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Docket Number 50-346

License Number NPF-3

Serial Number 2664

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United States Nuclear Regulatory Commission
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Subject: License Amendment Application to Revise Technical Specification 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$ (License Amendment Request No. 97-0007)

Ladies and Gentlemen:

Enclosed is an application for an amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specifications. The proposed change involves Technical Specification (TS) 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$.

Technical Specification (TS) 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 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). The test 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. The SR for the watertight enclosure provides assurance that the backup 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 change to SR 4.5.2.f would modify the presently specified 18 month surveillance frequency in SR 4.5.2.f.1 to a new specified frequency of 24 months. The proposed surveillance frequency change has been prepared in accordance with the NRC

A001

guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to accommodate a 24-Month Fuel Cycle."

The DBNPS is currently operating on a 24-month fuel cycle. The 18-month surveillance test for the watertight enclosure for valves DH-11 and DH-12 was last performed during the Twelfth Refueling Outage (12RFO) in May, 2000, and is next due in November, 2001, which is during the current operating Cycle 13. Utilizing the 25% surveillance interval extension under the provisions of Technical Specification 4.0.2, the surveillance test will reach its late date, (125% of the current 18-month interval) on March 25, 2002. The next refueling outage, 13RFO, is scheduled to commence soon thereafter. The proposed change would prevent an early plant shutdown to perform the test. Therefore, it is requested that this license amendment application be approved by the NRC by October 31, 2001 for the current operating cycle (Cycle 13).

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

Very truly yours,

MKL/laj

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, NRC/NRR Project Manager
D. J. Shipley, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
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APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached are the requested changes to the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3. Also included are the Safety Assessment and Significant Hazards Consideration and the Environmental Assessment.

The proposed changes (submitted under cover letter Serial Number 2664) concern:

Appendix A, Technical Specifications (TS):

3/4.5.2 Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

By: Guy G. Campbell
Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me this 30th day of March, 2001.

Nora L. Flood
Notary Public, State of Ohio - Nora L. Flood
My Commission expires September 4, 2002.

The following information is provided to support issuance of the requested change to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specification (TS) 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$.

A. Time Required to Implement: The License Amendment associated with this license amendment application is to be implemented within 120 days following NRC issuance.

B. Reason for Change (License Amendment Request Number 97-0007):

The proposed change to SR 4.5.2.f would modify the presently specified 18 month surveillance frequency in SR 4.5.2.f.1 to a new specified frequency of 24 months. The proposed surveillance frequency change has been prepared in accordance with the NRC guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to accommodate a 24-Month Fuel Cycle."

The DBNPS is currently operating on a 24-month fuel cycle. The 18-month surveillance test for the watertight enclosure for valves DH-11 and DH-12 was last performed during the Twelfth Refueling Outage (12RFO) in May, 2000, and is next due in November, 2001, which is during the current operating Cycle 13. Utilizing the 25% surveillance interval extension under the provisions of TS 4.0.2, the surveillance test will reach its late date, (125% of the current 18-month interval) on March 25, 2002. The next refueling outage, 13RFO, is scheduled to commence soon thereafter. The proposed change would prevent an early plant shutdown to perform the test. Therefore, it is requested that this license amendment application be approved by the NRC for the current operating cycle (Cycle 13).

C. Safety Assessment and Significant Hazards Consideration: See Attachment 1.

D. Environmental Assessment: See Attachment 2.

Docket Number 50-346
License Number NPF-3
Serial Number 2664
Attachment 1

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 97-0007

(20 pages follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 97-0007

TITLE:

Proposed Modification to the Davis-Besse Nuclear Power Station Unit Number 1, Facility Operating License NPF-3, Appendix A Technical Specifications, to Revise Technical Specification (TS) 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems – $T_{avg} \geq 280°F$, to Increase the Surveillance Interval for Surveillance Requirement 4.5.2.f.1.

