UNI'ED STATES
NUCLEAR REGI'LATORY COMMISSION
WASHI'IGTON, D. C. 20555

March 29, 1989

MEMORANDUM FOR: Charles E. Rossi, Dir·
Division of Operationa ɛnts

Lawrence Shao, Director
Division of Engineering and
Systems Technology

FROM: Gus C. Lainas, Acting Director
Division of Reactor Projects - I/II

SUBJECT: MCGUIRE DPO TECH''CAL SPECIFICATIONS (TACS 55435, 55436, and 67757)

By memorandum of March 15, 1989 to J. Sniezek, R. Licciardo discussed his concerns for the delayed closure of Technical Specification deficiencies arising from his Differing Professional Opinion (DPO) review of the 1984 "Proof and Review" version of the proposed McGuire Technical Specifications (TS). Mr. Licciardo also outlined the various reviews and actions taken in the past with respect to the numerous (380) original items identified, noting that in 1985, 220 of these had been divided into three groups: (1) generic, (2) plant specific, and (3) closed.

Subsequent actions toward resolution of the generic and plant specific concerns have proceeded in accordance with assignments in H. Thompson's memorandum of May 28, 1985. Specifically, the plant specific items were forwarded to Duke (and, in turn, to Westinghouse) for comment and for submittal of TS change requests as appropriate. Duke's reply of June 10, 1986, indicates that five of the items had potential impact on the TS and three on the FSAR, but deferred submittals pending NRC approval. Duke also recommended that three of the items be reclassified as generic.

The generic items from Thompson's 1985 memo and all of the Duke responses are in final stages of review by Reactor Systems Branch under TACS 55435/55436 (plant specific) and 67757 (generic). The review utilized technical assistance of an engineering consultant under completed Contract NRC-03-85-051. Additionally, the on going TS Improvement Program is in advanced stages with approved criteria as to content for the new STS. Mr. Licciardo's items have received continuing review by OTSB for potential application to the TS Improvement Program.

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Accordingly, I believe the basis now exists for prompt final resolution of these items. I look forward to our meeting of March 30 (at 8:00 am in S. Varga's office, 12-D-9) to discuss your plans to this end.

/s/

Gus C. Lainas, Acting Director
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc: J. Taylor
T. Murley
J. Sniezek
F. Miraglia
J. Partlow
S. Varga

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G. Lainas	14-E-4
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Duke Power Company
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Charlotte, N.C. 28242

Hal B. Tucker
Vice President
Nuclear Production
(704)373-4531

DUKE POWER

September 15, 1989

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

Subject: McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370
Requested Technical Specifications Changes
NRC Differing Professional Opinion Concerns (T.S. 3/4.3.2)
(TACS 55435 and 55436)

Gentlemen:

Attached (pursuant to 10CFR50.4 and 50.90) are license amendment requests to Appendix A, Technical Specifications, of Facility Operating Licenses NPF-9 and NPF-17 for McGuire Nuclear Station Units 1 and 2, respectively. The requested amendments correct an inadvertent error and more adequately define an Engineered Safety Features Actuation System Instrumentation trip setpoint/allowable value, add appropriate Engineered Safety Features response times, and revise an Engineered Safety Features response time. These changes result from certain concerns raised by the NRC in a Differing Professional Opinion (DPO) during review of the proposed original McGuire Units 1 and 2 combined Technical Specifications.

Attachment 1 contains the requested Technical Specification changes, Attachment 2 contains the justification and safety analysis to support the requested changes. Pursuant to 10CFR 50.91, Attachment 3 provides the analysis performed in accordance with the standards contained in 10CFR 50.92 which concludes that the proposed amendments do not involve a Significant Hazards Consideration, and Duke is forwarding a copy of this amendment request application and No Significant Hazards Consideration Analysis to the North Carolina Department of Human Resources. The requested amendments have been reviewed and have been determined to have no adverse safety or environmental impact.

