TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.*
~~R~~	At least once per 18 months.*
R	At least once per 24 months.*
S/U	Prior to each reactor startup.
N/A	Not applicable.

*In these Technical Specifications, 6 months is defined to be 184 days, ~~and~~ 18 months is defined to be 550 days, and 24 months is defined to be 730 days.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the Reactor Protection System instrumentation channels and bypasses 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

4.3.1.1.1 Each Reactor Protection System instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The total bypass function shall be demonstrated OPERABLE at least once per REFUELING INTERVAL ~~once per 18 months~~ during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per REFUELING INTERVAL ~~18 months~~. Each test shall include at least one channel per function such that all channels are tested at least once every N times the REFUELING INTERVAL ~~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.

TABLE 3.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. High Flux	4	2	3	1, 2	2#, 10
3. RC High Temperature	4	2	3	1, 2	3#, 10
4. Flux - ΔFlux - Flow	4	2(a)(b)	3	1, 2	2#, 10
5. RC Low Pressure	4	2(a)	3	1, 2	3#, 10
6. RC High Pressure	4	2	3	1, 2	3#, 10
7. RC Pressure-Temperature	4	2(a)	3	1, 2	3#, 10
8. High Flux/Number of Reactor Coolant Pumps On	4	2(a)(b)	3	1, 2	3#, 10
9. Containment High Pressure	4	2	3	1, 2	3#, 10
10. Intermediate Range, Neutron Flux and Rate	2	N/A	2(c)	1, 2 and *	4
11. Source Range, Neutron Flux and Rate					
A. Startup	2	N/A	2	2## and *	5
B. Shutdown	2	N/A	1	3, 4 and 5	6
12. Control Rod Drive Trip Breakers	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#, 8#
13. Reactor Trip Module	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
14. Shutdown Bypass High Pressure	4	2	3	2**, 3** 4**, 5**	6#
15. SCR Relays	2	2	2	1, 2 and *	9#

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TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

**When Shutdown Bypass is actuated.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above 10^{-10} amps on both Intermediate Range channels.

(a) Trip may be manually bypassed when RCS pressure ≤ 1820 psig by actuating Shutdown Bypass provided that:

(1) The High Flux Trip Setpoint is ≤ 5% of RATED THERMAL POWER,

(2) The Shutdown Bypass High Pressure Trip Setpoint of ≤ 1820 psig is imposed, and

(3) The Shutdown Bypass is removed when RCS pressure > 1820 psig.

(b) Trip may be manually bypassed when Specification 3.10.3 is in effect.

(c) The minimum channels OPERABLE requirement may be reduced to one when Specification 3.10.1 or 3.10.2 is in effect.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by 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/or open the control rod drive trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided both of the following conditions are satisfied:

a. The inoperable channel is placed in the bypassed or tripped condition within one hour.

b. Either, THERMAL POWER is restricted to ≤ 75% of RATED THERMAL POWER and the High Flux Trip Setpoint is reduced to ≤ 85% of RATED THERMAL POWER within 4 hours or the QUADRANT POWER TILT is monitored at least once per 12 hours.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and POWER OPERATION may proceed provided the inoperable channel is placed in the bypassed or tripped condition within one hour.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. $\leq$ 5% of RATED THERMAL POWER restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

b. > 5% of RATED THERMAL POWER, POWER OPERATION may continue.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. $\leq 10^{-10}$ amps on the Intermediate Range (IR) instrumentation, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10^{-10} amps on the IR instrumentation.

b. $> 10^{-10}$ amps on the IR instrumentation, operation may continue.

ACTION 6 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 within one hour and at least once per 12 hours thereafter.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. Within 1 hour:

1. Place the inoperable channel in the tripped condition, or

2. Remove power supplied to the control rod trip device associated with the inoperative channel.

b. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1, and the inoperable channel above may be bypassed for up to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1.1. The inoperable channel above may not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With one of the Reactor Trip Breaker diverse trip features (undervoltage or shunt trip devices) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

ACTION 9 - With one or both channels of SCR Relays inoperable, restore the channels to OPERABLE status during the next COLD SHUTDOWN exceeding 24 hours.

ACTION 10 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, within one hour, place one inoperable channel in trip and the second inoperable channel in bypass, and restore one of the inoperable channels to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and open the reactor trip breakers.

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FOR INFORMATION ONLY

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIMES** (seconds)
1. Manual Reactor Trip	Not Applicable
2. High Flux*	$\leq$ 0.266
3. RC High Temperature	Not Applicable
4. Flux - Δ Flux - Flow* - Variable Flow	$\leq$ 1.77
- Constant Flow	$\leq$ 0.266
5. RC Low Pressure	$\leq$ 0.341
6. RC High Pressure	$\leq$ 0.341
7. RC Pressure - Temperature - Constant Temperature	Not Applicable
8. High Flux/Number of Reactor Coolant Pumps On*	$\leq$ 0.631***
9. Containment High Pressure	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.

** Including sensor (except as noted), RPS instrument delay and the breaker delay.

*** A 0.24 sec delay time has been assumed for pump monitor.

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. High Flux	S	D(2), and Q(6,9)	N.A.	1, 2
3. RC High Temperature	S	R	SA(9)	1, 2
4. Flux - ΔFlux - Flow	S(4)	M(3) and Q(6,7,9)	N.A.	1, 2
5. RC Low Pressure	S	R	SA(9)	1, 2
6. RC High Pressure	S	R	SA(9)	1, 2
7. RC Pressure-Temperature	S	R	SA(9)	1, 2
8. High Flux/Number of Reactor Coolant Pumps On	S	Q(6,9)	N.A.	1, 2
9. Containment High Pressure	S	E~~R~~	SA(9)	1, 2
10. Intermediate Range, Neutron Flux and Rate	S	E~~R~~(6)	N.A.(5)	1, 2 and *
11. Source Range, Neutron Flux and Rate	S	E~~R~~(6)	N.A.(5)	2, 3, 4 and 5
12. Control Rod Drive Trip Breakers	N.A.	N.A.	M(8,9) and S/U(1)(8)	1, 2 and *
13. Reactor Trip Module Logic	N.A.	N.A.	M(9)	1, 2 and *
14. Shutdown Bypass High Pressure	S	R	SA(9)	2**, 3**, 4**, 5**
15. SCR Relays	N.A.	N.A.	R	1, 2 and *

ADDITIONAL CHANGES PREVIOUSLY
PROPOSED BY LETTER
Serial No. 2335 Date 5/28/96

TABLE 4.3-1 (Continued)

Notation

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - When THERMAL POWER [TP] is above 50% of RATED THERMAL POWER [RTP], and at steady state, compare out-of-core measured AXIAL POWER IMBALANCE [API_o] to incore measured AXIAL POWER IMBALANCE [API_I] as follows:

$$\frac{RTP}{TP}[API_o - API_I] = \text{Offset Error}$$

Recalibrate if the absolute value of the Offset Error is ≥ 2.5%

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - CHANNEL FUNCTIONAL TEST is not applicable. Verify at least one decade overlap prior to each reactor startup if not verified in previous 7 days.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once per 18 months.

(8) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of both the undervoltage and shunt trip devices of the Reactor Trip Breakers.

(9) - Performed on a STAGGERED TEST BASIS.

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

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INSTRUMENTATION

3/4.3.2 SAFETY SYSTEM INSTRUMENTATION

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Safety Features Actuation System (SFAS) functional units 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 a SFAS functional unit trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the functional unit inoperable and apply the applicable ACTION requirement of Table 3.3-3, until the functional unit is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With a SFAS functional unit inoperable, take the action shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each SFAS functional unit shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of functional units affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per REFUELING INTERVAL ~~18 months~~ during CHANNEL CALIBRATION testing of each functional unit affected by bypass operation.

4.3.2.1.3 The SAFETY FEATURES RESPONSE TIME of each SFAS function shall be demonstrated to be within the limit at least once per REFUELING INTERVAL ~~18 months~~. Each test shall include at least one functional unit per function such that all functional units are tested at least once every N times the REFUELING INTERVAL ~~18 months~~ where N is the total number of redundant functional units in a specific SFAS function as shown in the "Total No. of Units" Column of Table 3.3-3.

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TABLE 3.3-3

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF UNITS	UNITS TO TRIP	MINIMUM UNITS OPERABLE	APPLICABLE MODES	ACTION
1. INSTRUMENT STRINGS					
a. Containment Radiation - High	4	2	3	1,2,3,4,6****	10#
b. Containment Pressure - High	4	2	3	1, 2, 3	10#
c. Containment Pressure - High-High	4	2	3	1, 2, 3	10#
d. RCS Pressure - Low	4	2	3	1, 2, 3*	10#
e. RCS Pressure - Low-Low	4	2	3	1, 2, 3**	10#
f. BWST Level - Low-Low	4	2	3	1, 2, 3	10#
2. OUTPUT LOGIC					
a. Incident Level #1: Containment Isolation	2	1	2	1,2,3,4,6****	11
b. Incident Level #2: High Pressure Injection and Starting Diesel Generators	2	1	2	1, 2, 3, 4	11
c. Incident Level #3: Low Pressure Injection	2	1	2	1, 2, 3, 4	11
d. Incident Level #4: Containment Spray	2	1	2	1, 2, 3, 4	11
e. Incident Level #5: Containment Sump Recirculation Permissive	2	1	2	1, 2, 3, 4	11

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TABLE 3.3-3 (Continued)

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF UNITS	UNITS TO TRIP	MINIMUM UNITS OPERABLE	APPLICABLE MODES	ACTION
3. MANUAL ACTUATION					
a. SFAS (except Containment Spray and Emergency Sump Recirculation)	2	2	2	1,2,3,4,6****	12
b. Containment Spray	2	2	2	1,2,3,4	12
4. SEQUENCE LOGIC CHANNELS					
a. Sequencer	4	2/BUS	2/BUS	1,2,3,4	15#
b. Essential Bus Feeder Breaker Trip (90%)	4*****	2/BUS	2/BUS	1,2,3,4	15#
c. Diesel Generator Start, Load shed on Essential Bus (59%)	4	2/BUS	2/BUS	1,2,3,4	15#
5. INTERLOCK CHANNELS					
a. Decay Heat Isolation Valve	1	1	1	1,2,3	13#
b. Pressurizer Heaters	2	2	2	3******	14

TABLE 3.3-3 (Continued)
TABLE NOTATION

* Trip function may be bypassed in this MODE with RCS pressure below 1800 psig. Bypass shall be automatically removed when RCS pressure exceeds 1800 psig.

** Trip function may be bypassed in this MODE with RCS pressure below 600 psig. Bypass shall be automatically removed when RCS pressure exceeds 600 psig.

*** DELETED

**** This instrumentation, or the containment purge and exhaust system noble gas monitor (with the containment purge and exhaust system in operation), must be OPERABLE during CORE ALTERATIONS or movement of irradiated fuel within containment to meet the requirements of Technical Specification 3.9.4. When using the containment purge and exhaust system noble gas monitor, SFAS is not required to be OPERABLE in MODE 6.

***** All functional units may be bypassed for up to one minute when starting each Reactor Coolant Pump or Circulating Water Pump.

****** When either Decay Heat Isolation Valve is open.

\# The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 10 - With the number of OPERABLE functional units one less than the Total Number of Units, STARTUP and/or POWER OPERATION may proceed provided both of the following conditions are satisfied:

a. The inoperable functional unit is placed in the tripped condition within one hour.

b. The Minimum Units OPERABLE requirement is met; however, one additional functional unit may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 11 - With any component in the Output Logic inoperable, trip the associated components within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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TABLE 3.3-3 (Continued)

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE Units one less than the Total Number of Units, restore the inoperable functional unit 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 13 - a. With less than the Minimum Units OPERABLE and reactor coolant pressure $\geq$ 438 psig, both Decay Heat Isolation Valves (DH11 and DH12) shall be verified closed.

b. With Less than the Minimum Units OPERABLE and reactor coolant pressure < 438 psig operation may continue; however, the functional unit shall be OPERABLE prior to increasing reactor coolant pressure above 438 psig.

ACTION 14 - With less than the Minimum Units OPERABLE and reactor coolant pressure < 438 psig, operation may continue; however, the functional unit shall be OPERABLE prior to increasing reactor coolant pressure above 438 psig, or the inoperable functional unit shall be placed in the tripped state.

ACTION 15 - a. With the number of OPERABLE units one less than the Minimum Units Operable per Bus, place the inoperable unit in the tripped condition within one hour. For functional unit 4.a the sequencer shall be placed in the tripped condition by physical removal of the sequencer module. The inoperable functional unit may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

b. With the number of OPERABLE units two less than the Minimum Units Operable per Bus, declare inoperable the Emergency Diesel Generator associated with the functional units not meeting the required minimum units OPERABLE and take the ACTION required of Specification 3.8.1.1.

TABLE 3.3-4

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
INSTRUMENT STRINGS		
a. Containment Radiation	< 4 x Background at RATED THERMAL POWER	< 4x Background at RATED THERMAL POWER#
b. Containment Pressure - High	≤ 18.4 psia	≤ 18.52 psia#
c. Containment Pressure - High-High	≤ 38.4 psia	≤ 38.52 psia#
d. RCS Pressure - Low	≥ 1620.75 psig	≥ 1615.75 psig#
e. RCS Pressure - Low-Low	≥ 420.75 psig	≥ 415.75 psig#
f. BWST Level	≥ 89.5 and ≤ 100.5 in. H_2O	≥ 88.3 and ≤ 101.7 in. H_2O #
SEQUENCE LOGIC CHANNELS		
a. Essential Bus Feeder Breaker Trip (90%)	≥ 3744 volts for ≤ 7.8 sec	≥ 3558 volts ≤ 7.8 sec
b. Diesel Generator Start, Load Shed on Essential Bus (59%)	≥ 2071 and ≤ 2450 volts for 0.5 ± 0.1 sec	≥ 2071 and ≤ 2450 volts for 0.5 ± 0.1 sec#
INTERLOCK CHANNELS		
a. Decay Heat Isolation Valve and Pressurizer Heater	< 438 psig	< 443 psig#*

Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

* Referenced to the centerline of DH11 and DH12

TABLE 3.3-5

SAFETY FEATURES SYSTEM REPONSE TIMES

THIS PAGE PROVIDED FOR INFORMATION ONLY

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Fans	
1. Emergency Vent Fan	NA
2. Containment Cooler Fan	NA
b. HV & AC Isolation Valves	
1. ECCS Room	NA
2. Emergency Ventilation	NA
3. Containment Air Sample	NA
4. Containment Purge	NA
5. Pentration Room Purge	NA
c. Control Room HV & AC Units	NA
d. High Pressure Injection	
1. High Pressure Injection Pumps	NA
2. High Pressure Injection Valves	NA
e. Component Cooling Water	
1. Component Cooling Water Pumps	NA
2. Component Cooling Aux. Equip. Inlet Valves	NA
3. Component Cooling to Air Compressor Valves	NA
f. Service Water System	
1. Service Water Pumps	NA
2. Service Water From Component Cooling Heat Exchanger Isolation Valves	NA
g. Containment Spray Isolation Valves	NA
h. Emergency Diesel Generator	NA
i. Containment Isolation Valves	
1. Vacuum Relief	NA
2. Normal Sump	NA
3. RCS Letdown Delay Coil Outlet	NA
4. RCS Letdown High Temperature	NA

TABLE 3.3-5 (Continued)

SAFETY FEATURES SYSTEM RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
i. Containment Isolation Valves (cont'd)	
5. Pressurizer Sample	NA
6. Service Water to Cooling Water	NA
7. Vent Header	NA
8. Drain Tank	NA
9. Core Flood Tank Vent	NA
10. Core Flood Tank Fill	NA
11. Steam Generator Sample	NA
12. Quench Tank	NA
13. Emergency Sump	NA
14. RCP Seal Return	NA
15. Air Systems	NA
16. N_2 System	NA
17. Quench Tank Sample	NA
18. RCP Seal Inlet	NA
19. Core Flood Tank Sample	NA
20. RCP Standpipe Demin Water Supply	NA
21. Containment H_2 Dilution Inlet	NA
22. Containment H_2 Dilution Outlet	NA
j. BWST Outlet Valves	NA
k. Low Pressure Injection	
1. Decay Heat Pumps	NA
2. Low Pressure Injection Valves	NA
3. Decay Heat Pump Suction Valves	NA
4. Decay Heat Cooler Outlet Valves	NA
5. Decay Heat Cooler Bypass Valves	NA
l. Containment Spray Pump	NA
m. Component Cooling Isolation Valves	
1. Inlet to Containment	NA
2. Outlet from Containment	NA
3. Inlet to CRDM's	NA
4. CRDM Booster Pump Suction	NA
5. Component Cooling from Decay Heat Coolers	NA

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TABLE 3.3-5 (Continued)

SAFETY FEATURES SYSTEM RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Containment Pressure - High	
a. Fans	
1. Emergency Vent Fans	≤ 25*
2. Containment Cooler Fans	≤ 45*
b. HV & AC Isolation Valves	
1. ECCS Room	≤ 75*
2. Emergency Ventilation	≤ 75*
3. Containment Air Sample	≤ 30*
4. Containment Purge	≤ 15*
5. Penetration Room Purge	≤ 75*
c. Control Room HV & AC Units	≤ 10*
d. High Pressure Injection	
1. High Pressure Injection Pumps	≤ 30*
2. High Pressure Injection Valves	≤ 30*
e. Component Cooling Water	
1. Component Cooling Water Pumps	≤ 180*
2. Component Cooling Aux. Equip. Inlet Valves	≤ 180*
3. Component Cooling to Air Compressor Valves	≤ 180*
f. Service Water System	
1. Service Water Pumps	≤ 45*
2. Service Water From Component Cooling Heat Exchanger Isolation Valves	≤ NA*
g. Containment Spray Isolation Valves	≤ 80*
h. Emergency Diesel Generator	≤ 15*

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TABLE 3.3-5 (Continued)

SAFETY FEATURES SYSTEM RESPONSE TIMES

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INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Containment Pressure - High (Continued)	
i. Containment Isolation Valves	
1. Vacuum Relief	≤ 30*
2. Normal Sump	≤ 25*
3. RCS Letdown Delay Coil Outlet	≤ 30*
4. RCS Letdown High Temperature	≤ 30*
5. Pressurizer Sample	≤ 48*
6. Service Water to Cooling Water	≤ 45*
7. Vent Header	≤ 15*
8. Drain Tank	≤ 15*
9. Core Flood Tank Vent	≤ 15*
10. Core Flood Tank Fill	≤ 15*
11. Steam Generator Sample	≤ 15*
12. Quench Tank	≤ 15*
13. Emergency Sump	NA*
14. RCP Seal Return	≤ 45*
15. Air System	≤ 15*
16. N_2 System	≤ 15*
17. Quench Tank Sample	≤ 35*
18. RCP Seal Inlet	≤ 17*
19. Core Flood Tank Sample	≤ 15*
20. RCP Standpipe Demin Water Supply	≤ 15*
21. Containment H_2 Dilution Inlet	≤ 75*
22. Containment H_2 Dilution Outlet	≤ 75*
j. BWST Outlet Valves	NA*
k. Low Pressure Injection	
1. Decay Heat Pumps	≤ 30*
2. Low Pressure Injection Valves	≤ NA*
3. Decay Heat Pump Suction Valves	≤ NA
4. Decay Heat Cooler Outlet Valves	≤ NA*
5. Decay Heat Cooler Bypass Valves	≤ NA*
3. Containment Pressure--High-High	
a. Containment Spray Pump	≤ 80*
b. Component Cooling Isolation Valves	
1. Inlet to Containment	≤ 25*
2. Outlet from Containment	≤ 25*