DESCRIPTION:

Background

Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1 Technical Specification (TS) Limiting Condition for Operation (LCO) 3.5.2 requires two independent Emergency Core Cooling Systems (ECCS) subsystems to be operable in Mode 1 (Power Operation) through Mode 3 (Hot Standby). 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 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). The test 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. The SR for the watertight enclosure provides assurance that the backup circulation flow path will be maintained to prevent boric acid concentration build-up and boric acid precipitation in the reactor vessel post-LOCA.

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 Loss-of-Coolant Accident (LOCA), and the valves' electric motor operators, by themselves, are not qualified for submergence. The current TS requirement is that the motor operators must remain unsubmerged for a period of up to 7 days following a LOCA to ensure their operability to open the valves to provide a circulation flow path for reactor coolant. As described later in this license amendment request, the watertight enclosure for the valves consists of a concrete walled pit covered with a reinforced steel top, sealed in place.

The proposed change would modify the presently specified 18 month surveillance frequency in SR 4.5.2.f.1 to a new specified frequency of at least once per Refueling Interval. A "Refueling Interval" is presently defined by TS Definition 1.42 as "a period of time ≤ 730 days."

The DBNPS is currently operating on a 24-month fuel cycle. The proposed change is desired to avoid the need for a shutdown and cooldown solely to perform the test. The 18-month surveillance test was last performed during the Twelfth Refueling Outage (12RFO) in May, 2000, and is next due in November, 2001, which is during the current operating Cycle 13. Utilizing the 25% surveillance interval extension under the provisions of Specification 4.0.2, the surveillance test will reach its late date, (125% of the current 18-month interval) on March 25, 2002. The next refueling outage, 13RFO, is scheduled to commence soon thereafter. Therefore, the proposed change would prevent an early plant shutdown to perform the test.

As described below, the proposed surveillance frequency change has been prepared in accordance with the NRC guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

The proposed change is shown on the attached marked-up Operating License Appendix A, Technical Specification page.

Licensing Basis for Vacuum Test of Watertight Enclosure

The NRC guidance provided by Generic Letter 91-04 was utilized in the preparation of this license amendment application. This included the completion of a review of the licensing basis for SR 4.5.2.f.1.

Surveillance Requirement 4.5.2.f.1 is a non-standard, plant-specific requirement that was added to the DBNPS Technical Specifications, 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 valve pit cover, 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."

The proposed revision to SR 4.5.2.f.1 would increase the time between tests, however, with the DBNPS currently operating on a 24-month fuel cycle, testing would be retained on a refueling outage frequency schedule, consistent with the original licensing basis.

The periodic surveillance test is typically performed near the end of each refueling outage, prior to entry into Mode 4 (Hot Shutdown). With the exception of the infrequent use of a small inspection port, the sealant mechanisms are in a passive, undisturbed state, with no interfacing moving parts that would cause seal wear or damage, from the time the watertight enclosure successfully passes its vacuum leakrate surveillance test near the end of a refueling outage, through the operating cycle, to the beginning of the following refueling outage. In addition, the sealant materials are not subject to degradation under normal operating conditions or under expected post-LOCA conditions, as discussed under the "Maintenance

Records Review" section below. Furthermore, the location of the valves and watertight enclosure are within the containment building itself and not normally accessed during plant operation, thereby preventing the likelihood of inadvertent damage. Based on these factors, the likelihood of the watertight enclosure developing a significant leak during plant operation over the extended interval between refueling outages is considered to be small, and the original licensing basis of SR 4.5.2.f.1 is preserved.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The activity affected by the proposed revision involves performance of the vacuum leakage rate surveillance test of the watertight enclosure for Decay Heat Removal (DHR) System valves DH-11 and DH-12. These valves are part of the decay heat /low-pressure injection (LPI) system. Changes are proposed to the required frequency of performance of the test.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

The LPI system provides a source of borated water directly to the reactor vessel following a large break LOCA. It also is used to provide long term core cooling by recirculating water from the containment emergency sump to the core. It can be used to supply borated water to the suction of the High Pressure Injection Pumps, to supply auxiliary spray to the pressurizer and to provide long term boric acid dilution for the reactor vessel. In its normal mode of operation, the system is used to remove heat from the Reactor Coolant System (RCS) and decay heat from the core during plant cooldowns and shutdowns.

