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Docket Number 50-346

10 CFR 50.90

License Number NPF-3

Serial Number 2834

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United States Nuclear Regulatory Commission
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Subject: Davis-Besse Nuclear Power Station
License Amendment Application to Modify Technical Specification 3/4.5.2,
Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$
(License Amendment Request No. 03-0004)

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, an amendment is requested for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The proposed amendment would modify Technical Specification (TS) 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$.

Limiting Condition for Operation (LCO) 3.5.2 requires two independent Emergency Core Cooling Systems (ECCS) Subsystems to be operable. Surveillance Requirement (SR) 4.5.2.f requires each ECCS Subsystem to be demonstrated operable by performing a vacuum leakage rate test of the watertight enclosure for Decay Heat Removal System valves DH-11 and DH-12 that assures the electric motor operators on valves DH-11 and DH-12 will not be flooded for at least seven (7) days following a Loss-of-Coolant Accident (LOCA). This SR provides assurance that a circulation flow path will be maintained to prevent boric acid concentration build-up and boric acid precipitation in the reactor vessel post-LOCA.

The proposed amendment would allow the relocation of SR 4.5.2.f to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM). Future changes to the relocated specification, such as a planned extension of the current 18-month surveillance interval, will be evaluated under the requirements of Section 50.59 of Title 10 of the Code of Federal Regulations (10 CFR), and the NRC will be informed of these changes in accordance

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with the requirements of 10 CFR 50.59(d)(2), as applicable, and the USAR update requirements of 10 CFR 50.71(e).

Enclosure 1 to this letter contains the technical justification for these proposed changes and the proposed no significant hazards consideration determination. Approval of the proposed amendment is requested by February 6, 2004. Once approved, the amendment shall be implemented within 120 days.

The proposed changes have been reviewed by the DBNPS Station Review Board and Company Nuclear Review Board.

Should you have any questions or require additional information, please contact Mr. Kevin L. Ostrowski, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

MKL

Enclosures

cc: Regional Administrator, NRC Region III
J. B. Hopkins, NRC/NRR Senior Project Manager
D. J. Shipley, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
C. S. Thomas, NRC Region III, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

APPLICATION FOR AMENDMENT

TO FACILITY OPERATING LICENSE NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This submittal requests changes to the Davis-Besse Nuclear Power Station Unit Number 1, Facility Operating License Number NPF-3. The statements contained in this submittal, including its associated enclosures and attachments, are true and correct to the best of my knowledge and belief.

I declare under penalty of perjury that I am authorized by the FirstEnergy Nuclear Operating Company to make this request and the foregoing is true and correct.

Executed on: 8/11/03

By: Lew W. Myers

Lew W. Myers, Chief Operating Officer

Docket Number 50-346
License Number NPF-3
Serial Number 2834
Enclosure 1

DAVIS-BESSE NUCLEAR POWER STATION
EVALUATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 03-0004

(21 pages follow)

DAVIS-BESSE NUCLEAR POWER STATION
EVALUATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 03-0004

Subject: License Amendment Application to Modify Technical Specification 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$

1.0 DESCRIPTION

This letter is a request to amend the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3.

The proposed change would revise Technical Specification (TS) 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$, to relocate Surveillance Requirement (SR) 4.5.2.f to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM).

2.0 PROPOSED CHANGE

Limiting Condition for Operation (LCO) 3.5.2 requires two independent Emergency Core Cooling Systems (ECCS) Subsystems to be operable. Surveillance Requirement (SR) 4.5.2.f requires each ECCS Subsystem to be demonstrated operable by performing a vacuum leakage rate test of the watertight enclosure for Decay Heat Removal System valves DH-11 and DH-12 that assures the electric motor operators on valves DH-11 and DH-12 will not be flooded for at least seven (7) days following a Loss-of-Coolant Accident (LOCA). This SR provides assurance that a circulation flow path will be maintained to prevent boric acid concentration build-up and boric acid precipitation in the reactor vessel post-LOCA.

The proposed amendment would allow the relocation of Technical Specification Surveillance Requirement 4.5.2.f to the DBNPS Updated Safety Analysis Report Technical Requirements Manual. Surveillance Requirement 4.5.2.f would be relocated to the USAR TRM at the time of the implementation of the approved license amendment. Future changes to the relocated specification, such as a planned extension of the current 18-month surveillance interval, will be evaluated under the requirements of Section 50.59 of Title 10 of the Code of Federal Regulations (10 CFR), and the NRC will be informed of these changes in accordance with the requirements of 10 CFR 50.59(d)(2), as applicable, and the USAR update requirements of 10 CFR 50.71(e).

There is a discussion of SR 4.5.2.f included in the Bases associated with TS 3/4.5.2. In conjunction with, but separate from this license amendment application, the Bases discussion will also be relocated to the USAR TRM. This change will be made under the provisions of the DBNPS TS Bases Control program. The affected TS Bases pages, as currently existing, are included in Attachment 3 for information.

