Joseph M. Farley Nuclear Plant - Units 1 and 2
Technical Specification Changes Associated with
Relocation of Reactor Trip and Engineered Safety Feature
Actuation System Response Time Limits

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months.*. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

* Neutron detectors are exempt from response time testing.

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3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature 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.

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 shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months.

4.3.2.3 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 logic train such that both logic 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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BASES

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

The OPERABILITY of the Reactor Trip and Engineered Safety Feature Actuation System instrumentation and interlocks ensure that 1) the associated Engineered Safety Feature Actuation 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 is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes 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.

The Engineered Safety Features Actuation System interlocks perform the functions indicated below on increasing the required parameter, consistent with the setpoints listed in Table 3.3-4:

P-11 Defeats the manual block of safety injection actuation on low pressurizer pressure.

P-12 Defeats the manual block of safety injection actuation on low steam line pressure.

P-14 Trip of all feedwater pumps, turbine trip, closure of feedwater isolation valves and inhibits feedwater control valve modulation.

On decreasing the required parameter the opposite function is performed at reset setpoints, with the exception of P-12 as noted below:

P-12 Allows manual block of safety injection actuation on low steam line pressure. Causes steam line isolation on high steam flow. Affects steam dump blocks (i.e., prevents premature block of the noted function).

The measurement of response time at the specified frequencies provides assurance that the reactor trip and ESF actuation associated with each channel is completed within the time limit assumed in the accident analyses. Response time limits for the Reactor Trip System and Engineered Safety Features Actuation System are maintained in Tables 7.2-5 and 7.3-16 of the Farley FSAR, respectively. No credit was taken in the analyses for those channels with response times indicated as not applicable.

REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

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.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

Alarm/trip setpoints for the containment purge have been established for a purge rate of 5,000 scfm in all MODES and for purge rates of 25,000 scfm and 50,000 scfm in MODES 4, 5, and 6. The containment purge setpoints are based on a release in which Xe-133 and Kr-85 are the predominant isotopes, on the 10 CFR 20 Appendix B, Table 2, MPC values for these isotopes and on a X/Q of 5.6×10^{-6} sec/m^3 at the site boundary.

The Alarm/trip setpoint for the fuel storage pool area has been established based on a flow rate of 13,000 scfm; a release in which Xe-133 and Kr-85 are the predominant isotopes, on the 10 CFR 20, Appendix B, Table 2, MPC values for these isotopes and on a X/Q of 5.6×10^{-6} sec/m^3 at the site boundary.

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 of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$, $F^N_{\Delta H}$, and F_{xy} a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system. Full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months.* Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

* Neutron detectors are exempt from response time testing.

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3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature 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.

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 shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months.

4.3.2.3 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 logic train such that both logic 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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3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip and Engineered Safety Feature Actuation System instrumentation and interlocks ensure that 1) the associated Engineered Safety Feature Actuation 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 is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes 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.

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P-12 Allows manual block of safety injection actuation on low steam line pressure. Causes steam line isolation on high steam flow. Affects steam dump blocks (i.e., prevents premature block of the noted function).

The measurement of response time at the specified frequencies provides assurance that the reactor trip and ESF actuation associated with each channel is completed within the time limit assumed in the accident analyses. Response time limits for the Reactor Trip System and Engineered Safety Features Actuation System are maintained in Tables 7.2-5 and 7.3-16 of the Farley FSAR, respectively. No credit was taken in the analyses for those channels with response times indicated as not applicable.

REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

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.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

Alarm/trip setpoints for the containment purge have been established for a purge rate of 5,000 scfm in all MODES and for purge rates of 25,000 scfm and 50,000 scfm in MODES 4, 5, and 6. The containment purge setpoints are based on a release in which Xe-133 and Kr-85 are the predominant isotopes, on the 10 CFR 20 Appendix B, Table 2, MPC values for these isotopes and on a X/Q of 5.6×10^{-6} sec/m^3 at the site boundary.

The Alarm/trip setpoint for the fuel storage pool area has been established based on a flow rate of 13,000 scfm; a release in which Xe-133 and Kr-85 are the predominant isotopes, on the 10 CFR 20, Appendix B, Table 2, MPC values for these isotopes and on a X/Q of 5.6×10^{-6} sec/m^3 at the site boundary.

