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.



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

# INFORMATION ONLY

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 each REFUELING INTERVAL ~~per 18 months~~ by:

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 ≤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.

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.



Blind Note: SR 4.5.2.d.2.a, 4.5.2.e and 4.5.2.g.2 are the only portions of SR 4.5.2 addressed by this LAR. Other portions are addressed by separate LARs.

INFORMATION ONLY

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



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 each REFUELING INTERVAL ~~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 each REFUELING INTERVAL ~~per 18 months~~.



INFORMATION ONLY

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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 | ≥ 375 gpm at 400 psig* |
| Injection Leg 1-2 | ≥ 375 gpm at 400 psig* |
| Injection Leg 2-1 | ≥ 375 gpm at 400 psig* |
| Injection Leg 2-2 | ≥ 375 gpm at 400 psig* |

LPI System - Single Pump

| | |
|---|---|
| Injection Leg 1 | ≥ 2650 gpm at 100 psig** |
| Injection Leg 2 | ≥ 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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LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE decay heat (DH) pump,

b. One OPERABLE DH cooler, and

c. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) and manually transferring suction to the containment emergency sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of the DH pump, the DH cooler or the flow path from the BWST, restore at least one ECCS subsystem to OPERABLE status within one hour or maintain the Reactor Coolant System Tavg less than 280°F by use of alternate heat removal methods.

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

4.5.3 The ECCS subsystems shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.



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# INFORMATION ONLY

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two trains of auxiliary feedwater, each consisting of an auxiliary feedwater pump and associated flow path to both steam generators, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one train of auxiliary feedwater inoperable to either or both steam generator(s), restore the inoperable train to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With any Auxiliary Feed Pump Turbine Inlet Steam Pressure Interlocks inoperable, restore the inoperable interlocks to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

c. With steam generator inlet valve AF 599 or AF 608 closed, re-open the closed valve AF 599 or AF 608 within one hour or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each Auxiliary Feedwater train shall be demonstrated OPERABLE:

a. At least once per 92 days on a STAGGERED TEST BASIS by:*

1. Verifying the differential pressure of each steam turbine driven pump is greater than or equal to the required differential pressure at the specified recirculation flow rate. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

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* When conducting tests of an auxiliary feedwater train in MODES 1, 2, and 3 which require local manual realignment of valves that make the train inoperable, the Motor Driven Feedwater Pump and its associated flow paths shall be OPERABLE per Specification 3.7.1.7 during the performance of this surveillance. If the Motor Driven Feedwater Pump or an associated flow path is inoperable, a dedicated individual shall be stationed at the realigned auxiliary feedwater train's valves (in communication with the control room) able to restore the valves to normal system OPERABLE status.



INFORMATION ONLY

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each valve (power operated or automatic) in the flow path is in its correct position.

2. Verifying that all manual valves in the auxiliary feedwater pump suction and discharge lines that affect the system's capacity to deliver water to the steam generator are locked in their proper position.

3. Verifying that valves CW 196, CW 197, FW 32, FW 91 and FW 106 are closed.

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Steam and Feedwater Rupture Control System actuation test signal.

2. Verifying that each pump starts automatically upon receipt of a Steam and Feedwater Rupture Control System actuation test signal. The provisions of Specification 4.0.4 are not applicable for entry in MODE 3.

3. Verifying that there is a flow path from each auxiliary feedwater pump to both steam generators by pumping water from the Condensate Storage Tank with each pump to both steam generators.

The flow paths shall be verified by either steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication. Verification of the Auxiliary Feedwater System's flow capacity is not required.

d. The Auxiliary Feed Pump Turbine Steam Generator Level Control System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months.

e. The Auxiliary Feed Pump Suction Pressure Interlocks shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months.

f. After any modification or repair to the Auxiliary Feedwater System that could affect the system's capability to deliver water to the steam generator, the affected flow path shall be demonstrated available as follows:

1. If the modification or repair is downstream of the test flow line, each auxiliary feed pump(s) associated with the affected flow path shall pump water from the Condensate Storage Tank to the steam generator(s) associated with the affected flow path; and the flow path availability will be verified by steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication.

2. If the modification or repair is upstream of the test flow line, the auxiliary feed pump shall pump water through the Auxiliary Feedwater System to the test flow line; and the flow path availability will be verified by flow indication in the test flow line.*

This Surveillance Testing shall be performed prior to entering MODE 3 if the modification is made in MODES 4, 5 or 6. Verification of the Auxiliary Feedwater System's flow capacity is not required.

g. Following each extended cold shutdown (> 30 days in MODE 5), by:

1. Verifying that there is a flow path from each auxiliary feedwater pump to both steam generators by pumping Condensate Storage Tank water with each pump to both steam generators. The flow paths shall be verified by either steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

Verification of the Auxiliary Feedwater System's flow capacity is not required.

4.7.1.2.2 The Auxiliary Feed Pump Turbine Inlet Steam Pressure Interlocks shall be demonstrated OPERABLE when the steam line pressure is greater than 275 psig, by performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months. The CHANNEL FUNCTIONAL TEST shall be performed within 24 hours after exceeding 275 psig during each plant startup, if the test has not been performed within the last 31 days.

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* When conducting tests of an auxiliary feedwater train in MODES 1, 2, and 3 which require local manual realignment of valves that make the train inoperable, the Motor Driven Feedwater Pump and its associated flow paths shall be OPERABLE per Specification 3.7.1.7 during the performance of this surveillance. If the Motor Driven Feedwater Pump or an associated flow path is inoperable, a dedicated individual shall be stationed at the realigned auxiliary feedwater train's valves (in communication with the control room) able to restore the valves to normal system OPERABLE status.



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PLANT SYSTEMS

MOTOR DRIVEN FEEDWATER PUMP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.7 The Motor Driven Feedwater Pump and associated flow paths to the Auxiliary Feedwater System shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the Motor Drive Feedwater Pump or its associated flow paths to the Auxiliary Feedwater System inoperable, restore to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.7 The required Motor Driven Feedwater Pump and flow paths to the Auxiliary Feedwater System shall be demonstrated OPERABLE:

a. Deleted

b. At least once per 31 days by:

1. When in MODE 1 with RATED THERMAL POWER greater than 40%, verifying that each manual valve in the Motor Driven Feedwater Pump suction and discharge lines that affect the system's capability to deliver water to the steam generators is locked in its proper position.

2. When in MODE 1 with RATED THERMAL POWER greater than 40%, verifying that each power operated valve in the flow path is in its correct position.

3. When in MODE 1 at RATED THERMAL POWER equal to or less than 40% or when in MODES 2 or 3, verifying that each valve (manual or power operated) in the Motor Driven Feedwater Pump flow path is able to be positioned locally for delivering flow to the Auxiliary Feedwater System.

(Ability is demonstrated by verifying the presence of handwheels for all manual valves and the presence of either handwheels or available power supply for motor operated valves.)

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 92 days and prior to entry into MODE 3 from MODE 4 (if not performed in the past 92 days) by:*

1. Verifying proper operation of each power operated and automatic valve in the Motor Driven Feedwater Pump flow path to the Auxiliary Feedwater System.

2. Verifying the Motor Driven Feedwater Pump starts from the Control Room. **

3. Verifying proper operation of the Motor Driven Feedwater Pump.**

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

1. Verifying that there is a flow path between the Motor Driven Feedwater Pump System and the Auxiliary Feedwater System by pumping water from the Condensate Storage Tanks to the steam generators. The flow path to the steam generators shall be verified prior to entering MODE 3 from MODE 4 by either steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication. Verification of Motor Driven Feedwater Pump System flow capacity is not required.

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* If the Motor Driven Feedwater Pump cannot be tested within the time period specified, due to being aligned to the Main Feedwater System, the Surveillance Requirement shall be met within 72 hours after the Motor Driven Feedwater Pump has been aligned to the Auxiliary Feedwater System for 1 hour.