3.0 BACKGROUND

The vacuum leakage rate test of the watertight enclosure for valves DH-11 and DH-12 (SR 4.5.2.f) is required to be performed: (1) At least once per 18 months, (2) After each opening of the watertight enclosure, and (3) After any maintenance on or modification to the watertight enclosure which could affect its integrity. As stated in the TS Bases, this SR ensures that, at a minimum, the assumptions used in the safety analyses are met and that subsystem operability is maintained.

A watertight enclosure is required for valves DH-11 and DH-12 because these valves are located in an area which would be flooded following a LOCA, and the valves' electric motor operators, by themselves, are not qualified for submergence. The current TS Surveillance Requirement requires a vacuum leakage rate test of the watertight enclosure for valves DH-11 and DH-12 to assure their motor operators would remain unsubmerged for a period of up to 7 days following a LOCA. This ensures that the valves will remain capable of opening, providing a circulation flow path for reactor coolant. This circulation flow path will prevent boric acid concentration build-up and boric acid precipitation in the reactor vessel post-LOCA.

As described in USAR Section 6.3.3.1.2.1, "Boron Precipitation Control," two active means of ensuring the chemical additive concentration remains below its solubility limit throughout the post-accident cooling period are provided, a primary boron precipitation control (BPC) method and a backup method. The primary BPC method currently utilizes the auxiliary pressurizer spray (APS) flow path. For this method, High Pressure Injection (HPI) Pump 2 takes suction from the discharge of Decay Heat (DH)/Low Pressure Injection (LPI) Pump 2 to supply water to the APS line via a tie-line, providing dilution flow to the core via the pressurizer.

The backup BPC method utilizes one of the two operating DH/LPI pumps taking suction from the DH "drop line" via valves DH-11 and DH-12, and discharges a low (throttled) flow rate into the reactor vessel via the core flood nozzles. The flow through the drop line allows forward flow through the reactor vessel, so that any amount of flow of relatively low concentration water from the LPI train aligned to the containment emergency sump will enter and dilute the boric acid in the core. The backup BPC method would only be utilized if the primary method is unavailable and if both DH/LPI pumps are functioning.

During the ongoing Thirteenth Refueling Outage (13RFO), a plant modification to provide an improved BPC method is planned for implementation. The plant modification would add a new cross-tie line and associated valving and instrumentation, allowing the discharge of either DH/LPI pump to backflow through the DH-11 and DH-12 drop line and into the reactor vessel. This flow path would become the new primary BPC method. In addition, the current primary BPC method would become the backup BPC method, and the current backup BPC method would no longer be used. However, since the improved BPC methodology would still utilize a flow path through valves DH-11 and DH-12, the watertight enclosure remains an important design feature.

The watertight enclosure consists of a stainless steel-lined concrete cavity located at the lower elevation of the containment vessel. It is approximately nineteen feet-six inches long by seven feet wide by seven feet deep. During 13RFO, extensive modifications were made to the watertight enclosure to improve its sealing capability. Quarter-inch thick stainless steel plates were anchor-bolted to the four walls and the floor. For the top, quarter-inch thick stainless steel plates were mounted to the underside of existing structural steel supports, below the existing carbon steel checker plate. Expansion joints were incorporated into the design to accommodate differential expansion. Welded construction was used for joints and seams of the new stainless steel liner. In addition, the supporting structure for the top plates and access openings was

reinforced to improve the leaktightness of the seals for the removable manway and access covers.

The DBNPS is currently operating on a 24-month fuel cycle, whereas SR 4.5.2.f is required to be performed at an 18-month frequency. Therefore, in the event that an outage of sufficient duration does not occur during an operating cycle, an early plant shutdown to perform the test may be necessary. Upon relocation of the SR to the USAR TRM, DBNPS personnel plan to perform an evaluation under the 10 CFR 50.59 process to determine if the current surveillance interval can be changed to a refueling interval frequency, based in large part on the modifications made to improve the sealing capability of the watertight enclosure. This would eliminate the need to perform the surveillance in mid-cycle outages, resulting in reduced radiation exposure to personnel.

The conversion of SR 4.5.2.f to a refueling interval frequency would be consistent with the original licensing basis. The SR is a non-standard, plant-specific requirement that was added to the DBNPS TSs, with an 18-month frequency, at the time the Operating License was issued in 1977. This requirement was added due to the design of the watertight enclosure, and was based upon performing testing at a refueling outage frequency rather than at a fixed absolute timespan. Specifically, Section 6.3.3.5, "Submerged Valves," of Supplement 1 to the NRC's Operating License Safety Evaluation Report (NUREG-0136) refers to the NRC staff's requirement to perform "an acceptable leakage test of this enclosure at each refueling interval."

