8-21-2003
Date

Notary Public

My commission expires: 7-10-2012
Date

SEAL

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ATTACHMENT 1

MARKED-UP TECHNICAL SPECIFICATIONS PAGES FOR CATAWBA

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------------NOTE------------------ The inoperable channel may be bypassed for up to 4 hours* for surveillance testing of other channels. --	
	E.1 Place channel in trip.	6 hours
	<u>OR</u>	
	E.2 Be in MODE 3.	12 hours
F. THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	2 hours
	<u>OR</u>	
	F.2 Increase THERMAL POWER to > P-10.	2 hours
G. THERMAL POWER > P-6 and < P-10, two Intermediate Range Neutron Flux channels inoperable.	G.1 -------------NOTE------------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. -------------------------------------	
	Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours

(continued)

* Insert 1

INSERT 1

For Unit 1, during the time that the channel I Over Power delta-T and Over Temperature delta-T trip functions are inoperable due to the failure of the hot leg resistance temperature detectors on the 'A' reactor coolant system hot leg, the Over Power delta-T and Over Temperature delta-T trip functions may be bypassed beyond the 4 hours up to 12 hours. This footnote is not applicable after the Unit 1 refueling outage, 1EOC14.

No changes to this page
For information only

Table 3.3.1-1 (page 1 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.14	NA	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	≤ 110.9% RTP	109% RTP
b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 27.1% RTP	25% RTP
3. Power Range Neutron Flux						
High Positive Rate	1,2	4	D	SR 3.3.1.7 SR 3.3.1.11	≤ 6.3% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 31% RTP	25% RTP
	2(d)	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 31% RTP	25% RTP
5. Source Range Neutron Flux	2(d)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.4 E5 cps	1.0 E5 cps
	3(a), 4(a), 5(a)	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.4 E5 cps	1.0 E5 cps
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 SR 3.3.1.17	Refer to Note 1 (Page 3.3.1-18)	Refer to Note 1 (Page 3.3.1-18)

(continued)

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

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Table 3.3.1-1 (page 2 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
7. Overpower 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 SR 3.3.1.17	Refer to Note 2 (Page 3.3.1-19)	Refer to Note 2 (Page 3.3.1-19)
8. Pressurizer Pressure						
a. Low	1(e)	4	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1938(f) psig	1945(f) 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	≤ 2399 psig	2385 psig
9. Pressurizer Water Level - High	1(e)	3	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.8%	92%
10. Reactor Coolant Flow - Low						
a. Single Loop	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	≥ 89.7%	91%
b. Two Loops	1(h)	3 per loop	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 89.7%	91%

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Time constants utilized in the lead-lag controller for Pressurizer Pressure - Low are 2 seconds for lead and 1 second for lag.

(g) Above the P-8 (Power Range Neutron Flux) interlock.

(h) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

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Table 3.3.1-1 (page 3 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
11. Undervoltage RCPs	1(e)	1 per bus	L	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 5016 V	5082 V
12. Underfrequency RCPs	1(e)	1 per bus	L	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 55.9 Hz	56.4 Hz
13. Steam Generator (SG) Water Level - Low Low	1,2	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 9% (Unit 1) ≥ 35.1% (Unit 2) of narrow range span	10.7% (Unit 1) 36.8% (Unit 2) of narrow range span
14. Turbine Trip						
a. Stop Valve EH Pressure Low	1(j)	4	N	SR 3.3.1.10 SR 3.3.1.15	≥ 500 psig	550 psig
b. Turbine Stop Valve Closure	1(j)	4	O	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open	NA
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	P	SR 3.3.1.5 SR 3.3.1.14	NA	NA

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(i) Not used.

(j) Above the P-9 (Power Range Neutron Flux) interlock.

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Table 3.3.1-1 (page 4 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
16. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2[(d)]	2	R	SR 3.3.1.11 SR 3.3.1.13	≥ 6E-11 amp	1E-10 amp
b. Low Power Reactor Trips Block, P-7	1	1 per train	S	SR 3.3.1.5	NA	NA
c. Power Range Neutron Flux, P-8	1	4	S	SR 3.3.1.11 SR 3.3.1.13	≤ 50.2% RTP	48% RTP
d. Power Range Neutron Flux, P-9	1	4	S	SR 3.3.1.11 SR 3.3.1.13	≤ 70% RTP	69% RTP
e. Power Range Neutron Flux, P-10	1,2	4	R	SR 3.3.1.11 SR 3.3.1.13	≥ 7.8% RTP and ≤ 12.2% RTP	10% RTP
f. Turbine Impulse Pressure, P-13	1	2	S	SR 3.3.1.12 SR 3.3.1.13	≤ 12.2% RTP turbine impulse pressure equivalent	10% RTP turbine impulse pressure equivalent
17. Reactor Trip Breakers[(k)]	1,2	2 trains	Q,U	SR 3.3.1.4	NA	NA
	3[(a)], 4[(a)], 5[(a)]	2 trains	C	SR 3.3.1.4	NA	NA
18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	T	SR 3.3.1.4	NA	NA
	3[(a)], 4[(a)], 5[(a)]	1 each per RTB	C	SR 3.3.1.4	NA	NA
19. Automatic Trip Logic	1,2	2 trains	P,U	SR 3.3.1.5	NA	NA
	3[(a)], 4[(a)], 5[(a)]	2 trains	C	SR 3.3.1.5	NA	NA

(continued)