Note that these requested changes are in accordance with our June 10, 1986 response to the plant-specific NRC DPO concerns, which were subsequently concurred with by the NRC, and should resolve the relevant concerns. Please be advised that the FSAR revisions identified in the June 10th response (i.e.. question nos. 4a&b, and 4c), which were also concurred with by the NRC, will be incorporated into the McGuire FSAR via the 1989 FSAR annual update. These requested Technical Specifications amendments and the upcoming FSAR update will complete all currently known Duke Power Company required actions regarding the DPO (other concerns contained in the DPO were either being considered by the NRC for generic resolution, had been closed by NRC internal review, or were still under review). It is anticipated that any other concerns will be addressed through McGuire implementation of the generic Technical Specification improvement efforts currently underway by industry (e.g., WOG MERITS/NRC TSIP).

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This matter was discussed with the NRC Resident Inspector and Mr. D. S. Hood of your staff at the June 21, 1989 Duke/NRC Interface Meeting. Should there be any questions concerning these amendment requests or if additional information is required, please advise.

Very truly yours,

Hal B. Tucker

PBN178/lcs

Attachments

xc: (w/attachments)
Mr. S. D. Ebneter
Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
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Mr. Dayne Brown, Chief
Radiation Protection Branch
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701 Barbour Drive
Raleigh, North Carolina 276[illegible]3-2008

Mr. Darl S. Hood
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. P. K. VanDoorn
NRC Resident Inspector
McGuire Nuclear Station

Hal B. Tucker, being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this revision to the McGuire Nuclear Station Technical Specifications, Appendix A to Facility Operating License Nos. NPF-9 and NPF-17; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

Hal B. Tucker, Vice President

Subscribed and sworn to before me this 15th day of September, 1989.

____

Notary Public

My Commission Expires:

____

ATTACHMENT 1

McGuire Unit 1 and 2 Technical Specifications Changes Requests

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3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) Instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation channel or interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS Instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

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TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3	15
d. Pressurizer Pressure - Low-Low	4	2	3	1, 2, 3#	19
e. Steam Line Pressure-Low					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (continued)					
b. Phase "B" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High	4	2	3	1, 2, 3	16
c. Purge and Exhaust Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) System	2	1	2	1, 2, 3	22
2) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
d. Negative Steam Line Pressure Rate - High					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	3##	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)
e. Steam Line Pressure - Low					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Turbine Trip & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relay	2	1	2	1, 2	21
b. Steam Generator Water Level--High-High	3/stm. gen.	2/stm. gen. in any operating stm gen.	2/stm. gen. in each operating stm. gen.	1, 2	15
c. Doghouse Water Level (Feedwater Isolation Only)	3/train/ Doghouse	2/train/ Doghouse	2/train/ Doghouse	1, 2	25
6. Containment Pressure Control System	8(4/train)	4/train	8	1, 2, 3, 4	26

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Auxiliary Feedwater					
a. Manual Initiation	2	1	2	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Stm. Gen. Water Level-Low-Low					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen in each operating stm. gen	1, 2, 3	19
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	2/motor driven pump	2/pump	2 of the same train/pump	1,2,3	24
	4/turbine driven pump	2/pump	2 of the same train/pump	1,2,3	24
e. Safety Injection Start Motor-Driven Pumps					

See Item 1. above for all Safety Injection initiating functions and requirements

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Auxiliary Feedwater (continued)					
f. Station Blackout (Note 1) Start Motor-Driven Pumps and Turbine-Driven Pump	6-3/Bus	2/Bus Either Bus	2/Bus	1, 2, 3	19
g. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	2-1/MFWP	2-1/MFWP	2-1/MFWP	1, 2#	27
8. Automatic Switchover to Recirculation					
RWST Level	3	2	2	1, 2, 3	15
9. Loss of Power					
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15a
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22
d. Steam Generator Level, P-14	3/stm gen.	2/stm gen. in any operating stm gen.	2/stm gen. in each operating stm gen.	1, 2, 3	20

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TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

##Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam pressure is not blocked.