** When conducting tests of the Motor Driven Feedwater Pump System in MODE 1 greater than 40% RATED THERMAL POWER which require local manual realignment of valves that make the system inoperable, both auxiliary feedwater pumps and their associated flow paths shall be OPERABLE per Specification 3.7.1.2 during the performance of this surveillance. If one auxiliary feedwater pump or flow path is inoperable, a dedicated individual shall be stationed at the realigned Motor Driven Feedwater Pump System's valves (in communication with the control room) able to restore the valves to normal system OPERABLE status.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying proper operation of the Motor Driven Feedwater Pump lube oil interlocks.

3. Verifying proper operation of manual valves by shifting the Motor Driven Feedwater Pump between the Main Feedwater System and the Auxiliary Feedwater System.

e. After any modification or repair to the Motor Driven Feedwater Pump System that could affect the system's capability to deliver water from the Condensate Storage Tanks to the Auxiliary Feedwater System, the affected flow path shall be demonstrated available as follows:*

1. If the modification or repair is in the Auxiliary Feedwater flow path downstream of the Motor Driven Feedwater Pump test flow line tie-in, the Motor Driven Feedwater Pump shall pump water from the Condensate Storage Tanks to the Auxiliary Feedwater System and the flow path availability will be verified by either steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication.

2. If the modification or repair is upstream of the Motor Driven Feedwater Pump test flow line tie-in, the Motor Driven Feedwater Pump shall pump water from the Condensate Storage Tanks to the test flow line and the flow path availability will be verified by Motor Driven Feedwater Pump Safety Grade Flow Indication.

f. Following each extended COLD SHUTDOWN (greater than 30 days in MODE 5), by:

1. Verifying that there is a flow path between the Motor Driven Feedwater System and the Auxiliary Feedwater System by pumping water from the Condensate Storage Tanks to the steam generators. The flow path to the steam generators shall be verified prior to entering MODE 3 from MODE 4 by either steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication. Verification of Motor Driven Feedwater Pump flow capacity is not required.

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* This surveillance testing shall be performed prior to entering MODE 3 from MODE 4 if the modification is made in MODES 4, 5, or 6. Verification of the Motor Driven Feedwater Pump flow capacity is not required.



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## PLANT SYSTEMS

## 3/4.7.3 COMPONENT COOLING WATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.3.1 Two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one component cooling water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.3.1 Each component cooling water loop shall be demonstrated OPERABLE:

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

b. At least once each REFUELING INTERVAL ~~per 18 months, during shutdown~~, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on an SFAS test signal.

   2. Verifying that each component cooling water emergency pump starts automatically on an SFAS test signal.

PLANT SYSTEMS

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 Two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one service water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

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

b. At least once each REFUELING INTERVAL ~~per 18 months, during shutdown,~~ by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on an SFAS test signal.

   2. Verifying that each service water emergency pump starts automatically on an SFAS test signal.

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### BASES

### 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 ≥ 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 $\geq 7.0$, 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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BASES (Continued)

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.



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## 3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 280°F from normal operating conditions in the event of a total loss of offsite power. The OPERABILITY of the Auxiliary Feed Pump Turbine Inlet Steam Pressure Interlocks is required only for high energy line break concerns and does not affect Auxiliary Feedwater System OPERABILITY.

The Condensate Storage Tanks are the non-safety-related primary source of the water for the Auxiliary Feedwater System. When the auxiliary feedwater pumps are needed and either the Condensate Storage Tanks are not available or have been emptied by the Auxiliary Feedwater System, a safety-related transfer system transfers the suction from the Condensate Storage Tanks to the Service Water System. The Service Water System is the safety-related secondary source of the water and must be available for the associated Auxiliary Feedwater System train to be OPERABLE. The transfer is initiated upon detection of a low suction pressure at the suction of the auxiliary feedwater pumps by suction pressure interlock switches. These pressure switches, upon sensing low suction pressure, will automatically transfer the suction of the auxiliary feedwater pumps to the Service Water System. On a sustained low-low suction pressure, additional Auxiliary Feedwater Pump Suction Pressure Interlocks will operate to close the steam supply valves to protect the turbine driven auxiliary feedwater pumps from cavitation. The steam supply valves will re-open automatically upon restoration of suction pressure to the pumps. Both the low and the low-low suction Auxiliary Feed Pump Suction Pressure Interlocks are required to be OPERABLE for OPERABILITY of the associated auxiliary feedwater train.

Each steam driven auxiliary feedwater pump is capable of delivering the required feedwater flow at the full open pressure of the Main Steam Safety Valves as assumed in the Updated Safety Analysis Report. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 280°F where the Decay Heat Removal System may be placed in operation. Each train of auxiliary feedwater must be capable of providing feedwater flow to each steam generator in order to be OPERABLE. However, the design of the system does not provide for feeding both steam generators simultaneously from one train.

When conducting tests of an auxiliary feedwater train in MODES 1, 2, or 3 which require local manual realignment of valves that make the train inoperable, a dedicated individual shall be stationed at the valves, in communication with the control room, able to restore the valves to normal system OPERABLE status. However, it is not required to have this dedicated individual stationed if the other train of the Auxiliary Feedwater System is OPERABLE and the Motor Driven Feedwater Pump System is OPERABLE pursuant to Technical Specification 3/4.7.1.7 because two sources of auxiliary feedwater to the steam generators are OPERABLE. In either situation, the Auxiliary Feedwater System train with the local manual realigned valves is inoperable and the Limiting Condition for Operation ACTION must be followed.

Closure of valve AF 599 or AF 608 will render both trains of the Auxiliary Feedwater System and the Motor Driven Feedwater Pump System inoperable. This is because closure of these valves would result in a complete loss of auxiliary feedwater to the steam generators for certain postulated feedwater or steam line breaks.

INFORMATION ONLY

(3/4.7.1.2 is continued on page B 3/4 7-2.)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (Continued)

Following any modifications or repairs to the Auxiliary Feedwater System piping from the Condensate Storage Tank through auxiliary feed pumps to the steam generators that could affect the system's capability to deliver water to the steam generators, following extended cold shutdown, a flow path verification test shall be performed. This test may be conducted in MODES 4, 5 or 6 using auxiliary steam to drive the auxiliary feed pumps turbine to demonstrate that the flow path exists from the Condensate Storage Tank to the steam generators via auxiliary feed pumps.

Verification of the turbine plant cooling water valves (CW 196 and CW 197), the startup feedwater pump suction valves (FW 32 and FW 91), and the startup feedwater pump discharge valve (FW 106) in the closed position is required to address the concerns associated with potential pipe failures in the auxiliary feedwater pump rooms, that could occur during operation of the startup feedwater pump.

3/4.7.1.3 CONDENSATE STORAGE TANKS

The OPERABILITY of the Condensate Storage Tanks with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 13 hours with steam discharge to atmosphere and to cooldown the Reactor Coolant System to less than 280° under normal conditions (i.e., no loss of offsite power). The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves



INFORMATION ONLY

within the closure times of the surveillance requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY - Deleted

3/4.7.1.7 MOTOR DRIVEN FEEDWATER PUMP SYSTEM

The OPERABILITY of the Motor Driven Feedwater Pump System ensures that the Reactor Coolant System can be cooled down from normal operating conditions in the event of the total loss of Main Feedwater and Auxiliary Feedwater Pumps.

The Motor Driven Feedwater Pump System must be capable of providing feedwater flow to each steam generator in order to be OPERABLE.

The Motor Driven Feedwater Pump flow capability ensures that adequate feedwater flow is available to remove Decay Heat and reduce the Reactor Coolant System temperature to where the Decay Heat System may be placed into operation.

When conducting tests of the Motor Driven Feedwater Pump System in MODE 1 at greater than 40% RATED THERMAL POWER which requires local manual realignment of valves which make the system inoperable, a dedicated individual shall be stationed at the realigned train's valves, in communication with the control room, able to restore the valves to normal system OPERABLE status. However, it is not required to have this dedicated individual stationed if both trains of the Auxiliary Feedwater System are OPERABLE pursuant to Technical Specification 3/4.7.1.2 because two sources of auxiliary feedwater to the steam generators are OPERABLE. In either situation, the Motor Driven Feedwater Pump System with the local manual realigned valves is inoperable and the Limiting Condition for Operation ACTION must be followed.