The safety function of Decay Heat Removal (DHR) isolation valves DH-11 and DH-12 is to isolate the RCS from the DHR system when the RCS pressure is greater than the pressure rating of the DHR system. These valves also provide a circulation flow path to prevent boric acid concentration build-up and boric acid precipitation post-LOCA. As described in USAR Section 6.3.3.1.2.1, "Boron Precipitation Control," this flowpath is utilized as the backup active boric acid precipitation control (BPC) method. Prior to the implementation of a plant modification in 12RFO (Spring 2000), which made significant improvements in the post-LOCA BPC methodology, the DH-11 and DH-12 flowpath was the primary BPC method. The primary BPC method now utilizes the pressurizer auxiliary spray flowpath. The backup BPC method would only be utilized if the primary method is unavailable and if both DHR/LPI pumps are functioning. However, in order to preserve the viability of the backup BPC method should the primary method fail and the watertight pit leak excessively, valves DH-11 and DH-12 would be opened soon after switchover from the borated water storage tank to the ECCS emergency sump, provided the RCS is within the design pressure and temperature range for the DHR piping and components.

The motor operators for valves DH-11 and DH-12 are located below the expected post-LOCA water level in containment and are not qualified for submergence by themselves. The watertight enclosure, also called the "decay heat valve pit," ensures that the motor operators on valves DH-11 and DH-12 will not be flooded for at least 7 days following a LOCA.

The purpose of the vacuum leakage testing activities required by SR 4.5.2.f is to ensure that the watertight enclosure is capable of performing its function.

The valve pit is a five-sided (four walls and bottom) concrete cavity with a reinforced steel decking closure on the top, located at the lower elevation of the containment vessel. The valve pit is approximately twenty feet long by seven feet wide by seven feet deep. There are several penetrations through the valve pit cover: access hatches to allow for personnel to enter the valve pit to conduct maintenance on equipment located in the valve pit; piping penetrations, including the Decay Heat Removal line exit from the valve pit, and a vent from the pit, extending above the maximum post-LOCA water level in containment, that serves to equalize pressure inside and outside the pit; electrical penetrations; and an inspection port penetration.

In addition to the valve pit cover penetrations, there is a steel-framed penetration made of reinforced steel plate in one of the walls of the valve pit where the Decay Heat Removal line enters.

The valve pit cover also serves as a portion of the 565' containment elevation floor decking, along a normal ingress/egress route for some refueling outage maintenance activities. As such, the valve pit cover is designed to support personnel and equipment loads.

The support steel for the cover plate along three of the four walls is comprised of angle shapes welded to create a ledge that is embedded in the concrete. This support steel is sealed to the concrete by a Bisco boot attached to the angle shapes and the concrete on the inside of the valve pit, and caulked with sealant at the seam between the structural steel and concrete on the exterior of the valve pit. The fourth wall and the penetration in the side wall of the valve pit are sealed to the concrete side walls by a 1/4" thick asbestos gasket that is compressed between the support steel and the concrete. The cover plate for the valve pit and the steel plate for the pipe penetration in the side wall are welded to the support steel. The gasket joint between the support steel and the concrete is then caulked with sealant on the exterior of the valve pit.

The access hatches in the valve pit cover are sealed by a 1-1/4" wide by 1/2" thick silicon rubber gasket that is compressed between the structural support steel and the access hatches. These sealed areas are then caulked with sealant on the exterior of the valve pit.

The 12-inch (nominal) DHR piping penetrates the valve pit via pipe sleeves that are welded to the steel plate. The seal of the pipe penetrations is accomplished by clamped connections formed by the banding of a Bisco pipe boot to the sleeve and the pipe. These sealed connections are then caulked with sealant at the seam in the Bisco pipe boot and on the exterior enclosure at the banded joints.

The 8-inch (nominal) piping for the valve pit vent is welded to the steel plate.

The electrical penetrations are formed by a threaded conduit connector that is welded to the cover plate. The conduit connectors are then filled with silicon foam to create a watertight seal of the penetration.

