FENOC
FirstEnergy Nuclear Operating Company

5501 North State Route 2
Oak Harbor, Ohio 43449

Mark B. Bezilla
Vice President - Nuclear

419-321-7676
Fax: 419-321-7582

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License Number NPF-3

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Subject: Davis-Besse Nuclear Power Station Technical Specifications Bases Update

Ladies and Gentlemen:

This letter is submitted in accordance with the requirements of the Davis-Besse Nuclear Power Station (DBNPS) Technical Specification (TS) 6.17, "Technical Specifications (TS) Bases Control Program." Technical Specification 6.17.d requires the FirstEnergy Nuclear Operating Company (FENOC) to submit to the NRC on a frequency consistent with 10 CFR 50.71(e), changes made to the TS Bases and implemented without NRC prior approval. This submittal reflects the changes to the Technical Specification Bases made and implemented by FENOC without prior NRC approval from October 1, 2002, through May 21, 2004. Technical Specification Bases changes issued by the NRC following adoption of the DBNPS Technical Specification Bases Control Program are not included in this submittal.

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

Very truly yours,

MAR
Enclosures

cc: J. L. Caldwell, Regional Administrator, NRC Region III
J. B. Hopkins, DB-1 NRC/NRR Senior Project Manager
C. S. Thomas, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

A001

Docket Number 50-346
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Enclosure 1

List of
Technical Specification Bases Changes
(For Information Only)

LAR No.	Affected TS Bases Section	Description
99-0004	3/4.3.1 and 3/4.3.2	This change supported License Amendment 259 to clarify the application of the 8-hour allowance for Safety Features Actuation System channel functional testing.
01-0012	3/4.4.4	This change supported License Amendment 255, which revised requirements for pressurizer level.
01-0017	3/4.0	This change supported License Amendment 254, which revised requirements for missed surveillances.
03-0002	3/4.5.2 and 3/4.5.3	This change supported License Amendment 256, which revised Emergency Core Cooling System flow testing requirements.
03-0003	3/4.8	This change clarified what is required for performance of Surveillance Requirement 4.8.1.1.1.b.
03-0006	3/4.3.1 and 3/4.3.2	This change clarified the discussion of design standards applicable to instrumentation systems.
03-0018	3/4.8	This change provided guidance on the configuration of transformers required to meet the requirements for a qualified offsite circuit.
03-0020	3/4.5.1	This change provided guidance on when Surveillance Requirement 4.5.1.d is to be performed.
04-0002	3/4.8	This change identified the auxiliary transformer backfeed circuit as a qualified offsite source provided certain criteria are met.
04-0006	3/4.3.1 and 3/4.3.2	This change provided guidance on when channel functional testing of the Manual Reactor Trip Function and the Control Rod Drive Trip Breakers is performed. Additionally, the change provided clarification of the applicability of requirements for the source range, neutron flux and rate instrumentation requirements.

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Enclosure 2

Revised Technical Specification Bases Pages
(13 pages follow)

Note: Revised pages incorporate all changes made within the time period for this update. Only the most recent change to a page is identified with revision bars.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The allowable tolerance for performing surveillance activities is sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval. It is not intended that the allowable tolerance be used as a convenience to repeatedly schedule the performance of surveillances at the allowable tolerance limit.

The allowable tolerance for performing surveillance activities also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified to be performed at least once each REFUELING INTERVAL. It is the intent that REFUELING INTERVAL surveillances be performed in an OPERATIONAL MODE consistent with safe plant operation.

4.0.3 This specification establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when either they are found or known to be inoperable although still meeting the Surveillance Requirements, or the requirements of the surveillance(s) are known to be not met between required surveillance performances.

Specification 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with Specification 4.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with required ACTIONS or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Specification 4.0.3 allows for the full delay period up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

Specification 4.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by required ACTIONS.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by Specification 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the ACTIONS for the applicable LCO Conditions must be entered immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the ACTIONS for the applicable LCO Conditions must be entered immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the time limits of the ACTIONS, restores compliance with this specification.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that: 1) inservice inspection of ASME Code Class 1, 2 and 3 components will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a, and 2) inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) and applicable Addenda as required by 10 CFR 50.55a.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and the ASME OM Code and their applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and the ASME OM Code and their applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME OM Code provision which allows pumps to be tested up to one week after return to normal operation.

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

The OPERABILITY of the RPS, SFAS and SFRCS instrumentation systems ensure that 1) the associated action and/or trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for RPS, SFAS and SFRCS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the required design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The response time limits for these instrumentation systems are located in the Updated Safety Analysis Report and are used to demonstrate OPERABILITY in accordance with each system's response time surveillance requirements.

As indicated in RPS Table 4.3-1 for Functional Units 1 and 12, a CHANNEL FUNCTIONAL TEST is required to be performed for the Manual Reactor Trip function and the CRD Trip Breakers function once prior to each reactor startup, i.e., prior to Mode 2 entry, if not performed within the previous 7 days. These surveillance requirements ensure the OPERABILITY of these Functional Units prior to achieving criticality.