When at 40% RATED THERMAL POWER or less and in MODES 1, 2, or 3, the Motor Driven Feedwater Pump System may be aligned to provide a flow path from the Deaerator Storage Tank through the Motor Driven Feedwater Pump to the Main Feedwater System. During this Motor Driven Feedwater Pump mode of operation, a flow path from the Condensate Storage Tanks through the Motor Driven Feedwater Pump to the Auxiliary Feedwater System shall be maintained with the ability for manual positioning of valves such that

the flow path can be established. The ability for local, manual operation is demonstrated by verifying the presence of the handwheels for all manual valves and the presence of either handwheels or available power supply for motor operated valves.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 110°F and 237 psig are based on a steam generator $RT_{NDT}$ of 40°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants" March 1974.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix "A", 10 CFR 50.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirements 4.5.2.d.2.a, 4.5.2.e, 4.5.2.g.2, and 4.5.3

1. A. Technical Specification (TS): 3/4.5.2, "Emergency Core Cooling Systems, ECCS Subsystems - Tavg $\geq$ 280°F" Surveillance Requirements (SR) 4.5.2.d.2.a, 4.5.2.e, and 4.5.2.g.2, and TS 3/4.5.3 "Emergency Core Cooling Systems, ECCS Subsystems - Tavg < 280°F" SR 4.5.3

   Note:

   The Safety Features Actuation System (SFAS) instrumentation and controls extend from the generating station variables to the input terminals of the safety features actuation control devices, such as motor controllers and solenoid valves. The subject surveillances cover actuated components which receive an SFAS signal. The surveillances do not include the SFAS instrumentation and controls. Applicable SFAS surveillances will be addressed under a separate License Amendment Request.

   B. Systems or Components:

   Emergency Core Cooling System

   C. Updated Safety Analysis Report (USAR) Section:

   6.3 Emergency Core Cooling System

2. Licensing Basis Review:

   A. Technical Specification SR 4.5.2.d.2.a requires that each ECCS subsystem be demonstrated operable at least once per 18 months by 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.

   It is proposed that in SR 4.5.2.d, the words "At least once per 18 months" be replaced with "At least once each REFUELING INTERVAL."

   Technical Specification SR 4.5.2.e requires that each ECCS subsystem be demonstrated operable at least once per eighteen months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal, and by verifying that each High Pressure Injection (HPI) and Low Pressure Injection (LPI) pump starts automatically upon receipt of a Safety Features Actuation System (SFAS) test signal.

It is proposed that in SR 4.5.2.e, the words "At least once per 18 months, during shutdown" be replaced with "At least once each REFUELING INTERVAL."

Technical Specification SR 4.5.2.g.2 requires verifying the correct position of each mechanical position stop for valves DH-14A and DH-14B at least once per 18 months.

It is proposed that in SR 4.5.2.g.2, the words "At least once per 18 months" be replaced with "At least once each 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.

The proposed changes are 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.

Technical Specification SR 4.5.3 states: "The ECCS subsystems shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2." Therefore the above referenced proposed changes to SR 4.5.2 (which is applicable in Modes 1, 2 and 3) also apply equally to SR 4.5.3 (which is applicable in Mode 4).

B. As described in USAR Section 6.3, "Emergency Core Cooling System," the ECCS is designed to mitigate the consequences of all breaks of the Reactor Coolant System (RCS) pressure boundary which result in loss of reactor coolant at a rate in excess of the capability of the Reactor Coolant Makeup System up to and including a break equivalent in area to the double-ended rupture of the largest pipe of the RCS. The ECCS is designed such that separate and independent flow paths are provided in the ECCS, and redundancy in the active components ensures that the required functions will be performed if a single active failure occurs. Separate essential power sources are supplied to the redundant active components, and separate SFAS channels are used to actuate the system.

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 Loss of Coolant Accident (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.

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.

Mechanical position stops for the Decay Heat (DH) Cooler 1-1 outlet flow control valve (DH-14B) and the DH Cooler 1-2 outlet flow control valve (DH-14A) are provided to ensure that proper ECCS flowrates will be maintained in the event of a LOCA.

The frequency at which the ECCS surveillance requirements are performed is not an initiator, nor a contributor, to the initiation of an accident described in the USAR.

C. The current surveillance interval of 18 months for SR 4.5.2.d.2.a, SR 4.5.2.e, and SR 4.5.3 was based on the guidance of NUREG-0103, Revision 0, June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," during the initial licensing of the DBNPS. The 18 month surveillance intervals were based upon an 18 month refueling frequency. It may be noted that although the DBNPS is not committed to Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," June 1974, Regulatory Position C.14 is similar to the requirements of SR 4.5.2.d.2.a. This inspection of the sump is performed ". . . every refueling period downtime," i.e., it is not associated with a particular frequency of refueling.

The original revision of the Standard Technical Specifications did not include a surveillance requirement for verifying the correct position of each mechanical position stop for ECCS throttle valves, hence the original DBNPS Technical Specifications did not include such a surveillance requirement. At the request of the NRC in NRC Letter Log Number 301, dated November 9, 1977, Toledo Edison (TE) proposed, by letter dated January 13, 1978, a license amendment request to add SR 4.5.2.g, which included a requirement to verify correct position of each mechanical position stop for valves DH-14A and DH-14B at least once per 18 months. The NRC approved this request by issuing Amendment 20 to the TS on October 2, 1979. The NRC's letter dated November 9, 1977, stated that ". . . periodic verification be made of these valve positions," and provided sample surveillances with a frequency of 18 months. The basis for the frequency of 18 months was not specifically discussed. However, 18 months was consistent with performance during each 18 month refueling interval which was similar to the existing NUREG-0103 Standard Technical Specifications.

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 will not be invalidated by increasing the surveillance interval for SR 4.5.2.d.2.a, SR 4.5.2.e, SR 4.5.2.g.2, and SR 4.5.3 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. DBNPS Technical Specifications issued with the original operating license by the NRC dated April 22, 1977.

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

v. NRC letter (Toledo Edison Log Number 301), dated November 9, 1977.

vi. Toledo Edison License Amendment Request dated January 13, 1978 (Toledo Edison Serial Number 413).

vii. Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," June 1974.

viii. NRC License Amendment No. 20 to Facility Operating License No. NPF-3, dated October 2, 1979 (Toledo Edison Log Number 441).

ix. USAR Section 6.3, "Emergency Core Cooling System," through Revision 19

3. Surveillance Data Review:

The applicable 18 month TS Surveillance Test results data 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.

Surveillance Data Review for SR 4.5.2.d.2.a:

A. The 18 month TS surveillance test results data for the containment emergency sump was reviewed for the period 5RFO through 9RFO.

B. The test results for all inspections of the containment emergency sump were satisfactory. No unsatisfactory conditions were noted.

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 no surveillance failures during the review period; the low potential for an increase in failure rates over the increased interval; and no known additional failure modes, the surveillance interval for SR 4.5.2.d.2.a can be increased from 18 months to 24 months. Furthermore, it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

Surveillance Data Review for SR 4.5.2.e.1:

A. The 18 month TS surveillance test results data for automatic valves in the flow path that actuate to a safety position on a safety injection test signal were reviewed. The following valves were reviewed for SR 4.5.2.e.1:

| | |
|---|---|
| DH7A | Borated Water Storage Tank Isolation Valve (line 2) |
| DH7B | Borated Water Storage Tank Isolation Valve (line 1) |
| DH9A | Decay Heat Pump 1-2 Suction from Emergency Sump Valve |
| DH9B | Decay Heat Pump 1-1 Suction from Emergency Sump Valve |
| DH13A | Decay Heat Cooler 1-2 Bypass Flow Control Valve |
| DH13B | Decay Heat Cooler 1-1 Bypass Flow Control Valve |
| DH14A | Decay Heat Cooler 1-2 Outlet Flow Control Valve |
| DH14B | Decay Heat Cooler 1-1 Outlet Flow Control Valve |
| DH2733 | Decay Heat Pump 1-1 Suction Valve |
| DH2734 | Decay Heat Pump 1-2 Suction Valve |
| HP2A | High Pressure Injection Line 2-1 Isolation Valve |
| HP2B | High Pressure Injection Line 2-2 Isolation Valve |
| HP2C | High Pressure Injection Line 1-1 Isolation Valve |
| HP2D | High Pressure Injection Line 1-2 Isolation Valve |

B. The test results for all automatic valves going to their safety position were satisfactory. No unsatisfactory conditions were noted. All valves except DH9A and DH9B are stroked quarterly for American Society of Mechanical Engineers (ASME) Code testing and no safety position failures were identified for this testing. No adverse trends were identified from the ASME Code testing.