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

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Table 3.3.1-1 (page 5 of 7)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following NOMINAL TRIP SETPOINT by more than 4.3% (Unit 1) and 4.5% (Unit 2) of RTP.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \left[T \frac{1}{(1+\tau_6 s)} - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is the measured RCS ΔT by loop narrow range RTDs, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP (allowed by Safety Analysis), ≤ 585.1°F (Unit 1) ≤ 590.8°F (Unit 2).
P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, = 2235 psig

K_1 = Overtemperature ΔT reactor NOMINAL TRIP SETPOINT, as presented in the COLR,

K_2 = Overtemperature ΔT reactor trip heatup setpoint penalty coefficient, as presented in the COLR,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient, as presented in the COLR,

τ_1, τ_2 = Time constants utilized in the lead-lag compensator for ΔT, as presented in the COLR,

τ_3 = Time constant utilized in the lag compensator for ΔT, as presented in the COLR,

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, as presented in the COLR,

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the COLR, and

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

(i) for $q_t - q_b$ between the "positive" and "negative" $f_1(\Delta I)$ breakpoints as presented in the COLR; $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the $f_1(\Delta I)$ "negative" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "negative" slope presented in the COLR; and

(continued)

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Table 3.3.1-1 (page 6 of 7)
Reactor Trip System Instrumentation

(iii) for each percent ΔI that the magnitude of $q_t - q_b$ is more positive than the $f_1(\Delta I)$ "positive" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "positive" slope presented in the COLR.

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following NOMINAL TRIP SETPOINT by more than 2.6% (Unit 1) and 3.1% (Unit 2) of RTP.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_0 \left\{K_4 - K_5\frac{\tau_7 s}{1+\tau_7 S}\left(\frac{1}{1+\tau_6 s}\right)T - K_6\left[T\frac{1}{1+\tau_6 s} - T^{*}\right] - f_2(\Delta I)\right\}$$

Where: ΔT is the measured RCS ΔT by loop narrow range RTDs, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T^{*} is the nominal T_{avg} at RTP (calibration temperature for ΔT instrumentation), $\le$ 585.1°F (Unit 1) $\le$ 590.8°F (Unit 2).

K_4 = Overpower ΔT reactor NOMINAL TRIP SETPOINT as presented in the COLR,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

K_6 = Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the COLR for $T > T^{*}$ and $K_6 = 0$ for $T \le T^{*}$,

τ_1, τ_2 = Time constants utilized in the lead-lag compensator for ΔT, as presented in the COLR,

τ_3 = Time constant utilized in the lag compensator for ΔT, as presented in the COLR,

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the COLR,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, as presented in the COLR, and

$f_2(\Delta I)$ = a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" $f_2(\Delta I)$ breakpoints as presented in the COLR; $f_2(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(continued)

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Table 3.3.1-1 (page 7 of 7)
Reactor Trip System Instrumentation

(ii) for each percent ΔI that the magnitude of $q_t - q_b$ is more negative than the $f_2(\Delta I)$ "negative" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "negative" slope presented in the COLR; and

(iii) for each percent ΔI that the magnitude of $q_t - q_b$ is more positive than the $f_2(\Delta I)$ "positive" breakpoint presented in the COLR, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "positive" slope presented in the COLR.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
J. One channel inoperable.	J.1	---NOTE--- The inoperable channel may be bypassed for up to 4 hours* for surveillance testing of other channels.	
		Place channel in trip.	6 hours
	OR		
	J.2	Be in MODE 3.	12 hours
K. One Main Feedwater Pumps trip channel inoperable.	K.1	Place channel in trip.	1 hour
	OR		
	K.2	Be in MODE 3.	7 hours
L. One channel inoperable.	L.1	---NOTE--- One channel may be bypassed for up to 2 hours for surveillance testing provided the other channel is OPERABLE.	
		Be in MODE 3.	6 hours

(continued)

* Insert 2

INSERT 2

For Unit 1, during the time that the reactor coolant system channel I T_{avg} input into the turbine trip and feedwater isolation on T_{avg}-Low coincident with Reactor Trip (P-4) channel is inoperable due to the failure of the hot leg resistance temperature detectors on the 'A' reactor coolant system hot leg, the reactor coolant system channel I T_{avg} input may be bypassed beyond the 4 hours up to 12 hours. This footnote is not applicable after the Unit 1 refueling outage, 1EOC14.

No changes to this page

Information only

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Safety Injection(b)						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 1.4 psig	1.2 psig
d. Pressurizer Pressure - Low	1,2,3(a)	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 1839 psig	1845 psig
2. Containment Spray						
a. Manual Initiation	1,2,3,4	1 per train, 2 trains	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 3.2 psig	3.0 psig
3. Containment Isolation(b)						
a. Phase A Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) The requirements of this Function are not applicable to Containment Purge Ventilation System and Hydrogen Purge System components, since the system containment isolation valves are sealed closed in MODES 1, 2, 3, and 4.

No changes to this page
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Table 3.3.2-1 (page 2 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
3. Containment Isolation (continued)						
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	1 per train, 2 trains	B	SR 3.3.2.8	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(3) Containment Pressure - High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 3.2 psig	3.0 psig
4. Steam Line Isolation						
a. Manual Initiation						
(1) System	1,2(b),3(b)	2 trains	F	SR 3.3.2.8	NA	NA
(2) Individual	1,2(b),3(b)	1 per line	G	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2(b),3(b)	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High High	1,2(b),3(b)	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 3.2 psig	3.0 psig
d. Steam Line Pressure						
(1) Low	1,2(b),3(a)(b)	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	≥ 744 psig	775 psig

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Except when all MSIVs are closed and de-activated.