4.0 TECHNICAL ANALYSIS

The proposed amendment would relocate SR 4.5.2.f from ECCS TS 3/4.5.2 to the USAR TRM. This SR provides assurance that at least one BPC method will be available to prevent boric acid concentration build-up and boric acid precipitation in the reactor vessel post-LOCA. The SR is a non-standard, plant-specific requirement that was added to the existing TS 3/4.5.2 at the time the Operating License was issued in 1977. A specific LCO for BPC methods was not added at that time. In addition, there is no LCO for BPC methods in the improved Standard Technical Specifications for Babcock and Wilcox-type plants (NUREG-1430).

The removal of the detail for performing this SR from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also, this change is acceptable because the detail in this SR is being relocated to the USAR TRM and will therefore continue to be adequately controlled. Any changes to the USAR TRM are made under the requirements of Section 50.59 of Title 10 of the Code of Federal Regulations (10 CFR), which ensures changes are properly evaluated. It is concluded that there is no adverse effect on plant safety as a result of this relocation.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Technical Specification Limiting Condition for Operation 3.5.2 requires two independent Emergency Core Cooling Systems (ECCS) Subsystems to be operable. Surveillance Requirement 4.5.2.f requires each ECCS Subsystem to be demonstrated operable by performing a vacuum leakage rate test of the watertight enclosure for Decay Heat Removal System 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 seven (7) days following a Loss-of-Coolant Accident (LOCA). This surveillance requirement provides reasonable assurance that a circulation flow path will be maintained to prevent boric acid concentration build-up and boric acid precipitation in the reactor vessel post-LOCA. The proposed amendment would relocate Surveillance Requirement 4.5.2.f to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM). Future changes to the relocated specification will be evaluated under the requirements of Section 50.59 of Title 10 of the Code of Federal Regulations (10 CFR), and the NRC will be informed of these changes in accordance with the requirements of 10 CFR 50.59(d)(2), as applicable. and the USAR update requirements of 10 CFR 50.71(e).

An evaluation has been performed to determine whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Under the proposed change, initial conditions and assumptions remain as previously analyzed for accidents in the Davis-Besse Nuclear Power Station Updated Safety Analysis Report. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   Under the proposed change, the manner in which the watertight enclosure is sealed and tested is not altered, and the operability requirements of the watertight enclosure for Decay Heat Removal System valves DH-11 and

DH-12 will continue to be adequately addressed by testing. No different accident initiators or failure mechanisms are introduced by the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   Since there are no new or significant changes to the initial conditions contributing to accident severity or consequences, there are no significant reductions in a margin of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

As stated in the DBNPS Updated Safety Analysis Report (USAR), Appendix 3D, "Conformance with the NRC General Design Criteria, Safety Guides, and Information Guides," the design of the Davis-Besse Nuclear Power Station meets the intent of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," as published in the Federal Register on February 20, 1971, and as amended in the Federal Register on July 7, 1971.

Regarding General Design Criterion (GDC) 35, "Emergency Core Cooling," USAR Appendix 3D states:

> A system to provide abundant emergency core cooling is provided. The system safety function is to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities are provided to assure that for onsite electric power system operation (assuming that offsite power is not available) and for offsite electric power system operation (assuming that onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Abundant emergency core cooling is provided by the low-pressure injection (decay heat removal), high-pressure injection, and the core flooding systems. These three systems make up an Emergency Core Cooling System (ECCS) that maintains core cooling in the event of a loss-of-coolant accident (LOCA).

Redundancy of components, power supplies, and initiation logic and separation of functions are provided so that a single failure does not prevent the ECCS from fulfilling its function. The ECCS may be operated from either onsite or offsite power supplies.

The primary function of the Emergency Core Cooling System is to deliver cooling water to the reactor core in the event of a LOCA. The system provides protection for all potential break sizes in the reactor coolant system pressure boundary piping up to and including the double-ended rupture of the largest pipe. In addition, breaks in the High Pressure Injection line and the Core Flood Tank line are postulated.

The basic design criteria for loss-of-coolant accident evaluation are as follows:

a. The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

b. The calculated total oxidation of the fuel cladding shall not exceed 17% of the total cladding before oxidation.

c. The amount of hydrogen generated from cladding metal-water reaction does not exceed 1% of the total amount of cladding in the reactor.

d. The core geometry is maintained in a state that is amenable to cooling.

e. The cladding temperature is reduced and maintained at an acceptably low value and decay heat is removed for extended periods of time.

For a rupture in the steam piping, the Emergency Core Cooling System adds shutdown reactivity, so that with minimum tripped rod worth and minimum ECCS operation, the reactor core does not return to criticality; thus, there is no core damage.