TABLE 3.3-5 (Continued)

SAFETY FEATURES SYSTEM RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
b. Component Cooling Isolation Valves (Continued)	
3. Inlet to CRDM's	≤ 35*
4. CRDM Booster Pump Suction	≤ 35*
5. Component Cooling from Decay Heat Cooler	NA
4. RCS Pressure-Low	
a. Fans	
1. Emergency Vent Fans	≤ 25*
2. Containment Cooler Fans	≤ 45*
b. HV & AC Isolation Valves	
1. ECCS Room	≤ 75*
2. Emergency Ventilation	≤ 75*
3. Containment Air Sample	≤ 30*
4. Containment Purge	≤ 15*
5. Penetration Room Purge	≤ 75*
c. Control Room HV & AC Units	≤ 10*
d. High Pressure Injection	
1. High Pressure Injection Pumps	≤ 30*
2. High Pressure Injection Valves	≤ 30*
e. Component Cooling Water	
1. Component Cooling Water Pumps	≤ 180*
2. Component Cooling Aux. Equipment Inlet Valves	≤ 180*
3. Component Cooling to Air Compressor Valves	≤ 180*
f. Service Water System	
1. Service Water Pumps	≤ 45*
2. Service Water from Component Cooling Heat Exchanger Isolation Valves	≤ NA*
g. Containment Spray Isolation Valves	≤ 80*
h. Emergency Diesel Generator	≤ 15*

THIS PAGE PROVIDED FOR INFORMATION ONLY

TABLE 3.3-5 (Continued)

SAFETY FEATURES SYSTEM RESPONSE TIMES

THIS PAGE PROVIDED FOR INFORMATION ONLY

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. RCS Pressure-Low (continued)	
i. Containment Isolation Valves	
1. Vacuum Relief	≤ 30*
2. Normal Sump	≤ 25*
3. RCS Letdown Delay Coil Outlet	≤ 30*
4. RCS Letdown High Temperature	≤ 30*
5. Pressurizer Sample	≤ 45*
6. Service Water to Cooling Water	≤ 45*
7. Vent Header	≤ 15*
8. Drain Tank	≤ 15*
9. Core Flood Tank Vent	≤ 15*
10. Core Flood Tank Fill	≤ 15*
11. Steam Generator Sample	≤ 15*
12. Quench Tank	≤ 15*
13. Emergency Sump	NA*
14. Air Systems	≤ 15*
15. N_2 System	≤ 15*
16. Quench Tank Sample	≤ 35*
17. Core Flood Tank Sample	≤ 15*
18. RCP Standpipe Demin Water Supply	≤ 15*
19. Containment H_2 Dilution Inlet	≤ 75*
20. Containment H_2 Dilution Outlet	≤ 75*
j. BWST Outlet Valves	NA*
5. RCS Pressure--Low-Low	
a. Low Pressure Injection	
1. Decay Heat Pumps	≤ 30*
2. Low Pressure Injection Valves	≤ NA*
3. Decay Heat Pump Suction Valves	≤ NA*
4. Decay Heat Cooler Outlet Valves	≤ NA*
5. Decay Heat Cooler Bypass Valves	≤ NA*
b. Component Cooling Isolation Valves	
1. Auxiliary Equipment Inlet	≤ 90*
2. Inlet to Air Compressor	≤ 90*
3. Component Cooling from Decay Heat Cooler	≤ NA*
c. Containment Isolation Valves	
1. RCP Seal Return	≤ 45*
2. RCP Seal Inlet	≤ 17*

TABLE 3.3-5 (Continued)

SAFETY FEATURES SYSTEM RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Containment Radiation - High	
a. Emergency Vent Fans	≤ 25*
b. HV & AC Isolation Valves	
1. ECCS Room	≤ 75*
2. Emergency Ventilation	≤ 75*
3. Containment Air Sample	≤ 30*
4. Containment Purge	≤ 15*
5. Penetration Room Purge	≤ 75*
c. Control Room HV & AC Units	≤ 10*

TABLE NOTATION

* Diesel generator starting and sequence loading delays included when applicable. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

THIS PAGE PROVIDED FOR INFORMATION ONLY

TABLE 4.3-2

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. INSTRUMENT STRINGS				
a. Containment Radiation - High	S	~~R~~E	M	1, 2, 3, 4, 6 #
b. Containment Pressure - High	S	~~R~~E	M(2)	1, 2, 3
c. Containment Pressure - High-High	S	~~R~~E	M(2)	1, 2, 3
d. RCS Pressure - Low	S	R	M	1, 2, 3
e. RCS Pressure - Low-Low	S	R	M	1, 2, 3
f. BWST Level - Low-Low	S	~~R~~E	M	1, 2, 3
2. OUTPUT LOGIC				
a. Incident Level #1: Containment Isolation	S	~~R~~E	M	1, 2, 3, 4, 6 #
b. Incident Level #2: High Pressure Injection and Starting Diesel Generators	S	~~R~~E	M	1, 2, 3, 4
c. Incident Level #3: Low Pressure Injection	S	~~R~~E	M	1, 2, 3, 4
d. Incident Level #4: Containment Spray	S	~~R~~E	M	1, 2, 3, 4
e. Incident Level #5: Containment Sump Recirculation Permissive	S	~~R~~E	M	1, 2, 3, 4
3. MANUAL ACTUATION				
a. SFAS (Except Containment Spray and Emergency Sump Recirculation)	NA	NA	M(1)	1, 2, 3, 4, 6 #
b. Containment Spray	NA	NA	M(1)	1, 2, 3
4. SEQUENCE LOGIC CHANNELS	S	NA	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. INTERLOCK CHANNELS				
a. Decay Heat Isolation Valve	S	R	**	1, 2, 3
b. Pressurizer Heater	S	R	**	3 ##

**See Specification 4.5.2.d.1

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per REFUELING INTERVAL ~~18 months during shutdown~~. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

(2) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either vacuum or pressure to the appropriate side of the transmitter.

\# These surveillance requirements in conjunction with those of Section 4.9.4 apply during CORE ALTERATIONS or movement of irradiated fuel within the containment only if using the SFAS area radiation monitors listed in Table 3.3-3, Items 1a, 2a, and 3a, in lieu of the containment purge and exhaust system noble gas monitor.

\## When either Decay Heat Isolation Valve is open.

INSTRUMENTATION

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.2 The Steam and Feedwater Rupture Control System (SFRCS) instrumentation channels shown in Table 3.3-11 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-12 and with RESPONSE TIMES as shown in Table 3.3-13.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With a SFRCS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-12, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-11, until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With a SFRCS instrumentation channel inoperable, take the action shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.2.2.1 Each SFRCS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST during the MODES and at the frequencies shown in Table 4.3-11.

4.3.2.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per REFUELING INTERVAL ~~18 months~~ during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.2.3 The STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM RESPONSE TIME of each SFRCS function shall be demonstrated to be within the limit at least once per REFUELING INTERVAL ~~18 months~~. Each test shall include at least one channel per function such that all channels are tested at least once every N times the REFUELING INTERVAL ~~18 months~~ where N is the total number of redundant channels in a specific SFRCS function as shown in the "Total No. of Channels" Column of Table 3.3-11.

TABLE 3.3-11

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ACTION
1. Main Steam Pressure Low Instrument Channels*	2	1	2	16#
a. PS 3689B Steam Line 1 Channel 1				
b. PS 3689D Steam Line 2 Channel 1				
c. PS 3689F Steam Line 1 Channel 1				
d. PS 3689H Steam Line 2 Channel 1				
e. PS 3687A Steam Line 2 Channel 2				
f. PS 3687C Steam Line 1 Channel 2				
g. PS 3687E Steam Line 2 Channel 2				
h. PS 3687G Steam Line 1 Channel 2				

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FOR INFORMATION ONLY

TABLE 3.3-11 (Continued)

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ACTION
2. Feedwater/Steam Generator Differential Pressure - High Instrument Channels	2	1	2	16#
a. PDS 2685A Feedwater/Steam Generator 2 Channel 2 PDS 2685B Feedwater/Steam Generator 2 Channel 2				
b. PDS 2685C Feedwater/Steam Generator 2 Channel 1 PDS 2685D Feedwater/Steam Generator 2 Channel 1				
c. PDS 2686A Feedwater/Steam Generator 1 Channel 1 PDS 2686B Feedwater/Steam Generator 1 Channel 1				
d. PDS 2686C Feedwater/Steam Generator 1 Channel 2 PDS 2686D Feedwater/Steam Generator 1 Channel 2				
3. Steam Generator Level - Low Instrument Channels	2	1	2	16#
a. LSLL SP9B8 Steam Generator 1 Channel 1 LSLL SP9B9 Steam Generator 1 Channel 1				
b. LSLL SP9A6 Steam Generator 2 Channel 1 LSLL SP9A7 Steam Generator 2 Channel 1				
c. LSLL SP9A8 Steam Generator 2 Channel 2 LSLL SP9A9 Steam Generator 2 Channel 2				

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TABLE 3.3-11 (Continued)

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ACTION
3. Steam Generator Level - Low Instrument Channels (continued)				
d. LSLL SP9B6 Steam Generator 1 Channel 2 LSLL SP9B7 Steam Generator 1 Channel 2				
4. Loss of RCP Channels	2	1	2	16#
5. Manual Initiation (Push buttons)				
a. Initiate AFPT #1	1	1	1	17
b. Initiate AFPT #2	1	1	1	17
c. Initiate AFPT #1 and Isolate SG #1	1	1	1	17
d. Initiate AFPT #2 and Isolate SG #2	1	1	1	17

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FOR INFORMATION ONLY

TABLE 3.3-11 (Continued)

TABLE NOTATION

* May be bypassed when steam pressure is below 750 psig. Bypass shall be automatically removed when the steam pressure exceeds 800 psig.

The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable section of the channel is placed in the tripped condition within 1 hour.

ACTION 17 - 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 the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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FOR INFORMATION ONLY

TABLE 3.3-12

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINTS	ALLOWABLE VALUES
1. Steam Line Pressure - Low	≥ 591.6 psig	≥ 591.6 psig* ≥ 586.6 psig**
2. Steam Generator Level - Low(1)	≥ 16.4"	≥ 15.6"* ≥ 12.9"**
3. Steam Generator Feedwater Differential Pressure - High (2)	≤ 197.6 psid	≤ 197.6 psid* ≤ 199.6 psid**
4. Reactor Coolant Pumps - Loss of	High ≤ 1384.6 amps Low ≥ 106.5 amps	≤ 1384.6 amps# ≥ 106.5 amps#

(1) Actual water level above the lower steam generator tubesheet.
(2) Where differential pressure is steam generator minus feedwater pressure.
*Allowable Value for CHANNEL FUNCTIONAL TEST
**Allowable Value for CHANNEL CALIBRATION
#Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION

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FOR INFORMATION ONLY

TABLE 3.3-13

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM RESPONSE TIMES

ACTUATED EQUIPMENT	RESPONSE TIME IN SECONDS
1. Auxiliary Feed Pump	≤ 40
2. Main Steam Isolation Valves*	
a. Main Steam Low Pressure Channels	≤ 6
b. Feedwater/Steam Generator High Differential Pressure Channels	≤ 6.5
3. Main Feedwater Valves	
a. Main Control	≤ 8
b. Startup Control	≤ 13
c. Stop Valve	≤ 16
4. Turbine Stop Valves**	≤ 1

* The response time is to be the time elapsed from the monitored variable exceeding the trip setpoint until the MSIV is fully closed.

** The response time is to be the time elapsed from the main steam line low pressure trip condition until the TSV is fully closed.

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TABLE 4.3-11

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Instrument Channel			
a. Steam Line Pressure - Low	S	RE	M
b. Steam Generator Level - Low	S	R	M
c. Steam Generator - Feedwater Differential Pressure - High	S	RE	M
d. Reactor Coolant Pumps - Loss of	S	RE	M
2. Manual Actuation	NA	NA	R

3/4.3.3 MONITORING INSTRUMENTATION

THIS PAGE PROVIDED FOR INFORMATION ONLY

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area Emergency Ventilation System Actuation	1	**	≤ 2 x background	0.1 - 10^7 mr/hr	22
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3, & 4	Not Applicable	10 - 10^6 cpm	21
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3, & 4	Not Applicable	10 - 10^6 cpm	21

THIS PAGE PROVIDED FOR INFORMATION ONLY

**With fuel in the storage pool or building

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 21 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

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FOR INFORMATION ONLY

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area Emergency Ventilation System Actuation	S	ER	M	**
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection	S	ER	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection	S	ER	M	1, 2, 3 & 4

**With fuel in the storage pool or building

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5.1 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

3.3.3.5.2 The control circuits and transfer switches required for a serious control room or cable spreading room fire shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With one or more control circuits or transfer switches required for a serious control room or cable spreading room fire inoperable, restore the inoperable circuit(s) or switch(es) to OPERABLE status within 30 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the circuit(s) or switch(es) to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

4.3.3.5.2 At least once per REFUELING INTERVAL ~~18 months~~, verify each control circuit and transfer switch required for a serious control room or cable spreading room fire is capable of performing the intended function.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	(a) 480v F&DC CH. 2 Switchgear Room	OPEN-CLOSE	(a) 1 (Trip Breaker A)
	(b) 480v E&DC CH. 1 Switchgear Room		(b) 1 (Trip Breaker B)
	(c) 480v F&DC CH. 2 Switchgear Room		(c) 1 (Trip Breaker C)
	(d) CRDC Cabinet Room		(d) 1 (Trip Breaker D)
2. Reactor Coolant Temperature - Hot Leg	Aux. Shutdown Panel	520-620 °F	1
3. Reactor Coolant System Pressure	Aux. Shutdown Panel	0-3000 psig	1
4. Pressurizer Level	Aux. Shutdown Panel	0-320 inches	1
5. Steam Generator Outlet Steam Pressure	Aux. Shutdown Panel	0-1200 psig	1/steam generator
6. Steam Generator Level Startup Range	Aux. Shutdown Panel	0-250 inches	1/steam generator
7. Control Rod Position Switches	Control Rod Drive Control Cabinets, System Logic Cabinet #4	0, 25, 50, 75 and 100%	1/rod

THIS PAGE PROVIDED
FOR INFORMATION ONLY

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	M	N.A.
2. Reactor Coolant Temperature-Hot Legs	M	R
3. Reactor Coolant System Pressure	M	R
4. Pressurizer Level	M	R
5. Steam Generator Outlet Steam Pressure	M	R
6. Steam Generator Startup Range Level	M	R
7. Control Rod Position Switches	M	N.A.

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INSTRUMENTATION

POST-ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring instrumentation channels less than the Minimum Channels OPERABLE required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-10.

THIS PAGE PROVIDED
FOR INFORMATION ONLY

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. SG Outlet Steam Pressure	1/Steam Generator
2. RC Loop Outlet Temperature	2/Loop
3. RC Loop Pressure	2/Loop
4. Pressurizer Level	2
5. SG Startup Range Level	2/Steam Generator
6. Containment Vessel Post-Accident Radiation	2
7. High Pressure Injection Flow	1/Channel
8. Low Pressure Injection (DHR) Flow	1/Channel
9. Auxiliary Feedwater Flow Rate	2/Steam Generator
10. RC System Subcooling Margin Monitor	1
11. PORV Position Indicator	1
12. PORV Block Valve Position Indicator	1
13. Pressurizer Safety Valve Position Indicator	1/Valve
14. BWST Level	3
15. Containment Normal Sump Level	1
16. Containment Wide Range Water Level	1

THIS PAGE PROVIDED FOR INFORMATION ONLY

TABLE 3.3-10 (Continued)

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
17. Containment Wide Range Pressure	1
18. Incore Thermocouples	2 per core quadrant
19. Reactor Coolant Hot Leg Level (Wide Range)	1
20. Neutron Flux (Wide Range)	1
21. Neutron Flux (Source Range)	1

THIS PAGE PROVIDED FOR INFORMATION ONLY

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. SG Outlet Steam Pressure	M	R
2. RC Loop Outlet Temperature	M	R
3. RC Loop Pressure	M	R
4. Pressurizer Level	M	R
5. SG Startup Range Level	M	R
6. Containment Vessel Post-Accident Radiation	M	R
7. High Pressure Injection Flow	M	E~~R~~
8. Low Pressure Injection (DHR) Flow	M	E~~R~~
9. Auxiliary Feedwater Flow Rate	M	E~~R~~
10. RC System Subcooling Margin Monitor	M	R
11. PORV Position Indicator	M	R
12. PORV Block Valve Position Indicator	M	R
13. Pressurizer Safety Valve Position Indicator	M	R
14. BWST Level	S	E~~R~~
15. Containment Normal Sump Level	M	R
16. Containment Wide Range Water Level	M	R

TABLE 4.3-10 (Continued)

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
17.	Containment Wide Range Pressure	M	R
18.	Incore Thermocouples	M	ER
19.	Reactor Coolant Hot Leg Level (Wide Range)	M	R
20.	Neutron Flux (Wide Range)	M	ER**
21.	Neutron Flux (Source Range)	M	ER**

**Neutron detectors may be excluded from CHANNEL CALIBRATION.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

THIS PAGE PROVIDED FOR INFORMATION ONLY

CORE FLOODING TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system core flooding tank (CFT) shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume between 7555 and 8004 gallons of borated water,

c. $\geq$ 2600 and $\leq$ 3500 ppm of boron, and

d. A nitrogen cover-pressure of between 575 and 625 psig.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one CFT inoperable because of boron concentration not within limits, restore the inoperable CFT to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and reduce the RCS pressure to less than 800 psig within the following 12 hours.

b. With any CFT inoperable for reasons other than boron concentration not within limits, restore the CFT to OPERABLE status within one hour or be in HOT STANDBY within the next 6 hours and reduce the RCS pressure to less than 800 psig within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each core flooding tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each tank isolation valve is open.

*With Reactor Coolant pressure > 800 psig.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days, and within 6 hours of each solution volume increase of ≥ 80 gallons that is not the result of addition from the borated water storage tank (BWST), by verifying the boron concentration of the CFT solution.

c. At least once per 31 days by verifying that power to the isolation valve operator is disconnected by locking the breakers in the open position.

d. At least once per REFUELING INTERVAL ~~18 months~~ by verifying that each core flooding tank isolation valve opens automatically and is interlocked against closing whenever the Reactor Coolant System pressure exceeds 800 psig.