If the plant is in MODE 2 or if the plant is in MODE 3, 4, or 5 with the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal, then Functional Unit 11.A of Table 3.3-1 applies. With THERMAL POWER level > 10^{-10} amps on both Intermediate Range channels, high voltage to the Source Range detectors may be de-energized. If the plant is in MODE 3, 4, or 5 with the control rod drive trip breakers not in the closed position or the control rod drive system not capable of withdrawal, Functional Unit 11.B of Table 3.3-1 applies. Applicability of Functional Units 11.A and 11.B is dependent upon the plant MODE and control rod drive system status and is not dependent on if the plant is in the process of a startup or a shutdown.

SFAS Table 3.3-3, ACTION 10, allows entry into this ACTION statement to be delayed for up to 8 hours when a functional unit is placed in an inoperable status solely for performance of a CHANNEL FUNCTIONAL TEST, provided at least two other corresponding functional units remain OPERABLE. The term "corresponding functional units" refers to the functional units (Total No. of Units column in Table 3.3-3) for the same trip setpoint in the other SFAS channels. For example, the corresponding functional units for a Containment Pressure - High unit are the other three Containment Pressure - High units. This 8-hour allowance provides a reasonable time to perform the required surveillance testing without having to enter the ACTION statement and implement the required ACTIONS.

SFRCS Table 3.3-11, ACTION 16, allows entry into this ACTION statement to be delayed for up to 8 hours when a channel is placed in an inoperable status solely for performance of a CHANNEL FUNCTIONAL TEST, provided the remaining actuation channel remains OPERABLE. This 8-hour allowance provides a reasonable time to perform the required surveillance testing without having to enter the ACTION statement and implement the required ACTIONS.

For the RPS, SFAS Table 3.3-4 Functional Unit Instrument Strings b, c, d, e, and f, and Interlock Channel a, and SFRCS Table 3.3-12 Functional Unit 2:

> Only the Allowable Value is specified for each Function. Nominal trip setpoints are specified in the setpoint analysis. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the specific

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

setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip parameter. These uncertainties are defined in the specific setpoint analysis.

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the specified Allowable Values. Any setpoint adjustment shall be consistent with the assumptions of the current specific setpoint analysis.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The frequency is justified by the assumption of an 18 or 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The measurement of response time at the specified frequencies provides assurance that the RPS, SFAS, and SFRCS action function associated with each channel is completed within the time limit assumed in the safety analyses.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

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

Surveillance Requirement 4.3.2.2.3 requires demonstration that each SFRCS function can be performed within the applicable SFRCS RESPONSE TIME. When this surveillance requirement can not be met due to an inoperable SFRCS-actuated component, the LCO ACTION associated with the inoperable actuated component should be entered. When the SFRCS RESPONSE TIME surveillance requirement can not be met due to inoperable components within the SFRCS, ACTION 16 of Table 3.3-11 should be followed.

The actuation logic for Functional Units 4.a., 4.b., and 4.c. of Table 3.3-3, Safety Features Actuation System Instrumentation, is designed to provide protection and actuation of a single train of safety features equipment, essential bus or emergency diesel generator. Collectively, Functional Units 4.a., 4.b., and 4.c. function to detect a degraded voltage condition on either of the two 4160 volt essential buses, shed connected loads, disconnect the affected bus(es) from the offsite power source and start the associated emergency diesel generator. In addition, if an SFAS actuation signal is present under these conditions, the sequencer channels for the two SFAS channels which actuate the train of safety features equipment powered by the affected bus will automatically sequence these loads onto the bus to prevent overloading of the emergency diesel generator. Functional Unit 4.a. has a total of four units, one associated with each SFAS

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

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

An SFRCS channel consists of 1) the sensing device(s), 2) associated logic and output relays, and 3) power sources. The SFRCS output signals that close the Main Feedwater Block Valves (FW-779 and FW-780) and trip the Anticipatory Reactor Trip System (ARTS) are not required to mitigate any accident and are not credited in any safety analysis. Therefore, LCO 3.3.2.2 does not apply to these functions.

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

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and pilot operated relief valve against water relief.

The low level limit is based on providing enough water volume to prevent the low level interlock from de-energizing the pressurizer heaters during steady state operations. The high level limit is based on providing enough steam volume to prevent water relief through the pressurizer relief valves during the most challenging anticipated pressurizer insurge transient, which is a loss of feedwater. Since prevention of water relief is a goal for abnormal transient operation, rather than a Safety Limit, the value for high pressurizer level is nominal and is not adjusted for instrument error.

The ACTION statement provides 1 hour to restore pressurizer level prior to requiring shutdown. The 1-hour completion time is considered to be a reasonable time for restoring pressurizer level to within limits.

The pilot operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the pilot operated relief valve minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken. A process equivalent to the inspection method described in Topical Report BAW-2120P will be used for inservice inspection of steam generator tube sleeves. This inspection will provide ensurance of RCS integrity.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 GPD through any one steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal

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.