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 of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$, $F^N_{\Delta H}$, and F_{xy} a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system. Full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

Enclosure 4

Hand-Marked Pages

Technical Specifications Changes Associated with
Relocation of Reactor Trip and Engineered Safety Feature
Actuation System Response Time Limits

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE. ~~with RESPONSE TIMES as shown in Table 3.3-2.~~

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months.* Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

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* NEUTRON DETECTORS ARE EXEMPT FROM RESPONSE TIME TESTING.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

DELETED

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Power Range, Neutron Flux	
a. High	≤ 0.5 seconds*
b. Low	Not Applicable
3. Power Range, Neutron Flux, High Positive Rate	Not Applicable
4. Power Range, Neutron Flux, High Negative Rate	Not Applicable
5. Intermediate Range, Neutron Flux	Not Applicable
6. Source Range, Neutron Flux	Not Applicable
7. Overtemperature ΔT	≤ 6.0 seconds*
8. Overpower ΔT	Not Applicable
9. Pressurizer Pressure--Low	≤ 2.0 seconds
10. Pressurizer Pressure--High	≤ 2.0 seconds
11. Pressurizer Water Level--High	Not Applicable

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* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. A. Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
13. Steam Generator Water Level--Low-Low	≤ 2.0 seconds
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Not Applicable
15. Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds
16. Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
17. Turbine Trip	
A. Low Auto Stop Oil Pressure	Not Applicable
B. Turbine Throttle Valve Closure	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable
20. Reactor Trip System Interlocks	Not Applicable
21. Reactor Trip Breakers	Not Applicable
22. Automatic Trip Logic	Not Applicable

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3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature 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 shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months.

4.3.2.3 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 logic train such that both logic 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 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Purge Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
Containment Air Coolers	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Purge Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Purge Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	$\leq 32.0^{(6)}$
d. Containment Isolation-Phase "A"	$\leq 17.0^{(4)}/27.0^{(5)}$
e. Containment Purge Isolation	≤ 5.0
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 77.0^{(4)}/87.0^{(5)}$
h. Containment Air Cooler Fan	≤ 27.4

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

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INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27.0(1)/12.0(4)
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 32.0(6)
d. Containment Isolation-Phase "A"	≤ 17.0(4)
e. Containment Purge Isolation	≤ 5.0
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	≤ 77.0(4)/87.0(1)
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤ 12.0(4)/22.0(5)
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 32.0(6)
d. Containment Isolation-Phase "A"	≤ 17.0(4)/27.0(5)
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	≤ 77.0(4)/87.0(5)
5. Steam Flow in Two Steam Lines-High Coincident with T_{avg}--Low-Low	
a. Steam Line Isolation	Not Applicable
6. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 12.0(4)/22.0(5)
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 32.0(6)
d. Containment Isolation-Phase "A"	≤ 17.0(4)/27.0(5)
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	≤ 77.0(4)/87.0(5)
h. Steam Line Isolation	≤ 7.0

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Containment Pressure--High-High	
a. Steam Line Isolation	≤ 7.0
8. Containment Pressure--High-High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
9. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 32.0(6)
10. Steam Generator Water Level -- Low-Low	
a. Motor-driven Aux. Feedwater Pumps(2)	≤ 60.0
b. Turbine-driven Aux. Feedwater Pump(3)	≤ 60.0
11. Undervoltage RCP	
a. Turbine-driven Aux. Feedwater Pump	≤ 60.0
12. S.I. Signal	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60.0
13. Trip of Main Feedwater Pump	
a. Motor-driven Aux. Feedwater Pumps	Not Applicable
14. Loss of Power	
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	(7)
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	(7)

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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 SI path and attainment of discharge pressure for centrifugal charging pumps, and RHR pumps.

(2) One 2/3 any Steam Generator

(3) On 2/3 in 2/3 Steam Generators

(4) Diesel generator starting and sequence loading delay not included. Offsite power available. Respone time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

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

(6) Verification shall include testing of all instrumentation, the isolation valves (MOV-3232A, 3232B, 3232C) and the control valves (FCV-478, 479, 488, 489, 498, 499). The isolation valves must function within 30 seconds and the control valves within 5 seconds.