THIS PAGE PROVIDED FOR INFORMATION ONLY

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - T_{avg} $\geq$ 280°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure injection (HPI) pump,

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

Revised by NRC Letter Dated
June 6, 1995

SURVEILLANCE REQUIREMENTS (continued)

b. At least once per 18 months, or prior to operation after ECCS piping has been drained by verifying that the ECCS piping is full of water by venting the ECCS pump casings and discharge piping high points.**

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment emergency sump and cause restriction of the pump suction during LOCA conditions. This visual inspection shall be performed.

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. For all areas of containment affected by an entry, at least once daily while work is ongoing and again during the final exit after completion of work (containment closeout) when CONTAINMENT INTEGRITY is established.

d. At least once per REFUELING INTERVAL ~~18 months~~ by:

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 2383 Date 9/17/96

1. Verifying that the interlocks:

a) Close DH-11 and DH-12 and deenergize the pressurizer heaters, if either DH-11 or DH-12 is open and a simulated reactor coolant system pressure which is greater than the trip setpoint (<438 psig) is applied. The interlock to close DH-11 and/or DH-12 is not required if the valve is closed and 480 V AC power is disconnected from its motor operators.

b) Prevent the opening of DH-11 and DH-12 when a simulated or actual reactor coolant system pressure which is greater than the trip setpoint (<438 psig) is applied.

2. a) A visual inspection of the containment emergency sump which verifies that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

b) Verifying that on a Borated Water Storage Tank (BWST) Low-Low Level interlock trip, with the motor operators for the BWST outlet isolation valves and the containment emergency sump recirculation valves energized, the BWST Outlet Valve HV-DH7A (HV-DH7B) automatically close in $\leq$75 seconds after the operator manually pushes the control switch to open the Containment Emergency Sump Valve HV-DH9A (HV-DH9B) which should be verified to open in $\leq$75 seconds.

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 2382 Date 9/12/96

3. Deleted

** The requirements of this Surveillance may be deferred until the Tenth Refueling Outage for the ECCS flowpath which does not have manual high point venting capability.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that a minimum of 290 cubic feet of trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

5. Deleted

6. Deleted

e. At least once per 18 months, during shutdown, by

1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

2. Verifying that each HPI and LPI pump starts automatically upon receipt of a SFAS test signal.

f. By performing a vacuum leakage rate test of the watertight enclosure for valves DH-11 and DH-12 that assures the motor operators on valves DH-11 and DH-12 will not be flooded for at least 7 days following a LOCA:

1. At least once per 18 months.

2. After each opening of the watertight enclosure.

3. After any maintenance on or modification to the watertight enclosure which could affect its integrity.

g. By verifying the correct position of each mechanical position stop for valves DH-14A and DH-14B.

1. Within 4 hours following completion of the opening of the valves to their mechanical position stop or following completion of maintenance on the valve when the LPI system is required to be OPERABLE.

2. At least once per 18 months.

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h. By performing a flow balance test, during shutdown, following completion of modifications to the HPI or LPI subsystems that alter the subsystem flow characteristics and verifying the following flow rates:

HPI System - Single Pump

Injection Leg 1-1 $\geq$ 375 gpm at 400 psig*
Injection Leg 1-2 $\geq$ 375 gpm at 400 psig*

Injection Leg 2-1 $\geq$ 375 gpm at 400 psig*
Injection Leg 2-2 $\geq$ 375 gpm at 400 psig*

LPI System - Single Pump

Injection Leg 1 $\geq$ 2650 gpm at 100 psig**
Injection Leg 2 $\geq$ 2650 gpm at 100 psig**

---

* Reactor coolant pressure at the HPI nozzle in the reactor coolant pump discharge.

** Reactor coolant pressure at the core flood nozzle on the reactor vessel.

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3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION

The OPERABILITY of the RPS, SFAS and SFRCS instrumentation systems ensure that 1) the associated action and/or trip will be initiated when the parameter monitored by each channel or combination thereof exceeds 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 RPS, SFAS and SFRCS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance 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 measurement of response time at the specified frequencies provides assurance that the RPS, SFAS, and SFRCS action function associated with each channel is completed within the time limit assumed in the safety 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 actuation logic for Functional Units 4.a., 4.b., and 4.c. of Table 3.3-3, Safety Features Actuation System Instrumentation, is designed to provide protection and actuation of a single train of safety features equipment, essential bus or emergency diesel generator. Collectively, Functional Units 4.a., 4.b., and 4.c. function to detect a degraded voltage condition on either of the two 4160 volt essential buses, shed connected loads, disconnect the affected bus(es) from the offsite power source and start the associated emergency diesel generator. In addition, if an SFAS actuation signal is present under these conditions, the sequencer channels for the two SFAS channels which actuate the train of safety features equipment powered by the affected bus will automatically sequence these loads onto the bus to prevent overloading of the emergency diesel generator. Functional Unit 4.a. has a

(Next page is B 3/4 3-1a)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION (Continued)

total of four units, one associated with each SFAS channel (i.e., two for each essential bus). Functional Units 4.b. and 4.c. each have a total of four units, (two associated with each essential bus); each unit consisting of two undervoltage relays and an auxiliary relay.

An SFRCS channel consists of 1) the sensing device(s), 2) associated logic and output relays (including Isolation of Main Feedwater Non Essential Valves and Turbine Trip), and 3) power sources.

The SFRCS response time for the turbine stop valve closure is based on the combined response times of main steam line low pressure sensors, logic cabinet delay for main steam line low pressure signals and closure time of the turbine stop valves. This SFRCS response time ensures that the auxiliary feedwater to the unaffected steam generator will not be isolated due to a SFRCS low pressure trip during a main steam line break accident.

Safety-grade anticipatory reactor trip is initiated by a turbine trip (above 45 percent of RATED THERMAL POWER) or trip of both main feedwater pump turbines. This anticipatory trip will operate in advance of the reactor coolant system high pressure reactor trip to reduce the peak reactor coolant system pressure and thus reduce challenges to the pilot operated relief valve. This anticipatory reactor trip system was installed to satisfy Item II.K.2.10 of NUREG-0737. The justification for the ARTS turbine trip arming level of 45% is given in BAW-1893, October, 1985.

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3/4.3 INSTRUMENTATION

BASES

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.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Bases Figures 3-1 and 3-2 for examples of acceptable minimum incore detector arrangements.

3/4.3.3.3 SEISMIC INSTRUMENTATION

Deleted

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

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3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of

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3/4.3 INSTRUMENTATION

BASES

REMOTE SHUTDOWN INSTRUMENTATION (Continued)

HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost.

SR 4.3.3.5.2 verifies that each Remote Shutdown System transfer switch and control circuit required for a serious control room or cable spreading room fire performs its intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. This will ensure that if the control room becomes inaccessible, the unit can be safely shutdown from the remote shutdown panel and the local control stations.

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The containment Hydrogen Analyzers, although they are considered part of the plant post-accident monitoring instrumentation, have their OPERABILITY requirements located in Specification 3/4.6.4.1, Hydrogen Analyzers.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS - Deleted

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION - Deleted

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3/4.5.1 CORE FLOODING TANKS

The OPERABILITY of each core flooding tank ensures that a sufficient volume of borated water will be immediately forced into the reactor vessel in the event the RCS pressure falls below the pressure of the tanks. This initial surge of water into the vessel provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on volume, boron concentration and pressure ensure that the assumptions used for core flooding tank injection in the safety analysis are met.

The tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The one hour limit for operation with a core flooding tank (CFT) inoperable for reasons other than boron concentration not within limits minimizes the time the plant is exposed to a possible LOCA event occurring with failure of a CFT, which may result in unacceptable peak cladding temperatures.

With boron concentration for one CFT not within limits, the condition must be corrected within 72 hours. The 72 hour limit was developed considering that the effects of reduced boron concentration on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the CFTs is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of both CFTs, the consequences are less severe than they would be if the contents of a CFT were not available for injection.

The completion times to bring the plant to a MODE in which the Limiting Condition for Operation (LCO) does not apply are reasonable based on operating experience. The completion times allow plant conditions to be changed in an orderly manner and without challenging plant systems.

CFT boron concentration sampling within 6 hours after an 80 gallon volume increase will identify whether inleakage from the RCS has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the borated water storage tank (BWST), because the water contained in the BWST is within CFT boron concentration requirements.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The operability of two independent ECCS subsystems with RCS average temperature $\geq$ 280°F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the core flooding tanks is capable of supplying sufficient core cooling to maintain the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

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With the RCS temperature below 280°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

The function of the trisodium phosphate dodecahydrate (TSP) contained in baskets located in the containment normal sump or on the 565' elevation of containment adjacent to the normal sump, is to neutralize the acidity of the post-LOCA borated water mixture during containment emergency sump recirculation. The borated water storage tank (BWST) borated water has a nominal pH value of approximately 5. Raising the borated water mixture to a pH value of 7 will ensure that chloride stress corrosion does not occur in austenitic stainless steels in the event that chloride levels increase as a result of contamination on the surfaces of the reactor containment building. Also, a pH of 7 is assumed for the containment emergency sump for iodine retention and removal post-LOCA by the containment spray system.

The Surveillance Requirement (SR) associated with TSP ensures that the minimum required volume of TSP is stored in the baskets. The minimum required volume of TSP is the volume that will achieve a post-LOCA borated water mixture pH of ≥ 7.0, conservatively considering the maximum possible sump water volume and the maximum possible boron concentration. The amount of TSP required is based on the mass of TSP needed to achieve the required pH. However, a required volume is verified by the SR, rather than the mass, since it is not feasible to weigh the entire amount of TSP in containment. The minimum required volume is based on the manufactured density of TSP (53 lb/ft^3). Since TSP can have a tendency to agglomerate from high humidity in the containment, the density may increase and the volume decrease during normal plant operation, however, solubility characteristics are not expected to change. Therefore, considering possible agglomeration and increase in density, verifying the minimum volume of TSP in containment is conservative with respect to ensuring the capability to achieve the minimum required pH. The minimum required volume of TSP to meet all analytical requirements is 250 ft^3. The surveillance requirement of 290 ft^3 includes 40 ft^3 of spare TSP as margin. Total basket capacity is 325 ft^3.

Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

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Containment Emergency Sump Recirculation Valves DH-9A and DH-9B are de-energized during MODES 1, 2, 3 and 4 to preclude postulated inadvertent opening of the valves in the event of a Control Room fire, which could result in draining the Borated Water Storage Tank to the Containment Emergency Sump and the loss of this water source for normal plant shutdown. Re-energization of DH-9A and DH-9B is permitted on an intermittent basis during MODES 1, 2, 3 and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

Borated Water Storage Tank (BWST) outlet isolation valves DH-7A and DH-7B are de-energized during MODES 1, 2, 3, and 4 to preclude postulated inadvertent closure of the valves in the event of a fire, which could result in a loss of the availability of the BWST. Re-energization of valves DH-7A and DH-7B is permitted on an intermittent basis during MODES 1, 2, 3, and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on the BWST minimum volume and boron concentration ensure that:

1) sufficient water is available within containment to permit recirculation cooling flow to the core following manual switchover to the recirculation mode, and

2) The reactor will remain at least 1% $\Delta k/k$ subcritical in the cold condition at 70°F, xenon free, while only crediting 50% of the control rods' worth following mixing of the BWST and the RCS water volumes.

These assumptions ensure that the reactor remains subcritical in the cold condition following mixing of the BWST and the RCS water volumes.

With either the BWST boron concentration or BWST borated water temperature not within limits, the condition must be corrected in eight hours. The eight hour limit to restore the temperature or boron concentration to within limits was developed considering the time required to change boron concentration or temperature and assuming that the contents of the BWST are still available for injection.

The bottom 4 inches of the BWST are not available, and the instrumentation is calibrated to reflect the available volume. The limits on water volume, and boron concentration ensure a pH value of between 7.0 and 11.0 of the solution sprayed within the containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.1.1.1, Table 4.3-1, Functional Unit 15

1. A. Technical Specification (TS) 3/4.3.1, "Reactor Protection System Instrumentation," Surveillance Requirement (SR):

   4.3.1.1.1, Table 4.3-1, Functional Unit 15, SCR Relays, Channel Functional Test

   B. Systems or Components:

   Control Rod Drive Control System (CRDCS)
   Silicon Controlled Rectifier (SCR) Relays

   C. Updated Safety Analysis Report (USAR) Sections:

   4.3 Nuclear Design
   5.2 Integrity of Reactor Coolant Pressure Boundary
   7.2 Reactor Protection System
   7.4 Systems Required for Safe Shutdown
   7.7 Control Systems

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.1.1.1 requires that a Channel Functional Test be performed for the Reactor Protection System (RPS) instrumentation channels at the frequency shown in TS Table 4.3-1. Table 4.3-1 specifies a frequency of "R" for Channel Functional Testing of the Silicon Controlled Rectifier (SCR) Relays (Functional Unit 15). The present TS Definition Table 1.2 defines Notation "R" as a frequency of "At least once per 18 months." Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   This License Amendment Request addresses in its main body a proposed change to TS Table 1.2 which would redefine Notation "R" to be a frequency of "At least once per 24 months." This change would result in increasing the surveillance interval for RPS SCR Relay Channel Functional Testing to "At least once per 24 months." This is consistent with the guidance provided in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

B. The purpose of the RPS is to initiate a reactor trip when a sensed parameter (or group of parameters) exceeds a setpoint value indicating the approach of an unsafe condition. In this manner, the reactor core is protected from exceeding design limits and the Reactor Coolant System (RCS) is protected from overpressurization. The RPS monitors the following generating station variables:

1. Total Out-of-core Neutron Flux
2. RCS Coolant Flow
3. RCS Pump Status
4. RCS Reactor Outlet Temperature
5. RCS Pressure
6. Containment Vessel Pressure
7. Out-of-core Neutron Flux Imbalances

The RPS is described in the DBNPS Updated Safety Analysis Report (USAR), Section 7.2, "Reactor Protection System." The RPS consists of four identical protection channels which are redundant and independent. Each channel is served by its own independent sensors which are physically isolated from the sensors of the other protective channels. Each sensor supplies an input signal to one or more signal processing strings in the RPS channel. Each signal processing string terminates in a bistable which electronically compares the processed signal with trip setpoints. All bistable contacts are connected in series.

In the normal untripped state, the contact associated with each bistable will be closed, thereby energizing the channel terminating relay. A trip of one of the channel bistables in one channel causes a half trip in each of the four RPS channels. The trip of a bistable in a second channel completes the two-out-of--four logic and causes a full trip of each RPS channel. The full trip of each channel deenergizes the undervoltage coils and undervoltage relays of the channel's respective Control Rod Drive (CRD) trip breaker in the Control Rod Drive Control System (CRDCS). Each RPS channel contains a reactor trip module which performs the two-out-of-four trip logic and provides the signals to open the channels' associated CRD trip breaker.

The function of the Control Rod Drive Control System - Trip Portion is to interrupt power to the control rod drive mechanisms to insert control rods upon receipt of a RPS, Anticipatory Reactor Trip System (ARTS), Diverse Scram System (DSS) or manual trip signal.

A trip signal from the RPS or ARTS is applied to the undervoltage coils and the undervoltage relays of the CRD trip breakers causing the breakers to trip open, thereby removing power from the control rod drive motors resulting in the insertion of control rods and a reactor trip.

The trip portion of the CRDCS is described in the DBNPS USAR, Section 7.4.1.1. The CRDCS trip logic is designed so that when power is removed from the control rod drive mechanisms, the roller nuts disengage from the lead screw, and a free-fall gravity insertion of the control rods occurs. Two diverse and independent trip methods, in series, are provided for removal of power to the mechanisms. First, a trip is initiated when power is interrupted to the undervoltage coils of the main A.C. feeder breakers and to the undervoltage relays in the shunt trip circuits. Second, a trip is initiated when the gating signals to the SCRs are interrupted. Since parallel power feeds are provided to the control rod drive mechanisms, interruption of both feeds is required for trip action in either method of trip. There are two CRD trip breakers per power feed, each associated with one of four Reactor Protection System channels.

The primary method of trip interrupts power to the CRD mechanism power supplies. Power circuit breakers equipped with instantaneous undervoltage coils and shunt trip devices are used as primary trip devices. The RPS channels energize the undervoltage coil of the breakers. A trip breaker can remain closed only if its undervoltage coil is energized. Upon loss of voltage at the undervoltage coil due to interruption by an RPS, ARTS or manual trip signal, the CRD trip breaker trips open. No external power is required to trip the breakers which have stored-energy trip mechanisms. The trip breakers must be manually reset once tripped. Breaker reset is possible only after the trip signal is reset. The shunt trip undervoltage relay is installed in parallel with the undervoltage coil of the CRD trip breaker. Voltage interruption due to a trip signal deenergizes the undervoltage relay energizing the shunt trip device which is powered from essential 125 VDC, thereby also tripping the CRD trip breaker.

The second trip method interrupts the gate control signals to the SCRs in each of the nine CRD mechanisms motor power supplies and the motor return power supply. The trip devices in this case are ten relays connected with their coils in parallel. Contacts of these relays interrupt the gate control signals to the SCRs in each power supply. When the gate signals are interrupted, the SCRs will revert to their open state on the next negative half-cycle of the applied A.C. voltage, thus removing all power at the outputs of the motor power supplies. Because the power supplies have redundant halves, two sets of ten relays each are provided. RPS channel 3 energizes one set of trip relays and RPS channel 4 energizes the other set through auxiliary relays in the breakers.

The trip relays can remain in their non-tripped state only if the associated RPS channel is energized. When an RPS channel trips, the associated trip relays deenergize, interrupting the SCR gate control signals.

The SCRs are not an initiator, nor contributor, to the initiation of an accident described in the USAR. The SCRs function to meet the design purpose of the RPS of initiating a reactor trip when a sensed parameter or group of parameters exceeds a setpoint value indicating the approach of an unsafe condition and thereby protecting the reactor core from exceeding design limits and the RCS from overpressurization. The trip parameters and trip setpoints were selected so that no core design limits are exceeded as a consequence of any anticipated operational occurrences.

C. The original Standard Technical Specifications upon which the DBNPS Technical Specifications were modeled, NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," did not include surveillance requirements for the SCR Relays. On March 27, 1987, Toledo Edison submitted a license amendment application proposing the addition of Channel Functional Testing surveillance requirements, on an 18 month frequency, for the SCR Relays. These proposed changes were in accordance with the Generic Letter 85-10, "Technical Specifications for Generic Letter 83-28, Items 4.3 and 4.4," dated May 23, 1985. Performing Channel Functional Testing on an eighteen month frequency was determined as sufficient since the SCR relays are a duplicative function. That is, the non-1E, non-seismic SCR relays are a backup tripping means to the Reactor Trip Breakers. The NRC approved the license amendment via issuance of Amendment No. 108, dated March 2, 1988.

As discussed above, the proposed changes follow the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. Based on the above review, it is concluded that the licensing basis for the SCR Relay Channel Functional Test SR will not be invalidated by increasing the surveillance interval from 18 months to 24 months and by continuing to allow the application of TS 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. USAR Section 7.2, "Reactor Protection System," through Revision 19.

v. USAR Section 7.4.1.1, "Control Rod Drive Control System - Trip Portion," through Revision 19.

vi. USAR Section 3D.1.16, "Criterion 20 - Protection System Functions," through Revision 19.

vii. NRC Generic Letter 85-10, "Technical Specifications for Generic Letter 83-28, Items 4.3 and 4.4," dated May 23, 1985 (Toledo Edison Log Number 1756).

viii. Toledo Edison letter to the NRC dated March 27, 1987 (Toledo Edison Serial Number 1312).

ix. NRC License Amendment No. 108 to Facility Operating License No. NPF-3, dated March 2, 1988 (Toledo Edison Log Number 2512).