It was noted that valve DH13A was documented as unable to be opened on two occasions when testing was performed. However, the safety position of valve DH13A is closed and the valve could, therefore, perform its safety function. These occurrences are further discussed under Section 4, "Maintenance Records Review," which describes a modification that has been implemented to correct this problem.

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 satisfactory results of tests performed 5RFO through 9RFO; a low potential for an increase in failure rates over the increased interval; and no known additional failure modes, the surveillance interval for SR 4.5.2.e.1 can be increased from 18 months to 24 months. Furthermore it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

<u>Surveillance Data Review for SR 4.5.2.e.2:</u>

A. The 18 month TS surveillance test results data for HPI and LPI pump automatic starts were reviewed for the period 5RFO through 9RFO.

B. The test results indicated no failures for the period 5RFO through 9RFO.

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 no failures during the review period; a low potential for an increase in failure rates over the increased interval; and no known additional failure modes, the surveillance interval for SR 4.5.2.e.2 can be increased from 18 months to 24 months. Furthermore it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

<u>Surveillance Data Review for SR 4.5.2.g.2:</u>

A. The 18 month TS surveillance test results data for valves DH14A and DH14B mechanical stop positions was reviewed for the period 5RFO through 9RFO.

B. The test results for all mechanical stop inspections for valves DH14A and DH14B were satisfactory. No unsatisfactory conditions were noted. In addition, the mechanical stops surveillance requirement is actually being fulfilled quarterly during the ASME Code valve stroke testing in accordance with SR 4.5.2.g.1.

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 no surveillance failures during the review period; a low potential for an increase in failure rates over the increased interval; and no known additional failure modes, the surveillance interval for SR 4.5.2.g.2 can be increased from 18 months to 24 months. Furthermore it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

<u>Surveillance Data Review for SR 4.5.3:</u>

Technical Specification SR 4.5.3 states: "The ECCS subsystems shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2." Therefore the above referenced Surveillance Data Review performed for SR 4.5.2 applies to SR 4.5.3.

E. References used in the Surveillance Data Review of SR 4.5.2.d.2.a, e and g.2

i. DBNPS procedure DB-SP-03134, "Containment Emergency Sump Visual Inspection"

ii. DBNPS procedure DB-SC-03114, "SFAS Integrated Response Time Tests"

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

iv. DBNPS procedure DB-PF-03206, "ECCS Valves Train 2 Quarterly Test"

v. DBNPS procedure DB-SP-03136, "Decay Heat Pump 1 Quarterly Pump and Valve Test"

4. Maintenance Records Review:

Maintenance records 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.

<u>Maintenance Records Review for SR 4.5.2.d.2.a:</u>

There are no maintenance activities applicable for the activities associated with this surveillance requirement. Therefore, a maintenance records review is not applicable.

<u>Maintenance Records Review for SR 4.5.2.e.1:</u>

A. A review was performed of maintenance records from 5RFO through 9RFO for SR 4.5.2.e.1 automatic valves.

B. Several failures and degradation were identified as discussed below. One TS inoperable condition was identified during Cycle 6 (1989) where valves HP2A, HP2B, HP2C, and HP2D would not stroke as required with a SFAS level 2 signal and a loss of offsite power. The HPI System was declared inoperable. As corrective action, time delay drop out relays were installed in the valve circuitry. After installation of the time delays, the valves were tested satisfactorily, and this problem has not recurred. This problem was a design issue and was independent of the fuel cycle length.

Valve HP2C's torque switch was replaced in 5RFO (1988), due to degradation. This degradation has not recurred and, based on engineering review, was not attributable to fuel cycle length.

Valve HP2B had a valve seat leaking by and upon disassembly in 7RFO (1991) was found to have a damaged disc in the valve. A new disc was installed. This valve is stroked quarterly under the ASME Code testing. The valve seat has not experienced leakage during the five years since replacement of the disc. This failure is not attributable to the fuel cycle length.

The manual isolation valve for the DH13A actuator has failed two times (1992 and 1995) which prevented operation of the valve. Valve DH13A is normally in its closed safety position during plant operation. When the failure was identified, Valve DH13A was in its safety position and was not affected other than the inability to stroke the valve from the control room. The valve fails to its safety position by use of a spring and does not need air for motive force to move to the safety position. The root cause of the failures was a design deficiency. Manual isolation valves used during the performance of the periodic surveillance testing would break internally resulting in a loss of air to DH13A but with no indication of the failure. Operations personnel would return the valve handle to its operating position, however, the internal portions of the valve would not move. The manual isolation valve will be replaced with a new design more appropriate for the application. This modification will be implemented during Cycle 11 (1 97). This valve is stroked quarterly under the ASME Code testing. A similar failure would be identified by the quarterly ASME Code testing regardless of the fuel cycle length.

In 5RFO, (1988) the electrical power breaker handle on valve DH9A malfunctioned and was replaced. No other similar failure has subsequently occurred on this breaker. This failure was independent of the fuel cycle length.

The failures listed above are not age related. Increasing the surveillance interval will not affect failures that are not age related, therefore, extending the surveillance interval from 18 months to 24 months will not affect failures of this type.

C. After reviewing and considering the type of failures and corrective actions discussed above, it is concluded that no additional actions are necessary or recommended to increase SR 4.5.2.e.1 from 18 to 24 months.

In addition to the TS surveillance requirements, testing on motor-operated valves is performed in accordance with Generic Letter 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance." Testing on valves HP2A through D resulted in identification of the need to increase these valves' actuator thrust. Section 5 presents a brief description of the modifications to these valves. No instances were identified that made the valve inoperable. The testing ensures the valves will perform their intended safety function.

D. Based on the past performance of these components and the actions taken to prevent recurrence of the failures described above, the low potential for increases in failure rates of these components under a longer test interval, and the introduction of no known new failure modes, it is concluded that it is acceptable to increase the surveillance interval from 18 to 24 months. Furthermore it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

Maintenance Records Review for SR 4.5.2.e.2:

A. A review was performed on refueling maintenance records that could affect HPI or LPI pump automatic starts for the period 5RFO through 9RFO.

B. In 7RFO (1991) and 8RFO (1993), breaker AD112 for LPI Pump 2 tripped during pump operation in the refueling outage. Consequently the pump was declared inoperable. Extensive testing and troubleshooting was performed. There is no evidence that the failure was age-related. No parts or components required replacement, and preventive maintenance did not identify any age-related degradation. The LPI Pump is only used during refueling, therefore, the breaker was exchanged with a breaker of the same design from a condensate pump cubicle in order to provide a longer period of continuous use. The condensate pumps are continuously running during the operating cycle. This breaker was used most of Cycle 9 for an operating condensate pump to identify any age related failure mechanisms and operated satisfactorily. The LPI pump breaker performed as required in both 9RFO and 10RFO.

A review of other maintenance activities for 5RFO through 9RFO indicates these activities would not have adversely impacted the LPI or HPI systems if performed on a 24 month fuel cycle.

C. As a result of the above review no additional actions are considered necessary to support an increase in the present TS surveillance interval from 18 months to 24 months.