(7) The respone time shall include the time delay associated with the undervoltage relays as determined in Table 3.3-4 plus an additional second associated with interposing relay and circuit operation.

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RESPONSE TIME LIMITS FOR THE REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM ARE MAINTAINED IN TABLES 7.2-5 AND 7.3-16 OF THE FARLEY FSAR, RESPECTIVELY.

REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the reactor trip and ESF 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.

3/4.3.3 MONITORING INSTRUMENTATION

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Alarm/trip setpoints for the containment purge have been established for a purge rate of 5,000 scfm in all MODES and for purge rates of 25,000 scfm and 50,000 scfm in MODES 4, 5, and 6. The containment purge setpoints are based on a release in which Xe-133 and Kr-85 are the predominent isotopes, on the 10 CFR 20, Appendix B, Table 2, MPC values for these isotopes and on a X/Q of 5.6×10^{-6} sec/m^3 at the site boundry.

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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 of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$, $F^N_{\Delta H}$, and F_{xy} a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system. Full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE. ~~with RESPONSE TIMES as shown in Table 3.3-2~~.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months.* Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

---

* NEUTRON DETECTORS ARE EXEMPT FROM RESPONSE TIME TESTING.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

DELETED

	FUNCTIONAL UNIT	RESPONSE TIME
1.	Manual Reactor Trip	Not Applicable
2.	Power Range, Neutron Flux	
	a. High	≤ 0.5 seconds*
	b. Low	Not Applicable
3.	Power Range, Neutron Flux, High Positive Rate	Not Applicable
4.	Power Range, Neutron Flux, High Negative Rate	Not Applicable
5.	Intermediate Range, Neutron Flux	Not Applicable
6.	Source Range, Neutron Flux	Not Applicable
7.	Overtemperature ΔT	≤ 6.0 seconds*
8.	Overpower ΔT	Not Applicable
9.	Pressurizer Pressure--Low	≤ 2.0 seconds
10.	Pressurizer Pressure--High	≤ 2.0 seconds
11.	Pressurizer Water Level--High	Not Applicable

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* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. A. Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
13. Steam Generator Water Level--Low-Low	≤ 2.0 seconds
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Not Applicable
15. Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds
16. Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
17. Turbine Trip	
A. Low Auto Stop Oil Pressure	Not Applicable
B. Turbine Throttle Valve Closure	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. Reactor Coolant Pump Breaker Position .rip	Not Applicable
20. Reactor Trip System Interlocks	Not Applicable
21. Reactor Trip Breakers	Not Applicable
22. Automatic Trip Logic	Not Applicable

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3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature 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 shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months.

4.3.2.3 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 logic train such that both logic 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 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

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INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Purge Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
Containment Air Coolers	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Purge Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Purge Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 27.0(1)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 32.0(6)
d. Containment Isolation-Phase "A"	$\leq$ 17.0(4)/27.0(5)
e. Containment Purge Isolation	$\leq$ 5.0
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 77.0(4)/87.0(5)
h. Containment Air Cooler Fan	$\leq$ 27.4

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/12.0$[illegible]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	$\leq 32.0^{(6)}$
d. Containment Isolation-Phase "A"	$\leq 17.0^{(4)}$
e. Containment Purge Isolation	≤ 5.0
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 77.0^{(4)}/87.0^{(1)}$
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 12.0^{(4)}/22.0^{(5)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	$\leq 32.0^{(6)}$
d. Containment Isolation-Phase "A"	$\leq 17.0^{(4)}/27.0^{(5)}$
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 77.0^{(4)}/87.0^{(5)}$
5. Steam Flow in Two Steam Lines-High Coincident with T_{avg}--Low-Low	
a. Steam Line Isolation	Not Applicable
6. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12.0^{(4)}/22.0^{(5)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	$\leq 32.0^{(6)}$
d. Containment Isolation-Phase "A"	$\leq 17.0^{(4)}/27.0^{(5)}$
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 77.0^{(4)}/87.0^{(5)}$
h. Steam Line Isolation	≤ 7.0

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Containment Pressure--High-High	
a. Steam Line Isolation	$\leq$ 7.0
8. Containment Pressure--High-High-High	
a. Containment Spray	$\leq$ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
9. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 32.0(6)
10. Steam Generator Water Level -- Low-Low	
a. Motor-driven Aux. Feedwater Pumps(2)	$\leq$ 60.0
b. Turbine-driven Aux. Feedwater Pump(3)	$\leq$ 60.0
11. Undervoltage RCP	
a. Turbine-driven Aux. Feedwater Pump	$\leq$ 60.0
12. S.I. Signal	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq$ 60.0
13. Trip of Main Feedwater Pumps	
a. Motor-driven Aux. Feedwater Pumps	Not Applicable
14. Loss of Power	
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	(7)
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	(7)

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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 SI path and attainment of discharge pressure for centrifugal charging pumps, and RHR pumps.