3. Surveillance Data Review:

A. The 18 month surveillance test results data for the SCR Relay Channel Functional Test were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results.

B. The test results indicate no failures over this time period for the components.

C. Based on a review of the 18 month surveillance test results, no additional actions are necessary or recommended to support the increase in the present surveillance interval.

The SCR Relays are checked as part of Reactor Trip Breaker Channel Functional Testing. These tests verify that the SCRs are functioning properly and provide additional opportunities to detect SCR abnormalities.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for the SCR Relay Channel Functional Test SR from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-MI-03011, "Channel Functional Test Of Reactor Trip Breaker B, RPS Channel 1 Reactor Trip Module Logic, and ARTS Channel 1 Output Logic."

ii. DBNPS Procedure DB-MI-03012, "Channel Functional Test Of Reactor Trip Breaker A, RPS Channel 2 Reactor Trip Module Logic, and ARTS Channel 2 Output Logic."

iii. DBNPS Procedure DB-MI-03013, "Channel Functional Test Of Reactor Trip Breaker D, RPS Channel 3 Reactor Trip Module Logic, and ARTS Channel 3 Output Logic."

iv. DBNPS Procedure DB-MI-03014, "Channel Functional Test Of Reactor Trip Breaker C, RPS Channel 4 Reactor Trip Module Logic, and ARTS Channel 4 Output Logic."

v. DBNPS Procedure DB-SC-03273, "CRD Independent SCR Functional Test."

4. Maintenance Records Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of maintenance activities.

B. The maintenance records review revealed the following:

In December, 1990, during Cycle 7, failure of the K11A motor return relay resulted in a reactor trip. Troubleshooting performed following the trip identified a failed contact in the relay. This failure was categorized as a random failure, and not age-related. The relay was replaced. In addition, in order to prevent reactor trips during surveillance testing, the applicable surveillance test procedures were revised to include prerequisite actions to ensure that K11A relay contacts are properly passing the associated motor return SCR gating signals. This event was reported to the NRC in Licensee Event Report (LER) 90-016-01. The potential for this type of failure would not be affected by converting from an 18 month to a 24 month cycle.

In September, 1994, during Cycle 9, while performing surveillance test DB-MI-03012, CRD electronic trip "C" did not automatically reset. The trip was manually reset. Subsequent investigation and troubleshooting could not cause the problem to repeat. However, investigation revealed that failure of an electrical contact to conduct in one of two model KHU17A11 Potter-Brumfield relays (K10A or K11A) could have caused the problem. Therefore, as a preventive measure, both relays were replaced. Further investigation to determine specifically which of the two relays caused the problem was not feasible due to the intermittent nature of the problem and due to the inaccessibility of the electronic trip "C" circuitry for troubleshooting. Troubleshooting of the electronic trip "C" circuitry would have required disassembly, which could have led to additional problems in the circuit. The only other similar recorded failure of this model relay in the CRD system was a random failure during Cycle 7, as described above. It is noted that a failure of this type does not place the plant at risk since it occurs only after the reactor trip breaker is opened then reset. However, a failure of this type results in an inconvenience since it requires the circuit to be manually reset. This problem has not recurred, and its consequences would have no impact on the RPS performing its safety function. Therefore, this occurrence does not not have an affect on converting from an 18 month to a 24 month fuel cycle.

C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for the SCR Relay Channel Functional Test from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Maintenance Work Order Records.

ii. DBNPS Potential Condition Adverse to Quality (PCAQ) Records.

iii. Licensee Event Report (LER) 90-016-01, "Reactor Trip Due to Group Rod Drop."

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.1.1.2

1. A. Technical Specification (TS) 3/4.3.1, "Reactor Protection System Instrumentation," Surveillance Requirement (SR):

   4.3.1.1.2

   B. Systems or Components:

RPS1RC2203	(Shutdown Bypass RPS Channel 1)
RPS1RC2205	(Shutdown Bypass Select RPS Channel 1)
RPS2RC2203	(Shutdown Bypass RPS Channel 2)
RPS2RC2205	(Shutdown Bypass Select RPS Channel 2)
RPS3RC2203	(Shutdown Bypass RPS Channel 3)
RPS3RC2205	(Shutdown Bypass Select RPS Channel 3)
RPS4RC2203	(Shutdown Bypass RPS Channel 4)
RPS4RC2205	(Shutdown Bypass Select RPS Channel 4)

   C. Updated Safety Analysis Report (USAR) Sections:

4.3	Nuclear Design
5.2	Integrity of Reactor Coolant Pressure Boundary
7.2	Reactor Protection System
7.4	Systems Required for Safe Shutdown
7.7.1	Control Systems - Description

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.1.1.2 requires that the total bypass function be demonstrated Operable at least once per 18 months during Channel Calibration testing of each channel affected by bypass operation. Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in SR 4.3.1.1.2 the words "at least once per 18 months" be replaced with "at least once per REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time $\leq$ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

B. The purpose of the RPS is to initiate a reactor trip when a sensed parameter (or group of parameters) exceeds a setpoint value indicating the approach of an unsafe condition. In this manner, the reactor core is protected from exceeding design limits and the Reactor Coolant System (RCS) is protected from overpressurization. The RPS monitors the following generating station variables:

1. Total Out-of-core Neutron Flux
2. RCS Coolant Flow
3. RCS Pump Status
4. RCS Reactor Outlet Temperature
5. RCS Pressure
6. Containment Vessel Pressure
7. Out-of-core Neutron Flux Imbalances

The RPS is described in the DBNPS Updated Safety Analysis Report (USAR), Section 7.2. The RPS consists of four identical protection channels which are redundant and independent. Each channel is served by its own independent sensors which are physically isolated from the sensors of the other protective channels. Each sensor supplies an input signal to one or more signal processing strings in the RPS channel. Each signal processing string terminates in a bistable which electronically compares the processed signal with trip setpoints. All bistable contacts are connected in series.

In the normal untripped state, the contact associated with each bistable will be closed, thereby energizing the channel terminating relay. A trip of one of the channel bistables in one channel causes a half trip in each of the four RPS channels. The trip of a bistable in a second channel completes the two-out-of-four logic and causes a full trip of each RPS channel. The full trip of each channel deenergizes the undervoltage coils and undervoltage relays of the channel's respective Control Rod Drive (CRD) trip breaker in the Control Rod Drive Control System (CRDCS). Each RPS channel contains a reactor trip module which performs the two-out-of-four trip logic and provides the signals to open the channels' associated CRD trip breaker.

The RPS channel bypass is a maintenance bypass and permits the testing and maintenance of a single channel during power operations. With the bypass in effect, the three remaining channels provide the necessary protection. Since only two channel trips are required to cause a reactor trip, a single failure will not prevent the RPS from fulfilling its protective function.

The RPS is a de-energize-to-trip system. Therefore, if power is lost to a channel, that channel will trip, reducing the system trip coincidence to one-out-of-three. In the event that a module, which performs a protective function is removed from its rack, that RPS channel will trip (unless that channel is bypassed).

A shutdown bypass is provided to allow rod withdrawal testing with the unit shutdown. To initiate the bypass the operator must turn a key switch in each RPS channel. Turning the key switch removes the following trips from the logic train: power/imbalance/flow, power/pumps, variable low RC pressure-temperature, and low RC pressure. The following trips are unaffected: high flux, high RC temperature, high RC pressure, and high containment pressure. The key switch also inserts the shutdown bypass high pressure trip. The setpoint of this trip is lower than the setpoint of the normal low pressure trip.

During normal operation the shutdown bypass high pressure trip bistable is normally tripped since operating pressure is greater than the trip setpoint.

If the operator initiates the shutdown bypass with the unit at power, that RPS channel trips. The procedure for effecting this bypass is to wait until primary pressure is below the trip setpoint and the plant is shut down. The operator is then free to reset the tripped bistable and to turn the key switch in each channel.

Initiation of the shutdown bypass is continuously indicated locally on the RPS cabinets and on the station annunciator board.

The shutdown bypass is considered the total bypass for the purposes of SR 4.3.1.1.2.

The RPS is not an initiator, nor contributor, to the initiation of an accident described in the USAR. The trip parameters and trip setpoints were selected so that no core design limits are exceeded as a consequence of any anticipated operational occurrences. Further, the shutdown bypass is utilized when the unit is in a shutdown condition and initiation of the shutdown bypass while at power would cause a trip of the affected RPS channel. The RPS is designed so that no single failure can prevent the RPS from performing its protective function.

C. The current surveillance interval of 18 months was based on the guidance of NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," during the initial licensing of the DBNPS. As discussed above, the proposed change follows the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis of the RPS total bypass function SR will not be invalidated by increasing the surveillance interval for SR 4.3.2.1.2 from 18 months to 24 months and by continuing to allow application of TS 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. USAR Section 7.2, "Reactor Protection System," through Revision 19.

v. USAR Section 7.4.1.1, "Control Rod Drive Control System - Trip Portion," through Revision 19.

vi. USAR Section 3D.1.16, "Criterion 20 - Protection System Functions," through Revision 19.

3. Surveillance Data Review:

A. The 18 month surveillance test results data for SR 4.3.1.1.2 were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985 and covers five refueling outages and four operating cycles of test results.

B. The test results indicate no failures over this time period for the components.

C. Based on a review of the 18 month surveillance test results data, no additional actions are necessary or recommended to support an increase in the present surveillance interval.

In addition, it is noted that there is a parallel contact pair operated off the same relay coil (in the shutdown bypass auxiliary relay module) that implements the bypass function, which provides a dim to bright lamp indication when the shutdown bypass key switch module is placed in the bypass position. The bright lamp indication is presently verified by procedure, providing an opportunity to detect a failure immediately upon entering shutdown bypass for a given RPS channel.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.3.1.1.2 from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-MI-03051, "Channel Calibration of 58A-ISP/TRC02B2/3B2 RCS Pressure and Temperature to RPS Channel 1."

ii. DBNPS Procedure DB-MI-03052, "Channel Calibration of 58A-ISP/TRC02A2/3A4 RCS Pressure and Temperature to RPS Channel 2."

iii. DBNPS Procedure DB-MI-03053, "Channel Calibration of 58A-ISP/TRC02B1/3B4 RCS Pressure and Temperature to RPS Channel 3."

iv. DBNPS Procedure DB-MI-03054, "Channel Calibration of 58A-ISP/TRC02A1/3A2 RCS Pressure and Temperature to RPS Channel 4."

v. DBNPS Procedure DB-OP-06403, "RPS and NI Operating Procedure."

4. Maintenance Records Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of maintenance activities.

B. No failures were noted during this time period for these components.

C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support an increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.3.1.1.2 from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Maintenance Work Order Records.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.1.1.3

1. A. Technical Specification (TS) 3/4.3.1, "Reactor Protection System Instrumentation," Surveillance Requirement (SR):

   4.3.1.1.3

   B. Systems or Components:

   Reactor Protection System instrumentation components required for the functional units with response times specified in TS Table 3.3-2.

   C. Updated Safety Analysis Report (USAR) Sections:

4.3	Nuclear Design
5.2	Integrity of Reactor Coolant Pressure Boundary
7.2	Reactor Protection System

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.1.1.3 requires that the Reactor Protection System (RPS) Response Time of each reactor trip function be demonstrated to be within its limit at least once per 18 months. Surveillance Requirement 4.3.1.1.3 also specifies that each test shall include at least 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. Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in SR 4.3.1.1.3 the words "at least once per 18 months" be replaced with "at least once per REFUELING INTERVAL," and the words "once every N times 18 months" be replaced with "once every N times the REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time ≤ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

   B. The purpose of the RPS is to initiate a reactor trip when a sensed parameter (or group of parameters) exceeds a setpoint value indicating the approach of an unsafe condition. In this manner, the reactor core is protected from exceeding design limits and the Reactor Coolant System (RCS) is protected from overpressurization. The RPS monitors the following generating station variables:

1. Total Out-of-core Neutron Flux
2. RCS Coolant Flow
3. RCS Pump Status
4. RCS Reactor Outlet Temperature
5. RCS Pressure
6. Containment Vessel Pressure
7. Out-of-core Neutron Flux Imbalances

The RPS is described in the USAR Section 7.2, Reactor Protection System. The RPS consists of four identical protection channels which are redundant and independent. Each channel is served by its own independent sensors which are physically isolated from the sensors of the other protective channels. Each sensor supplies an input signal to one or more signal processing strings in the RPS channel. Each signal processing string terminates in a bistable which electronically compares the processed signal with trip setpoints. All bistable contacts are connected in series.

In the normal untripped state, the contact associated with each bistable will be closed, thereby energizing the channel terminating relay. A trip of one of the channel bistables in one channel causes a half trip in each of the four RPS channels. The trip of a bistable in a second channel completes the two-out-of-four logic and causes a full trip of each RPS channel. The full trip of each channel deenergizes the undervoltage coils and undervoltage relays of the channel's respective Control Rod Drive (CRD) trip breaker in the Control Rod Drive Control System (CRDCS). Each RPS channel contains a reactor trip module which performs the two-out-of-four trip logic and provides the signals to open the channels' associated CRD trip breaker.

Technical Specification Definition 1.25 defines RPS Response Time as that time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until power interruption at the control rod drive breakers.

As stated in TS Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, the measurement of RPS Response Time at the specified frequencies provides assurance that the RPS action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

As further stated in the TS Bases, 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 RPS is not an initiator, nor contributor, to the initiation of an accident described in the USAR. The trip parameters and trip setpoints were selected so that no core design limits are exceeded as a consequence of any anticipated operational occurrences. The RPS is designed so that no single failure can prevent the RPS from performing its protective function.

C. The NRC Operating License Safety Evaluation Report, NUREG-0136, dated December 1976 stated in Section 7.2.1, "Protection System Response Time Testing," that the RPS response time tests would be repeated each refueling outage, but not less frequently than every 18 months, and that the TS would reflect this requirement. The current surveillance interval of 18 months for the RPS instrumentation functional units response times listed in TS Table 3.3-2 also follows on the guidance of NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors." As discussed above, the proposed change follows the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991, by demonstrating historical surveillance and maintenance data support a request that the licensing basis be revised to allow response time testing on a 24-month frequency.

D. As a result of the above review, it is concluded that the licensing basis of the Reactor Protection System Response Time SR will not be invalidated by increasing the surveillance interval for SR 4.3.1.1.3 from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a nonroutine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. USAR Section 7.2, "Reactor Protection System," through Revision 19.

3. Surveillance Data Review:

A. The 18 month surveillance test results data for SR 4.3.1.1.3 were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results.

B. The test results indicate that there was one instance of TS overall response time limits being exceeded over this time period.

In November, 1994, during 9RFO, both the Reactor Coolant (RC) low pressure and RC high pressure response times for RPS Channel 4 exceeded their TS overall response time limit. The overall response time is calculated by adding measurements of the RPS cabinet modules' response time to the response time for the pressure transmitter and the reactor trip breaker. The RPS cabinet modules include buffer amplifier, bistable, and reactor trip modules. The failure was determined to be due to an incorrect buffer amplifier module type being installed during 6RFO. Post-maintenance testing following installation of the incorrect buffer amplifier did not reveal a problem since the overall response time was within limits. Performance of RPS Channel 4 RC pressure string response time was not required again until 9RFO. During the 9RFO test performance, the response times measured for the RPS cabinet modules alone exceeded the TS overall response time limit, without the addition of the pressure transmitter and reactor trip breaker response times. The surveillance test failure during 9RFO was a configuration control issue rather than a hardware failure. This event was reported to the NRC in Licensee Event Report (LER) 94-005-00. Changing the fuel cycle length from 18 to 24 months would have no effect on this type of occurrence where an incorrect buffer amplifier was installed.

C. Based on a review of the 18 month surveillance test results data, no additional actions are necessary or recommended to support an increase in the TS surveillance interval.

No evidence exists in the trend data to suggest that the response times of these components increase with time at a rate that would jeopardize successful surveillance testing at the increased test interval.

It is noted that there are very few failure modes for the components being tested that can only be identified by response time testing, and not by channel calibration or functional testing. These tests provide opportunities, in addition to response time testing, to detect problems.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.3.1.1.3 from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-SC-03160, "RPS Overall Response Time Calculations Channel 1."

ii. DBNPS Procedure DB-SC-03161, "RPS Overall Response Time Calculations Channel 2."

iii. DBNPS Procedure DB-SC-03162, "RPS Overall Response Time Calculations Channel 3."

iv. DBNPS Procedure DB-SC-03163, "RPS Overall Response Time Calculations Channel 4."

v. DBNPS Procedure DB-ME-03020, "Reactor Trip Breaker Response Time Test."

vi. DBNPS Procedure DB-MI-03005, "Time Response Test of Reactor Protection System Channel 1."

vii. DBNPS Procedure DB-MI-03006, "Time Response Test of Reactor Protection System Channel 2."

viii. DBNPS Procedure DB-MI-03007, "Time Response Test of Reactor Protection System Channel 3."

ix. DBNPS Procedure DB-MI-03008, "Time Response Test of Reactor Protection System Channel 4."

x. DBNPS Procedure DB-MI-03009, "Reactor Protection System On-Line Response Time Data Collection."

xi. DBNPS Procedure DB-MI-03205, "Channel Functional Test/Calibration and Response Time of RCP Monitor (RC 3601) to SFRCS Channel 1 and RPS Channel 1."

xii. DBNPS Procedure DB-MI-03206, "Channel Functional Test/Calibration and Response Time of RCP Monitor (RC 3603) to SFRCS Channel 3 and RPS Channel 3."

xiii. DBNPS Procedure DB-MI-03207, "Channel Functional Test/Calibration and Response Time of RCP Monitor (RC 3602) to SFRCS Channel 2 and RPS Channel 2."

xiv. DBNPS Procedure DB-MI-03208, "Channel Functional Test/Calibration and Response Time of RCP Monitor (RC 3604) to SFRCS Channel 4 and RPS Channel 4."

xv. Licensee Event Report (LER) 94-005-00, "RPS Channel 4 Response Times Exceeded."

xvi. NUREG-0136, NRC Safety Evaluation Report for the Davis-Besse Nuclear Power Station Operating License NPF-3, dated December, 1976.

4. Maintenance Records Review:

A. The 18 month maintenance records for the RPS instrumentation components with response times listed in Table 3.3-2 were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results. Maintenance activities that involved measurements and/or tests of RPS instrumentation response times were evaluated.

B. Three failures of Agastat Model 7022PJ relays used in the Reactor Coolant Pump (RCP) monitoring circuits have been identified during refueling outage preventive maintenance (PM) activities. No other component response time failures have been discovered during refueling outage maintenance activities.

During 6RFO (1990), Agastat relay X24 in RC3604, RCP Monitor Relay Panel Channel 4, was found to be defective. Its drop out time was very erratic, ranging from 0.06 to several seconds. The desired setting is 0.042 seconds. The relay was replaced with an identical one.