The inspection port penetration consists of a 4-inch (nominal) pipe stub welded to the pit cover and sealed with a 4-inch (nominal) adapter and cap ("Kamlok" coupling) manufactured by Dover Corporation/OPW Division. This coupling is designed to be removed and reinstalled, and includes an integral viton gasket for sealing.

EFFECTS ON SAFETY:

Historical surveillance test data and maintenance records were reviewed in evaluating the effect on safety. The results are summarized below. A perspective on the impact of the proposed changes on risk is also provided below.

Surveillance Data Review

The 18-month TS surveillance test results data for the watertight enclosure were reviewed for the period of the Fifth Refueling Outage (5RFO) through 12RFO. In addition, the results of surveillance tests performed in May, 1997 (Cycle 11 forced outage), and in May, 1999 (Cycle 12 mid-cycle outage) were reviewed. This period spans more than an eleven-year period, and includes eight refueling outages.

As specified in SR 4.5.2.f, the surveillance test is performed by vacuum leak test. The initial test vacuum creates a differential pressure across the enclosure equal to the expected pressure created by the post-LOCA containment water level, thereby verifying the structural adequacy of the enclosure. Confirmation that the valve pit is sufficiently leaktight is through subsequent observation of the vacuum decay.

The periodic surveillance test is typically performed near the end of each refueling outage, prior to entry into Mode 4. There are several reasons why the test is performed at this time:

- The vacuum test is best performed in Modes 5 or 6 for proper test conditions. During these Modes, the Decay Heat Removal System is typically in service. In order to perform the surveillance test within the assumptions of the calculation that determined the test acceptance criteria, it is important that the containment air temperature and the valve pit air temperature be as constant as possible during the air vacuum test. Such constant temperature conditions are much more difficult to achieve at the beginning of an outage than near the end of an outage since decay heat loads are changing much more rapidly at the beginning of an outage. Due to the changing decay heat load at the beginning of an outage, the temperature of the fluid in the Decay Heat Removal System line running through the valve pit varies, resulting in more air temperature variation in the valve pit, and thereby potentially affecting the results of the vacuum test.

- Maintenance activities are typically required to be performed on the components in the valve pit as well as on the valve pit itself. In accordance with SR 4.5.2.f.2, when the

watertight enclosure is opened, the vacuum test is required. Scheduling a single test near the end of the refueling outage, following completion of all maintenance activities, satisfies both SR 4.5.2.f.1 and SR 4.5.2.f.2.

- As previously described, the valve pit cover serves as a portion of the 565' containment elevation floor decking, along a normal ingress/egress route for some refueling outage maintenance activities. This could potentially make the valve pit cover seals susceptible to inadvertent damage due to the personnel/equipment traffic during the refueling outage. Scheduling the test near the end of the refueling outage minimizes the potential need to reperform the test should the seal become damaged after performing an earlier test.

A review of the surveillance test history shows that the vacuum leak surveillance test has been successfully performed each refueling outage during the review period, as required prior to plant startup. The surveillance test was also successfully performed during the May, 1997 Cycle 11 forced outage, and the May, 1999 Cycle 12 mid-cycle outage.

In the event that initial surveillance test results indicate excessive leakage, additional maintenance activities are performed on the valve pit seals, typically consisting of application of additional sealant material to areas where leaks are detected via soap bubble testing with the valve pit slightly pressurized. Pressurization testing is further described under the "Maintenance Records Review" section below.

It is noted that since the vacuum leak surveillance testing is as-left, as opposed to as-found, the potential effects of the proposed increase in the surveillance interval cannot be determined based on this data. However, as described in further detail below, the valve pit seal materials are not subject to degradation under normal operating conditions or under expected post-LOCA conditions, therefore a significant increase in failure rate due to the proposed change would not be expected. Furthermore, some as-found pressurization tests have been conducted in the past with favorable results, as discussed in the "Maintenance Records Review" section below.

The potential failure modes that may result in the watertight enclosure losing leaktight integrity are physical damage to the boots or seals, breakdown of the sealant as a result of chemical contact, radiation, aging or fatigue, or a physical change in the structure that is sealed. The proposed lengthening of the surveillance interval does not result in any changes to the environment of the sealing materials, nor does it cause a physical change to the valve pit sealing, therefore no new failure mechanisms are introduced. Following final sealing near the end of the refueling outage, with the exception of the infrequent use of a small inspection port, the watertight enclosure remains in a passive, undisturbed state with no interfacing moving parts that could lead to a loss of leakage integrity.