**These values left blank pending NRC approval of three loop operation.

Note 1: Turbine driven auxiliary feedwater pump will not start on a blackout signal coincident with a safety injection signal.

ACTION STATEMENTS

ACTION 14 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 15a With the number of OPERABLE channels less than the total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour. With more than one channel inoperable, enter Specification 3.8.1.1.

ACTION 16 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

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TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.11 and Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the action required by Specification 3.7.1.4.

ACTION 24 - With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.7.1.2. With the channels associated with more than one auxiliary feedwater pump inoperable, immediately declare the associated auxiliary feedwater pumps inoperable and take the action required by Specification 3.7.1.2.

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ACTION 25 - With one of the two trains of doghouse water level instrumentation inoperable (less than the minimum required number of channels operable), restore the inoperable train to operable status in 72 hours. After 72 hours with one train inoperable, or within one hour with 2 trains inoperable, monitor doghouse water level in the affected doghouse continuously until both trains are restored to operable status.

ACTION 26 - With any of the eight channels inoperable, place the inoperable channel(s) in the start permissive mode within one hour and apply the applicable action statement (Containment Spray - T.S. 3.6.2, Containment Air Return/Hydrogen Skimmer - T.S. 3.6.5.6).

ACTION 27 - With the number of OPERABLE channels one less than the total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water.		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High	$\leq$ 1.1 psig	$\leq$ 1.2 psig
d. Pressurizer Pressure--Low-Low	$\geq$ 1845 psig	$\geq$ 1835 psig
e. Steam Line Pressure - Low	$\geq$ 585 psig	$\geq$ 565 psig
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
c. Purge and Exhaust Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
d. Negative Steam Line Pressure Rate - High	$\leq$ ~~-~~100 psi~~/sec~~ WITH A RATE/LAG FUNCTION TIME CONSTANT $\geq$ 50 SECONDS	$\leq$ ~~-~~120 psi~~/sec~~ WITH A RATE/LAG FUNCTION TIME CONSTANT $\geq$ 50 SECONDS
e. Steam Line Pressure - Low	$\geq$ 585 psig	$\geq$ 565 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	$\leq$ 82% of narrow range instrument span each steam generator	$\leq$ 83% of narrow range instrument span each steam generator
c. Doghouse Water Level-High (Feedwater Isolation Only)	12"	13"
6. Containment Pressure Control System		
Start Permissive/Termination (SP/T)	0.3 $\leq$ SP/T $\leq$ 0.4 PSIG	0.25 $\leq$ SP/T $\leq$ 0.45 PSIG

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	$\geq$ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 40.0% of span at 100% of RATED THERMAL POWER.	$\geq$ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 39.0% of span at 100% of RATED THERMAL POWER.
2) Start Turbine-Driven Pumps	$\geq$12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 40.0% of span at 100% of RATED THERMAL POWER.	$\geq$ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to $\geq$ 39.0% of span at 100% of RATED THERMAL POWER.
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	$\geq$ 2 psig	$\geq$ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
f. Station Blackout - Start Motor-Driven Pumps and Turbine-Driven Pump (Note 1)	3464 ± 173 volts with a 8.5 ± 0.5 second time delay	$\geq$ 3200 volts
g. Trip of Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Automatic Switchover to Recirculation		
RWST Level	$\geq$ 90 inches	$\geq$ 80 inches
9. Loss of Power		
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	3464 ± 173 volts with a 8.5 ± 0.5 second time delay	$\geq$ 3200 volts
10. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	$\leq$ 1955 psig	$\leq$ 1965 psig
b. T_{avg}, P-12	$\geq$ 553°F	$\geq$ 551°F
c. Reactor Trip, P-4	N.A.	N.A.
d. Steam Generator Level, P-14	See Item 5. above for all Trip Setpoints and Allowable Values.	

Note 1: The turbine driven pump will not start on a blackout signal coincident with a safety injection signal.