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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	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
(2) Negative Rate - High	3(b)(c)	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	≤ 122.8(d) psi	100(d) psi
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2(e)	2 trains	I	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. SG Water Level - High High (P-14)	1,2(e)	4 per SG	J	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	≤ 85.6% (Unit 1) ≤ 78.9% (Unit 2)	83.9% (Unit 1) 77.1% (Unit 2)
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. T_{avg}-Low	1,2(e)	4	J	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≥ 561°F	564°F
coincident with Reactor Trip, P-4	Refer to Function 8.a (Reactor Trip, P-4) for all initiation functions and requirements.					
e. Doghouse Water Level - High High	1,2(e)	2 per doghouse	L	SR 3.3.2.8	≤ 12 inches above 577 ft floor level	11 inches above 577 ft floor level
f. Trip of all main feedwater pumps	1,2(a)	3 per MFW pump	K	SR 3.3.2.8	NA	NA

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Except when all MSIVs are closed and de-activated.

(c) Trip function automatically blocked above P-11 (Pressurizer Pressure) interlock and may be blocked below P-11 when Steam Line Isolation Steam Line Pressure - Low is not blocked.

(d) Time constant utilized in the rate/lag controller is ≥ 50 seconds.

(e) Except when all MFIVs, MFCVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

No changes to this page
Information Only

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	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. 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	≥ 9% (Unit 1) ≥ 35.1% (Unit 2)	10.7% (Unit 1) 36.8% (Unit 2)
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. Loss of Offsite Power	1,2,3	3 per bus	D	SR 3.3.2.3 SR 3.3.2.9 SR 3.3.2.10	≥ 3242 V	3500 V
e. Trip of all Main Feedwater Pumps	1,2(a)	3 per pump	K	SR 3.3.2.8 SR 3.3.2.10	NA	NA
f. Auxiliary Feedwater Pump Train A and Train B Suction Transfer on Suction Pressure - Low	1,2,3	3 per train	M*	SR 3.3.2.8 SR 3.3.2.10	A) ≥ 9.5 psig B) ≥ 5.2 psig (Unit 1) ≥ 5.0 psig (Unit 2)	A) 10.5 psig B) 6.2 psig (Unit 1) 6.0 psig (Unit 2)
7. Automatic Switchover to Containment Sump						
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. Refueling Water Storage Tank (RWST) Level – Low	1,2,3,4	4	N	SR 3.3.2.1 SR 3.3.2.7 SR 3.3.2.9 SR 3.3.2.10	≥ 162.4 inches	177.15 inches
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.
*If more than one channel of Auxiliary Feedwater Suction Pressure – Low for one train becomes inoperable, immediately enter the applicable Condition(s) and Required Action(s) for the associated AFW train made inoperable by the inoperable channels. This is a one time only change for Unit 1 in support of the activities associated with the replacement of pressure switch 1CAPS5232.

No Changes to this page
Information Only

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
8. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	1 per train, 2 trains	F	SR 3.3.2.8	NA	NA
b. Pressurizer Pressure, P-11	1,2,3	3	O	SR 3.3.2.5 SR 3.3.2.9	≥ 1944 and ≤ 1966 psig	1955 psig
c. T_{avg} - Low Low, P-12	1,2,3	1 per loop	O	SR 3.3.2.5 SR 3.3.2.9	≥ 550°F	553°F
9. Containment Pressure Control System						
a. Start Permissive	1,2,3,4	4 per train	P	SR 3.3.2.1 SR 3.3.2.7 SR 3.3.2.9	≤ 0.45 psid	0.4 psid
b. Termination	1,2,3,4	4 per train	P	SR 3.3.2.1 SR 3.3.2.7 SR 3.3.2.9	≥ 0.25 psid	0.3 psid
10. Nuclear Service Water Suction Transfer - Low Pit Level	1,2,3,4	3 per pit	Q,R	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.11	≥ El. 555.4 ft	El. 557.5 ft

ATTACHMENT 2

REPRINTED TS PAGES FOR CATAWBA

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	-----------------NOTE----------------- The inoperable channel may be bypassed for up to 4 hours* for surveillance testing of other channels. --	
	E.1 Place channel in trip.	6 hours
	OR	
	E.2 Be in MODE 3.	12 hours
F. THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	2 hours
	OR	
	F.2 Increase THERMAL POWER to > P-10.	2 hours
G. THERMAL POWER > P-6 and < P-10, two Intermediate Range Neutron Flux channels inoperable.	G.1 -------------NOTE------------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. ---------------------------------	
	Suspend operations involving positive reactivity additions.	Immediately
	AND	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours

(continued)

*For Unit 1, during the time that the channel I Over Power delta-T and Over Temperature delta-T trip functions are inoperable due to the failure of the hot leg resistance temperature detectors on the 'A' reactor coolant system hot leg, the Over Power delta-T and Over Temperature delta-T trip functions may be bypassed beyond the 4 hours up to 12 hours. This footnote is not applicable after the Unit 1 refueling outage, 1EOC14.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
J. One channel inoperable.	J.1	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours* for surveillance testing of other channels. ------------------------------ Place channel in trip.	6 hours
	<u>OR</u>		
	J.2	Be in MODE 3.	12 hours
K. One Main Feedwater Pumps trip channel inoperable.	K.1	Place channel in trip.	1 hour
	<u>OR</u>		
	K.2	Be in MODE 3.	7 hours
L. One channel inoperable.	L.1	------------NOTE------------ One channel may be bypassed for up to 2 hours for surveillance testing provided the other channel is OPERABLE. ------------------------------ Be in MODE 3.	6 hours

(continued)

*For Unit 1, during the time that the reactor coolant system channel I T_{avg} input into the turbine trip and feedwater isolation on T_{avg}-Low coincident with Reactor Trip (P-4) channel is inoperable due to the failure of the hot leg resistance temperature detectors on the 'A' reactor coolant system hot leg, the reactor coolant system channel I T_{avg} input may be bypassed beyond the 4 hours up to 12 hours. This footnote is not applicable after the Unit 1 refueling outage, 1EOC14.