D. Based on the historical good performance of the HPI and LPI pumps, the low potential for increases in failure rates of these components under a longer interval, and no known new failure modes, it is concluded that it is acceptable to increase the surveillance interval of SR 4.5.2.e.2 from 18 to 24 months. Furthermore it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

Maintenance Records Review for SR 4.5.2.g.2

A. A review was performed for the mechanical stop components for the period 5RFO through 9RFO.

B. No failures were identified for the mechanical stops.

C. As a result of the above, no additional actions are necessary to support an increase in the present TS surveillance interval from 18 months to 24 months.

D. Based on the above, the performance frequency for SR 4.5.2.g.2 can be increased from 18 months to 24 months. Furthermore it remains acceptable to allow the continued application of TS 4.0.2 on a non-routine basis.

Maintenance Records Review for SR 4.5.3:

Technical Specification SR 4.5.3 states: "The ECCS subsystems shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2." Therefore the above referenced Maintenance Records Review for SR 4.5.2 applies to SR 4.5.3.

E. References:

i. DBNPS Maintenance Work Order Records

ii. NRC Generic Letter 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance," June 28, 1989.

5. Other Information Regarding SR 4.5.2.e.1

Motor-operated valve testing identified the need to increase the actuator thrust for valves HP2A through D. This was done under a modification. A modification was also implemented in 9RFO to change the control scheme for valves HP2A through D, from torque control to limit control over 90% of the disk travel. This change complemented the modification that increased the valves' actuator thrust. These modifications provide sufficient thrust margin for all accident conditions. In addition, a torque closed feature was implemented to prevent jamming the valves' disk into the seat. These modifications provided further improvements to plant valves for even greater reliability and performance.

## Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.7.1.2.1.c

1. A. Technical Specification (TS): 3/4.7.1.2, "Auxiliary Feedwater System," Surveillance Requirement (SR) 4.7.1.2.1.c

   Note:

   The Steam and Feedwater Rupture Control System (SFRCS) instrumentation and controls extend from the generating station variables to the input terminals of the SFRCS actuation control devices, such as motor controllers and solenoid valves. The subject surveillances cover actuated components which receive an SFRCS signal. The surveillances do not include the SFRCS instrumentation and controls. Applicable SFRCS surveillances will be addressed under a separate License Amendment Request.

   B. Systems or Components:

   Auxiliary Feedwater System

   C. Updated Safety Analysis Report (USAR) Sections:
   9.2.7 Auxiliary Feedwater System
   15.2.8 Loss of Normal Feedwater

2. Licensing Basis Review:

   A. Technical Specification SR 4.7.1.2.1.c requires that each Auxiliary Feedwater train be demonstrated operable at least once per 18 months by:

      1) Verifying that each automatic valve in the flow path actuates to its correct position on a Steam and Feedwater Rupture Control System actuation test signal.

      2) Verifying that each pump starts automatically upon receipt of a Steam and Feedwater Rupture Control System actuation test signal.

      3) Verifying that there is a flow path from each auxiliary feedwater pump to both steam generators by pumping water from the Condensate Storage Tank with each pump to both steam generators. The flow paths shall be verified by either steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication. Verification of the Auxiliary Feedwater System's flow capacity is not required.

   TS 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 4.7.1.2.1.c the words "At least once per 18 months" be replaced with "At least once each 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. As described in USAR Section 9.2.7, the Auxiliary Feedwater (AFW) System is designed to provide feedwater to the steam generators when the turbine-driven main feedwater pumps are not available or following a loss of normal and reserve electric power. On station shutdown, the auxiliary feedwater pumps can be used to remove decay heat until the decay heat removal system can be placed in service. The auxiliary feedwater system consists of two steam turbine-driven feedwater pumps, condensate storage tanks, suction and discharge water piping, steam piping, valves, and associated instrumentation and controls.

As described in USAR Section 7.4.1.3, the Auxiliary Feedwater System is automatically started by the Steam and Feedwater Line Rupture Control System (SFRCS) in the event of a main steam line or main feedwater line rupture, on the loss of both main feed pumps, or on the loss of all four reactor coolant pumps.

The Auxiliary Feedwater System is not an initiator, nor a contributor to the initiation of an accident described in the USAR. The two safety-grade AFW pumps are capable of being actuated and controlled by safety-grade signals that ensure the availability of feedwater to at least one steam generator, under the assumed conditions of a single failure.

C. The current surveillance interval of 18 months was based on the guidance of NUREG-0103, Revision 0, June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," during the initial licensing of the DBNPS. 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 will not be invalidated by increasing the surveillance interval for SR 4.7.1.2.1.c 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 7.4.1.3, "Steam and Feedwater Line Rupture Control System," through Revision 19.

v. USAR Section 9.2.7, "Auxiliary Feedwater System," through Revision 19.

vi. USAR Section 15.2.8, "Loss of Normal Feedwater," through Revision 19.

3. Surveillance Data Review:

A. The 18 month TS surveillance test results data for SR 4.7.1.2.1.c were reviewed 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 indicate that no failures occurred over this time period for these components.

C. Based on a review of the 18-month surveillance test results data, no additional actions are necessary or recommended to extend the surveillance interval from 18 months to 24 months.

Certain components of the Auxiliary Feedwater System are also tested monthly under the ASME Code Section XI Inservice Test Program while the plant is on line. These tests include pump flow and pressure verifications, valve stroke time testing, and system interlock verifications.

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, the introduction of no new failure modes, and inclusion of certain components in the Inservice Test Program, it is concluded that it is acceptable to increase the surveillance interval for SR 4.7.1.2.1.c 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-SP-03155, "AFW Train 1 Flow Path To SG Verification"
DB-SP-03164, "AFW Train 2 Flow Path To SG Verification"
DB-SP-03147, "AFW Train 1 Forward Flow Check Valve Test"
DB-SP-03148, "AFW Train 2 Forward Flow Check Valve Test"
DB-SC-03261, "Integrated Test Of SFRCS Actuation CH 1"
DB-SC-03262, "Integrated Test Of SFRCS Actuation CH 2"

4. Maintenance Records Review:

A. The 18-month maintenance records for the AFW system components were reviewed 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 maintenance activities.

B. The maintenance records indicate several failures which resulted in components being TS inoperable and one instance of significant degradation during this time period for the AFW System.

In September of 1988 (5RFO), AF3869, AFW Pump 1-1 to Steam Generator 1-2 Stop Valve, failed to open due to the valve being stuck in its seat. During this event the valve motor operator torque switch pin sheared and allowed the motor to run to failure while the valve was stuck. The valve and operator were replaced. Since the replacement, no similar problems have occurred with the replacement valve. This valve is included in the DBNPS MOV program. This program made detailed reviews of each MOV's operating requirements and tested and adjusted each valve's settings for operation. This program presently requires annual inspections of the equipment and has improved the overall reliability of the valve. Additionally, this valve is currently stroked monthly while the plant is on line under the Surveillance and Test Program, which will identify any similar failures. Therefore, there is no impact from this failure on the proposed change to a 24-month fuel cycle.

In March of 1992 (Cycle 8), the valve position indication circuit for valve MS106, Main Steam Line 1 to AFW Pump Turbine 1-1 Isolation Valve, shorted while replacing the local closed lamp, which had failed. This caused a loss of control power to the valve due to a blown fuse. The fuse was replaced to restore control power. In September of that same year this local indication lamp was deleted due to High Energy Line Break concerns. In August of 1994 (Cycle 9), a similar event occurred, in that the closed indication lamp in the control room illuminated brightly, failed, and caused the power isolation fuse to open while the valve was being repositioned. This fuse and lamp were replaced. Based upon engineering review and evaluation, this event was classified as a random failure. No other occurrences of loss of control power due to indication circuit problems have occurred. Both of these failures

occurred during operating cycles and were not associated with the length of the fuel cycle. No additional risk to this circuit will occur as a result of a 24 month cycle.