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Containment Isolation	
Phase "A" Isolation	N.A.
Phase "B" Isolation	N.A.
Purge and Exhaust Isolation	N.A.
d. Steam Line Isolation	N.A.
e. Feedwater Isolation	N.A.
f. Auxiliary Feedwater	N.A.
g. Nuclear Service Water	N.A.
h. Component Cooling Water	N.A.
i. Reactor Trip (from SI)	N.A.
j. Start Diesel Generators	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27^{(1)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 9
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	~~N.A.~~ ≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
i. Start Diesel Generators	≤ 11

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low-Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 9
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	~~N.A.~~ ≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water System	$\leq 76^{(1)}/65^{(3)}$
h. Component Cooling Water	$\leq 76^{(1)}/65^{(3)}$
i. Start Diesel Generators	≤ 11
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12^{(3)}/22^{(4)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 9
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	~~N.A.~~ ≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Steam Line Isolation	≤ 7
i. Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
j. Start Diesel Generators	≤ 11
5. Containment Pressure--High-High	
a. Containment Spray	≤ 45
b. Containment Isolation-Phase "B"	N.A.
c. Steam Line Isolation	≤ 7
6. Steam Generator Water Level--High-High	
a. Turbine Trip	N.A.
b. Feedwater Isolation	≤ ~~13~~ 9

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Steam Generator Water Level LowLow	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-driven Auxiliary Feedwater Pumps	≤ 60
8. Negative Steam Line Pressure Rate - High	
Steam Line Isolation	≤ 7
9. Start Permissive	
Containment Pressure Control System	N.A.
10. Termination	
Containment Pressure Control System	N.A.
11. Auxiliary Feedwater Suction Pressure - Low	
Auxiliary Feedwater Pumps (Suction Supply Automatic Realignment)	≤ 13
12. RWST Level	
Automatic Switchover to Recirculation	≤ 60
13. Station Blackout	
a. Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Start Turbine-Driven Auxiliary Feedwater Pump (6)	≤ 60
14. Trip of Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
15. Loss of Power	
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	≤ 11

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TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps, Safety Injection and RHR pumps.

(2) Valves 1KC305B and 1KC315B for Unit 1 and Valves 2KC305B and 2KC315B for Unit 2 are exceptions to the response times listed in the table. The following response times in seconds are the required values for these valves for the initiating signal and function indicated:

2.d	$\leq 30^{(3)}/40^{(4)}$
3.d	$\leq 30^{(3)}$
4.d	$\leq 30^{(3)}/40^{(4)}$

(3) Diesel generator starting and sequence loading delays <u>not</u> included. Offsite power available. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps and Safety Injection pumps.

(4) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps and Safety Injection pumps.

(5) Response time for motor-driven auxiliary feedwater pumps on all Safety Injection signal shall be less than or equal to 60 seconds. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for auxiliary feedwater pumps.

(6) The turbine driven pump does not start on a blackout signal coincident with a safety injection signal.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection		See Item 1. above for all Safety Injection Surveillance Requirements.						
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection		See Item 1. above for all Safety Injection Surveillance Requirements.						

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Negative Steam Line Pressure Rate-High	S	R	M	N.A.	N.A.	N.A.	N.A.	3
e. Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Doghouse Water Level-High (Feedwater Isolation Only)	S	N.A	N.A	R	N.A.	N.A.	N.A.	1, 2
6. Containment Pressure Control System								
Start Permissive/ Termination	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A	N.A.	1, 2, 3
d. Auxiliary Feedwater Suction Pressure-Low	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
f. Station Blackout	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
8. Automatic Switchover to Recirculation								
RSWT Level	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
9. Loss of Power								
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Level, P-14	S	R	M	N.A.	M(1)	M(1)	[illegible]	1, 2, 3

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TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

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3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip and Engineered Safety Features Actuation System instrumentation and interlocks ensure that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features Instrumentation and (3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. (Implementation of quarterly testing of RTS is being postponed until after approval of a similar testing interval for ESFAS.) The NRC Safety Evaluation Report for WCAP-10271 was provided in a letter dated February 21, 1985 from C. O. Thomas (NRC) to J. J. Sheppard (WOG-CP&L).