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

Background:

Introduction

The Catawba Nuclear Station (CNS) is a Westinghouse 4-loop pressurized water reactor. The Reactor Protection System (RPS) automatically limits reactor operation to within a safe region by shutting down the reactor whenever the limits of the region are approached. The safe operating region is defined by several considerations such as mechanical/ hydraulic limitations on equipment and heat transfer phenomena. Therefore the RPS keeps surveillance on process variables which are directly related to equipment mechanical limitations, such as pressure, pressurizer water level (to prevent water discharge through safety valves, and uncovering heaters) and also on variables which directly affect the heat transfer capability of the reactor (e.g. flow and reactor coolant temperatures). Still other parameters utilized in the RPS are calculated from various process variables. Whenever a direct process or calculated variable exceeds a setpoint the reactor will be shutdown in order to protect against either gross damage to fuel cladding or loss of system integrity which could lead to release of radioactive fission products into the containment.

The RPS consists of sensors and analog circuitry, consisting of two to four redundant channels, which monitors various plant parameters; and digital circuitry, consisting of two redundant logic trains, which receives inputs from the analog protection channels to complete the logic necessary to automatically open the reactor trip breakers. Each of the two logic trains, A and B, is capable of opening a separate and independent reactor trip breaker, RTA and RTB, respectively.

The RPS is capable of being tested during power operation. Where only parts of the system are tested at any one time, the testing sequence provides the necessary overlap between the parts to assure complete system operation. The RPS is designed to permit periodic testing of the analog channel portion of the RPS during reactor power operation without initiating a protective action unless a trip condition actually exists. This is because of the coincidence logic required for reactor trip. These tests may be performed at any plant power from cold shutdown to full power. Before starting any of these tests with the plant at power, all redundant reactor trip channels associated with the function to be tested must be in the normal (untripped) mode in order to avoid spurious trips.

The Engineered Safety Features Actuation System (ESFAS) using selected plant parameters determines whether or not predetermined safety limits are being exceeded and, if they are, combines the signals into logic matrices sensitive to combinations indicative of primary or secondary system boundary ruptures (Class III or IV faults). Once the required logic combination is completed, the system sends actuation signals to the appropriate Engineered Safety Features components. The ESFAS consists of two discrete portions of circuitry: 1) An analog portion consisting of three to four redundant channels per parameter or variable to monitor various plant parameters such as reactor coolant system (RCS) and steam system pressures, temperatures, and flows and containment pressures; and 2) a digital portion consisting of two redundant logic trains which receive inputs from the analog protection channels and perform the logic needed to actuate the Engineered Safety Features.

One of the inputs into the RPS and ESFAS is temperature measurement of the RCS. The RCS temperature measurement is accomplished by individual hot and cold leg loop temperature signals that are obtained using resistance temperature detectors (RTDs) installed in each reactor coolant loop. The hot leg temperature measurement on each loop is accomplished with three fast response narrow range RTDs mounted in thermowells, spatially located 120° around the hot leg. One fast response narrow range RTD is located in each cold leg at the discharge of the reactor coolant pump. The cold leg temperature measurement, together with the average hot leg temperature measurement obtained from the three hot leg temperatures, is used to calculate reactor coolant loop delta-T and T-average. These signals are then processed through both protection and control circuits for the RPS and ESFAS.

On August 7, 2003, Maintenance performed a CHAR (Characterization) test on the Unit 1 "A" reactor coolant system (RCS) loop hot leg narrow range resistance temperature detectors (RTDs). Upon return to service, it was discovered that two of the three RTDs were reading off scale high. Measurements taken of the RTDs indicate an open circuit for RTDs A1 and A3. These RTDs were damaged and will need to be replaced. Upon determination of the failure of the RTDs, the associated functions were placed in the tripped condition as required by TS and within the required TS completion time. Current CNS TS would allow continued operation in this condition indefinitely. These RTDs are not able to be replaced at power. Currently, the next scheduled opportunity to be in a mode to allow replacement

is during the upcoming Unit 1 refueling outage (1EOC14) this fall.

Catawba has reviewed scheduled TS surveillances that could be affected by this failure and they fall into three (3) categories. The first category is the TS surveillances that can be extended and not exceed their grace period. These TS surveillances will be rescheduled to occur in the next Unit 1 refueling outage. The second category is those TS surveillances that will have their frequency plus grace period expire before the RTD replacement that would require the inoperable channel to be bypassed. These TS surveillances can be completed within four (4) hours. In order to complete these surveillances, the inoperable channel would have to be bypassed. The TS provide Notes that allow bypassing an inoperable channel for up to 4 hours for surveillance testing of other channels. The third category of TS surveillances is those that require the inoperable channel to be bypassed but cannot be completed within the 4 hour time frame allowed by current TS. These surveillances include power range nuclear instrumentation channel COTs and delta-T protection channel III COT.