(2) On 2/3 any Steam Generator

(3) On 2/3 in 2/3 Steam Generators

(4) Diesel generator starting and sequence loading delay <u>not</u> included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifigal charging pumps.

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

(6) Verification shall include testing of all instrumentation, the isolation valves (MOV-3232A, 3232B, 3232C) and the control valves (FCV-478, 479, 488, 489, 498, 499). The isolation valves must function within 30 seconds and the control valves within 5 seconds.

(7) The response time shall include the time delay associated with the undervoltage relays as determined in Table 3.3-4 plus an additional second associated with interposing relay and circuit operation.

DELETED

RESPONSE TIME LIMITS FOR THE REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM ARE MAINTAINED IN TABLES 7.2-5 AND 7.3-16 OF THE FARLEY FSAR, RESPECTIVELY.

REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the reactor trip and ESF 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.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

Alarm/trip setpoints for the containment purge have been established for a purge rate of 5,000 scfm in all MODES and for purge rates of 25,000 scfm and 50,000 scfm in MODES 4, 5, and 6. The containment purge setpoints are based on a release in which Xe-133 and Kr-85 are the predominent isotopes, on the 10 CFR 20, Appendix B, Table 2, MPC values for these isotopes and on a X/Q of 5.6×10^{-6} sec/m^3 at the site boundry.

The Alarm/trip setpoint for the fuel storage pool area has been established based on a flow rate of 13,000 scfm; a release in which Xe-133 and Kr-85 are the predominent isotopes, on the 10 CFR 20, Appendix B, Table 2, MPC values for these isotopes and on a X/Q of 5.6×10^{-6} sec/m^3 at the site boundry.

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 of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$, $F^N_{\Delta H}$, and F_{xy} a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system. Full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

ENCLOSURE 5

Proposed FSAR Mark-ups Associated with
Relocation of Reactor Trip and Engineered Safety Feature
Actuation System Response Time Tables

7.2.3.1 Inservice Tests and Inspections

Periodic surveillance of the reactor trip system is performed to ensure proper protective action. This surveillance consists of checks, calibrations, ~~and~~ channel functional testing, AND RESPONSE TIME TESTING which are summarized as follows:

A. Checks

A check consists of a qualitative determination of acceptability by observation of channel behavior during operation. It includes comparison of the channel with other independent channels measuring the same variable. Failures such as blown instrument fuses, defective indicators, or faulted amplifiers are noticeable by simple observation of the functioning of the instrument or system. Furthermore, in many cases such failures are revealed by alarm or annunciator action, and a check supplements this type of surveillance.

B. Calibration

A channel calibration consists of adjustment of channel output such that it responds, within acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration encompasses the entire channel including alarm and/or trip; it also includes the channel functional testing discussed below. Thus, the calibration ensures the acquisition and presentation of accurate information.

C. Channel Functional Testing

A channel functional test consists of injecting a simulated signal into the signal conditioning portion of the channel to verify its operability, including alarm and/or trip initiating action.

D. INSERT 1

The minimum frequency for checks, calibration, CHANNEL FUNCTIONAL TESTING, and RESPONSE TIME testing are defined in the plant technical specifications.

7.2.3.2 Periodic Testing of the Nuclear Instrumentation System

The following periodic tests of the nuclear instrumentation system are performed:

INSERT 1

Response time testing consists of any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined in the Technical Specifications. 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 measurement of response time at the specified frequencies provides assurance that the reactor trip associated with each channel is completed within the time limit assumed in the accident analyses. The response time limits for the reactor trip system are provided in Table 7.2-5.