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in-place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the

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3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, and (10) nuclear service water pumps start and automatic valves position.

Technical Specifications for the Reactor Trip Breakers and the Reactor Trip Bypass Breakers are based upon NRC Generic Letter 85-09 "Technical Specifications for Generic Letter 83-28, Item 4.3," dated May 23, 1985.

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REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 Defeats the manual block of Safety Injection actuation on low pressurizer pressure and low steamline pressure and defeats steamline isolation on negative steamline pressure rate. Defeats the manual block of the motor-driven auxiliary feedwater pumps on trip of main feedwater pumps and low-low steam generator water level.

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the steam dump system. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the steam dump system.

P-14 On increasing steam generator level, P-14 automatically trips all feedwater isolation valves and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to intiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution

ATTACHMENT 2

Justification and Safety Analysis

The requested changes to Technical Specification 3/4.3.2 correct an inadvertent error and more adequately define an Engineered Safety Features Actuation System Instrumentation trip setpoint/allowable value, add appropriate Engineered Safety Features response times, and revise an Engineered Safety Features response time. The changes result from certain concerns raised by the NRC during review of the proposed original McGuire units 1 and 2 combined Technical Specifications.

Background/Justification:

Mr. T. M. Novak's (NRC/ONRR) July 9, 1985 letter to Mr. H. B. Tucker (DPC) indicated that a review of the McGuire Unit 1 and 2 Technical Specifications was being conducted in response to concerns raised by a member of the NRC staff in a differing professional opinion (DPO) resulting from a review of the proof and review copy of the McGuire Units 1/2 combined Technical Specifications which existed in mid-January 1983. Duke Power Company's comments were requested on certain plant-specific concerns contained in the DPO.

By Mr. H. B. Tucker's letter to Mr. H. R. Denton (NRC/ONRR) dated June 10, 1986, Duke Power Company responded to those concerns. It was noted that the response had potential plant-specific impacts on the station's Technical Specifications [i.e. question nos. 6a, 7d (and 7i, 7k), and 7n], and that Duke would pursue appropriate plant-specific Technical Specifications revisions following NRC concurrence with positions contained therein. Consequently, Mr. D. S. Hood's (NRC/ONRR) letter of May 16, 1989 indicated that the NRC staff had reviewed the responses which were identified as possibly impacting the McGuire Technical Specifications, and found the items to be suitable candidates for Technical Specifications amendments.

Accordingly, these requested Technical Specifications changes are being submitted in accordance with the May 16, 1989 NRC letter.

Description of Requested Changes and Bases/Safety Analysis:

I. Technical Specifications Table 3.3-4, Item 4.d

Technical Specifications Table 3.3-4 specifies the Engineered Safety Features Actuation System Instrumentation trip setpoints and allowable values for various functional units. The table's functional unit no. 4 item d addresses Negative Steam Line Pressure Rate-High (for Steam Line Isolation). The requested change would delete the negative sign in front of the trip setpoint and allowable value values, and correct the per second portion of the values' descriptors.

The negative signs in front of the values were inadvertently included in the Technical Specifications when they were written. Since the fact that the values are negative is evidenced by the item's description, i.e. "Negative Steam Line Pressure Rate-High" (emphasis added), it is clear that the negative signs are not appropriate.

The values' descriptors are changed to more adequately define the values' conservatisms by stating "with a rate/lag function time constant ≥50 seconds". This is in accordance with the provisions of the Westinghouse Reactor Protection System/Engineered Safety Features Actuation System Setpoint Methodology utilized for McGuire (ref. W. O. Parker (DPC) letter to H. R. Denton (NRC) dated October 8, 1981).