Maintenance Records Review

A maintenance records review was performed for the watertight enclosure for the period of 5RFO through 12RFO. This period spans more than an eleven-year period, and includes eight refueling outages.

The routine watertight enclosure maintenance, conducted during a refueling outage, consists of three activities: removing hatches to provide access to equipment located in the valve pit; inspecting and repairing any physical damage to the seal materials that result from the cover plates also functioning as a floor deck; and detecting and sealing leaks.

As an optional maintenance activity, the watertight enclosure can be slightly pressurized. Soap bubbles are then used to detect any gross leakage. This pressurization test may cause failures of the seal that would not necessarily have occurred during performance of the SR vacuum test; nevertheless, it is a very useful method of determining the location of leakage. Any adverse effects as a result of the pressurization test would be detected in the SR vacuum test, which is conducted later in the outage.

A Maintenance Work Order (MWO) record review indicated that during 5RFO, prior to performing any maintenance or testing on the watertight enclosure, mechanical damage was observed on at least one boot seal. The cause and actual time of occurrence of this mechanical damage was not determined. However, since there are no moving parts in this area during plant operation, it is likely that the damage occurred during the refueling outage.

For the reasons discussed above, as-found vacuum testing of the enclosure has not historically been performed. However, as-found pressurization testing has been performed intermittently in the past in an effort to identify any potential leakage paths, and the following data is available:

> During 6RFO, an as-found (i.e., prior to any maintenance on the valve pit) pressurization test was performed. The MWO record indicates that there were no leaks found. During 7RFO and 8RFO, as-found pressurization tests were also conducted. The MWO records do not indicate the discovery of any leakage.
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> During 11RFO, an as-found pressurization test was performed and a minor leak was found. However, the quantity of the leakage was indeterminate, and the cause of the leak could have been due to the pressurization test itself. Therefore, these results are judged as inconclusive.

Although this as-found pressurization test data is limited, it does provide indication of the acceptable performance of the enclosure between refueling outages.

A review of the MWOs did not identify any defects/failure of seal materials typically associated with aging or fatigue.

The concrete at the structural steel-to-concrete interface at the top of the valve pit, as well as at the penetration at the wall of the valve pit (where the Decay Heat Removal line enters), has experienced highly localized spalling and cracking. Accordingly, during a plant outage in May, 1997, Belzona Super Metal 1111 was applied to these areas to re-establish the steel-to-concrete contours. GE RTV 106 caulking was then applied to seal the areas where the Belzona material was applied. Restoring the contours with the Belzona material reduced

the surface area of the opening sealed by the caulking material, resulting in higher stability for the sealed joint.

The Belzona Super Metal 1111 material is a polymer with superior tensile strength when compared to concrete. The material has been tested as a coating for steel in accordance with ASTM D 3911, "Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions," and ASTM D 4082, "Standard Test Methods for Effects of Gamma Radiation on Coatings for Use in Light-Water Nuclear Power Plants," for DBA conditions that exceed the DBNPS post-accident containment conditions. These attributes provide confidence that the material is suitable for the DBNPS application.

In assessing the suitability of the Belzona Super Metal 1111 material for the DBNPS application, it was noted that the material exhibited cracking when it underwent the previously described qualification testing as a coating on steel. However, the test acceptance criteria stated that cracking is not considered a failure unless accompanied by delamination or a loss of adhesion, and no delamination or loss of adhesion occurred. Since the test results demonstrated that the Belzona material is not subject to tearing or delamination from the surface when subjected to environmental conditions that exceeded the DBNPS DBA conditions, this provides assurance that the material will remain in place for the DBNPS application. Further, as previously described, for the DBNPS application, the Belzona material is sealed with GE RTV 106 caulking, providing added assurance that if any cracking of the Belzona material did occur, or if any separation of the Belzona material from the concrete did occur, the capability of the joint to prevent leakage would not be impacted.