Catawba is requesting a change, applicable for a limited time period, to the Notes for remedial actions to change the time allowed to bypass the inoperable channel from 4 hours to 12 hours. The affected functions are as follows:

- TS Table 3.3.1-1, Function 6 - Overtemperature delta-T
- TS Table 3.3.1-1, Function 7 - Overpower delta-T
- TS Table 3.3.2-1, Function 5.d - Turbine Trip and Feedwater Isolation - T_{avg} Low coincident with Reactor Trip, P-4

WCAP-14333 provides the justification for increasing the bypass times for testing in the RPS instrumentation and ESFAS instrumentation Technical Specifications. The NRC issued a Safety Evaluation on July 15, 1998 approving WCAP-14333. These changes have been incorporated in Technical Specification Task Force (TSTF) traveler, TSTF-418, revision 2, RPS and ESFAS Test Times and Completion Times (WCAP-14333).

Description of Proposed Changes

Duke Energy proposes to change the Catawba Nuclear Station (CNS) Unit 1 TS requirements, applicable for a limited time period, to permit relaxation of the allowed bypass test times for Limiting Condition for Operation (LCOs) 3.3.1,

Reactor Trip System (RTS) Instrumentation and 3.3.2, Engineered Safety Feature System Actuation System (ESFAS) Instrumentation. These relaxations are consistent with those that have been generically approved in WCAP-14333-P-A, Revision 1, Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times." The proposed changes are consistent with approved TSTF-418, Revision 2.

The TS changes for Catawba are as follows:

LCO 3.3.1, RTS Instrumentation		
Affected Condition	**Affected Instrumentation**	**Proposed Change**
Condition E	• Overtemperature Delta-T • Overpower Delta-T	Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours.

LCO 3.3.2, ESFAS Instrumentation		
Affected Condition	**Affected Instrumentation**	**Proposed Change**
Condition J	• Turbine Trip and Feedwater Isolation o T_{avg} - Low coincident with Reactor Trip, P-4	Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours.

The following footnote will be added to Condition E of TS 3.3.1.

For Unit 1, during the time that the channel I Over Power delta-T and Over Temperature delta-T trip functions are inoperable due to the failure of the hot leg resistance temperature detectors on the 'A' reactor coolant system hot leg, the Over Power delta-T and Over Temperature delta-T trip functions may be bypassed beyond the 4 hours up to 12 hours. This footnote is not applicable after the Unit 1 refueling outage, 1EOC14.

The following footnote will be added to Condition J of TS 3.3.2.

For Unit 1, during the time that the reactor coolant system channel I T_{avg} input into the turbine trip and feedwater isolation on T_{avg}-Low coincident with Reactor

Trip (P-4) channel is inoperable due to the failure of the hot leg resistance temperature detectors on the 'A' reactor coolant system hot leg, the reactor coolant system channel I T_{avg} input may be bypassed beyond the 4 hours up to 12 hours. This footnote is not applicable after the Unit 1 refueling outage, 1EOC14.

These footnotes will allow Catawba to perform required surveillances without challenging plant protective systems and allow sufficient time to reduce the possibility of human error.

Technical Analysis

Catawba proposes to increase the time allowed to bypass an inoperable channel from 4 hours to 12 hours, applicable for a limited time period. This proposed revision will be used for a short time and only for those surveillances that cannot be completed within 4 hours and if not performed would require a shutdown of Unit 1. Catawba does not have an installed bypass capability. Procedures are being developing to perform this function. These procedures will be validated on a loop simulator prior to use. This loop simulator simulates a single channel and can be used to simulate bypassing that channel. These procedures will also be used to bypass the inoperable channel to perform TS surveillance testing for those tests that can be performed in less than 4 hours. Therefore, Catawba will have the procedures in place and verified to perform the bypass of inoperable channels.

Catawba understands that continued operation with one channel of overpressure and overtemperature delta-T reactor trip signal in the tripped condition, places Unit 1 in a condition where an additional channel failure would result in a reactor trip.

WCAP-14333-P-A, Revision 1 provides the technical basis and methodology for extending the bypass test times as described above. A risk-informed approach was used to justify these changes. The approach is consistent with that recommended by the Nuclear Regulatory Commission in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant- Specific Changes to the Licensing Basis," and Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." The impact of the proposed test time and allowed outage time changes on core damage frequency,

incremental core damage probability, and large early release frequency was assessed by the WCAP. The results of the risk evaluation demonstrated that the results conform to the acceptance guidelines provided in Regulatory Guides 1.174 and 1.177

The Westinghouse Owners Group Technical Specification Optimization Program (WOG TOP) evaluated changes to the bypass times. The NRC approved increasing the bypass test times for the analog channels. A probabilistic risk assessment approach was used in these analyses which included assessing the impact of the changes on signal availability and plant safety. The justification for the acceptability of the changes was the small impact the changes had on plant safety.

Placing a channel in bypass for additional time does reduce the availability of signals to initiate component actuation for event mitigation when required, but as shown in WCAP-14333, the impact on safety is small due to the availability of other signals or operator action to trip the reactor or cause component actuation. Therefore, these proposed changes should reduce the potential for inadvertent reactor trips and inadvertent equipment actuations due to human error or spurious actuation.

The NRC Safety Evaluation for WCAP-14333-P concludes that "the risk analysis in WCAP-14333-P supports the proposed TS changes and is acceptable, subject to the following conditions which must be addressed in referencing licensee's plant-specific license amendment requests:

1. Confirm the applicability of the WCAP-14333-P analyses for their plant.
2. Address the Tier 2 and Tier 3 analyses including the CRMP (Configuration Risk Management Program) insights, by confirming that these insights are incorporated into the referenced licensee's decision making process before taking equipment out of service."

The requirements for the above conditions are addressed in Attachments 6 and 7. Attachment 6 discusses the applicability of the WCAP-14333 analyses to Catawba Nuclear Station Unit 1. Attachment 7 addresses the Tier 2 and 3 requirements. The purpose of the Tier 2 requirements is to ensure the plant risk does not increase to unacceptable levels if multiple components are out of service simultaneously.