TABLE ~~3.3-2~~ 7.2-5 (SHEET 1 OF 2)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

	FUNCTIONAL UNIT	RESPONSE TIME
1.	Manual Reactor Trip	Not Applicable
2.	Power Range, Neutron Flux	
	a. High	≤ 0.5 seconds*
	b. Low	Not Applicable
3.	Power Range, Neutron Flux, High Positive Rate	Not Applicable
4.	Power Range, Neutron Flux, High Negative Rate	Not Applicable
5.	Intermediate Range, Neutron Flux	Not Applicable
6.	Source Range, Neutron Flux	Not Applicable
7.	Overtemperature ΔT	≤ 6.0 seconds*
8.	Overpower ΔT	Not Applicable
9.	Pressurizer Pressure--Low	≤ 2.0 seconds
10.	Pressurizer Pressure--High	≤ 2.0 seconds
11.	Pressurizer Water Level--High	Not Applicable

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

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TABLE ~~3.3-2 (Continued)~~ 7.2-5 (SHEET 2 OF 2)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. A. Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	$\leq$ 1.0 seconds
13. Steam Generator Water Level--Low-Low	$\leq$ 2.0 seconds
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Not Applicable
15. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.2 seconds
16. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.6 seconds
17. Turbine Trip	
A. Low Auto Stop Oil Pressure	Not Applicable
B. Turbine Throttle Valve Closure	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable
20. Reactor Trip System Interlocks	Not Applicable
21. Reactor Trip Breakers	Not Applicable
22. Automatic Trip Logic	Not Applicable

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plant condition changes, the test frequency is accelerated to accommodate the situation until the marginal performance is resolved.

D. The test interval discussed in paragraph 5.2 of IEEE 338-1971 is developed primarily on past operating experience and modified, if necessary, to ensure that system and subsystem protection is reliably provided. Analytical methods for determining reliability are not used to determine test interval.

7.3.2.6 Evaluation of Compliance with IEEE 344-1971

The seismic testing, as set forth in paragraph 7.2.1.10, IEEE 338-1971,[9] IEEE 344-1971,[10] WCAP-7706,[11] and WCAP-7705,[3] conforms to the guidelines set forth in IEEE 344-1971.[10]

7.3.2.7 RESPONSE TIME TESTING ← INSERT 2

7.3.2.~~7~~8 Further Considerations

In addition to the considerations given above, a loss of instrument air or loss of component cooling water to vital equipment has been considered. Assuming no other accident conditions, neither cause safety limits, as given in the technical specifications, to be exceeded. Likewise, loss of either one of the two will not adversely affect the core or the reactor coolant system, nor will it prevent an orderly shutdown if this is necessary. Furthermore, all pneumatically operated valves and controls will assume a preferred operating position upon loss of instrument air. It is also noted that, for conservatism during the accident analyses (chapter 15), credit is not taken for the instrument air systems nor any control system benefit.

7.3.2.~~8~~9 Summary

The effectiveness of the ESFAS is evaluated in chapter 15, based upon the ability of the system to contain the effects of Condition III and IV faults, including LOCAs and steam break accidents. The ESFAS parameters are based upon the component performance specifications which are given by the manufacturer or verified by test for each component. Appropriate factors to account for uncertainties in the data are factored into the constants characterizing the system.

The ESFAS must detect Condition III and IV faults and generate signals which actuate the engineered safety features. The system must sense the accident condition and generate the

INSERT 2

Response time testing consists of any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined in the Technical Specifications. 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 measurement of response time at the specified frequencies provides assurance that the reactor trip associated with each channel is completed within the time limit assumed in the accident analyses. The response time limits for the Engineered Safety Feature Actuation System are provided in Table 7.3-16.

TABLE ~~3.3-5~~ 7.3-16 (SHEET 1 of 4)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Purge Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
Containment Air Coolers	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Purge Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Purge Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 27.0(1)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 32.0(6)
d. Containment Isolation-Phase "A"	$\leq$ 17.0(4)/27.0(5)
e. Containment Purge Isolation	$\leq$ 5.0
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 77.0(4)/87.0(5)
h. Containment Air Cooler Fan	$\leq$ 27.4