The surveillance testing performed during the May, 1997 forced outage and during subsequent outages has confirmed the adequacy of Belzona Super Metal 1111 for the general service application (as a material suitable to restore to essentially original contour the concrete to steel interface). There has been no evidence of loss of adhesion to the concrete or steel in the areas where the Belzona material was applied, and no evidence of any breakdown of the material.

As stated previously, with the exception of the infrequent use of a small inspection port, the watertight enclosure is in a passive, undisturbed state during plant operation, and there are no interfacing moving parts that could lead to a loss of leakage integrity.

The average ambient temperature in the area of the valve pit is 74 °F. The estimated total radiation dose for the valve pit area is 9.58E6 rads, which includes a 40-year normal dose of 6.94E4 rads, followed by a 7-day post-LOCA dose comprised of 4.47E5 rads airborne gamma, 3.88E6 rads sump water gamma, 2.35E6 rads airborne beta, and 2.83E6 rads sump water beta. As described in USAR Section 3.11.2, "Methodology for Development of LOCA and HELB Environments," the maximum post-LOCA temperature and pressure is 283 °F and 36.95 psig, respectively, and equipment inside containment is exposed to chemical spray consisting of up to 2800 ppm boron. The top of the valve pit would be submerged beneath approximately 7 feet-2 inches of water post-LOCA. All of the materials used in sealing the valve pit (i.e., sealants, gaskets, and boot material) have been evaluated for these conditions.

These reviews have demonstrated that under the normal operating conditions that these materials are subjected to, there will be no deleterious effect on the materials. Furthermore, the materials have been evaluated for post-LOCA conditions to verify that the materials will remain capable of performing their intended function of sealing the valve pit.

Accordingly, no additional actions are necessary or recommended for the proposed increase in the present surveillance interval.

Risk Perspective

A summary of DBNPS Calculation C-NSA-099.16-26 (Reference 7) was previously provided to the NRC in support of a request for an exemption from 10 CFR 50, Appendix K, for BPC methodology (Reference 8). Calculation C-NSA-099.16-26 provided a detailed analysis of the core damage frequency (CDF) and the large early release frequency (LERF) associated with the failure of post-LOCA methods of boric acid precipitation control (BPC). The results of the NRC review of the calculation are provided in the safety evaluation issued with the exemption (Reference 9).

Calculation C-NSA-099.16-26 obtained a bounding frequency of 1.1E-7 for sequences in which failure of BPC led to core damage. This evaluation was bounding because it was based on several conservative assumptions, including not crediting passive methods of BPC, and using LOCA frequencies that would represent a broader range of LOCA sizes and locations than would actually have the potential to require BPC.

To support LAR 97-0007, Calculation C-NSA-099.16-26 was revised to provide an estimate of the CDF increase associated with the proposed SR 4.5.2.f test interval increase. To perform this analysis a failure rate for the watertight enclosure was estimated based on results of as-found pressurization testing. A failure rate of approximately 1.0E-5 / hour was estimated based on the as-found pressurization testing performed during 6RFO, 7RFO, 8RFO and 11RFO. Due to the uncertainty of this failure rate evaluation, sensitivity calculations were performed using a failure rate a factor of two higher and lower than 1.0E-5 / hour. Using these failure rates, the CDF was calculated with an 18 month test interval and a 24 month test interval. The results of this calculation, even assuming the higher failure rate of 2.0E-5 / hour, demonstrate that the increase in CDF due to the increased test interval is less than 1.0E-8 / year. Compared to the current DBNPS average CDF of about 1.5E-5 / year, the contribution from extending the test interval is negligible.

The negligible contribution to CDF from increasing the decay heat watertight enclosure test interval is consistent with previous analysis. Calculation C-NSA-099.16-26 previously determined the increase in CDF due to the unavailability of the backup method of BPC, which relies on the use of the decay heat drop line, to be about 2.0E-7 / year. Therefore, the CDF contribution from increasing the watertight enclosure test interval would be expected to be a fraction of the increase due to the total unavailability of the backup method.