Risk Reduction Measures

During the time period that Unit 1 operates with the RCS loop 'A' hot leg RTDs inoperable Catawba will be implementing several actions to reduce overall risk to the plant. These activities are intended to ensure that both planned and emergent work is carefully reviewed to understand its impact on plant operation given the current configuration on Unit 1. Listed below is a summary of some of the risk reduction measures that will be implemented while the Unit 1 RCS loop 'A' hot leg RTDs are inoperable. These actions will be applied to Unit 1 and as applicable to Unit 2 until Unit 1 is in a mode where the affected TS are no longer applicable. These actions being taken are above and beyond our normal work and scheduling process.

1. Increased reviews of Unit 1 scheduled items to include daily and weekly activity review by Operations staff personnel. These reviews would be in addition to routine reviews by Work Control Department and the Operations shift review of the activity schedule. These reviews provide an additional check of upcoming work activities to ensure they do not challenge Unit 1 in its current configuration.
2. Operations management has held discussions with the Operating crews on the importance of comprehensive review of emergent work.
3. The 7300 process protection system cabinets and the power range nuclear instrumentation cabinets have been roped off to restrict entry without the Operations shift permission.
4. On Unit 1, perform thermography of 7300 cabinets and vital inverters on a periodic basis.
5. On Unit 1, implementation of new risk management tool (matrix) to assist in the review of work activities by the plant.
6. A site wide communication has been made to remind plant personnel of restriction on the use of electronic devices in restricted areas.
7. Currently, welding activities inside the protective area have been restricted until further evaluation. This evaluation will be done on a case by case basis in conjunction with item 1 above.

Precedent Licensing Actions for TSTF-418

This proposed license amendment is modeled after those changes approved by the NRC in Technical Specification Task Force (TSTF) traveler 418, revision 2.

The following plants have submitted TS amendments based on WCAP 14333.

Plant	Submittal Date	NRC Approval Date
Indian Point 2	05/05/1999	11/30/2000
North Anna	05/06/1999	03/09/2000
Vogtle	10/13/1999	12/22/2000
South Texas	05/30/2001	03/19/2002

References:

1. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"
2. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications"
3. WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times"
4. Letter, K.J. Vavrek, Westinghouse, to Westinghouse Owners Group Licensing Subcommittee Representatives, "Implementation Guideline for WCAP-14333-P-A, Rev. 1 (Proprietary), "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times" (MUHP-3054), Dated December 2, 1998

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

No Significant Hazards Consideration Determination

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or

3. Involve a significant reduction in a margin of safety.

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes to the bypass test time do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the reactor protection system and engineered safety feature actuation system (RPS and ESFAS) signals. The RPS and ESFAS will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation.

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. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation

exposures. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

Therefore, the proposed changes do not result in a significant increase in the consequences of an accident previously evaluated.

Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant that will introduce any new accident causal mechanisms. In addition, the changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice. Therefore, the possibility of a new or different malfunction of safety related equipment is not created.

Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by these changes. Redundant RPS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and ESFAS is also maintained. The signals credited as primary or secondary and the operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside design basis. Placing a channel in bypass for additional time does reduce the availability of signals to initiate component actuation for event mitigation

when required, but as shown in WCAP-14333, the impact on safety is small due to the availability of other signals or operator action to trip the reactor or cause component actuation.

Therefore, it is concluded that this change does not involve a significant reduction in the margin of safety.

Based upon the preceding discussion, Duke Energy has concluded that the proposed amendment does not involve a significant hazards consideration.

ATTACHMENT 5

ENVIRONMENTAL ANALYSIS

Environmental Analysis

Pursuant to 10 CFR 51.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

Implementation of this amendment will have no adverse impact upon the Catawba units; neither will it contribute to any additional quantity or type of effluent being available for adverse environmental impact or personnel exposure.

It has been determined there is:

1. No significant hazards consideration,

2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and

3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, this amendment to the Catawba TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

ATTACHMENT 6

APPLICABILITY OF WCAP-14333-P ANALYSES TO CATAWBA

APPLICABILITY OF WCAP-14333-P ANALYSES TO CNS

In accordance with the guidance provided in letter WOG-98-245, "Implementation Guideline for WCAP-14333-P-A, Rev. 1 (Proprietary), "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," dated December 2, 1998; the information provided in the attached tables demonstrates the applicability of the generic WCAP-14333 analysis to the Catawba Nuclear Station Unit 1.

Table 1
WCAP-14333 Implementation Guidelines: Applicability of the Analysis
General Parameters

Parameter	WCAP-14333 Analysis Assumptions	Plant Specific Parameter
Logic Cabinet Type (1)	Relay and SSPS	SSPS
Component Test Intervals (2)		
• Analog Channels	3 months	92 Days
• Logic Cabinets (SSPS)	2 months	31 Days (Staggered Test Basis)
• Logic Cabinets (Relay)	1 month	NA
• Master Relays (SSPS)	2 months	31 Days (Staggered Test Basis)
• Master Relays (Relay)	1 month	NA
• Slave Relays	3 months	92 Days
• Reactor Trip Breakers	2 months	31 Days (Staggered Test Basis)
Analog Channel Calibrations (3)		
• Done at-power	Yes	Yes
• Interval	18 months	18 Months
Typical At-Power Maintenance Intervals (4)		
• Analog Channels	24 months	Equal to or greater than
• Logic Cabinets (SSPS)	18 months	Equal to or greater than
• Logic Cabinets (Relay)	12 months	NA
• Master Relays (SSPS)	infrequent (5)	Infrequent
• Master Relays (Relay)	infrequent (5)	NA
• Slave Relays	infrequent (5)	Infrequent
• Reactor trip breakers	12 months	Equal to or greater than
AMSAC (6)	Credited for AFW pump start	Yes, provides AFW pump start.
Total Transient Event Frequency (7)	3.6	1.4 events/reactor year
ATWS Contribution to CDF (current PRA model) (8)	8.4E-06	3.3E-07 events/reactor year
Total CDF from Internal Events (current PRA model) (9)	5.8E-05	5.2E-05 events/reactor year
Total CDF from Internal Events (IPE) (10)	Not Applicable	5.9E-05 events/reactor year

