REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual safety and regulating rod drop time from the fully withdrawn position shall be $\leq$ 1.58 seconds from power interruption at the control rod drive cabinets to 3/4 insertion with:

a. $T_{avg} \geq 525°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any safety or regulating rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 and 2.

b. With the rod drop times within limits but determined with less than 4 reactor coolant pumps operating, operation may proceed provided that THERMAL POWER is restricted to less than or equal to the THERMAL POWER allowable for the reactor coolant pump combination operating at the time of rod drop time measurement.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of safety and regulating rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once each REFUELING INTERVAL ~~every 18 months~~.

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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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June 6, 1995

SURVEILLANCE REQUIREMENTS (continued)

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

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

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

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

d. At least once per 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 $\leq$75 seconds after the operator manually pushes the control switch to open the Containment Emergency Sump Valve HV-DH9A (HV-DH9B) which should be verified to open in $\leq$75 seconds.

   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.~~

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

5. Deleted

6. Deleted

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

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

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

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

1. At least once per 18 months.

2. After each opening of the watertight enclosure.

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

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

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

2. At least once per 18 months.

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

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

LPI System - Single Pump

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

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

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

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3/4.1.2 BORATION SYSTEMS (Continued)

The requirement for a minimum available volume of 482,778 gallons of borated water in the BWST ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4; therefore, the larger volume of borated water is specified.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 70°F. This condition requires either 900 gallons of 7875 ppm borated water from the BAAS or 3,000 gallons of 2600 ppm borated water from the BWST.

The bottom 4 inches of the BWST are not available, and the instrumentation is calibrated to reflect the available volume. All of the boric acid addition tank volume is available. The limits on water volume, and boron concentration ensure a pH value of between 7.0 and 11.0 of the solution recirculated within 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.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the safety analysis.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

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3/4.1.3. MOVABLE CONTROL ASSEMBLIES (Continued)

The maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analyses. Measurement with $T_{avg} \geq 525°F$ and with reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied. A 1.5% group average position uncertainty is applied to the rod index curves. Therefore, the position indicators must be capable of supporting this accuracy. The Surveillance Requirement ensures this accuracy by keeping the RPI calibrated to a "known" position as indicated by the API. Using the API as a "known" position is valid provided two consecutive reed switches are not inoperable. Having one entire string (i.e., every other reed switch) inoperable is acceptable.

A specific surveillance of the reed switches is not required because:

1) When one or more reed switch fails closed, a large API indication of asymmetry occurs.

2) Two failed open reed switches in series result in a large indication of asymmetry.

3) Failed open reed switches not in series (up to every other switch) are bounded by the analysis.

Therefore, a reed switch condition not bounded by the analysis will be indicated by API system asymmetry indications.

Technical Specification 3.1.3.8 provides the ability to prevent excessive power peaking by transient xenon at RATED THERMAL POWER. Operating restrictions resulting from transient xenon power peaking, including xenon-free startup, are inherently included in the limits of Sections 3.1.3.6 (Regulating Rod Insertion Limits), 3.1.3.9 (Axial Power Shaping Rod Insertion Limits), and 3.2.1 (Axial Power Imbalance) for transient peaking behavior bounded by the following factors. For the period of cycle operation where regulating rod groups 6 and 7 are allowed to be inserted at RATED THERMAL POWER, an 8% peaking increase is applied at or above 92% FP. An 18% increase is applied below 92% FP. For operation where only regulating rod group 7 is allowed to be inserted at RATED THERMAL POWER, a 5% peaking increase is applied at or above 92% FP and a 13% increase is applied below 92% FP.

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

If these values, checked every cycle, conservatively bound the peaking effects of all transient xenon, then the need for any hold at a power level cutoff below RATED THERMAL POWER is precluded. If not, either the power level at which the requirements of Section 3.1.3.8 must be satisfied or the above-listed factors will be suitably adjusted to preserve the LOCA linear heat rate limits.

The limitation on axial power shaping rod insertion is necessary to ensure that power peaking limits are not exceeded.

3/4.5.1 CORE FLOODING TANKS

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

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

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

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

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

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

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

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

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

EMERGENCY CORE COOLING SYSTEMS

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BASES

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

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

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

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

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

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

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

3/4.5.4 BORATED WATER STORAGE TANK

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

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

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

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

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

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

Summary of Licensing Basis, Surveillance Data, and Maintenance Record Reviews for Surveillance Requirement 4.1.3.4.c

1. A. Technical Specification (TS) 3/4.1.3.4, "Reactivity Control Systems - Rod Drop Time," Surveillance Requirement (SR):

   4.1.3.4.c

   B. Systems or Components:

   Control Rod Assemblies - Safety and Regulating Groups

   C. Updated Safety Analysis Report (USAR) Sections:

4.3	Nuclear Design
Appendix 4B	Reload Report
7.4.1.1	Control Rod Drive Control System (CRDCS) - Trip Portion
7.7.1.3	CRDCS - Without Trip Portion
Table 15.1-2	Parameters Applicable to All Accidents in the Accident Analysis

2. Licensing Basis Review:

   A. Technical Specification SR 4.1.3.4.c requires that the rod drop time of safety and regulating rods be demonstrated through measurement prior to reactor criticality at least once every 18 months. 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 4.1.3.4.c, the words "At least once every 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 ≤ 730 days." 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. Core reactivity is controlled by Control Rod Assemblies (CRAs) and soluble boron in the coolant. Sufficient CRA worth is available to shut the reactor down with at least a 1% Dk/k subcritical margin in the hot condition at any time during the cycle with the most reactive CRA stuck in the fully withdrawn position.