The purpose of Surveillance Requirement 4.5.1.d is to ensure that when the LCO is applicable (MODE 1, MODE 2, and MODE 3 with Reactor Coolant System pressure greater than 800 psig), the CFT isolation valves will be automatically opened and interlocked against closing. Since the bistables which provide input to the automatic opening and interlock functions actuate prior to the associated pressure instruments reaching 800 psig, it is intended that this Surveillance Requirement be performed in MODE 3 prior to the associated pressure instruments reaching 800 psig.

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

BASES (Continued)

The surveillance requirement for throttle valve position stops provides 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.

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME OM Code. This type of testing may be accomplished by measuring the pump's developed head at only one point of the pump's characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant accident analysis. Surveillance Requirements are specified in the Inservice Testing Program, which encompasses the ASME OM Code. The ASME OM Code and Technical Specification 4.0.5 provide the activities and frequencies necessary to satisfy the requirements.

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

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

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

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

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

BASES

The OPERABILITY of the A.C. and D.C. power sources and associated distribution Systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General design Criterion 17 of Appendix "A" to 10 CFR 50.

Qualified offsite to onsite circuits are those that are described in the USAR and are part of the licensing basis for the plant.

An OPERABLE qualified offsite to onsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E essential buses.

An OPERABLE qualified offsite to onsite circuit consists of:

1. One OPERABLE 345 kV transmission line
2. One OPERABLE 345 - 13.8 kV startup transformer, or an OPERABLE main transformer and unit auxiliary transformer with the generator links removed ("backfeed" alignment), as described below
3. One OPERABLE 13.8 kV bus, and
4. One OPERABLE 13.8 - 4.16 kV bus tie transformer as described below.

An OPERABLE qualified circuit from the transmission line through the main transformer and unit auxiliary transformer exists when the following conditions are met:

1. The plant is in MODE 3, 4, 5, or 6.
2. Each OPERABLE 13.8 kV bus is powered from the unit auxiliary transformer, and
3. If a startup transformer is OPERABLE, each OPERABLE 13.8 kV bus is configured to permit automatic transfer to an OPERABLE 345 - 13.8 kV startup transformer.

Typically, the electrical power reserve source selector switches are selected to the two different startup transformers. However, under certain conditions it is appropriate to select both switches to the same startup transformer. The circuit in which the startup transformer does not have a reserve source selector switch pre-selected to it must still meet the requirements of having its 345 kV transmission line, startup transformer, 13.8 kV bus and bus tie transformer OPERABLE (unless backfeeding through the unit auxiliary transformer).

In the case where a 13.8 kV bus is powered from a startup transformer, the reserve source selector switch should be selected to the opposite startup transformer (when that transformer is OPERABLE).

In MODES 1, 2, 3, and 4, additional restrictions apply to the configuration of the electrical power distribution system to ensure that adequate voltage is available for each of the required loads. Any time less than two 345 kV – 13.8 kV transformer circuits or less than two 13.8 kV – 4.16 kV transformers are OPERABLE, at least one qualified offsite to onsite circuit is not OPERABLE, and the appropriate ACTION statement must be entered:

Number of OPERABLE 345 kV – 13.8 kV Transformer Circuits	Number of OPERABLE 13.8 kV – 4.16 kV Bus Tie Transformers	Number of OPERABLE Qualified Offsite to Onsite Circuits MODES 1, 2, 3, 4	Number of OPERABLE Qualified Offsite to Onsite Circuits MODES 5, 6
2	1	1	1
2	0	0	0
1	2	1	1
1	1	0	1
1	0	0	0
0	0, 1, 2	0	0

The essential 4.16 kV buses remain OPERABLE while energized with one 13.8 kV - 4.16 kV bus tie transformer inoperable.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one of each of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

Surveillance Requirement 4.8.1.1.1.b is performed at least once each REFUELING INTERVAL during shutdown by (1) demonstrating the capability of transferring (both manually from the control room and automatically) each 13.8 kV bus power supply from the unit auxiliary transformer to each startup transformer circuit, and from each startup transformer circuit to the other startup transformer circuit, and (2) demonstrating the capability of transferring manually from each startup transformer circuit to the circuit through the main transformer and unit auxiliary transformer.

Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.c.4 verify proper starting of the Emergency Diesel Generators from standby conditions. Verification that an Emergency Diesel Generator has achieved a frequency of 60 Hz within the required time constraints meets the requirement for verifying the Emergency Diesel Generator has accelerated to 900 RPM.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the facility status.

BASES

The Surveillance Requirements for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance, Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants", February 1978, and IEEE Std. 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead - Acid Batteries for Stationary Applications," except that certain tests will be performed at least once each REFUELING INTERVAL.

Battery degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance discharge or modified performance discharge test, or is below 90% of the manufacturer's rated capacity.

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cell's float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current of less than two amps is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery. Exceptions to the specific gravity requirements are taken to allow for the normal deviations experienced after a battery discharge and subsequent recharge associated with a service, performance discharge, or modified performance discharge test. The specific gravity deviations are recognized and discussed in IEEE Std. 450-1995.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-1 is permitted for up to seven days. During this seven-day period: (1) the allowable value for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

Docket Number 50-346
License Number NPF-3
Serial Number 3059
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.

Commitment	**Due Date**
None	N/A