A loss of valve MS106 function also occurred in April of 1993 during the 8RFO. This event was the result of improper valve operation by plant personnel. This valve was manually seated in the closed position to ensure proper isolation for work. During the restoration from this activity, the valve was not manually opened and then electrically closed, in order to prevent damaging the valve, as required by station procedures. The valve was opened electrically and the valve disk pulled from the operator shaft. The valve was repaired. Training was conducted to emphasize the need to closely follow station procedures when operating motor operated valves. No similar valve failures have occurred since this event. Valve MS106 is included in the DBNPS MOV program to help ensure its reliability. The MOV program currently requires an annual inspection of valve MS106. Additionally, this valve is currently stroked monthly while the plant is on line under the Surveillance and Test Program, which will identify future failures should they occur. The failure of valve MS106 was associated with work performed during an outage and is not attributable to fuel cycle length.

In September of 1993, the random failure of an Agastat timing relay resulted in the failure of valve SW1383, Service Water Supply to AFP 1-2, to open on low Auxiliary Feedwater Pump Number 2 suction pressure during a monthly interlock surveillance test. This relay and all other similar relays are now periodically replaced under the Preventive Maintenance Program to ensure age-related failures of these components do not recur. Additionally, the monthly surveillance testing of this circuit will discover similar failures if they occur. Since repair of this circuit, no similar failures have occurred. Therefore, there is no impact on the proposed change to a 24-month fuel cycle.

During 1993 (8RFO), the air-operated steam admission valves for the AFW Pump Turbines were replaced to improve the steam shut-off capability of these components. During this change-out, the associated air regulators were also replaced with a different style. The air regulator for AFW Train 1 failed high in May of 1994. This resulted in a degradation of the train of AFW since the increased air pressure slowed down the response time of the train by increasing the stroke time of the steam admission valve. The style of the air regulator was changed from a plastic to a metal diaphragm to improve the temperature sensitivity of this component. The Train 1 and 2 regulators were replaced. Surveillance testing performed in accordance with SR 4.7.1.2.1.a on a 92-day staggered test basis would provide the opportunity to discover similar failures if they occurred. Also the operators currently record the valve diaphragm pressures on a daily basis. A narrow band of acceptability has been established. Readings that fall outside the band would be discovered. These components are periodically replaced under the Preventive Maintenance Program. No similar problems have occurred since this event and no additional actions are necessary to support an increase 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 from 18-months to 24-months.

D. Based on the historical good performance of these components, the corrective actions taken to further improve system performance, 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 4.7.1.2.1.c 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 Orders.

ii. DBNPS Potential Condition Adverse to Quality Reports.

5. Other Information:

A. There have been generic industry problems with governor valves sticking, however, the DBNPS has had good success with the chrome-plated AFW governor valve stems which were installed in 1987. Although there has been no valve sticking since the installation of the chrome-plated valve stems, there has been some chrome flaking and pitting of these components after five years of service. This flaking and pitting is minor and will not impact the operation of the valves. As an additional measure of conservative operation these valves are currently inspected each refueling outage. The successful use of the chrome plated governor valve stems over a five year interval shows that an increased operating cycle length will have no adverse effect. A significant improvement in governor valve stem conditions has been achieved by the reduction of steam leak-by of the steam admission valves which are just upstream of the governor valves. Without a moist environment for the governor valves, the degradation rate of the valve stems is significantly reduced over that of the previous conditions. Surveillance testing which is performed in accordance with SR 4.7.1.2.1.a on a 92-day staggered test basis, would provide the opportunity to identify a governor valve binding problem. Therefore, the proposed change to a 24-month fuel cycle will have no impact on the reliability of these valves.

B. The check valves in the main steam supply lines to the AFW Pump Turbines, MS734 and MS735, have had a history of pressure-pulse-induced valve tapping. To date, these valves have remained operable and able to perform their safety function, however, they have required maintenance each outage. In order to protect the valve seats from tapping, down-stream steam traps were placed in bypass to initiate steam flow through the valves in order to protect their seats from damage by reducing the magnitude of the valve disk to seat impacts with steam flow. With steam flowing through the check valves, the valve disk to

seat impacts are now reduced compared to previous cycles. A plant modification which is planned for completion by 11RFO will replace the current steam trap bypass method by routing the steam flow to a feed water heater to recover the lost energy. Either the interim changes in plant operation or the above modification, provide assurance that the check valves, MS734 and MS735 will remain operable for an extended operating cycle. It is acceptable to increase their inspection interval from 18 months to 24 months.

C. During Cycle 10, increased leakage of the shaft steam seal on the No. 2 AFW Pump Turbine was noted. The leakage resulted in an increased accumulation of water in the turbine's governor end bearing oil. The reason this problem occurred was that a gland pipe, attached to the gland housing, was misaligned and it placed a rotational force on the gland case. This force broke the seal between the turbine case and the gland seal assembly. This problem was a maintenance reassembly issue which was corrected during 10RFO and was not related to oper ; cycle length.

The bearing oil is periodically sampled and replaced during the operating cycle to ensure the proper operation of the AFW Pump turbine steam seals. The lube oil sampling and replacement program will continue to be performed during an increased operating cycle length. Therefore, there is no concern with changing to a 24-month cycle.

## Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.7.1.7.d

1. A. Technical Specification (TS): 3/4.7.1.7, "Motor Driven Feedwater Pump System," Surveillance Requirement (SR) 4.7.1.7.d

   B. Systems or Components:

   Motor Driven Feedwater Pump System

   C. Updated Safety Analysis Report (USAR) Sections:
   9.2.8 Motor Driven Feedwater Pump
   10.4.7.2, Condensate and Feedwater Systems - System Description

2. Licensing Basis Review:

   A. Technical Specification SR 4.7.1.7.d requires that the Motor Driven Feedwater Pump and flow paths to the Auxiliary Feedwater System be demonstrated operable at least once per 18 months by:

      1. Verifying that there is a flow path between the Motor Driven Feedwater Pump System and the Auxiliary Feedwater System by pumping water from the Condensate Storage Tanks to the steam generators. The flow path to the steam generators shall be verified prior to entering Mode 3 from Mode 4 by either steam generator level change or Auxiliary Feedwater Safety Grade Flow Indication. Verification of Motor Driven Feedwater Pump System flow capacity is not required.

      2. Verifying proper operation of the Motor Driven Feedwater Pump lube oil interlocks.

      3. Verifying proper operation of manual valves by shifting the Motor Driven Feedwater Pump between the Main Feedwater System and the Auxiliary Feedwater System.

   TS 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 4.7.1.7.d, the words "At least once per 18 months" be replaced with "at least once each 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. As described in USAR Section 9.2.8, the Motor Driven Feedwater Pump (MDFP) provides feedwater to the steam generators during normal plant startup and shutdown. The MDFP is also designed to provide a backup supply of feedwater to the steam generator in the event of a total loss of both auxiliary and main feedwater. During plant operation when reactor power is greater than 40%, the MDFP is aligned as a backup auxiliary feedwater pump capable of delivering water to both steam generators.

The MDFP is non-safety related. However, the pump provides a diverse means of supplying auxiliary feedwater to the steam generators and thus functions as a backup to the nuclear safety related auxiliary feedwater system.

The MDFP is not an initiator, nor a contributor to the initiation of an accident described in the USAR.

C. The MDFP System was not part of the original DBNPS plant design, but was added during the DBNPS's mid cycle shutdown which commenced in 1985. The current surveillance interval of 18 months was contained in License Amendment 103, which added the Limiting Conditions for Operation and Surveillance Requirements for the new MDFP System. The 18-month surveillance interval was chosen based upon the surveillance being performed during a refueling outage, similar to the Auxiliary Feedwater System under TS 3/4.7.1.2.

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 will not be invalidated by increasing the surveillance interval for SR 4.7.1.7.d 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. NRC License Amendment No. 103 to Facility Operating License No. NPF-3, dated September 2, 1987 (Toledo Edison Log Number 2384).

iv. USAR Section 9.2.8, "Motor Driven Feedwater Pump," through Revision 19.

v. USAR Section 10.4.7.2, "Condensate and Feedwater Systems - System Description," through Revision 19.