During 6RFO (1990), Agastat relay X43 in RC3603, RCP Monitor Relay Panel Channel 3, was found to be defective. Some of its contacts did not change state when the relay was operated. The relay was replaced with an identical one.

During 7RFO (1991), Agastat relay X24 in RC3604 had inconsistent drop out times that would change without moving the dial setting. The as-found drop out time was an unusually slow 0.224 seconds. The relay was replaced with an identical one.

Based on engineering review, these three cases are categorized as random failures. Further, the X43 relay failure, would have been detected through the monthly channel functional test. In addition, the first X24 relay failure, which occurred in 6RFO, would likely have been detected during the monthly channel functional test if the drop out time was several seconds, since such a slow drop out time would likely be identified by the test performer. The unusually slow X24 relay drop out time which was observed during 7RFO would not likely have been identified by the monthly functional channel test. However, since the relay response time accounts for only a portion of the overall channel response time, the unusually slow relay drop out time would not likely have caused the channel to fail its refueling surveillance test acceptance criteria.

The failure of a single Agastat relay in an RCP monitoring circuit would only prevent the associated RPS channel from sensing the loss of one RCP (i.e., it would still sense the loss of the other three RCPs). The RPS power/pumps trip function is the primary trip for only one USAR accident, loss of forced RC flow (USAR section 15.2.5). In that accident all four RCPs are lost. With one Agastat relay failed in an RPS channel, that channel would sense three pumps lost, which would still result in that RPS channel being tripped because the three pumps lost setpoint is the same as the four pumps lost setpoint ($\leq$ 0% full power).

There are 16 Agastat Model 7022PJ relays used in the RCP monitoring circuits, and no failures have occurred since 1991. Based on industry experience and engineering judgment, the failure rate for this model relay is reasonable.

C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support an increase in the present surveillance interval for the following reasons.

As stated above, the three failures of the Agastat Model 7022PJ relays used in the RCP monitoring circuits were random failures, and the failure rate over this time period is reasonable based on industry experience and engineering review. In addition, a 24 month test frequency is acceptable because plant operating experience shows that random failures of instrumentation components which can cause response time degradation but not channel failure, are infrequent occurrences. Furthermore, the failure of a particular parameter's functional unit will not affect the other three RPS channels or other parameters in the same channel. Therefore, there remains sufficient instrumentation component redundancy for the RPS instrumentation components to perform their safety function.

D. Based on the historical performance of these components, taking into consideration the low number of component response time failures that would not be identified by on-line surveillance testing or other on-line means, the low potential for increases in failure rates of these components under a longer test interval, the number of redundant components, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.3.1.1.3 from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Maintenance Work Order Records.

5. Other Information:

Reactor trip breakers are required by Technical Specifications to be response time tested only once every fourth refueling outage, however, they are actually tested each refueling outage as post-maintenance testing for breaker PM activities. This surveillance test is performed with the breaker installed in its cubicle. If the breakers experienced response time degradation over the course of an operating cycle, and this degradation was corrected by performance of the PM activities, then the surveillance test results would not show any degradation since the tests are always performed shortly after the PM activities are completed. Under this scenario, a 24 month fuel cycle would allow six additional months for potential breaker response time degradation to occur.

To demonstrate that this degradation is not occurring, PM data on breaker response time tests conducted with the breakers out of their cubicles was evaluated. These tests are performed before and after the PM activities. While they do not exactly replicate the response time test with the breaker installed in its cubicle, the results can be used to determine whether or not PM performance significantly improves breaker response time (i.e., whether or not breaker response times are significantly degrading over the course of an operating cycle). For the undervoltage device trip, for 38 samples, the mean change in response time from before the PM to after the PM was +0.63 milliseconds, with a standard deviation of 3.69 milliseconds. For the shunt trip device, for 42 samples, the mean change in response time was -1.55 milliseconds, with a standard deviation of 0.60 milliseconds. These times are all short compared to the typical response times obtained during the surveillance test (30 to 60 milliseconds), therefore, the PM activities have very little, if any, affect on breaker response time.

Therefore, it can be concluded that breaker response times have very little, or no degradation over the course of an 18 month operating cycle, and it is unlikely that a 24 month fuel cycle would produce unacceptable reactor trip breaker response time performance.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.2.1.1, Table 4.3-2, Functional Unit 3

1. A. Technical Specification (TS) 3/4.3.2.1, "Safety Features Actuation System Instrumentation," Surveillance Requirement (SR):

   4.3.2.1.1, Table 4.3-2, Functional Unit 3, Channel Functional Test, Note(1)

   B. Systems or Components:

   SFAS manual actuation handswitches and related circuitry, as described in further detail in Section 3.A below.

   C. Updated Safety Analysis Report (USAR) Sections:

   4.3.5.2 Safety Features Actuation System Instrumentation (SFAS)
   6.3.1.4 System Short- and Long-Term Capability
   7.3 Safety Features Actuation System (SFAS)
   7.6.1.1 Normal Decay Heat Removal Valve Control System

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.2.1.1 requires that a Channel Functional Test be performed for the Safety Features Actuation System (SFAS) functional units at the frequency shown in TS Table 4.3-2. Note (1) of Table 4.3-2, which applies to Functional Unit 3, requires that manual actuation switches be tested "at least once per 18 months during shutdown." Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in Note (1) to Table 4.3-2, the words at "least once per 18 months during shutdown" be replaced with "at least once per REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time $\leq$ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

B. The design goal of the SFAS is to automatically prevent or limit fission product and energy release from the core, to isolate the containment vessel and to initiate the operation of the Engineered Safety Features (ESF) equipment in the event of a loss of coolant accident (LOCA). The SFAS will automatically sequence the protective action by loading equipment in steps to the Emergency Diesel Generators (EDGs) if normal or reserve power is not available to the 4.16kV essential bus(es) coincident with an SFAS initiation signal. As described in USAR Section 6.3.1.4, the Emergency Core Cooling System (ECCS) design basis assumes simultaneous loss of normal and reserve (offsite) power with a LOCA.

As described in USAR Section 7.3, the SFAS consists of four identical redundant sensing and logic channels and two identical redundant actuation channels. Each sensing channel includes analog circuits with analog isolation devices and each logic channel includes trip bistable modules with digital (opto-electronic) isolation devices. The isolated output of the trip bistable module is used to comprise coincidence matrices with the terminating relays within the actuation channel of the SFAS. These opto-electronic isolation devices also provide isolation between channels. The trip bistables monitor the station variables and normally feed continuous electrical signals into two-out-of-four coincidence matrices. Should any of the station variables exceed their trip setpoints, the corresponding bistables in each of the four channels will trip and cease sending output signals. Should two of the four channel bistables monitoring the same station variable cease to send output signals, the corresponding normally energized terminating relays on all channels will trip. The terminating relays of sensing and logic Channels 1 and 3, must both be deenergized to activate safety actuation Channel 1. Similarly, sensing and logic Channels 2 and 4 are deenergized to activate actuation Channel 2. The terminating relays act on the actuation control devices such as motor controllers and solenoid valves.

Four manual trip switches are provided at the main control board; two switches to activate the two redundant Containment Vessel Spray Systems and two switches to activate the two redundant SFAS channels exclusive of the Containment Spray Systems and the CV Emergency Sump Recirculation Systems.

C. The current surveillance interval of 18 months was based on the guidance of NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," during the initial licensing of the DBNPS. As discussed above, the proposed change follows the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis of the SFAS manual actuation switches Channel Functional Test will not be invalidated by increasing the surveillance interval from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. USAR Section 6.3.1.4, "System Short- and Long-Term Capability," through Revision 19.

v. USAR Section 7.3, "Safety Features Actuation System (SFAS)," through Revision 19.

3. Surveillance Data Review:

A. The 18 month surveillance test results data for the SFAS manual actuation handswitches and related circuitry were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results.

As described in TS Table 4.3-2, Functional Unit 3, including Note (1), the scope of this SR is the SFAS (Except Containment Spray and Emergency Sump Recirculation) and the Containment Spray handswitch circuitry not tested under the monthly Channel Functional Test. The components affected consist of the handswitches mounted on the main control panel and the isolator relay / logic input gate circuit within each of the 68 affected output modules in the system. These components are checked by the SFAS Integrated Time Response Test, DBNPS Procedure DB-SC-03114, on an 18 month interval.

B. The test results indicated one failure over this time period for the components.

In October, 1988, during 5RFO, Containment Spray System Train 1 manual actuation push-button HIS202CA failed to operate properly. The root cause of the switch failure was attributed to the switch contact assembly separating from the push-button, apparently due to the switch assembly not being properly locked into place. The loose mounting may have been from poor initial installation or from personnel inadvertently bumping the switch while performing maintenance in the cabinet. The switch was properly reassembled and successfully retested. The switch has demonstrated reliable operation during the subsequent tests performed during 6RFO through 9RFO. The inoperability of the switch would not have prevented the automatic actuation of the components by SFAS, or the manual initiation of SFAS components from either the SFAS output modules or individual component handswitches.

C. There have been no further occurrences of handswitches failing to actuate since the above-mentioned failure in 1988. The handswitch mountings were re-checked in October, 1996, and verified to be tight. Based on this verification, the cause of the previous loose mountings is likely not a strictly time-dependent (possibly vibration induced) characteristic of the mounting screws. No further action is considered necessary under the change from an 18 month to a 24 month fuel cycle.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for for the manual actuation Channel Functional Test SR from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-SC-03110, "SFAS Channel 1 Functional Test."

ii. DBNPS Procedure DB-SC-03111, "SFAS Channel 2 Functional Test."

iii. DBNPS Procedure DB-SC-03112, "SFAS Channel 3 Functional Test."

iv. DBNPS Procedure DB-SC-03113, "SFAS Channel 4 Functional Test."

v. DBNPS Procedure DB-SC-03114, "SFAS Integrated Time Response Test."

4. Maintenance Records Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results.

The scope of the maintenance review focused on the handswitches mounted on the main control panel and the isolator relay / logic input gate circuit within each of the 68 affected output modules in the system.

B. With the exception of the handswitch failure previously identified and discussed in Section 3.B above, there were no failures.

C. No additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for for the manual actuation Channel Functional Test SR from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Maintenance Work Order Records.

ii. DBNPS Potential Condition Adverse to Quality (PCAQ) Records.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.2.1.2

1. A. Technical Specification (TS) 3/4.3.2.1, "Safety Features Actuation System Instrumentation," Surveillance Requirement (SR):

   4.3.2.1.2

   B. Systems or Components:

   Components required to add and remove Reactor Coolant System (RCS) pressure blocks, as described in further detail in Section 3.A below.

   C. Updated Safety Analysis Report (USAR) Sections:

   4.3.5.2 Safety Features Actuation System Instrumentation (SFAS)
   6.3.1.4 System Short- and Long-Term Capability
   7.3 Safety Features Actuation System (SFAS)
   7.6.1.1 Normal Decay Heat Removal Valve Control System

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.2.1.2 requires that the total bypass function shall be demonstrated Operable at least once per 18 months during Channel Calibration testing of each functional unit affected by bypass operation. Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in SR 4.3.2.1.2 the words "at least once per 18 months" be replaced with "at least once per REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time $\leq$ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

   B. The design goal of the Safety Features Actuation System (SFAS) is to automatically prevent or limit fission product and energy release from the core, to isolate the containment vessel and to initiate the operation of the Engineered Safety Features (ESF) equipment in the event of a loss of coolant accident (LOCA). The

SFAS will automatically sequence the protective action by loading equipment in steps to the Emergency Diesel Generators (EDGs) if normal or reserve power is not available to the 4.16kV essential bus(es) coincident with an SFAS initiation signal. As described in USAR Section 6.3.1.4, the Emergency Core Cooling System (ECCS) design basis assumes simultaneous loss of normal and reserve (offsite) power with a LOCA.

As described in USAR Section 7.3, the SFAS consists of four identical redundant sensing and logic channels and two identical redundant actuation channels. Each sensing channel includes analog circuits with analog isolation devices and each logic channel includes trip bistable modules with digital (opto-electronic) isolation devices. The isolated output of the trip bistable module is used to comprise coincidence matrices with the terminating relays within the actuation channel of the SFAS. These opto-electronic isolation devices also provide isolation between channels. The trip bistables monitor the station variables and normally feed continuous electrical signals into two-out-of-four coincidence matrices. Should any of the station variables exceed their trip setpoints, the corresponding bistables in each of the four channels will trip and cease sending output signals. Should two of the four channel bistables monitoring the same station variable cease to send output signals, the corresponding normally energized terminating relays on all channels will trip. The terminating relays of sensing and logic Channels 1 and 3, must both be deenergized to activate safety actuation Channel 1. Similarly, sensing and logic Channels 2 and 4 are deenergized to activate actuation Channel 2. The terminating relays act on the actuation control devices such as motor controllers and solenoid valves.

Each sensing and logic channel of SFAS includes two operating or total bypasses, one for the Reactor Coolant System (RCS) Pressure - Low signal, and the other for the RCS Pressure - Low-Low signal. These bypasses allow depressurization of the RCS without initiating the RCS pressure trips. The bypasses consist of eight push-buttons located at the main control console, two for each channel, and related sensing channel components. These bypasses can only be actuated manually and only when the RCS pressure is below 1800 psig or 600 psig respectively. The bypasses are automatically reset before the Reactor Coolant pressure exceeds 1800 psig or 600 psig, respectively.

A minimum of three of the four push-buttons of the RCS Pressure - Low or RCS Pressure - Low-Low bypasses must be actuated to effectively bypass the Reactor Coolant pressure trips. Indications that bypassing is permissible and that bypassing has been effected are provided at the main control console, in the SFAS cabinets, and at the station computer and station annunciator windows.

The SFAS is not an initiator, nor a contributor, to the initiation of an accident described in the USAR. The SFAS equipment is designed to allow single failure without preventing the system from performing the required operation.

C. The current surveillance interval of 18 months was based on the guidance of NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," during the initial licensing of the DBNPS. As discussed above, the proposed change follows the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis of the SFAS total bypass function will not be invalidated by increasing the surveillance interval for SR 4.3.2.1.2 from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. USAR Section 6.3.1.4, "System Short- and Long-Term Capability," through Revision 19.

v. USAR Section 7.3, "Safety Features Actuation System (SFAS)," through Revision 19.

3. Surveillance Data Review:

A. The 18 month surveillance test results data for SR 4.3.2.1.2 were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results.

The scope of this SR includes the components required to add the two RCS pressure blocks (in each SFAS channel) and to automatically remove the blocks prior to reaching the TS Table 3.3-3 specified values of 1800 or 600 psig, as appropriate. Except for the pressure transmitters, these components (consisting of an analog amplifier and two trip and two block bistables in each channel) are checked for the proper setpoint (as measured by a test input voltage) and proper capability of adding and automatically clearing the blocks in the monthly functional tests, DBNPS Procedures DB-SC-03110 through 03113. The 18 month calibration procedures, DB-MI-03131 through 3134 verify the setpoints from the transmitter.

On-line functional tests verify the operation of every aspect of the function except for the transmitter signal, which is assessed shiftly via Channel Checks.

B. There were five tests where RCS Pressure - Low reset values failed to meet their acceptance criteria in one of the four channels:

Channel 1 - November, 1988 (5RFO)
June, 1990 (6RFO)

Channel 2 - May, 1990 (6RFO)

Channel 4 - October, 1988 (5RFO)
September, 1991 (7RFO)

In addition, there were two tests where RCS Pressure - Low-Low reset values failed to meet their acceptance criteria:

Channel 1 - November, 1988 (5RFO)
June, 1990 (6RFO)

Subsequent investigation revealed that the block bistable field settings had tolerances that allowed them to be left at the TS limit. As a result, any drift in the nonconservative direction for the instrument string could result in the test acceptance criteria being violated. To address this cause, in November 1994, during 9RFO, the bistable field setpoints were lowered. Since this time, no further failures have occurred.

C. Based on a review of the 18 month surveillance test results data, and the action implemented during 9RFO, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical performance of these components (taking into account the one type of problem identified and the corrective actions taken), the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.3.2.1.2 from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-SC-03110, "SFAS Channel 1 Functional Test."

ii. DBNPS Procedure DB-SC-03111, "SFAS Channel 2 Functional Test."

iii. DBNPS Procedure DB-SC-03112, "SFAS Channel 3 Functional Test."

iv. DBNPS Procedure DB-SC-03113, "SFAS Channel 4 Functional Test."

v. DBNPS Procedure DB-MI-03131, "Channel Calibration of 64B-ISPRC02B4 Reactor Coolant Loop 1 Hot Leg Wide Range Pressure SFAS Channel 1."

vi. DBNPS Procedure DB-MI-03132, "Channel Calibration of 64B-ISPRC02A4 Reactor Coolant Loop 2 Hot Leg Wide Range Pressure SFAS Channel 2."

vii. DBNPS Procedure DB-MI-03133, "Channel Calibration of 64B-ISPRC02B3 Reactor Coolant Loop 1 Hot Leg Wide Range Pressure SFAS Channel 3."

viii. DBNPS Procedure DB-MI-03134, "Channel Calibration of 64B-ISPRC02A3 Reactor Coolant Loop 2 Hot Leg Wide Range Pressure SFAS Channel 4."

ix. DBNPS Potential Condition Adverse to Quality (PCAQ) Records.

x. Request for Assistance (RFA) 94-0365.

xi. DBNPS Calculation C-ICE-048.01-002, "SFAS Reactor Coolant Pressure Actuation Setpoint."

xii. Instrument Drift Study Summary for SFAS RCS Pressure Low (and Low Low) Trip Function.

xiii. Setpoint Change Request (SCR) 94-5007.

4. Maintenance Records Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results.

The logic for the bypasses is tested via monthly on-line tests. Since these tests are conducted on a more frequent basis than a refueling interval, the length of the refueling interval has no impact. Any problems identified in the monthly on-line tests would be addressed under the TS Action statement. Therefore, relative to the 24 month cycle conversion, the scope of the maintenance records review focused on the components which are not checked via the monthly on-line tests; i.e. the four transmitters.

Other than o-ring and transmitter replacement preventive maintenance activities due to the equipment qualification program, there are no normally scheduled 18 month maintenance activities for these transmitters.

B. In October, 1991, during 7RFO, pressure transmitter PTRC02B3 was replaced due to it reading low throughout the previous operating cycle, as identified during shiftly checks. As-found data was not taken with this "failed transmitter", however, it is likely that the TS limits for the blocks would not have been met for this single channel. Based on engineering review, this occurrence was considered not to be related to the length of the test interval and it is noted that the pressure transmitter failure was discovered on-line, not during surveillance testing or maintenance activities. The other three SFAS channels were not affected by this occurrence and would have functioned as designed within TS limits. Therefore, the blocks would have performed their function.

C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical performance of these components (taking into account the corrective actions taken), the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.3.2.1.2 from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Maintenance Work Order Records.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.2.1.3

1. A. Technical Specification (TS) 3/4.3.2.1, "Safety Features Actuation System Instrumentation," Surveillance Requirement (SR):

   4.3.2.1.3

   B. Systems or Components:

   Safety Features System components with response times as specified in TS Table 3.3-5.