Note that the above two changes are also internally (within the Technical Specifications) consistent with other similar type values signs/descriptors in Technical Specifications, e.g., Table 2.1-1, Item 4. These clarifying changes are administrative in nature, and do not involve changes to the actual values themselves or the manner in which they are implemented/used. These changes are in accordance with the response to DPO Question No. 6a.

II. Technical Specification Table 3.3-5, Items 2.e, 3.e, and 4.e

Technical Specifications Table 3.3-5 specifies the Engineered Safety Features Response Times. The table's Initiating Signal and Function nos. 2, 3, and 4 items e addresses Containment Purge and Exhaust Isolation (for Containment Pressure-High, Pressurizer Pressure-Low-Low, and Steam Line Pressure-Low, respectively). The requested change would show a response time of ≤4 seconds for each of these three items, which currently indicate that response time is "N.A." (i.e. not applicable).

The current Technical Specifications indications of N.A. for these three items (indicating that no credit was taken in the licensing basis accident analyses for those channels) are incorrect since response times have been used to minimize offsite consequences of any condition occurring while containment purge and exhaust is being used. Section 15.B.2 of the McGuire FSAR considers the case of a LOCA concurrent with lower containment pressure relief. The results of the additional offsite dose due to this accident are presented in FSAR Table 15.0.12-1. One of the parameters used to evaluate this case is the isolation time for the Containment Air Release and Addition (VQ) System valves which are used in venting lower containment (FSAR section 9.5.12.3 indicates that these valves auto close on a containment isolation). FSAR Table 15.B.2-1 indicates the isolation time for these valves is 4 seconds. Isolation times greater than 4 seconds (or no isolation at all) could result in offsite consequences in excess of that assumed in plant accident analysis. Therefore, the current Technical Specifications are non-conservative with respect to the licensing basis.

Consequently, the Technical Specifications are being changed to reflect the ≤4 second response times to ensure compliance with the assumptions used in the plant licensing basis analysis. Note that the ≤4 second response times are consistent with FSAR section 9.5.12.3 which indicates that these valves have a 3 second closure time, and the allowable 1 second for generating an ESF response as indicated in FSAR section 7.3.1.2.6. These changes are more conservative than the current Technical Specifications since they constitute an additional restriction not presently included in the Technical Specifications. These changes are in accordance with the response to DPO questions nos. 7d, 7i, and 7k.

III. Technical Specifications Table 3.3-5, Item 6.b

Technical Specifications Table 3.3-5 Item 6.b specifies the Engineered Safety Features Response Time addressing Feedwater Isolation (for Steam Generator Water Level--High-High). The requested change would lower the required response time to ≤9 seconds (from the currently specified ≤13 seconds).

The licensing basis safety analysis depending on this response time is excessive feedwater flow at full power, analyzed in FSAR section 15.1.2. FSAR Table 15.1.2-1 gives the sequence of events for this analysis. The High-High Steam Generator Water Level setpoint is reached at 27 seconds with feedwater isolation occurring at 36 seconds (i.e., 9 seconds later). For another 4 seconds in isolation time (i.e., ≤ 13 seconds), an additional mass increase beyond that assumed in the analysis could be expected. This additional feedwater level can affect the consequences of the event at power, if there has been a trip, with potential effects on reactivity control (e.g., power restoration) and/or overfill of the steam generator to cause water ingress (i.e., flooding) into the main steam lines. Additionally, it can have consequences of potentially larger importance for the event occurring from subcritical zero power. Therefore, the current Technical Specification value of ≤13 seconds is less conservative than the licensing basis.

Consequently, to ensure compliance with the assumptions used in the safety analysis, the Technical Specification value is being changed to ≤9 seconds. Note that this 9 seconds value is consistent with the values used for feedwater isolation on other ESF initiating signals (ref. Technical Specifications Table 3.3-5 items 2.c, 3.c, and 4.c). This change is more conservative than the current Technical Specification since it is more limiting than the ≤13 second value. This change is in accordance with the response to DPO question no. 7n.