The large early release frequency (LERF) for all BPC failures was very small. For the changes in test interval, the LERF would be less than 1.0E-11 / year. The very small LERF

contribution is to be expected because the reactor coolant system is depressurized and, at a minimum, the BWST would be injected.

In conclusion, the contribution to both CDF and LERF from the proposed change to SR 4.5.2.f is negligible.

Conclusion

Based on the historical good performance of the watertight enclosure, the low potential for significant increases in failure rates of the watertight enclosure under a longer interval between tests, the introduction of no new failure modes, and the insignificant impact on risk, it is concluded that it is acceptable to increase the surveillance interval for TS SR 4.5.2.f.1 from 18 to 24 months and that there is no adverse effect on nuclear safety.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazard exists due to a proposed amendment to an Operating License for a facility. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed changes. Initial conditions and assumptions remain as previously analyzed for accidents in the Davis-Besse Nuclear Power Station Updated Safety Analysis Report.

The proposed changes would increase the surveillance test interval in Technical Specification 4.5.2.f.1 from 18 to 24 months for the vacuum leakage rate test of the watertight enclosure for Decay Heat Removal System valves DH-11 and DH-12. The surveillance data and maintenance records have been reviewed and support an increase in the surveillance test interval from 18 to 24 months based on the low potential for a significant increase in the failure rate of the watertight enclosure due to an increased surveillance interval, and based on the introduction of no new failure modes. The proposed change to the surveillance interval has been evaluated consistent with the NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. The watertight enclosure and its condition do not contribute to the initiation of any accident. Therefore, the probability of any accident previously evaluated is not increased.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the integrity of the watertight enclosure sealing mechanisms has been evaluated, and it has been determined that the sealing mechanisms will remain intact for the proposed increased surveillance interval. Therefore, there is assurance that the backup boric acid precipitation control flow path will remain available, so that there will be no impact on the source term, containment isolation or radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not alter the manner in which the watertight enclosure is sealed or tested, and the operability requirements of Decay Heat Removal System valves DH-11 and DH-12 will continue to be adequately addressed by Surveillance Requirement 4.5.2.f.1.

   No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval. An increase in the surveillance test interval from 18 to 24 months is justified based on the low potential for a significant increase in the failure rate of the watertight enclosure due to an increased surveillance interval, and based on the introduction of no new failure modes.

   No different accident initiators or failure mechanisms are introduced by the proposed change. Thus, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Not involve a significant reduction in a margin of safety.

   An increase in the surveillance test interval from 18 to 24 months is justified based on the low potential for a significant increase in the failure rate of the watertight enclosure due to an increased surveillance interval, and based on the introduction of no new failure modes.

   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.

CONCLUSION:

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

ATTACHMENT:

Attached are the proposed marked-up changes to the Operating License.

REFERENCES:

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment 244.

2. DBNPS Updated Safety Analysis Report through Revision 22.

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

4. NUREG-0136, NRC Safety Evaluation Report for the Davis-Besse Nuclear Power Station Unit 1 Operating License, December, 1976, and Supplement 1, April, 1977.

5. ASTM D 3911, "Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions."

6. ASTM D 4082, "Standard Test Methods for Effects of Gamma Radiation on Coatings for Use in Light-Water Nuclear Power Plants."

7. DBNPS Calculation C-NSA-099.16-26, "Long Term Boron Dilution Modification 97-0074," Revision 0, January, 2000, through Revision 4, December, 2000.

8. DBNPS letter dated March 15, 2000 (DBNPS Serial Number 2633), as supplemented by letter dated April 3, 2000 (DBNPS Serial Number 2652).

9. NRC letter dated May 5, 2000 (DBNPS Log Number 5659)(TAC No. MA7831).

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4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation.

The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed.

The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable (equipment inoperability) outage time limits of the ACTION requirements are less than 24 hours.

Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a.

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

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ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 2667 Date 11/9/00

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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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

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, 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 each REFUELING INTERVAL~~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.

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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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 $\geq$ 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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BASES

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 2667 Date 11/9/00

3/4.5.1 CORE FLOODING TANKS

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

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

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

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

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

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

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

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

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

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

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

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

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

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.