TABLE ~~3.3-5 (Continued)~~ 7.3-16 (SHEET 2 OF 4) ~~- PAGE 8 OF 11 -~~

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/12.0^{(4)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	$\leq 32.0^{(6)}$
d. Containment Isolation-Phase "A"	$\leq 17.0^{(4)}$
e. Containment Purge Isolation	≤ 5.0
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 77.0^{(4)}/87.0^{(1)}$
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 12.0^{(4)}/22.0^{(5)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	$\leq 32.0^{(6)}$
d. Containment Isolation-Phase "A"	$\leq 17.0^{(4)}/27.0^{(5)}$
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 77.0^{(4)}/87.0^{(5)}$
5. Steam Flow in Two Steam Lines-High Coincident with T_{avg}--Low-Low	
a. Steam Line Isolation	Not Applicable
6. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12.0^{(4)}/22.0^{(5)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	$\leq 32.0^{(6)}$
d. Containment Isolation-Phase "A"	$\leq 17.0^{(4)}/27.0^{(5)}$
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 77.0^{(4)}/87.0^{(5)}$
h. Steam Line Isolation	≤ 7.0

TABLE ~~3.3-5 (Continued)~~ 7.3-16 (SHEET 3 OF 4) PAGE 9 OF 11

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Containment Pressure--High-High	
a. Steam Line Isolation	$\leq$ 7.0
8. Containment Pressure--High-High-High	
a. Containment Spray	$\leq$ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
9. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 32.0(6)
10. Steam Generator Water Level -- Low-Low	
a. Motor-driven Aux. Feedwater Pumps(2)	$\leq$ 60.0
b. Turbine-driven Aux. Feedwater Pump(3)	$\leq$ 60.0
11. Undervoltage RCP	
a. Turbine-driven Aux. Feedwater Pump	$\leq$ 60.0
12. S.I. Signal	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq$ 60.0
13. Trip of Main Feedwater Pumps	
a. Motor-driven Aux. Feedwater Pumps	Not Applicable
14. Loss of Power	
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	(7)
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	(7)

TABLE ~~3.3-5 (Continued)~~ 7.3-16 (SHEET 4 OF 4)

TABLE NOTATION

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

(2) One 2/3 any Steam Generator

(3) On 2/3 in 2/3 Steam Generators

(4) Diesel generator starting and sequence loading delay not included. Offsite power available. Respone time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

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

(6) Verification shall include testing of all instrumentation, the isolation valves (MOV-3232A, 3232B, 3232C) and the control valves (FCV-478, 479, 488, 489, 498, 499). The isolation valves must function within 30 seconds and the control valves within 5 seconds.

(7) The respone time shall include the time delay associated with the undervoltage relays as determined in Table 3.3-4 OF THE TECHNICAL SPECIFICATIONS plus an additional second associated with interposing relay and circuit operation.

A reactor trip signal acts to open two trip breakers connected in series which feed power to the control rod drive mechanisms (CRDMs). The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies (RCCAs) which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from when the monitored parameter reaches the trip setpoint until the rods are free and begin to fall. The setpoint study is performed in the course of finalizing the design of the plant; however, many of the accident analyses in this chapter conservatively do not take credit for the control systems.

Table 15.1-3 refers to the overtemperature and overpower ΔT trip shown in figure 15.1-1A.

These trip setpoints bound the transition cores and a full core of VANTAGE 5 fuel. The associated OTΔT f(ΔI) penalty is shown in figure 15.1-1B.

For all the reactor trips, the difference between the trip setpoints assumed in the analysis and the nominal trip setpoints account for instrumentation channel error and setpoint error. The plant technical specifications specify the nominal trip setpoints. RESPONSE TIME LIMITS FOR THE REACTOR TRIP SYSTEMS ARE MAINTAINED IN TABLE 7.2-5. The calibration of protection system channels and the periodic determination of instrument response times are in accordance with the plant technical specifications.

15.1.4 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS

The VANTAGE 5 fuel design features, the modified safety analysis assumptions, and the application of new methodologies (i.e., RTDP, WRB-1, and WRB-2) as discussed in section 4.4 (with respect to the changes associated with the instrument uncertainties for the NSSS control parameters of power, pressure, temperature, and flow) are covered in reference 2.

15.1.5 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTIC

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time from the start of insertion up to the dashpot entry or approximately 85 percent of the rod cluster travel. For accident analyses, it is conservatively assumed that the insertion time to dashpot entry is