   C. Updated Safety Analysis Report (USAR) Sections:

   4.3.5.2 Safety Features Actuation System Instrumentation (SFAS)
   6.3.1.4 System Short- and Long-Term Capability
   7.3 Safety Features Actuation System (SFAS)
   7.6.1.1 Normal Decay Heat Removal Valve Control System

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.2.1.3 requires that the Safety Features Response Time of each Safety Features Actuation System (SFAS) function be demonstrated within the limit at least once per 18 months. Surveillance Requirement 4.3.2.1.3 also specifies that each test shall include at least one functional unit per function such that all functional units are tested at least once every N times 18 months where N is the total number of redundant functional units in a specific SFAS function. Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in SR 4.3.2.1.3 the words "at least once per 18 months" be replaced with "at least once per REFUELING INTERVAL," and the words "once every N times 18 months" be replaced with "once every N times the REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time ≤ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

B. The design goal of the SFAS is to automatically prevent or limit fission product and energy release from the core, to isolate the containment vessel and to initiate the operation of the Engineered Safety Features (ESF) equipment in the event of a loss of coolant accident (LOCA). The SFAS will automatically sequence the protective action by loading equipment in steps to the Emergency Diesel Generators (EDGs) if normal or reserve power is not available to the 4.16kV essential bus(es) coincident with an SFAS initiation signal. As described USAR Section 6.3.1.4, the Emergency Core Cooling System (ECCS) design basis assumes simultaneous loss of normal and reserve (offsite) power with a LOCA.

As described in USAR Section 7.3, the SFAS consists of four identical redundant sensing and logic channels and two identical redundant actuation channels. Each sensing channel includes analog circuits with analog isolation devices and each logic channel includes trip bistable modules with digital (opto-electronic) isolation devices. The isolated output of the trip bistable module is used to comprise coincidence matrices with the terminating relays within the actuation channel of the SFAS. These opto-electronic isolation devices also provide isolation between channels. The trip bistables monitor the station variables and normally feed continuous electrical signals into two-out-of-four coincidence matrices. Should any of the station variables exceed their trip setpoints, the corresponding bistables in each of the four channels will trip and cease sending output signals. Should two of the four channel bistables monitoring the same station variable cease to send output signals, the corresponding normally energized terminating relays on all channels will trip. The terminating relays of sensing and logic Channels 1 and 3, must both be deenergized to activate safety actuation Channel 1. Similarly, sensing and logic Channels 2 and 4 are deenergized to activate actuation Channel 2. The terminating relays act on the actuation control devices such as motor controllers and solenoid valves.

Technical Specification Definition 1.26 defines Safety Feature Response Time as that time interval from when the monitored parameter exceeds its SFAS actuation setpoint at the channel sensor until the safety features equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

As stated in TS Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, the measurement of SFAS Response Time at the specified frequencies provides assurance that the SFAS action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

As further stated in the TS Bases, 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.

C. The current surveillance interval of 18 months was based on the guidance of NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," during the initial licensing of the DBNPS. As discussed above, the proposed change follows the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis of the SFAS response time SR will not be invalidated by increasing the surveillance interval for SR 4.3.2.1.3 from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. USAR Section 6.3.1.4, "System Short- and Long-Term Capability," through Revision 19.

v. USAR Section 7.3, "Safety Features Actuation System (SFAS)," through Revision 19.

3. Surveillance Data Review:

A. The 18 month surveillance test results data for SR 4.3.2.1.3 were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results.

The scope of this SR includes all sensors, logic components, and actuated equipment having a required response time within TS Table 3.3-5.

B. In November, 1994, during 9RFO, a test deficiency was identified during performance of DBNPS Procedure DB-SC-03114, "SFAS Integrated Time Response Test." The response times for ECCS Room Isolation Damper Operator HV5716A and Control Room Pneumatic Operator HV5716B, were found to be substantially above the expected time based on previous test results, although still within the TS requirement. Subsequent investigation revealed that the dampers were binding, apparently due to a lack of periodic stroking. As corrective action, a requirement to stroke the dampers was added as a quarterly periodic test (DBNPS Procedure DB-SS-04046).

No other test deficiencies which could potentially impact the value of the response times were noted for performance of procedure DB-SC-03114. Test deficiencies in which components were prevented from actuating due to reasons unrelated to the response time of the components were outside the scope of this license amendment request. These types of deficiencies are considered in the surveillance and maintenance data reviews of the license amendment requests for the associated component. These types of problems would continue to be detected and corrected regardless of the response time surveillance test frequency.

C. Based on a review of the 18 month surveillance test results data, no further actions are necessary or recommended to support this increase in the present surveillance interval. The test deficiency, as described above for HV5716A/B, was the only equipment problem related to response time degradation found during the review. This problem was resolved to avoid recurrence in the future.

It is also significant to note that many of the components' performance characteristics that could lead to response time degradation are checked frequently via other surveillance testing. This provides additional opportunities to detect and correct problems which could affect response time.

D. Based on the lack of response time degradation found during these reviews, a minimal number of test deficiencies, the low potential for increases in failure rates under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to extend the surveillance interval for SR 4.3.2.1.3 from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-SC-03114, "SFAS Integrated Time Response Test."

ii. DBNPS Procedure DB-SC-03109, "SFAS Overall Response Time Calculation."

iii. DBNPS Procedure DB-MI-03115, "Response Time Test of 59A-ISP2000 Containment Pressure SFAS Channel 1."

iv. DBNPS Procedure DB-MI-03116, "Response Time Test of 59A-ISP2001 Containment Pressure SFAS Channel 2."

v. DBNPS Procedure DB-MI-03117, "Response Time Test of 59A-ISP2002 Containment Pressure SFAS Channel 3."

vi. DBNPS Procedure DB-MI-03118, "Response Time Test of 59A-ISP2003 Containment Pressure SFAS Channel 4."

vii. DBNPS Procedure DB-MI-03125, "Response Time Test of 79A-ISR2004 Containment Radiation SFAS Channel 1."

viii. DBNPS Procedure DB-MI-03126, "Response Time Test of 79A-ISR2005 Containment Radiation SFAS Channel 2."

ix. DBNPS Procedure DB-MI-03127, "Response Time Test of 79A-ISR2006 Containment Radiation SFAS Channel 3."

x. DBNPS Procedure DB-MI-03128, "Response Time Test of 79A-ISR2007 Containment Radiation SFAS Channel 4."

xi. DBNPS Procedure DB-MI-03135, "Response Time Test of 64B-ISPRC02B4 Reactor Coolant Loop 1 Hot Leg Wide Range Pressure SFAS Channel 1."

xii. DBNPS Procedure DB-MI-03136, "Response Time Test of 64B-ISPRC02A4 Reactor Coolant Loop 2 Hot Leg Wide Range Pressure SFAS Channel 2."

xiii. DBNPS Procedure DB-MI-03137, "Response Time Test of 64B-ISPRC02B3 Reactor Coolant Loop 1 Hot Leg Wide Range Pressure SFAS Channel 3."

xiv. DBNPS Procedure DB-MI-03138, "Response Time Test of 64B-ISPRC02A3 Reactor Coolant Loop 2 Hot Leg Wide Range Pressure SFAS Channel 4."

xv. DBNPS Procedure DB-SS-04046, "ECCS Room Isolation Damper Quarterly Test."

xvi. DBNPS USAR Table 7.3-2, "Periodic Tests on SFAS and Actuated Equipment," through Revision 19.

4. Maintenance Records Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results.

This maintenance review covered actuated equipment which are response time tested at a frequency less than that required by Inservice Inspection and Testing Program requirements of the ASME Boiler and Pressure Vessel Code (reference TS 4.0.5). Any response time problems associated with this more frequently tested equipment will continue to be detected by the Inservice Inspection and Testing Program and will not be affected by the less frequent testing for SR 4.3.2.1.3.

B. The problems identified during this maintenance records review as potentially affecting component response time (other than the complete failure of the component to actuate, which would be found by other more frequent checks) are described as follows:

1. Control Room Isolation Damper, HV5311E

During 5RFO (1988), damper HV5311E was found to have a slower than normal closure time (7.32 seconds) which, when combined with the SFAS Logic Actuation Time, would have resulted in exceeding the required response time of 10 seconds required for the Control Room Heating, Ventilation and Air Conditioning Units by TS Table 3.3-5. The root cause was determined to be excessive dirt. The damper was cleaned and lubricated and was successfully retested. A preventive maintenance (PM) activity was put in place to clean, lubricate, check for air leakage, and ensure smooth stroking of the damper during the fuel cycle. This problem has not recurred, and this condition would not be affected by increasing the fuel cycle from 18 to 24 months.

2. Emergency Diesel Generator (EDG) 1

The EDG-1 response time, inclusive of output breaker closure but exclusive of SFAS Actuation logic time, slowly lengthened from approximately 9 seconds in October, 1988 (5RFO), to approximately 10 seconds in May 1990 (6RFO) and October, 1991 (7RFO), and to approximately 11 seconds in April, 1993 (8RFO) and November, 1994 (9RFO). After a generator inspection and major rebuild during 10RFO, it was determined that the likely cause was a combination of a 1988 over-excitation event, poor initial insulation, and age. The rebuild lowered the response time to approximately 7 seconds. The TS response time limit of 15 seconds (inclusive of sensor/logic response time) was never exceeded. This type of problem, as demonstrated above, was identified well before the TS limit was reached and would also be identified early under a 24 month fuel cycle. Therefore, increasing the fuel cycle to 24 months would have no adverse effect.

C. Based on a review of the 18 month maintenance records, no additional actions, other than those described above, are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical performance of these components (taking into account the corrective actions taken), the low potential for an increase in failure rates under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to extend the surveillance interval for SR 4.3.2.1.3 from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS ASME "Pump and Valve Basis Document, Vol. III."

ii. DBNPS Maintenance Work Order Records.

iii. DBNPS Potential Condition Adverse to Quality (PCAQ) Records.

iv. DBNPS USAR Table 7.3-2, "Periodic Tests on SFAS and Actuated Equipment," through Revision 19.

v. DBNPS Procedure DB-PF-00002, "Preventive Maintenance Program."

vi. DBNPS Procedure DB-SS-04046, "ECCS Room Isolation Damper Quarterly Test."

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.2.2.1, Table 4.3-11, Functional Unit 2

1. A. Technical Specification (TS) 3/4.3.2.2, "Steam and Feedwater Rupture Control System Instrumentation," Surveillance Requirement (SR):

   4.3.2.2.1, Table 4.3-11, Functional Unit 2, Manual Actuation, Channel Functional Test

   B. Systems or Components:

   SFRCS Manual Actuation Switches

   C. Updated Safety Analysis Report (USAR) Sections:

   5.5.2 Steam Generators
   7.4.1.3 Steam and Feedwater Line Rupture Control System (SFRCS)
   7.4.2.3 Steam and Feedwater Line Rupture Control System (SFRCS)
   10.3 Main Steam Supply System

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.2.2.1 requires that a Channel Functional Test be performed for the Steam and Feedwater Rupture Control System (SFRCS) instrument channels at the frequency shown in TS Table 4.3-11. Table 4.3-11 specifies a frequency of "R" for Channel Functional Testing of Manual Actuation. The present TS Definition Table 1.2 defines Notation "R" as a frequency of "At least once per 18 months." Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   This License Amendment Request addresses in its main body a proposed change to TS Table 1.2 which would redefine Notation "R" to be a frequency of "At least once per 24 months." This change would result in increasing the surveillance interval for SFRCS Manual Actuation Channel Functional Testing to "At least once per 24 months." This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

   B. The design goal of the SFRCS is to limit release of high energy steam, to automatically start the Auxiliary Feedwater System in the event of a main steam line or main feedwater line rupture, to automatically start the Auxiliary Feedwater System on the loss of both main feed pumps or the loss of all four Reactor Coolant pumps, and to prevent steam generator overfill and subsequent spillover into the main steam lines. The SFRCS also provides a trip signal to the Anticipatory Reactor Trip System.

The SFRCS consists of two identical redundant and independent channels. Each channel consists of two logic trains. The logic trains are identical and are maintained separate and independent.

Manual Actuation of SFRCS is accomplished via four push-button switches (two per channel) located on the main control board. The function of each of these switches is as follows:

1a. Initiate Auxiliary Feedwater (AFW) flow from SFRCS actuation Channel 1 taking steam from steam generator (SG) 1 and providing flow to SG 1.

1b. Initiate AFW flow from SFRCS actuation Channel 2 taking steam from SG 2 and providing flow to SG 2.

2a. Provide AFW flow as described in 1a and, in addition, isolate SG 1.

2b. Provide AFW flow as described in 1b and, in addition, isolate SG 2.

The SFRCS is not an initiator, nor a contributor, to the initiation of an accident described in the USAR. The SFRCS and SFRCS actuated equipment are designed to allow single failure without preventing the system from performing its required function. The SFRCS is designed to withstand physical damage or loss of function caused by earthquakes. The SFRCS will mitigate the consequences of an accident by initiating Auxiliary Feedwater, or isolating a ruptured steam generator and redirecting Auxiliary Feedwater flow to the intact steam generator.

C. The original Standard Technical Specifications upon which the DBNPS Technical Specifications were modeled, NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," did not include a comparable surveillance requirement for SFRCS Manual Actuation. The current surveillance interval of 18 months for SFRCS Manual Actuation Channel Functional Testing was included in the original operating license and Technical Specifications issued for the DBNPS, dated April 22, 1977, and was similar to other instrumentation surveillance frequencies performed on an 18 month refueling basis. The proposed changes follow the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis of the SFRCS Manual Actuation switches Channel Functional Test will not be invalidated by increasing the surveillance interval from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. NUREG-0136, Safety Evaluation Report for The Davis-Besse Nuclear Power Station, Unit 1, dated December 1976 and Supplement No.1.

v. DBNPS Operating License Appendix A Technical Specifications dated April 22, 1977.

vi. DBNPS USAR Section 7.4.1.3, "Steam and Feedwater Line Rupture Control System (SFRCS)," through Revision 19.

vii. DBNPS USAR Section 7.4.2.3, "Steam and Feedwater Line Rupture Control System (SFRCS)," through Revision 19.

3. Surveillance Data Review:

A. The 18 month surveillance test results data for the SFRCS manual actuation switches were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results. In addition, this time period provides representative data for the new SFRCS control system which was installed in 1988.

B. The test results indicate no failures over this time period for the components.

C. Based on a review of the 18 month surveillance test results data, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to extend the surveillance interval from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-SC-03261, "Integrated Test of SFRCS Actuation Channel 1."

ii. DBNPS Procedure DB-SC-03262, "Integrated Test of SFRCS Actuation Channel 2."

4. Maintenance Records Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results. In addition, this time period provides representative data for the new SFRCS control system which was installed in 1988.

B. The maintenance records review concerned items that may be affected by going from an 18 to 24 month refueling cycle. This review searched for failures that were identified during outage maintenance and testing, taking into account that any failure that could be identified while the plant is on-line will continue to be identified in a timely manner regardless of the refueling frequency.

The SFRCS had no refueling outage failures of the manual actuation function during this time period.

C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to extend the surveillance interval from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Maintenance Work Order Records.

ii. DBNPS Potential Condition Adverse to Quality (PCAQ) Records.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.2.2.2

1. A. Technical Specification (TS): 3/4.3.2.2, "Steam and Feedwater Rupture Control System Instrumentation," Surveillance Requirement (SR):

   4.3.2.2.2

   B. Systems or Components:

   Steam and Feedwater Rupture Control System (SFRCS) Bypasses

   C. Updated Safety Analysis Report (USAR) Sections:

   5.5.2 Steam Generators
   7.4.1.3 Steam and Feedwater Line Rupture Control System (SFRCS)
   7.4.2.3 Steam and Feedwater Line Rupture Control System (SFRCS)
   10.3 Main Steam Supply System

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.2.2.2 requires that the Steam and Feedwater Rupture Control System (SFRCS) total bypass function be demonstrated Operable at least once per 18 months during Channel Calibration testing of each channel affected by bypass operation. Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in SR 4.3.2.2.2 the words "at least once per 18 months" be replaced with "at least once per REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time $\leq$ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

   B. The design goal of the SFRCS is to limit release of high energy steam, to automatically start the Auxiliary Feedwater System in the event of a main steam line or main feedwater line rupture, to automatically start the Auxiliary Feedwater System on the loss of both main feed pumps or the loss of all four Reactor Coolant pumps, and to prevent steam generator overfill and subsequent spillover into the main steam lines. The SFRCS also provides a trip signal to the Anticipatory Reactor Trip System.

The SFRCS consists of two identical redundant and independent channels. Each channel consists of two logic trains. The logic trains are identical and are maintained separate and independent.

The operating or total bypass function of SFRCS is a two-out-of-two logic provided to allow bypassing each SFRCS channel to prevent initiation under normal cooldown conditions when the main steam line pressure drops below the SFRCS setpoint. The channel bypasses are automatically reset by a one-out-of-two logic before the main steam line pressure exceeds the setpoint value.

The SFRCS is not an initiator, nor a contributor, to the initiation of an accident described in the USAR. The SFRCS and SFRCS actuated equipment are designed to allow single failure without preventing the system from performing the required operation. The SFRCS is designed to withstand physical damage or loss of function caused by earthquakes. The SFRCS will mitigate the consequences of an accident by initiating Auxiliary Feedwater, or isolating a ruptured steam generator and redirecting Auxiliary Feedwater flow to the intact steam generator.

C. The original Standard Technical Specifications upon which the DBNPS Technical Specifications were modeled, NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," did not include a comparable surveillance requirement for the SFRCS total bypass function. The current surveillance interval of 18 months for SFRCS total bypass testing was included in the original operating license and Technical Specifications issued for the DBNPS, dated April 22, 1977, and was similar to other instrumentation surveillance frequencies performed on an 18 month refueling basis. The proposed changes follow the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis of the SFRCS total bypass function will not be invalidated by increasing the surveillance interval for SR 4.3.2.2.2 from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. NUREG-0136, Safety Evaluation Report for The Davis-Besse Nuclear Power Station, Unit 1, dated December 1976 and Supplement No.1.

v. DBNPS Operating License Appendix A Technical Specifications dated April 22, 1977.

vi. DBNPS USAR Section 7.4.1.3, "Steam and Feedwater Line Rupture Control System (SFRCS)," through Revision 19.

vii. DBNPS USAR Section 7.4.2.3, "Steam and Feedwater Line Rupture Control System (SFRCS)," through Revision 19.

3. Surveillance Data Review:

A. The 18 month surveillance test results data for SR 4.3.2.2.2 were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results. In addition, this time period provides representative data for the new SFRCS control system which was installed in 1988.

Sensor and logic testing is performed under channel functional testing. Since the channel functional tests are performed on-line monthly, they are not affected by an increase in the operating cycle length, and hence were not included in the scope of the surveillance data review.