3. Surveillance Data Review:

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

Note that surveillance data for SR 4.7.1.7.d.1 was not reviewed during the performance of this surveillance data review. The purpose of SR 4.7.1.7.d.1 is to verify that a flow path exists between the Motor Driven Feedwater Pump System and the Auxiliary Feedwater System utilizing steam generator level change or safety grade flow indication. All valves in the flowpath are locked open and controlled by station administrative procedures, therefore, the flowpath, once established will not change. Automatic valve actuation and Motor Driven Feedwater Pump flow rate verification is not included in SR 4.7.1.7.d. Surveillances to verify proper operation of each power operated and automatic valve and the Motor Driven Feedwater Pump itself are performed once per 92 days by SR 4.7.1.7.c and will not be revised as part of this license amendment request.

B. The surveillance test results for 4.7.1.7.d indicate no test failures due to equipment failure or malfunction.

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 surveillance interval.

D. Based on the historical good performance of these components, no introduction of new failure modes and low potential for increased failure rates, it is concluded that it is acceptable to increase the surveillance interval for SR 4.7.1.7.d from 18 months to 24 months. 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-SS-03092, "Motor Driven Feedpump Refueling Test."

4. Maintenance Records Review:

A. The maintenance records for the components were reviewed 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 maintenance activities.

B. A review of the past maintenance performed on the components associated with TS SR 4.7.1.7.d did not identify any failures or degradation.

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 R 4.7.1.7.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 Maintenance Work Order Records.

ii. DBNPS Potential Conditions Adverse to Quality Reports.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.7.3.1.b

1. A. Technical Specification (TS): 3/4.7.3.1, "Component Cooling Water System," Surveillance Requirement (SR) 4.7.3.1.b

   Note:

   The Safety Features Actuation System (SFAS) instrumentation and controls extend from the generating station variables to the input terminals of the safety features actuation control devices, such as motor controllers and solenoid valves. The subject surveillances cover actuated components which receive an SFAS signal. The surveillances do not include the SFAS instrumentation and controls. Applicable SFAS surveillances will be addressed under a separate License Amendment Request.

   B. Systems or Components:

   Component Cooling Water System

   C. Updated Safety Analysis Report (USAR) Section:
   9.2.2 Component Cooling Water System

2. Licensing Basis Review:

   A. Technical Specification SR 4.7.3.1.b requires that each component cooling water loop be verified operable 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 an SFAS test signal.

      2. Verifying that each component cooling water emergency pump starts automatically on an SFAS test signal.

   TS 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 4.7.3.1.b the words "At least once per 18 months, during shutdown" be replaced with "At least once each 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. As described in USAR Section 9.2.2, the Component Cooling Water (CCW) System is designed to provide cooling water to reactor auxiliaries and the Emergency Core Cooling System (ECCS) during normal station operation and Design Basis Accident (DBA) conditions. The components of the system are designed on the basis of removing the maximum heat load during normal station operation with 85°F service water temperature, and removing maximum heat loads from ECCS components during DBA conditions with service water at the ultimate heat sink conditions.

The part of the system required during DBA conditions is separated into two redundant loops, with each loop capable of supplying 100 percent of the cooling water required under those conditions.

Three CCW pumps and heat exchangers are provided so that any one of the pump heat exchanger units can be removed from service for maintenance or repair without reducing the capability or redundancy of the system. Thus, the third pump can take the place of either No. 1 or No. 2 pump in all respects.

Under DBA conditions, one CCW pump runs in each loop and nonessential components are isolated from the system. No single failure in a loop affects the other loop.

The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses. The CCW System is not an initiator, nor a contributor, to the initiation of an accident described in the USAR.

C. The current surveillance intervals of 18 months were based on the guidance of NUREG-0103, Revision 0, 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 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 will not be invalidated by increasing the surveillance interval for SR 4.7.3.1.b 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 9.2.2, "Component Cooling Water System," through Revision 19.

3. Surveillance Data Review:

A. The 18-month TS surveillance test results data for the surveillance requirement were reviewed 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 reviewed test results indicate one failure occurred over this time period for the reviewed components.

The CCW Outlet valve from Decay Heat Removal Cooler 1-1, CC1467, failed to stroke to its SFAS position during performance of the surveillance test during 8RFO (1993). The root cause of the valve actuation failure was a broken air line between the regulator and the open piston. The broken air line and associated fittings were replaced and the valve was retested satisfactorily. This valve stroked properly during surveillance testing performed during 9RFO, and no other problems have been since identified with this valve. This failure was not associated with the length of the fuel cycle.

In general, unless plant conditions or other circumstances prohibit valve stroking at power, CCW automatic valves are stroked quarterly under the ASME Code Section XI Inservice Testing Program to ensure valve operability. Such testing would discover problems such as the above-mentioned actuator problem which occurred with valve CC1467.

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, the introduction of no new failure modes, and the inclusion in the Inservice Test Program, it is concluded that it is acceptable to increase the surveillance interval for TS 4.7.3.1.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 Second Interval Inservice Test Program, Volume 1, Revision 02.

ii. DBNPS Procedures:

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

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

DB-SP-03091, "Component Cooling Water Pump 2 Refueling Test."

DB-SP-03092, "Component Cooling Water Pump 3 Refueling Test."

4. Maintenance Records Review:

A. A maintenance records summary for the components were reviewed 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 maintenance activities.

B. Review of the maintenance record summary indicated the following failures:

CCW Pump 3 Bearing: During the 5RFO (1988), the CCW Pump 3 motor was found to have high motor temperatures which were verified by surface pyrometer checks. The outboard bearing was replaced and the motor retested. The bearing failed again and was disassembled. During the repairs it was determined that the shaft diameter was below minimum allowable, requiring the rotating element to be replaced. Additionally, it was determined that the motor thermocouple was inserted too far, which displaced the slinger ring and prevented oil from properly lubricating the bearing. Repairs were made to the motor and the pump was successfully retested. This problem has not recurred. This maintenance was performed during a refueling outage, but could have been performed during the operating cycle since two trains of CCW, as required by the TS, would still have been available. Therefore, it is acceptable to increase the surveillance interval to 24 months.

Decay Heat Cooling: During the 5RFO (1988), both decay heat removal trains were declared inoperable due to a potential for the decay heat removal cooler component cooling water outlet isolation valves (CC1467 and CC1469) to stray from their fail-safe position on loss of air following an SFAS actuation. To correct this deficiency, a modification was developed which installed a detent to lock the valve in the fail-safe position. This problem was the result of a design deficiency which has been satisfactorily remedied. The cause of this problem was not attributable to cycle length, therefore, there would be no effect from a change to a 24-month 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 SR 4.7.3.1.b from 18 to 24 months and that there would be 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:

1. DBNPS Maintenance Work Orders.

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.7.4.1.b

1. A. Technical Specification (TS): 3/4.7.4.1, "Service Water System," Surveillance Requirement (SR) 4.7.4.1.b

Note:

The Safety Features Actuation System (SFAS) instrumentation and controls extend from the generating station variables to the input terminals of the safety features actuation control devices, such as motor controllers and solenoid valves. The subject surveillances cover actuated components which receive an SFAS signal. The surveillances do not include the SFAS instrumentation and controls. Applicable SFAS surveillances will be addressed under a separate License Amendment Request.

B. Systems or Components:

Service Water System

C. Updated Safety Analysis Report (USAR) Section:
9.2.1 Service Water System

2. Licensing Basis Review:

A. Technical Specification SR 4.7.4.1.b requires that each service water loop be verified operable 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 an SFAS test signal.

2. Verifying that each service water emergency pump starts automatically on an SFAS test signal.

TS 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 4.7.4.1.b the words "At least once per 18 months, during shutdown" be replaced with "At least once each 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. As described in USAR Section 9.2.1, the Service Water (SW) System is designed to serve two functions during station operation. The first function is to supply cooling water to the component cooling heat exchangers, the containment air coolers, and the cooling water heat exchangers in the turbine building during normal operation. The second function is to provide, through automatic valve sequencing, a redundant supply path to the engineered safety features components during an emergency. Only one path, with one service water pump, is necessary to provide adequate cooling during this mode of operation.