   There are 53 CRAs arranged in seven groups: four rod groups function as safety groups and three rod groups function as regulating groups.

Each of the seven groups may be assigned from four to twelve CRAs. The safety groups are normally fully withdrawn when the reactor is operating at power. The regulating groups serve as the principle reactor reactivity control medium.

The Control Rod Drive Control System (CRDCS) utilizes Control Rod Drive Mechanisms (CRDMs) to position groups of CRAs at the desired locations. Upon receipt of an automatic or manual trip command signal the CRDMs are de-energized, causing the CRAs to drop into the core, shutting the reactor down. The surveillance test verifies a maximum travel time of 1.58 seconds for a 3/4-insertion of CRAs. As stated in TS Bases 3/4.1.3, the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analyses. The CRDCS, including the CRAs, 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 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 for the rod drop time surveillance requirement will not be invalidated by increasing the surveillance interval for SR 4.1.3.4.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 4.3, "Nuclear Design," through Revision 19.

v. USAR Appendix 4B, "Reload Report," through Revision 19.

vi. USAR Section 7.4.1.1, "Control Rod Drive Control System (CRDCS) - Trip Portion," through Revision 19.

vii. USAR Section 7.7.1.3, "CRDCS - Without Trip Portion," through Revision 19.

viii. USAR Table 15.1-2, "Parameters Applicable to All Accidents in the Accident Analysis," through Revision 19.

3. Surveillance Data Review:

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

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

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

D. Based on the historical good performance of these components, the low potential for significant 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 TS 4.1.3.4.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 Procedure DB-SC-03270, "Control Rod Assembly Insertion Time Test."

4. Maintenance Records Review:

A. The 18 month 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. No failures that would have resulted in the components being TS inoperable were noted, and there were no significant degradation cases during this time period for these components.

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

D. Based on the historical good performance of these components, the low potential for significant 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 TS 4.1.3.4.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 Order Records.

5. Other Information:

Another Babcock and Wilcox-type plant which had increased the length of its fuel cycle experienced control rod drop time difficulties. These difficulties were determined to be caused by the presence of crud in the ball check valves of the CRDMs. This plant has a "Type A" CRDM design. The DBNPS has a "Type C" CRDM design which has not experienced these problems. In addition, with the implementation of longer fuel cycles, the DBNPS has established chemistry controls designed to minimize crud formation. Accordingly, no degradation of rod drop times due to excessive crud formation is anticipated over the longer fuel cycles.

The CRDMs are not normally removed from the reactor vessel head nor disassembled during refueling operations. However, during 10RFO, the CRDM in core location N-12 was removed and disassembled for the purpose of performing a life extension inspection for the Babcock and Wilcox Owners Group (BWOG). The inspection revealed a broken race on the radial bearing, a crack in the synchronizer bearing outer race pin pocket, and a crack in the leadscrew nut leaf spring. Followup inspections revealed no additional leaf spring cracks on other CRDMs. The defective components were replaced. An evaluation concluded that the defective parts would not have prevented insertion of the control rods into the reactor core. The root cause of the bearing failures was determined to be the result of an overload condition, most likely from impact during handling either during initial installation or during the recent inspection. The root cause of the leaf spring failure was also determined to be the result of an overload condition, which may have been caused from excessive forces applied or embrittlement. Fuel cycle length is not believed to be a contributor to the failures. The CRDMs installed at the DBNPS are original equipment and have been in operation for approximately 20 years.

As described in NRC Information Notice (IN) 96-12, two plants have experienced failures of control rods to fully insert into relatively high burnup fuel assemblies following a reactor trip. The fuel utilized at these plants is of a different design than the DBNPS and is supplied by a different fuel vendor. The respective Owner's Group is pursuing the root cause identification of these events. The DBNPS, in conjunction with the Babcock and Wilcox Owner's Group, is following this issue. Other B&W plants which have extended their operating cycles have not experienced control rod drop time difficulties of the type described in IN 96-12.

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

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

   4.5.2.b

   B. Systems or Components:

   Not Applicable --
   This SR ensures that the ECCS piping is full of water by venting the ECCS pump casings and discharge piping high points.

   C. Updated Safety Analysis Report (USAR) Sections:

   6.3 Emergency Core Cooling System

2. Licensing Basis Review:

   A. Technical Specification SR 4.5.2.b requires that the ECCS piping be verified to be full of water by venting the ECCS pump casings and discharge piping high points at least once every 18 months or prior to operation after the ECCS piping has been drained. 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 4.5.2.b, 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." This is consistent with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Technical Specification 4.0.2 would continue to apply which would allow increasing the new surveillance interval on a non-routine basis from 24 months to 30 months.