B. The test results indicate no failures over this time period for the components.

C. Based on a review of the 18 month surveillance test results data, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to extend the surveillance interval from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-SC-03261, "Integrated Test of SFRCS Actuation Channel 1."

ii. DBNPS Procedure DB-SC-03262, "Integrated Test of SFRCS Actuation Channel 2."

iii. DBNPS Procedure DB-MI-03201, "Channel Functional Test and Calibration of SFRCS Actuation Channel 1 Pressure Inputs PS-3689B, PS-3689D, PS-3689F, PS-3689H, PS-3689K, PS-3689L, PS-3689M, and PS-3689N."

iv. DBNPS Procedure DB-MI-03202, "Channel Functional Test and Calibration of SFRCS Actuation Channel 2 Pressure Inputs PS-3687A, PS-3687C, PS-3687E, PS-3687G, PS-3687K, PS-3687L, PS-3687M, and PS-3687N."

v. DBNPS Procedure DB-MI-03209, "Channel Functional Test of SFRCS Actuation Channel 1 Logic."

vi. DBNPS Procedure DB-MI-03210, "Channel Functional Test of SFRCS Actuation Channel 2 Logic."

vii. DBNPS Procedure DB-MI-03211, "Channel Functional Test of SFRCS Actuation Channel 1 Logic for Mode 1."

viii. DBNPS Procedure DB-MI-03212, "Channel Functional Test of SFRCS Actuation Channel 2 Logic for Mode 1."

4. Maintenance Records Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results. In addition, this time period provides representative data for the new SFRCS control system which was installed in 1988.

B. The maintenance records review concerned items that may be affected by going from an 18 to 24 month refueling cycle. This review searched for failures that were identified during outage maintenance and testing, taking into account that any failures that could be identified while the plant is on-line will continue to be identified in a timely manner regardless of the refueling frequency.

The SFRCS and its actuated devices had no refueling outage failures of the bypass function during this time period.

C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to extend the surveillance interval from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Maintenance Work Order Records.

ii. DBNPS Potential Condition Adverse to Quality (PCAQ) Records.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.2.2.3

1. A. Technical Specification (TS): 3/4.3.2.2, "Steam and Feedwater Rupture Control System Instrumentation," Surveillance Requirement (SR):

   4.3.2.2.3

   B. Systems or Components:

   Steam and Feedwater Rupture Control System instrumentation and components required for actuation as specified in TS Table 3.3-13.

   C. Updated Safety Analysis Report (USAR) Sections:

   5.5.2 Steam Generators
   7.4.1.3 Steam and Feedwater Line Rupture Control System (SFRCS)
   7.4.2.3 Steam and Feedwater Line Rupture Control System (SFRCS)
   10.3 Main Steam Supply System

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.2.2.3 requires that the Steam and Feedwater Rupture Control System (SFRCS) Response Time of each SFRCS function be demonstrated to be within the limit at least once per 18 months. Surveillance Requirement 4.3.2.2.3 also specifies that each test shall include at least 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 SFRCS function. Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in SR 4.3.2.2.3 the words "at least once per 18 months" be replaced with "at least per REFUELING INTERVAL," and the words "once every N times 18 months" be replaced with "once every N times the REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time $\leq$ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

B. The design goal of the SFRCS is to limit release of high energy steam, to automatically start the Auxiliary Feedwater System in the event of a main steam line or main feedwater line rupture, to automatically start the Auxiliary Feedwater System on the loss of both main feed pumps or the loss of all four Reactor Coolant pumps, and to prevent steam generator overfill and subsequent spillover into the main steam lines. The SFRCS also provides a trip signal to the Anticipatory Reactor Trip System.

The SFRCS consists of two identical redundant and independent channels. Each channel consists of two logic trains. The logic trains are identical and are maintained separate and independent.

Technical Specification Definition 1.28 defines SFRCS Response Time as that time interval from when the monitored parameter exceeds its SFRCS actuation setpoint at the channel sensor until the equipment is capable of performing its safety function.

The measurement of response times at the indicated frequency provides assurance that the SFRCS function associated with each channel is completed within the time limits assumed in the safety analysis.

As stated in TS Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, the measurement of SFRCS response time at the specified frequencies provides assurance that the SFRCS action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

As further stated in the TS Bases, 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 SFRCS is not an initiator, nor a contributor, to the initiation of an accident described in the USAR. The SFRCS and SFRCS actuated equipment are designed to allow single failure without preventing the system from performing the required operation. The SFRCS is designed to withstand physical damage or loss of function caused by earthquakes. The SFRCS will mitigate the consequences of an accident by initiating Auxiliary Feedwater, or isolating a ruptured steam generator and redirecting Auxiliary Feedwater flow to the intact steam generator.

C. The original Standard Technical Specifications upon which the DBNPS Technical Specifications were modeled, NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," did not include a comparable surveillance requirement for SFRCS Response Time testing. The current surveillance interval of 18 months for SFRCS Response Time testing was included in the original operating license and Technical Specifications issued for the DBNPS, dated April 22, 1977, and was similar to other instrumentation surveillance frequencies performed on an 18 month refueling basis. The proposed changes follow the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis of the SFRCS Response Time testing will not be invalidated by increasing the surveillance interval for SR 4.3.2.2.3 from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. NUREG-0136, Safety Evaluation Report for The Davis-Besse Nuclear Power Station, Unit 1, dated December 1976 and Supplement No. 1.

v. DBNPS Operating License Appendix A Technical Specifications dated April 22, 1977.

vi. DBNPS USAR Section 7.4.1.3, "Steam and Feedwater Line Rupture Control System (SFRCS)," through Revision 19.

vii. DBNPS USAR Section 7.4.2.3, "Steam and Feedwater Line Rupture Control System (SFRCS)," through Revision 19.

3. Surveillance Data Review:

A. The 18 month surveillance test results data for SR 4.3.2.2.3 were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results. In addition, this time period provides representative data for the new SFRCS control system which was installed in 1988.

B. The test results indicate no failures over this time period for the components. The response time data is random, showing no adverse trend.

In June, 1990, during 6RFO, Main Steam Line #2 Isolation Valve, MS100, failed to stroke within its time limit during post-maintenance surveillance testing. The valve had been disassembled to investigate an unidentified noise, however no abnormalities were noted. The valve was successfully retested following adjustment. This failure was a result of outage activities and not associated with the length of the fuel cycle.

C. Based on a review of the 18 month surveillance test results data, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval and the introduction of no new failure modes, it is concluded that it is acceptable to extend the surveillance interval from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedures:

DB-SC-03255, "SFRCS Overall Response Time Calculation."

DB-SP-03444, "SFRCS Channel 1 Trip of MS100 and MS 101."

DB-SP-03445, "SFRCS Channel 2 Trip of MS100 and MS 101."

4. Maintenance Records Review:

   A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985, and covers five refueling outages and four operating cycles of test results. In addition, this time period provides representative data for the new SFRCS control system which was installed in 1988.

   B. The maintenance records review concerned items that may be affected by going from a 18 to 24 month refueling cycle. This review searched for failures that were identified during outage maintenance and testing which could affect SFRCS response times, taking into account that any failures that could be identified while the plant is on-line will continue to be identified in a timely manner regardless of the refueling frequency.

      The SFRCS had no refueling outage failures of response time components/strings during this period.

   C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

   D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval and the introduction of no new failure modes, it is concluded that it is acceptable to extend the surveillance interval from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

   E. References:

      i. DBNPS Maintenance Work Order Records.

      ii. DBNPS Potential Condition Adverse to Quality (PCAQ) Records.

5. Other Information:

   A. In 1990, the limitorque actuator for Service Water Strainer 1-3 Drain Valve, SW1381, failed due to spring pack hydraulic lock. Grease trapped in the cavity created by the spring cartridge cap and the outer thrust washer of the torque spring assembly caused the hydraulic lock. When grease is packed in this area of the actuator and the valve is stroked in the closed direction, the worm, spring pack and bearing cartridge stem move toward the spring cartridge cap as the valve seats. The incompressible grease prevents bearing cartridge movement and spring pack compression, resulting in failure of the torque switch to operate, sometimes stalling the motor and damaging the valve and operator.

The following valves, which are all SFRCS actuated components, were identified as also being susceptible to this type of failure:

MS603	SG 1-2 Drain Line Isolation Valve
MS611	SG 1-1 Drain Line Isolation Valve
AF3869	Auxiliary Feedwater Pump 1-1 to SG 1-2 Stop Valve
AF3870	Auxiliary Feedwater Pump 1-1 to SG 1-1 Stop Valve
AF3871	Auxiliary Feedwater Pump 1-2 to SG 1-1 Stop Valve
AF3872	Auxiliary Feedwater Pump 1-2 to SG 1-2 Stop Valve

Modifications were performed on these valves to alleviate the hydraulic lock problem by implementing a maintenance program update provided by Limitorque. A relief path was drilled inside the actuator, and the spring pack locknut and thrust washer were modified. Since completion of these modifications to address this problem, the problem has not recurred and there is no predicted adverse impact from increasing the fuel cycle length.

B. In September, 1991, during 7RFO, an air leak was discovered in solenoid valve SV100D in the air control system for Main Steam Isolation Valve (MSIV) #2, MS100. The root cause of this failure was determined to be an under-estimated normal operating temperature for the thermal life calculation for the solenoid valves, which resulted in an over-estimated qualified life for these components. The solenoid valves were replaced, and in the future will be replaced prior to the end of their environmentally qualified life. It is noted that an air leak would not prevent MS100 from fulfilling its safety function. On a loss of air, MS100 fails to its closed position. Based on these completed changes, there will be no adverse impact from increasing the fuel cycle length.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews
for Surveillance Requirement 4.3.3.5.2

1. A. Technical Specification (TS) 3/4.3.3.5.2, " Remote Shutdown Instrumentation," Surveillance Requirement (SR):

   4.3.3.5.2

   B. Systems or Components:

   Control Circuits and Transfer Switches Required for a Serious Control Room or Cable Spreading Room Fire.

   C. Updated Safety Analysis Report (USAR) Sections:

   3D.1.15 Criterion 19 - Control Room
   7.4.1.6 Auxiliary Shutdown Panel
   7.4.2.5 Auxiliary Shutdown Panel (ASP)

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.3.5.2 requires verification that at least once per 18 months each control circuit and transfer switch required for a serious control room or cable spreading room fire is capable of performing the intended function. Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in SR 4.3.3.5.2 the words "at least once per 18 months" be replaced with "at least once per REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time $\leq$ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

   B. The transfer switches are used to meet 10 CFR 50 Appendix R, Fire Protection, requirements. The switches transfer control of safe shutdown components from the control room to the local location when a serious control room or cable spreading room fire renders the control room uninhabitable and control of the components is required in order to achieve and maintain safe shutdown of the plant.

The purpose of testing the transfer switches is to verify on a periodic basis that the switches are capable of performing their intended functions. This testing demonstrates that the equipment operates from the local control station when the transfer or isolation switch is placed in the "local" position and that the equipment cannot be operated from the control room. This testing also demonstrates that the equipment operates from the control room when the transfer switch is returned to the normal position.

The control circuits and transfer switches are not an initiator, nor a contributor, to the initiation of an accident described in the USAR. The control circuits and transfer switches are not credited in the mitigation of an accident described in the USAR. However, use of the control circuits and transfer switches is credited in the DBNPS Fire Hazard Analysis Report (FHAR).

C. The original Standard Technical Specifications upon which the DBNPS Technical Specifications were modeled, NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," did not include a comparable surveillance requirement for transfer switch testing. In addition, a surveillance requirement for transfer switch testing was not included in the original operating license and Technical Specifications issued for the DBNPS, dated April 22, 1977.

On December 23, 1992, Toledo Edison submitted a license amendment application (Toledo Edison Serial Number 2101) proposing the addition of SR 4.3.3.5.2. As stated in the license amendment application, the proposed SR was based on NUREG-1430, Revision 0, dated September 28, 1992, "Improved Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," and meets the guidelines of Item 8(j) of Enclosure 1 to NRC Generic Letter 81-12, "Fire Protection Rule," dated February 20, 1981. On June 14, 1994, the NRC approved the Toledo Edison request via issuance of License Amendment Number 187 (Toledo Edison Log Number 4229).

The current 18 month surveillance interval was selected to provide flexibility in scheduling testing in those cases where shutdown conditions are required to complete testing. The proposed changes follow the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis for the control circuits and transfer switches required for a serious control room or cable spreading room fire will not be invalidated by increasing the surveillance interval for SR 4.3.3.5.2 from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. "Improved Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-1430, Revision 0, dated September 28, 1992.

v. NUREG-0136, Safety Evaluation Report for the Davis-Besse Nuclear Power Station, Unit 1, dated December 1976 and Supplement No. 1.

vi. Toledo Edison License Amendment Request dated December 23, 1992 (Toledo Edison Serial Number 2101).

vii. NRC License Amendment No. 187 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1, dated June 14, 1994 (Toledo Edison Log Number 4229).

viii. DBNPS Fire Hazard Analysis Report (FHAR), Revision 15.

ix. NRC Generic Letter 81-12, "Fire Protection Rule," dated February 20, 1981.

3. Surveillance Data Review:

A. As noted above, SR 4.3.3.5.2 was added to the TS by License Amendment 187, which was issued June 14, 1994. Amendment 187 was implemented on October 19, 1994, prior to startup from 9RFO. Therefore, the completed testing history for most of the switches is accordingly limited to one test for each switch, with the exception of switches HIS1382B and HSICS38B, which are tested monthly in conjunction with the Auxiliary Feedwater Train surveillance testing. The 18 month surveillance test results data for the associated transfer switches were reviewed for the period of June 14, 1994 to January 1, 1996.

However, these components had, in general, been installed since the startup of the plant in 1977, and the test results demonstrate their capability to maintain good performance over periods of time greater than a 24 month fuel cycle.

B. The test results indicate that one failure occurred over this time period for the components. On October 14, 1994, during 9RFO, the 480V Essential Unit Substation E1 transfer switch at breaker BCE11 failed to operate properly during the performance of surveillance testing. With the transfer switch in its "emergency" position, breaker BCE11 failed to trip when breaker AC1CE11 was tripped. These breakers, when closed, supply power to E1 from 4160V essential bus C1. The cause of the problem was the termination lug on contact 15 interfering with the movable part of the switch contact, thus preventing contact 15 from closing properly. The lug was adjusted and the switch was successfully retested. This failure did not affect the ability of the switch to perform its specified function since the breaker BCE11 close circuit was unaffected. Further, breaker BCE11 remained capable of being manually tripped.

C. Based on the review of the 18-month surveillance test results data, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the demonstrated good performance of these components over long periods of time greater than 24 months, their reliable design, the low potential for increases in failure rates of these components under a longer test interval and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.3.3.5.2 from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedures:

DB-PF-03205, "ECCS Valves Train 1 Quarterly Test."

DB-PF-03386, "Makeup Valve Refueling Interval Test."

DB-SC-03003, "Testing of Appendix R Circuits for AC1CE11, AC1CE12, BCE11, BCE12."

DB-SC-03074, "EDG-1, ABDC1, and AC103 Appendix R Test."

DB-SP-03018, "Service Water Pump 1 Refueling Test."

DB-SP-03032, "Service Water Pump 3 Refueling Test."

DB-SP-03090, "Component Cooling Water Pump 1 Refueling Test."

DB-SP-03136, "DH Pump 1 Quarterly Pump and Valve Test."

DB-SP-03152, "AFW Train 1 Level Control, Interlock, and Flow Transmitter Test."

DB-SP-03294, "CAC-1 Monthly Test."

DB-SP-03363, "Pressurizer PORV Cycle Test."

DB-SP-03371, "Quarterly Makeup Pump 1 Inservice Test and Inspection."

4. Maintenance Record Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985 and covers five refueling outages and four operating cycles of test results.

B. The maintenance records review found only one failure. This failure was for the 480V Essential Unit Substation E1 transfer switch at breaker BCE11, which was previously discussed under Section 3.B. There have been no other reported failures of this switch or any of the other associated switches or circuits during this time period.

The basic design of these switches contributes to their reliable and dependable operation. However, two switches, HIS1382B and HSICS38B, are General Electric Type "SBM". This type of switch is used in several other applications at the DBNPS. Following the observation of degradation of the "cam follower" switch part used in some of these other applications, selected "SBM" switches were scheduled for replacement with an upgraded "SBM" switch containing a different material for the "cam follower" switch part. Switches HIS1382B and HSICS38B were replaced during 10RFO with the upgraded "SBM" switches.

C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval of SR 4.3.3.5.2 from 18 to 24 months and that there is no adverse effect on safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Maintenance Work Order Records.

ii. DBNPS Potential Condition Adverse to Quality (PCAQ) Records.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.3.3.6, Table 4.3-10, Instrument 12

1. A. Technical Specification (TS) 3/4.3.3.6, "Post-Accident Monitoring Instrumentation," Surveillance Requirement (SR):

   4.3.3.6, Table 4.3-10, Instrument 12, Channel Calibration

   Note: Channel Calibrations for other instruments in Table 4.3-10 which will be performed on a 24 month refueling basis are addressed in a separate license amendment request.

   B. Systems or Components:

   Power (Pilot) Operated Relief Valve (PORV) Block Valve Position Indication.

   C. Updated Safety Analysis Report (USAR) Sections:

5.2.2.3	Overpressure Protection
5.5.10	Pressurizer
5.6	Instrumentation Application RC System
7.7	Control Systems
7.13.3.7	PORV and Pressurizer Safety Valves Position Indicators

2. Licensing Basis Review:

   A. Technical Specification SR 4.3.3.6 requires that a Channel Calibration be performed for each post-accident monitoring instrumentation channel at the frequency shown in TS Table 4.3-10. Table 4.3-10 specifies a frequency of "R" for Channel Calibration testing of the PORV Block Valve Position Indicator (Instrument 12). Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   This License Amendment Request addresses in its main body a proposed change to TS Table 1.2 which would redefine Notation "R" to be a frequency of "At least once per 24 months." This change would result in increasing the surveillance interval for the PORV Block Valve Position Indicator Channel Calibration to "At least once per 24 months." This is consistent with the guidance provided in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

B. The PORV Block Valve is provided to isolate the PORV flowpath should the PORV be inoperable in the open position or to isolate the PORV with a leaking disk. This prevents uncontrolled depressurization and excessive leakage of reactor coolant. The PORV Block Valve position is determined by a limit switch on the valve. Position indication is provided in the control room.

The PORV Block Valve is not an initiator, nor a contributor, to the initiation of an accident described in the USAR. The PORV Blo Valve is provided to isolate the PORV flowpath, but is not credite n the mitigation of an accident described in the USAR.

C. The original Standard Technical Specifications upon which the DBNPS Technical Specifications were modeled, NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," did not include a comparable surveillance requirement for the PORV Block Valve Position Indicator Channel Calibration. In addition, such a surveillance requirement was not included in the original operating license and Technical Specifications issued for the DBNPS, dated April 22, 1977.

On September 16, 1980, Toledo Edison submitted a license amendment application (Toledo Edison Serial Number 650) proposing the addition of a Channel Calibration requirement for the PORV Block Valve Position Indication, with a refueling (18 month) frequency. These changes were submitted in accordance with "PWR Model Technical Specifications for NUREG-0578 TMI-2 Lessons Learned, Category A Items," transmitted as Enclosure 1 to the NRC letter dated July 2, 1980 (Toledo Edison Log Number 581). These Model TSs specified refueling (18 month) frequency for the PORV Block Valve Position Indication Channel Calibration. On March 24, 1981, the NRC approved the Toledo Edison request via issuance of License Amendment Number 37 (Toledo Edison Log Number 681).