Notes for Table 1

1. The analysis is applicable to Catawba Unit 1 because both types of logic cabinets are supported by WCAP-14333.
2. Catawba Unit 1 test intervals are equal to or greater than those used in WCAP-14333; therefore, the analysis is applicable.
3. Catawba Unit 1 channel calibrations are at-power and the calibration interval is equal to or greater than that used in WCAP-14333, therefore, the analysis is applicable.
4. The maintenance intervals for Catawba Unit 1 are equal to or greater than those used in WCAP-14333; therefore, the analysis is applicable.
5. Only corrective maintenance is done on the master and slave relays. The maintenance interval on typical relays is relatively long, that is, experience has shown they do not typically completely fail. Failure of slave relays usually involves failure of individual contacts. Fill in "infrequent" if this is consistent with your plant experience. If "infrequent" slave relay failures are the norm, then the WCAP-14333 analysis is applicable to your plant.
6. The Catawba AMSAC design will initiate AFW pump start for the Motor Driven AFW pumps; therefore the WCAP-14333 analysis is applicable to Catawba. However, the Catawba PRA does not credit its function due to associated small contribution to core damage frequency.
7. Includes total frequency for internal initiators requiring a reactor trip signal to be generated for event mitigation. This is required to assess the importance of ATWS events to CDF. Does not include events initiated by a reactor trip (i.e. reactor trip initiating event frequency is excluded).
8. Indicates the ATWS contribution to core damage frequency (from at-power, internal events). This is required to determine if the ATWS event is a large contributor to CDF.
9. Indicates the total CDF from internal events (including internal flooding) for the most recent PRA model update. This is required for comparison to the NRC's risk-informed CDF acceptance guidelines.
10. Indicates the total CDF from internal events from the IPE model (submitted to the NRC in response to Generic Letter 88-20). These values differ from the most recent PRA model update CDF so a concise list of reasons, in bulletized form, describing the differences between the models that account for the change in CDF, is provided. (See attached.)

WCAP-14333 Implementation Guidelines: Applicability of the Analysis
General Parameters
Table 1 Note 10
Additional Information

Plant model changes for Catawba Nuclear Station since IPE:

- Added backup cooling to the high head safety injection centrifugal charging pumps.
- Overall model component and logic review and update.
- Updated human error reliability data.
- Updated common cause data.
- Updated plant specific data.
- Updated initiating event frequencies.
- Updated system notebooks.
- Updated generic data.

Table 2
WCAP-14333 Implementation Guideline: Applicability of Analysis (Cont'd)
Reactor Trip Actuation Signals

Event	WCAP-14333 and WCAP-15376 Analysis Assumption	Plant Specific Parameter (1)
Large LOCA	Not Required	NA
Medium LOCA	Not Required	NA
Small LOCA	Nondiverse (2) w/OA (3)	Agree
Steam Generator Tube Rupture	Nondiverse w/OA	Agree
Interfacing System LOCA	Not Required	NA
Reactor Vessel Rupture	Not Required	NA
Secondary Side Breaks	Nondiverse w/OA	Agree
Transient Events, such as:	Diverse (4) w/OA	
- Positive Reactivity Insertion		Agree
- Loss of Reactor Coolant Flow		Agree
- Total or Partial Loss of Main Feedwater		Agree
- Loss of Condenser		Agree
- Turbine Trip		Agree
- Loss of DC Bus		Not Agree(5)
- Loss of Vital AC Bus		Not Agree(5)
- Loss of Instrument Air		Agree
- Spurious Safety Injection		Agree
- Inadvertent Opening of a Steam Valve		Agree
Reactor Trip	Generated by RPS	Agree
Loss of Offsite Power	Not Required by RPS	NA
Station Blackout	Not Required by RPS	NA
Loss of Service Water or Component Cooling Water	Nondiverse w/OA	Agree

Notes:
1. Use of "agree" for an event indicates that the WCAP-14333 analysis is applicable to Catawba.
2. Nondiverse means that (at least) one signal will be generated to initiate reactor trip for the event.
3. OA indicates that an operator could take action to initiate reactor trip for the event, that is, there is sufficient time for action and procedures are in place that will instruct the operator to take action.
4. Diverse means that (at least) two signals will be generated to initiate reactor trip for the event.
5. There is no direct reactor trip from loss of DC bus or Vital AC bus at Catawba.