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

Since the ECCS is normally in a standby, non-operating mode, the ECCS pump casings and discharge piping high point vents are periodically vented to minimize the potential for the flow path piping to develop voids. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly upon demand.

The frequency at which the ECCS piping is vented is not an initiator, nor a contributor, to the initiation of an accident described in the USAR.

C. The original Standard Technical Specifications upon which the DBNPS Technical Specifications were modeled, NUREG-0103, Revision 0, June 1, 1976, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," did not include a surveillance requirement for ECCS venting. However, according to Section 6.3.2 of Supplement 1 to the NRC Safety Evaluation Report for the DBNPS Operating License (NUREG-0136), the DBNPS was requested to adopt a surveillance requirement to verify that the ECCS piping is water solid, to minimize the potential for water hammer. Accordingly, the original DBNPS Technical Specifications issued by the NRC on April 22, 1977 as Appendix A to the Operating License included SR 4.5.2.b, with a 31 day frequency.

On October 23, 1978, Toledo Edison submitted a License Amendment Request to change the required frequency for SR 4.5.2.b from 31 days to 18 months. The bases for this request was the fact that once the system is filled and vented, a minimum positive head of 30 feet of water on that part of the system open to the Borated Water Storage Tank (BWST) will prevent air build-up or any subsequent water hammer potential. In addition, the LAR referred to an analysis summarized in the DBNPS Final Safety Analysis Report (FSAR) that the portion of the system not open to the BWST, downstream of the normally closed High Pressure Injection line isolation valves HP2B, HP2C, and HP2D, would not experience unacceptable forces in the event of system actuation, even if the lines downstream of the valves were completely void of water. It was noted that a fourth High Pressure Injection line isolation valve, HP2A, was in the normal makeup line and, therefore, would be maintained full at all times. Further, the LAR noted that the proposed decrease in surveillance frequency would result in lower occupational radiation dosage without a reduction in margin of safety provided by the ECCS.

In a January 26, 1979 supplemental letter to the October 23, 1978 LAR, Toledo Edison noted that the High Pressure Injection lines downstream of valves HP2B, HP2C, and HP2D were vented after approximately two years of plant operation, and that no air or gas was observed leaving the lines.

On July 2, 1980, the NRC issued License Amendment No. 25, as requested by the October 23, 1978 LAR, changing the required frequency of SR 4.5.2.b to 18 months.

During the Sixth Refueling Outage (6RFO), which commenced in January, 1990, a plant modification was implemented which changed the normal makeup flow path connection from downstream of valve HP2A to downstream of valve HP2B. The impact of this change on the ECCS venting capability

downstream of valve HP2A was not recognized at the time. This condition was discovered on March 4, 1996 and was reported to the NRC as Licensee Event Report 96-001 on April 3, 1996. A plant modification to add a vent on the high point downstream of valve HP2A was implemented during the Tenth Refueling Outage (10RFO) in May, 1996. In addition, a calculation was performed to verify that the line downstream of valve HP2A would not experience unacceptable forces in the event of system actuation, even if the line were completely void of water.

The bases for extending the SR 4.5.2.b interval from 31 days to 18 months equally applies to extending the interval to 24 months, as proposed by this LAR. In addition, the proposed changes follow the guidance of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle," dated April 2, 1991.

D. As a result of the above review, it is concluded that the licensing basis for the ECCS venting surveillance requirement will not be invalidated by increasing the surveillance intervals for SR 4.5.2.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. DBNPS Operating License Appendix A Technical Specifications dated April 22, 1977.

v. NUREG-0136 dated December, 1976 (DBNPS Unit 1 Safety Evaluation Report), including Supplement 1, dated April, 1977.

vi. Toledo Edison License Amendment Request dated October 23, 1978 (Toledo Edison Serial Number 462).

vii. Toledo Edison Supplemental Letter dated January 26, 1979 (Toledo Edison Serial Number 482).

viii. DBNPS Final Safety Analysis Report Position 5.5.1, Revision 16, October 1975.

ix. NRC License Amendment No. 25 to Facility Operating License No. NPF-3, dated July 2, 1980 (Toledo Edison Log Number 575).

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

xi. Plant Modification 89-066, ECCS (HPI) Makeup and Purification.

xii. Plant Modification 96-006, HPI Injection Line High Points.

xiii. DBNPS Calculation T3-2B-I Revision C1.

xiv. DBNPS Calculation T3-2B-II Revision C2.

xv. DBNPS Calculation 56B3 Revision C2.

xvi. DBNPS Licensee Event Report (LER) 96-001 dated April 3, 1996.

3. Surveillance Data Review:

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

B. The test results for all identified tests were satisfactory.

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

E. References:

i. DBNPS Procedure DB-SP-03212, "Venting of ECCS Piping."

4. Maintenance Records Review:

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