Conclusions:

These requested Technical Specifications changes clarify an Engineered Safety Features Actuation System Instrumentation trip setpoint/allowable value, and correct Non-Conservative Engineered Safety Features response times. Note that no changes to the Technical Specifications Bases section are necessitated by the requested amendments. Based upon the preceding justification, Duke Power Company concludes that the requested amendments are necessary to adequately define the trip setpoint/allowable value and to ensure compliance with the plant licensing basis safety analyses. Based upon the preceding safety analysis, Duke Power Company concludes that the requested amendments will not be inimical to the health and safety of the public, and would actually help ensure the health and safety of the public. Further, the requested amendments have been reviewed and concurred with by Westinghouse, who performed the relevant licensing basis analyses. In addition, the NRC has previously reviewed the requested changes and deemed them suitable candidates for Technical Specifications amendments.

ATTACHMENT 3

Analysis of Significant Hazards Consideration

Introduction:

As required by 10CFR 50.91, this analysis is provided concerning whether the requested amendments involve significant hazards considerations, as defined by 10CFR 50.92. Standards for determination that an amendment request involves no significant hazards considerations are if operation of the facility in accordance with the requested 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.

The requested amendments correct an inadvertent error and more adequately define a Engineered Safety Features Actuation System (ESFAS) Instrumentation trip setpoint/allowable value, add appropriate Engineered Safety Features (ESF) response times, and revise an Engineered Safety Features response time.

Analysis:

I. Technical Specifications Table 3.3-4, Item 4.d

The requested changes to delete the negative signs in front of the Negative Steam Line Pressure Rate-High (for Steam Line Isolation) ESFAS instrumentation trip setpoint and allowable value, and alter the the values' descriptors to more adequately define the values' conservatisms, are clarifying administrative changes only and do not involve changes to the actual values themselves or the manner in which they are implemented/used. Thus, the requested changes would not involve a significant increase in the probability of an accident previously evaluated or create the possibility of a new or different kind of accident from any accident previously evaluated since they can have no effect on accident causal mechanisms. Likewise, the changes would not involve a significant increase in the consequences of an accident or a significant reduction in a margin of safety since the actual values remain the same. Consequently, they clearly involve no significant hazards considerations as they do not impact the three standards referenced above.

The Commission has provided examples of amendments likely to involve no significant hazards considerations (48FR14870). One example of this type is (i), "A purely administrative change to Technical Specifications: For example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature". This example can be applied to these requested changes.

II/III. Technical Specifications Table 3.3-5, Items 2.e, 3.e, 4.e, and 6.b

The requested changes to add appropriate ESF response times addressing Containment Purge and Exhaust Isolation (for Containment Pressure-High, Pressurizer Pressure-Low-Low, and Steam Line Pressure-Low), and lower the required ESF response time addressing Feedwater Isolation (for Steam Generator Water Level--High-High), are conservative changes to ensure compliance with the plant's licensing basis safety analyses. The requested changes would not involve a significant increase in the probability of an accident previously evaluated or create the possibility of a new or different kind of accident from any accident previously evaluated since they only address previously evaluated accidents' aspects, and therefore involve no accident causal mechanisms not previously evaluated. Similarly, sir e the requested changes conform to the previously evaluated accident assumptions, they would not involve a significant increase in the consequences of an accident or a significant reduction in a margin of safety (they would ensure previously evaluated accidents' consequences and margins of safety were maintained - rather, the lack of these changes would possibly increase the consequences of accidents previously evaluated and reduce the margins of safety). As these changes are more restrictive/limiting than the current Technical Specifications, they clearly involve no significant hazards considerations as they can have no adverse effect on the three standards referenced above.

Another Commission provided example of actions not likely to involve a significant hazards consideration is (ii), "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: For example, a more stringent surveillance requirement". Since the requested changes are more restrictive/limiting, the above cited example can be applied to these amendments.

Conclusions:

Based on the preceding analyses, Duke Power Company concludes that the requested amendments do not involve a significant hazards consideration.