The Seismic Class I service water pumps are sized to provide cooling water to the component cooling heat exchangers, containment air coolers, and the emergency core cooling system room cooling coils. Two redundant pumps, of 100 percent capacity each, are provided to back up the operating pump.

The service water system also provides a backup source of water to the auxiliary feedwater system and the Motor Driven Feedwater pump (MDFP). During normal operation service water discharge provides makeup for the circulating water system.

The portion of the system required for emergency operation, including the intake structure, is designed to the ASME Code, Section III, Nuclear Class 3 and Seismic Class I, as applicable.

Three service water pumps are part of the system. They are installed in the intake structure and use Lake Erie as a source of water. The three pumps are piped to two separate interconnected but isolated supply paths.

The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analysis. The SW System is not an initiator, nor a contributor, to the initiation of an accident described in the USAR.

C. The current surveillance intervals of 18 months were based on the guidance of NUREG-0103, Revision 0, June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," during the initial licensing of the DBNPS. 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 will not be invalidated by increasing the surveillance interval for SR 4.7.4.1.b 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 9.2.1, "Service Water System," through Revision 19.

3. Surveillance Data Review:

A. The 18-month TS surveillance test results data for the surveillance requirements were reviewed 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 18 month surveillance tests revealed only one component failure, which is attributed to a design error made by the vendor of the supplied equipment.

The CCW Heat Exchanger 1-1 Outlet Control Valve, SW1424, failed to stroke open during the performance of a surveillance test during 6RFO (1990). This valve had just been installed during 6RFO, replacing the previous valve design with an improved ball valve design. The newly installed valve was removed and inspected. The root cause of the failure was a deficiency in the design of the seat spring which generated high seat spring torque. The seat design was modified and the valve was reinstalled and successfully retested. This valve has performed successfully during subsequent testing. This failure was not related to the length of an operating 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 an increase in the failure rates of these components under a longer test interval, the introduction of no new failure modes, and inclusion in the Inservice Test Program, it is concluded that it is acceptable to increase the surveillance interval for SR 4.7.4.1.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 Second Interval Inservice Test Program, Volume 1 Revision 02.

ii. DBNPS Procedures:

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

DB-SP-03024, "Service Water Pump 2 Refueling Test."

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

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

4. Maintenance Records Review:

A. Maintenance records for the components were reviewed 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 maintenance activities.

B. The maintenance records review indicated the following failures and deficiencies. Three of the failures and both deficiencies are attributed to either vendor design limitations or manufacturing defects of the supplied equipment.

<u>Failures</u>

1. The CCW Heat Exchanger 1-3 Outlet Control Valve, SW1429 failed to go full open in Cycle 9 (1993 - 1994) when placing it into service. Troubleshooting of the valve concluded that the frictional forces which occur when the valve remains in position for extended periods of time caused the actuator maximum torque to be exceeded, which stalled the actuator. An engineering analysis of this condition determined that the vendor supplied Teflon packing used in this type of valve will "take a set" if allowed to remain in one position too long. This control valve is one of three which are used for throttling in order to control flow through the CCW heat exchangers. In order to prevent this problem from recurring, the valves which are not in service are exercised periodically during the fuel cycle. This problem has not recurred. Since this exercising is performed independently of the fuel cycle length, there is no impact due to a cycle length change from 18 to 24 months.

2. The CCW Heat Exchanger 1-2 Outlet Control Valve, SW1434 failed to go full open during Cycle 9 (1993-1994) following the transferring of SW loops. The failure was attributed to a random failure of the temperature controller TIC 1434. It was subsequently replaced and recalibrated with the plant online during the operating cycle. The controller has operated properly following its replacement. This failure was identified during the operating cycle, remedied on line and, therefore, is not affected by the length of the operating cycle.

3. The Service Water Pump 1-3 Supply Breaker from 4160V Bus C1, Breaker ACD4 failed to close when required during 5RFO (1988). The charging motor cutoff switch was replaced and the breaker was retested successfully. The breaker operated properly until Cycle 6 (1989) when ACD4 did not close as required. Investigation concluded that the breaker failed to close due to high resistance contacts on the motor cutout switch, caused by a misaligned switch. The breaker was repaired and successfully retested. The maintenance records summary indicates that the breaker has performed properly since the repair. This failure was not dependent on operating cycle length.

4. The Service Water Pump 1-3 Supply Breaker from 4160V Bus D1, Breaker ACD5 failed to close when required. Follow-up investigations discovered a broken linkage which connects the auxiliary switch to the pole operating shaft. The root cause of the linkage failure was due to a faulty spot weld made during its manufacture. The linkage was replaced and breaker successfully retested during Cycle 6 (1989). The records summary indicates the breaker has performed properly since the repairs. This failure was not dependent on operating cycle length.

5. The Service Water Pump 1-3 shaft failed while in operation during Cycle 8 (1991). The root cause of the failure was attributed to a machining flaw created during the vendor's manufacturing of the shaft. The shaft was rebuilt, installed and successfully retested. Prevertive maintenance activities are presently in place to rebuild the pumps every 5 years. These activities are performed during the operating cycles and therefore, do not require a refueling outage to complete. As discussed earlier the DBNPS requires two SW pumps but has three available for use. Thus, the performance of the pump rebuild is not impacted by the length of the operating cycle.

6. The Service Water Loop 1 to the Turbine Plant Cooling Water Heat Exchanger Supply Header Isolation Valve, SW1399 torqued out when opening during Cycle 5. The subsequent troubleshooting monitcr-

ed torque readings and resulted in adjustment of the torque switch settings while the plant was on-line. The valve was retested and declared operable. No further deficiencies with valve SW1399 opening have been reported in the maintenance record summary. This failure was identified during an operating cycle, remedied with the plant on line and, therefore, was not associated with the length of the operating cycle.

7. During a routine isolation for maintenance, the CCW Heat Exchanger 1-3 Outlet Control Valve, SW1429 failed to go to the full open position during Cycle 8 (1991). Inspection of the valve internals discovered the failure was caused by a misalignment of the ball and seat which was attributed to a vendor assembly error. The valve was repaired, reassembled, and tested successfully. This failure was not dependent upon operating cycle length.

In general, unless plant conditions or other circumstances prohibit valve stroking at power, Service Water automatic valves are stroked quarterly under the ASME Code Section XI Inservice Testing Program to ensure valve operability. Such testing would discover problems such as the above-mentioned failures of the CCW Heat Exchanger 1-3 Outlet Control Valve, SW1429.

Deficiencies

1. The CCW Heat Exchanger 1-3 Outlet Control Valve, SW1429 was observed to not go fully closed during operating Cycle 9 (1993). However, the valve would have performed its safety function since it is designed to fail open. The valve was disassembled and inspected. The inspection found no significant deficiencies, and the valve was reassembled, reinstalled and retested. This problem has not recurred.

2. The CCW Heat Exchanger 1-1 Outlet Control Valve, SW1424 was also observed to not go fully closed under full system differential pressure during operating Cycle 10 (1995). Troubleshooting activities concluded that the actuator design limitations contributed to the valve not achieving a full shutoff position. However, the valve would have performed its safety function since it is designed to fail open.

In both of these situations, the valves were operable and would have performed their required safety functions because these valves are used in a throttling application and are designed to fail open.

As previously mentioned, these valves are exercised periodically during the 18-month operating cycle in order to ensure their continued operability. Therefore, the performance of these valves would not be adversely impacted by increasing the fuel cycle to 24 months.

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 continued ability to identify similar problems and remedy such problems during an operating cycle, the past performance of these components and the actions taken to prevent recurrence of the failures described above, the low potential for an increase in the failure rate 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 of 4.7.4.1.b from 18 to 24 months and that there is no adverse effect on 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 Orders.

ii. Potential Conditions Adverse to Quality Reports.