Reference: USAR Chapter 6 and 10 CFR 50.46

Pursuant to 10 CFR Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," paragraph (a)(1)(ii), the Davis-Besse Nuclear Power Station Emergency Core Cooling System (ECCS) is modeled in conformance with the required and acceptable features of 10 CFR 50, Appendix K, "ECCS Evaluation Models." Compliance with the 10 CFR 50.46 acceptance criteria is described in USAR Section 6.3, "Emergency Core Cooling System." One of the 10 CFR 50.46 acceptance criteria pertains to establishment of a mode of long-term core cooling. USAR Section 6.3.3.1.2.1, "Boron Precipitation Control," describes the methodology to ensure that the boric acid concentration in the reactor vessel will remain dilute throughout the post-accident cooling period, thereby ensuring that the long-term cooling acceptance criterion is satisfied.

Under the proposed license amendment application, Surveillance Requirement 4.5.2.f, which is related to the boron precipitation control methodology, is being relocated from the Technical Specifications to the USAR Technical Requirements Manual. Compliance with the intent of 10 CFR 50, Appendix A, GDC 35, and the requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K are unaffected by the proposed relocation of the Surveillance Requirement.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment No. 254.

2. DBNPS Updated Safety Analysis Report through Revision No. 23.

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

4. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

5. 10 CFR 50.59, "Changes, Tests, and Experiments."

6. NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants," Revision 2.

7. Engineering Change Request (ECR) 03-0146-00, "Boron Precipitation Control Modification."

8.0 ATTACHMENTS

1. Proposed Mark-Up of Technical Specification Pages

2. Proposed Retyped Technical Specification Pages

3. Technical Specification Bases Pages

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Attachment 1

PROPOSED MARK-UP
OF
TECHNICAL SPECIFICATION PAGES

(4 pages follow)

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 HPI train inoperable, restore the inoperable HPI train to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With one LPI train or its associated decay heat cooler inoperable, restore the inoperable equipment to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

c. 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.

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
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THIS PAGE PROVIDED FOR INFORMATION ONLY

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once each REFUELING INTERVAL, 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 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 Allowable Value (<328 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 Allowable Value (<328 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

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, 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.~~

~~The inspection port on the watertight enclosure may be opened without requiring performance of the vacuum leakage rate test, to perform inspections. After use, the inspection port must be verified as closed in its correct position. Provisions of TS 3.0.3 are not applicable during these inspections.~~Deleted

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.

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**

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
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* 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.

PROPOSED RETYPED
TECHNICAL SPECIFICATION PAGES

(1 page follows)

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, 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. Deleted

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.

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Attachment 3

TECHNICAL SPECIFICATION BASES PAGES

(4 pages follow)

Note: The Bases pages are provided for information only.

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. Each ECCS subsystem consists of one High Pressure Injection (HPI) train, one Low Pressure Injection (LPI) train (including the associated decay heat cooler), and the necessary piping, valves, instrumentation and controls to provide the required flowpaths from the Borated Water Storage

Tank (BWST) or the Containment Emergency Sump to the reactor vessel. 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. With RCS average temperature ≥ 280°F, the Limiting Condition for Operation (LCO) requires the OPERABILITY of a number of independent trains, the inoperability of one component in a train does not necessarily render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this LCO is to maintain a combination of equipment such that 100% of the safety injection flow equivalent to 100% of a single subsystem remains available. This allows increased flexibility in plant operations under circumstances when components in opposite subsystems are inoperable.

With one or more components inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS subsystem is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

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.

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.

Decay Heat Removal System valves DH-11 and DH-12 are located in an area that would be flooded following a LOCA. These valves are located in a watertight enclosure to ensure their operability up to seven days following a LOCA. Surveillance Requirements are provided to verify the acceptable leak tightness of this enclosure. An inspection port is located on this watertight enclosure, which is typically used for performing inspections inside the enclosure. During the vacuum leakage rate test, the inspection port is in a closed position and subject to the test. This inspection port may be subsequently opened for use in viewing inside the enclosure. Opening this inspection port will not require performance of the vacuum leakage rate test because of the design of the closure fitting, which will preclude leakage under LOCA conditions, when properly installed. Proper installation includes independent verification.

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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.

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.

The Decay Heat Isolation Valve and Pressurizer Heater Interlock setpoint is based on preventing over-pressurization of the Decay Heat Removal System normal suction line piping. The value stated is the RCS pressure at the sensing instrument's tap. It has been adjusted to reflect the elevation difference between the sensor's location and the pipe of concern.

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 (500,100 gallons of borated water, conservatively rounded up from the calculated value of 500,051 gallons) 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

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COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THE SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY FOR INFORMATION AND ARE NOT REGULATORY COMMITMENTS. PLEASE NOTIFY THE MANAGER – REGULATORY AFFAIRS (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	DUE DATE
Relocate Surveillance Requirement (SR) 4.5.2.f to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM).	Upon implementation of the approved license amendment.