Table 3
WCAP-14333 Implementation Guideline: Applicability of Analysis (Cont'd)
Engineered Safety Features Actuation Signals

Safety Function	Event	WCAP-14333 and WCAP-15376 Analysis Assumption	Plant Specific Parameter (1)
Safety Injection	Large LOCA	Nondiverse (2)	Agree
	Medium LOCA	Nondiverse, OA (3) by SI switch on main control board	Agree
	Small LOCA	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree
	Interfacing Systems LOCA	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree
	SG Tube Rupture	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree
	Secondary Side Breaks	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree
Auxiliary Feedwater Pump Start	Events generating SI signal Transient events	Pump actuation on SI signal Nondiverse, AMSAC, operator action	Agree Agree
Main Feedwater Isolation	Secondary Side Breaks	Nondiverse	Agree
Steamline Isolation	Secondary Side Breaks	Nondiverse	Agree
Containment Spray Actuation	All events	Nondiverse signal	Agree
Containment Isolation	All events	From SI signal	Agree
Containment Cooling	All events	From SI signal	Agree

Notes:
1. Use of "agree" for an event indicates that the WCAP-14333 analysis is applicable to Catawba.
2. Nondiverse means that (at least) one signal will be generated to initiate the engineered safety feature noted for the event.
3. OA indicates that an operator could take action to initiate the engineered safety feature for the event, that is, there is sufficient time for action and procedures are in place that will instruct the operator to take action.

ATTACHMENT 7

TIER 2 AND TIER 3 ANALYSES

NRC Condition and Limitation

Address the Tier 2 and Tier 3 analyses including risk significant configuration insights and confirm that these insights are incorporated into the plant-specific configuration risk management program.

Duke Response to NRC Condition and Limitation

The Tier 2 requirements of RG 1.177 state that the licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed TS change. Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations associated with the proposed change and provides reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed TS change is out-of- service.

The Tier 3 requirements of RG 1.177 require the licensee to develop a program that ensures that the risk impact of out-of- service equipment is appropriately evaluated prior to performing any maintenance activity. Tier 3 provides for the establishment of an overall Configuration Risk Management Program (CRMP) and confirmation that its insights are incorporated into the decision-making process before taking equipment out-of-service prior to or during the completion time. Tier 3 provides additional coverage on the basis of any additional risk-significant configurations that may be encountered during maintenance scheduling over extended periods of plant operation. Tier 3 guidance is satisfied by 10 CFR 50.65(a)(4) which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance, testing, and corrective and preventive maintenance.

The Tier 2 and Tier 3 requirements of RG 1.177 have been addressed at Duke for Catawba.

Tier 2 Assessment: Risk-significant Plant Equipment Outage Configurations

Attachment 3 discusses the risk reduction actions that will be in effect during the time period that this TS change will be in place. These measures have been put in place to increase the review of work activities to ensure that they have been reviewed thoroughly and that proper management review has taken place in light of the configuration that Unit 1 is currently in. These measures have

been put in place to reduce the time that Unit 1 could be in risk-significant configuration to those times where absolutely necessary. These measures will ensure that additional reviews by plant management will occur in those circumstances.

Tier 3 Assessment: Maintenance Rule Configuration Control

10 CFR 50.65 (a)(4), RG 1.182[1], and NUMARC 93-01[2] require that prior to performing maintenance activities, risk assessments shall be performed to assess and manage the increase in risk that may result from proposed maintenance activities. These requirements are applicable for all plant modes. NUMARC 91-06[3] requires utilities to assess and manage the risks that occur during the performance of outages.

Duke has several Work Process Manual procedures and Nuclear System Directives that are in place at Catawba Nuclear Stations to ensure the requirements of the Maintenance Rule are implemented. The key documents are as follows:

- Nuclear System Directive 415, "Operational Risk Management (Modes 1-3) per 10 CFR 50.65 (a.4)," Revision 1, April, 2002.
- Nuclear System Directive 403, "Shutdown Risk Management (Modes 4, 5, 6, and No-Mode) per 10 CFR 50.65 (a.4)," Revision 11, December, 2002.
- Work Process Manual, WPM-609, "Innage Risk Assessment Utilizing ORAM-SENTINEL," Revision 7, December, 2002.
- Work Process Manual, WPM-608, "Outage Risk Assessment Utilizing ORAM-SENTINEL," Revision 6, July, 2002.

The documents listed above are used to address the Maintenance Rule requirement and the on-line (and off-line) Maintenance Policy requirement to control the safety impact of combinations of equipment removed from service. They assure that the risk associated with the various plant configurations planned during at-power or shutdown conditions are assessed prior to entry into these configurations and appropriately managed while the plant is in these various configurations. More specifically, the Nuclear System Directives address the process, define the program, and state individual group responsibilities to ensure compliance with the Maintenance Rule.

The Work Process Manual procedures provide a consistent process for utilizing the computerized software assessment tool, ORAM-SENTINEL, which manages the risk associated with equipment inoperability. ORAM-SENTINEL is a Windows-based computer program designed by the Electric Power Research Institute as a tool for plant personnel to

use to analyze and manage the risk associated with all risk significant work activities including assessment of combinations of equipment removed from service. It is independent of the requirements of Technical Specifications and Selected Licensee Commitments.

The ORAM-SENTINEL models for Catawba are based on a "blended" approach of probabilistic (the full at power PRA models are utilized) and traditional deterministic approaches. The results of the risk assessment include a prioritized listing of equipment to return to service, a prioritized listing of equipment to remain in service, and potential contingency considerations.

Additionally, prior to the release of work for execution, Operations personnel must consider the effects of severe weather and grid instabilities on plant operation. This qualitative evaluation is inherent of the duties of the Work Control Center Senior Reactor Operator (SRO). Responses to actual plant risk due to severe weather or grid instabilities are programmatically incorporated into applicable plant emergency or response procedures.

Attachment 3 describes additional risk reduction measures that have been taken to ensure that the current status of Unit 1 is reviewed when reviewing upcoming work activities.

References:

1. NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000.

2. NUMARC 93-01, Revision 3, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," July 2000.

3. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.