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

ENVIRONMENTAL ASSESSMENT
FOR
LICENSE AMENDMENT REQUEST NUMBER 97-0007

Identification of Proposed Action

Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1 Technical Specification (TS) 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 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). The test 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. This SR ensures that, at a minimum, the assumptions used in the safety analyses are met and that subsystem operability is maintained. The SR for the watertight enclosure provides assurance that the backup circulation flow path will be maintained to prevent boric acid concentration build-up and boric acid precipitation in the reactor vessel post-LOCA.

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 are not qualified for submergence by themselves. The current TS requirement is that the motor operators must remain unsubmerged for a period of up to 7 days following a LOCA to ensure their operability to open the valves to provide a circulation flow path for reactor coolant.

The proposed change to SR 4.5.2.f would modify the presently specified 18 month surveillance frequency in SR 4.5.2.f.1 to a new specified frequency of at least once per 24 months. The DBNPS is currently operating on a 24-month fuel cycle.

Need for the Proposed Action

The proposed change to the test surveillance interval is desired to avoid the need for a shutdown and cooldown solely to perform the test. The 18-month surveillance test was last performed during the Twelfth Refueling Outage (12RFO) in May, 2000, and is next due in November, 2001, which is during the current operating Cycle 13. The surveillance test will reach its late date, (125% of the current 18-month interval) on March 25, 2002. The next refueling outage, 13RFO, is scheduled to commence soon

thereafter. Therefore, the proposed change would prevent an early plant shutdown to perform the test.

Environmental Impact of the Proposed Action

As described in the Safety Assessment and Significant Hazards Consideration (SASHC) for the proposed license amendment application, the DBNPS has determined that the structures, systems, and components which could be affected by the proposed license amendment will continue to be capable of performing their safety functions.

The proposed license amendment application involves a change to a requirement with respect to the use of plant components located within the restricted area as defined in 10 CFR Part 20. As discussed in the SASHC, this proposed license amendment does not involve a significant hazards consideration. The proposed changes do not alter source terms, containment isolation, or allowable releases. In addition, the proposed changes do not involve an increase in the amounts, and no change in the types, of any radiological effluents that may be allowed to be released offsite. Furthermore, there is no increase in the individual or cumulative occupational radiation exposure.

With regard to potential non-radiological impacts, the proposed license amendment involves no increase in the amounts or change in the types of any non-radiological effluents that may be released offsite, and has no other environmental impact.

Based on the above, the DBNPS concludes that there are no significant radiological or non-radiological environmental impacts associated with the proposed license amendment.

Alternatives to the Proposed Action

Since the DBNPS has concluded that the environmental effects of the proposed action are not significant, any alternatives will have only similar or greater environmental impacts. The principal alternative would be to not grant the license amendment. Since the environmental impacts of the proposed action are not significant, denial of the proposed license amendment would not significantly reduce the environmental impacts attributable to the plant.

Alternative Use of Resources

This action does not involve the use of resources not previously considered in the Final Environmental Statement Related to the Operation of the Davis-Besse Nuclear Power Station, Unit Number 1 (NUREG 75/097).

Finding of No Significant Impact

The DBNPS has reviewed the proposed license amendment against the categorical exclusion criteria of 10 CFR 51.22(c)(9) for an environmental assessment. As demonstrated in the proposed license amendment's SASHC, the proposed changes do not involve a significant hazards consideration. In addition, the proposed changes do not increase the types or amounts of effluents that may be released offsite, and do not increase individual or cumulative occupational radiation exposures. Accordingly, the DBNPS finds that the proposed license amendment, if approved by the Nuclear Regulatory Commission, will have no significant impact on the environment and that no environmental assessment is required.

Docket Number 50-346
License Number NPF-3
Serial Number 2664
Enclosure 2

PROPOSED TECHNICAL SPECIFICATION CHANGES
REVISION BAR FORMAT

(1 page follows)

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, 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 each REFUELING INTERVAL.

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.

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.

Docket Number 50-346
License Number NPF-3
Serial Number 2664
Enclosure 3

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
None	N/A