The changes proposed by this license amendment application follow the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis for the PORV Block Valve Position Indicator Channel Calibration will not be invalidated by increasing the surveillance interval from 18 months to 24 months and by continuing to allow the application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. DBNPS Operating License Appendix A Technical Specifications dated April 22, 1977.

v. NUREG-0136, Safety Evaluation Report for the Davis-Besse Nuclear Power Station, Unit 1, dated December 1976 and Supplement No.1.

vi. NRC letter dated July 2, 1980 (Toledo Edison Log Number 581).

vii. Toledo Edison License Amendment Request dated September 16, 1980 (Toledo Edison Serial Number 650).

viii. NRC License Amendment No. 37 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1, dated March 24, 1981 (Toledo Edison Log Number 681).

3. Surveillance Data Review:

A. The 18 month surveillance test results data for the PORV block valve position indication were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985 and covers five refueling outages and four operating cycles of test results.

B. The test results indicate no failures over this time period for these components.

C. Based on a review of the 18 month surveillance test results, no additional actions are necessary or recommended to support the increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for the PORV Block Valve Position Indicator Channel Calibration from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

   i. DBNPS Procedure DB-MI-03742, "Pressurizer Power Relief Valve Channel Calibration Check."

   ii. DBNPS Procedure DB-SC-03180, "Remote Shutdown, Post Accident Monitoring Instrumentation Monthly Channel Check."

4. Maintenance Records Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985 and covers five refueling outages and four operating cycles of test results.

B. The maintenance records indicate no failures during this time period for these components.

C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support this increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for the PORV Block Valve Position Indicator Channel Calibration from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

   i. DBNPS Maintenance Work Order Records.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.5.1.d

1. A. Technical Specification (TS) 3/4.5.1, "Emergency Core Cooling Systems (ECCS), Core Flooding Tanks," Surveillance Requirement (SR):

   4.5.1.d

   B. Systems or Components:

   CF1A (CFT Discharge Isolation Valve)
   CF1B (CFT Discharge Isolation Valve)
   PTRC02A4 (SFAS Channel 2 Wide Range Pressure Transmitter)
   PTRC02B3 (SFAS Channel 3 Wide Range Pressure Transmitter)

   C. Updated Safety Analysis Report (USAR) Sections:

   5.2.2.3 Overpressure Protection
   6.3.2.15 Core Flooding Tank Isolation Valve Control Circuits
   7.6.1.2 Core Flood Tank Isolation Valve Control System
   7.6.2.2 Core Flood Tank Isolation Valve Control System

2. Licensing Basis Review:

   A. Technical Specification SR 4.5.1.d requires verification at least once per 18 months that each core flooding tank isolation valve opens automatically and is interlocked against closing whenever the Reactor Coolant System (RCS) pressure exceeds 800 psig. Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in SR 4.5.1.d the words "at least once per 18 months" be replaced with "at least once per REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time ≤ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

   B. The core flooding tanks are part of the Emergency Core Cooling System. Each of the two core flooding tanks, located inside containment, provide a sufficient volume of borated water to be immediately forced into the reactor vessel by a nitrogen cover-pressure in the event the Reactor Coolant System (RCS) pressure falls below the pressure of the tanks (between 575 and 625 psig). This initial surge of water into the vessel provides the initial cooling mechanism during large RCS pipe ruptures. No instrumentation actuation or operator action is required for this injection. The discharge pipe from each core flooding tank (CFT) is attached directly to a reactor vessel core flooding nozzle. Each core flooding line at the outlet of the CFT contains an electrically operated isolation valve adjacent to the tank and two in-line check valves in series. The isolation valves at the CFT outlet are open when RCS pressure is above 800 psig. During power operation, when RCS pressure is higher than the core flooding system pressure, the two check valves in the line to the CFT prevent high pressure reactor coolant from entering the CFT.

Two independent computer alarms, one for each isolation valve, are provided from contacts on the motor operator to indicate when the isolation valve is not fully open and from two-out-of-four wide-range reactor coolant pressure sensors when they sense a pressure in excess of 725 psig. Two redundant computer and station annunciator alarms are also provided, one for each isolation valve, using contacts mounted on the yoke of the isolation valve and redundant, independent reactor coolant pressure signals to indicate when the valve is not fully open and pressure is greater than 750 psig. If the isolation valve has not been opened by the operator previously, the Safety Features Actuation System (SFAS) will automatically give an opening signal to the valve at 800 psig. The isolation valve is interlocked to prevent inadvertent closing when the reactor coolant pressure is above 800 psig. The open circuit is not inhibited by this interlock.

The core flooding tank isolation valve interlock is not an initiator, nor contributor, to the initiation of an accident described in the USAR. The isolation valve interlock functions to assure that the capability of the core flood tanks to inject their full volume into the reactor during accident conditions and provide the operator with system status indication in the control room.

C. The current surveillance interval of 18 months was based on the guidance of NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," during the initial licensing of the DBNPS. As discussed above, the proposed change follows the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis of SR 4.5.1.d will not be invalidated by increasing the surveillance interval from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. NUREG-0136, Safety Evaluation Report for the Davis-Besse Nuclear Power Station, dated December 1976 and Supplement No. 1.

3. Surveillance Data Review:

A. The 18 month TS surveillance test results data for the CFT isolation valves CF1B and CF1A were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. Also, the RCS pressure input to the referenced interlock string calibration results were similarly evaluated for the period of 5RFO through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985 and covers five refueling outages and four operating cycles of test results.

B. The test results review indicated one failure over this time period for these components, and several other noteworthy issues.

While conducting routine monthly channel checks during power operations in Cycle 7, the valve CF1B interlock's RCS pressure input from pressure transmitter PTRC02B3 was observed to have drifted low. The transmitter was subsequently replaced during 7RFO (1991) and has demonstrated acceptable performance following its replacement. This failure was not considered to be related to the length of the test interval. It is noted that the failure was discovered on-line, not during refueling surveillance testing or refueling maintenance activities.

The routine 18 month calibration performed on the pressure transmitter PTRC02B3 string in November, 1988, during 5RFO, observed the "as-found" RCS pressure value slightly out of tolerance low (15 psig), in the conservative direction. The string was recalibrated successfully and based on available as-found data, has been observed to be in calibration during subsequent calibration procedures.

Additionally, two situations of minor bistable drift were experienced over the time period:

- The CF1A bistable, PSH7529A/BA213, was found slightly out of tolerance (0.001 volt) during the performance of functional test procedure DB-SC-03111 in July, 1990. The bistable was recalibrated and has been observed to function properly in subsequent testing. It should be noted that the bistable deficiency did not make the interlock inoperable and did not prevent the interlock from performing its required function.

- The CF1B bistable, PSH7539A/BA313, was replaced in November, 1988 after being observed to be slightly out of tolerance in the conservative direction. The bistable has performed satisfactorily in subsequent testing. It should be noted that the bistable deficiency did not make the interlock inoperable and did not prevent the interlock from performing its required function.

C. Based on a review of the 18 month surveillance test results data and calibration activity results, no additional actions are necessary or recommended to support the increase in the present surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.5.1.d from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-SP-03175, "Core Flood Tank Isolation Valve Interlock Test."

ii. DBNPS Procedure DB-MI-03132, "Channel Calibration of 64B-ISPRC02A4 Reactor Coolant Loop 2 Hot Leg Wide Range Pressure SFAS Channel 2."

iii. DBNPS Procedure DB-MI-03133, "Channel Calibration of 64B-ISPRC02B3 Reactor Coolant Loop 1 Hot Leg Wide Range Pressure SFAS Channel 3."

iv. DBNPS Procedure DB-SC-03111, "SFAS Channel 2 Functional Test."

v. DBNPS Potential Condition Adverse to Quality (PCAQ) Report Records.

4. Maintenance Records Review:

   A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985 and covers five refueling outages and four operating cycles of test results.

   B. The maintenance records indicate one failure (i.e., would have resulted in being TS inoperable) and several minor deficiencies during this time period for these components.

   The one failure and several minor deficiencies were previously discussed and evaluated in Section 3.B above, and found not to be impacted by an increased fuel cycle length.

   A minor deficiency with the limit switch for isolation valve CF1B, was observed in February 1992, during Cycle 8 power operations. The limit switch rotor on the motor operator was found to be slightly out of adjustment. The limit switch rotor was readjusted and has performed properly following the readjustment.

   There have been deficiencies related to valve stem packing leakage for valves CF1A and CF1B. These deficiencies have typically been found while conducting refueling outage inspections inside containment. Subsequent remedial measures (e.g., tightening packing) are typically taken during the respective outages, when possible. Engineering review of documentation related to the prior packing leakage deficiencies has concluded that the packing leakage has not adversely impacted these valves. Also, the level of the Core Flood Tanks is monitored as are leakage detection instrumentation inside containment. Therefore, based on engineering review it is concluded that there will not be any resulting adverse impact on these valves from increasing the length of the refueling cycle to 24 months.

   The RCS pressure transmitters which provide input to valves' CF1A and CF1B interlocks are periodically replaced as part of the Environmental Qualification Program: PTRC02A4 transmitter replacement is every 54 months; and for PTRC02B3, O-ring replacement is every 60 months, and transmitter replacement every 81 months. Additionally, the motor operators for valves CF1A and CF1B have 18 month refueling based preventive maintenance activities performed. Each of these preventive maintenance activities will be reviewed as part of the ongoing PM Program evaluations to address the increase in the refueling interval.

C. Based on a review of the referenced maintenance records, no additional actions are necessary or recommended to support this increase in the present refueling interval.

D. Based on the above results from an engineering review of the historical performance of these components, the low potential for increases in failure rates of these components under a longer refueling interval and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.5.1.d from 18 months to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it is acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References

i. DBNPS Maintenance Work Order Records.

ii. DBNPS Potential Condition Adverse to Quality (PCAQ) Records.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.5.2.d.2.b

1. A. Technical Specification (TS) 3/4.5.2, "Emergency Core Cooling Systems, ECCS Subsystems - $T_{avg} \geq 280°F$," Surveillance Requirement (SR):

   4.5.2.d.2.b

   Note: The other TS 3/4.5.2 Surveillance Requirements needing changes to support conversion from an 18 month to 24 month fuel cycle are addressed by separate License Amendment Requests.

   B. Systems or Components:

   DH7A (BWST Outlet Valve)
   DH7B (BWST Outlet Valve)
   DH9A (Containment Emergency Sump Valve)
   DH9B (Containment Emergency Sump Valve)

   C. Updated Safety Analysis Report (USAR) Sections:

6.2.1.3.2	Containment Pressure Transient Analysis Break Spectrum
6.2.2.2.2	Containment Spray System
6.3	Emergency Core Cooling System
7.3	Safety Features Actuation System
7.6.1.3	Containment Spray Pump Anti-Cavitation Control System
9.3.5	Decay Heat Removal System

2. Licensing Basis Review:

   A. Technical Specification SR 4.5.2.d.2.b requires verification at least once per 18 months that on a Borated Water Storage Tank (BWST) Low-Low Level interlock trip, with the motor operators for the BWST outlet isolation valves and the containment emergency sump recirculation valves energized, the BWST Outlet Valve HV-DH7A (HV-DH7B) automatically close in ≤ 75 seconds after the operator manually pushes the control switch to open the Containment Emergency Sump Valve HV-DH9A (HV-DH9B) which should be verified to open in ≤ 75 seconds. Technical Specification 4.0.2 is applicable which allows increasing the surveillance interval on a non-routine basis from 18 months to 22.5 months.

   It is proposed that in SR 4.5.2.d the words "at least once per 18 months" be replaced with "at least once per REFUELING INTERVAL." A separate License Amendment Request (LAR 95-0018; DBNPS letter Serial Number 2342) proposes that "REFUELING INTERVAL" be defined as "a period of time ≤ 730 days" for the 24 month fuel cycle. This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

B. The Borated Water Storage Tank (BWST) is located outside the containment vessel and the auxiliary building. It contains a minimum of 2600 ppm boron in solution and is used both for emergency core injection and filling the refueling canal during refueling. The BWST also supplies borated water for emergency cooling by the containment spray system, decay heat removal/low pressure injection system, and high pressure injection system. It also supplies makeup water to the spent fuel pool cooling system and can serve as source for the makeup pumps.

As stated in TS Bases 3/4.5.4, Borated Water Storage Tank, the Operability of the BWST as part of the Emergency Core Cooling System (ECCS) ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a Loss of Coolant Accident (LOCA). The limits on the BWST minimum volume and boron concentration ensure that:

1) sufficient water is available within containment to permit recirculation cooling flow to the core following manual switchover to the recirculation mode, and

2) The reactor will remain at least 1% delta-k/k subcritical in the cold condition at 70°F, xenon free, while only crediting 50% of the control rods' worth following mixing of the BWST and the Reactor Coolant System (RCS) water volumes.

These assumptions ensure that the reactor remains subcritical in the cold condition following mixing of the BWST and the RCS water volumes.

Following a LOCA, after the BWST has been exhausted, the containment vessel emergency sump will serve continuous injection of the reactor coolant, through the low pressure injection/decay heat pump, into the Reactor Coolant System. This will maintain long term core cooling by recirculating the spilled reactor coolant back to the reactor vessel and/or through the containment spray pump, into the containment vessel atmosphere to remove the heat and decrease the pressure and temperature in the containment vessel.

In the event of a LOCA, the mechanism by which the Emergency Core Cooling System operates is totally automatic from the time of the SFAS signal until the BWST is depleted. Before the BWST is depleted, the operator is required to re-energize the BWST outlet isolation valves (DH7A and DH7B) and the containment emergency sump recirculation valves (DH9A and DH9B). These four valves are de-energized during plant Modes 1 through 4 to preclude an inadvertent change of position in the event of a fire. Energization of these four valves is required in order to switchover suction from the BWST to the containment emergency sump by manual actions.

Valves DH7A and DH7B receive Safety Features Actuation System (SFAS) Level 2 and Level 5 signals. The SFAS Level 2 signal (indicative of high containment pressure or low reactor coolant system pressure) sends an open signal to these normally open valves (as well as a close signal to normally closed valves DH9A and DH9B) to ensure that the BWST is available to the High Pressure Injection Pump, Low Pressure Injection Pump and Containment Spray Pump suctions. The SFAS Level 5 signal (BWST low-low level) sends a permissive signal to allow the manual opening of valves DH9A and DH9B, thus permitting the shifting of pump suctions to the containment emergency sump. The BWST outlet valves close automatically when the sump valves open.

The BWST Low-Low Level interlock trip is not an initiator, nor contributor, to the initiation of an accident described in the USAR. The interlock functions to provide for the manual switching of Emergency Core Cooling System suction from the BWST to the containment emergency sump upon depletion of the BWST inventory.

C. The original Standard Technical Specifications upon which the DBNPS Technical Specifications were modeled, NUREG-0103, Revision 0, dated June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," did not include a comparable surveillance requirement to the present SR 4.5.2.d.2.b for the BWST Low-Low Level interlock trip. The DBNPS was originally licensed to require an automatic switchover from the BWST to the containment emergency sump during a LOCA after the BWST reaches a low level. Hence, the present SR 4.5.2.d.2.b was not included in the original operating license and Technical Specifications issued for the DBNPS, dated April 22, 1977.

   On January 5, 1981, Toledo Edison submitted a license amendment application (Toledo Edison Serial Number 675) proposing the conversion to manual switchover of the BWST to the containment emergency sump in lieu of automatic switchover. The intent of this change was to preclude potential damage due to inadvertent or premature alignment of ECCS to recirculation modes of operation. On January 24, 1981, the NRC approved the Toledo Edison request via issuance of License Amendment Number 36 (Toledo Edison Log Number 651). The January 24, 1981 NRC letter noted that Toledo Edison had agreed to submit a follow-up license amendment request proposing additional related surveillance requirements.

On February 24, 1981, Toledo Edison submitted a license amendment application (Toledo Edison Serial Number 691) proposing the addition of a surveillance requirement for the BWST Low-Low Level interlock trip, with an 18 month frequency. On June 1, 1981, the NRC approved the Toledo Edison request via issuance of License Amendment Number 40 (Toledo Edison Log Number 726).

The changes proposed by this license amendment application follow the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis for the BWST Low-Low Level interlock trip will not be invalidated by increasing the surveillance interval for SR 4.5.2.d.2.b from 18 months to 24 months and by continuing to allow application of Technical Specification 4.0.2 on a non-routine basis.

E. References:

i. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 211.

ii. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

iii. "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," NUREG-0103, Revision 0, dated June 1, 1976.

iv. DBNPS Operating License Appendix A Technical Specifications dated April 22, 1977.

v. NUREG-0136, Safety Evaluation Report for the Davis-Besse Nuclear Power Station, Unit 1, dated December 1976 and Supplement No. 1.

vi. Toledo Edison License Amendment Request dated January 5, 1981 (Toledo Edison Serial Number 675).

vii. NRC License Amendment No. 36 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1, dated January 24, 1981 (Toledo Edison Log Number 651).

viii. Toledo Edison License Amendment Request dated February 24, 1981 (Toledo Edison Serial Number 691).

ix. NRC License Amendment No. 40 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1, dated June 1, 1981 (Toledo Edison Log Number 726).

3. Surveillance Data Review:

A. The 18 month surveillance test results data for SR 4.5.2.d.2.b were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985 and covers five refueling outages and four operating cycles of test results.

B. The test results indicate no surveillance failures over this time period for these components.

C. Based on the review of the 18 month surveillance test results data, no additional actions are necessary or recommended to support the increase in the present 18 month surveillance interval.

D. Based on the historical good performance of these components, the low potential for increases in failure rates of these components under a longer test interval and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval for SR 4.5.2.d.2.b from 18 to 24 months and that there is no adverse effect on nuclear safety. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Procedure DB-SC-03114, "SFAS Integrated Response Time Tests."

ii. DBNPS Procedure DB-SP-03208, "DH7B/DH9B and CS1530 Valve Test - Train 1."

iii. DBNPS Procedure DB-SP-03209, "DH7A/DH9A and CS1531 Valve Test - Train 2."

4. Maintenance Records Review:

A. The 18 month maintenance records for the components were reviewed for the period of the Fifth Refueling Outage (5RFO) through 9RFO. This time period was selected because it reflects the major plant improvements after June 1985 and covers five refueling outages and four operating cycles of test results.

B. The maintenance records review identified the following:

In March, 1988, during 5RFO, the breaker handle on valve DH9A malfunctioned and was replaced. Based on engineering review, this occurrence was categorized as a random failure, and not an age-related failure. This problem has not recurred.

C. Based on a review of the 18 month maintenance records, no additional actions are necessary or recommended to support an increase in the present surveillance interval.

D. Based on the overall good performance of these components, the low potential for increases in failure rates, and the introduction of no new failure modes, it is concluded that it is acceptable to increase the surveillance interval from 18 months to 24 months. Furthermore, it is acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

E. References:

i. DBNPS Maintenance Work Order (MWO) Records.