December 16, 1993

Docket No. 50-346

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Mr. Louis F. Storz
Vice President, Nuclear - Davis-Besse
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449

Dear Mr. Storz:

SUBJECT: AMENDMENT NO. 182 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NO. M84912)

The Commission has issued Amendment No. 182 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment revises the Technical Specifications in response to your application dated November 9, 1992, as supplemented on November 22, 1993.

This amendment revises the Technical Specifications to allow the de-energization of the borated water storage tank outlet isolation valves in the open position during operational Modes 1, 2, 3, and 4.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by Jon B. Hopkins

Jon B. Hopkins, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 182 to License No. NPF-3
2. Safety Evaluation

CP-1

cc w/enclosures:
See next page

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cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TOLEDO EDISON COMPANY

CENTERIOR SERVICE COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 182
License No. NPF-3

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Toledo Edison Company, Centerior Service Company, and the Cleveland Electric Illuminating Company (the licensees) dated November 9, 1992, as supplemented on November 22, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

(a) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 182, are hereby incorporated in the license. The Toledo Edison Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

Jon B. Hopkins, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: December 16, 1993

cc:

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# ATTACHMENT TO LICENSE AMENDMENT NO. 182

# FACILITY OPERATING LICENSE NO. NPF-3

# DOCKET NO. 50-346

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

| Remove | Insert |
|---|---|
| 3/4 5-3 | 3/4 5-3 |
| 3/4 5-4 | 3/4 5-4 |
| B 3/4 5-2 | B 3/4 5-2 |
| B 3/4 6-3 | B 3/4 6-3 |

## 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.

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

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

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

   2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

   1. Verifying that the interlocks:

      a) Close DH-11 and DH-12 and deenergize the pressurizer heaters, if either DH-11 or DH-12 is open and a simulated reactor coolant system pressure which is greater than the trip setpoint (<438 psig) is applied. The interlock to close DH-11 and/or DH-12 is not required if the valve is closed and 480 V AC power is disconnected from its motor operators.

      b) Prevent the opening of DH-11 and DH-12 when a simulated or actual reactor coolant system pressure which is greater than the trip setpoint (<438 psig) is applied.

   2. a) A visual inspection of the containment emergency sump which verifies that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

      b) Verifying that on a Borated Water Storage Tank (BWST) Low-Low Level interlock trip, with the motor operators for the BWST outlet isolation valves and the containment emergency sump recirculation valves energized, the BWST Outlet Valve HV-DH7A (HV-DH7B) automatically close in ≤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. Verifying a total leak rate ≤ 20 gallons per hour for the LPI system at:

      a) Normal operating pressure or hydrostatic test pressure of ≥150 psig for those parts of the system downstream of the pump suction isolation valve, and

      b) ≥45 psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.

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

### BASES

### 3/4.5.1 CORE FLOODING TANKS

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

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

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

The limits for operation with a core flooding tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

### 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.

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 decay heat removal system leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase of the low pressure injection will not be exceeded.

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.

## 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 BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the BWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analysis.

The bottom 4 inches of the borated water storage tank 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.

leakage rate are consistent with the assumptions used in the safety analyses. The leak rate surveillance requirements assure that the leakage assumed for the system during the recirculation phase will not be exceeded.

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.

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 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 valves 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.

### 3/4.6.2.2 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

### 3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the required time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Containment isolation valves and their required isolation times are addressed in the USAR. The opening of a closed inoperable containment isolation valve on an intermittent basis during plant operation is permitted under administrative control. Operating procedures identify those valves which may be opened under administrative control as well as the safety precautions which must be taken when opening valves under such controls.

## 3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the Hydrogen Analyzers, Containment Hydrogen Dilution System, and Hydrogen Purge System ensures that this equipment will be available to maintain the maximum hydrogen concentration within the containment vessel at or below three volume percent following a LOCA.

The two redundant Hydrogen Analyzers determine the content of hydrogen within the containment vessel. The Hydrogen Analyzers, although they have their OPERABILITY requirements in this Specification, are considered part of the post-accident monitoring instrumentation of Specification 3/4.3.3.6, Post-Accident Monitoring Instrumentation.

The Containment Hydrogen Dilution (CHD) System consists of two full capacity, redundant, rotary, positive displacement type blowers to supply air to the containment. The CHD System controls the hydrogen concentration by the addition of air to the containment vessel, resulting in a pressurization of the containment and suppression of the hydrogen volume fraction.

The Containment Hydrogen Purge System Filter Unit functions as a backup to the CHD System and is designed to release air from the containment atmosphere through a HEPA filter and charcoal filter prior to discharge to the station vent.

## 3/4.6.5 SHIELD BUILDING

### 3/4.6.5.1 EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the emergency ventilation systems ensures that containment vessel leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 182 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

CENTERIOR SERVICE COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

## 1.0 INTRODUCTION

By letter dated November 9, 1992, as supplemented on November 22, 1993, Toledo Edison Company (the licensee) proposed to change the Technical Specifications (TS) for the Davis-Besse Nuclear Power Station, Unit 1. The proposed TS change reflects de-energization of the Borated Water Storage Tank (BWST) outlet isolation valves DH-7A and DH-7B in their open position during operational modes 1, 2, 3, and 4. The licensee will revise the following TS to reflect the proposed change:

(1) TS 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280$ °F;

(2) TS Bases 3/4.5.2 and 3/4.5.3 Emergency Core Cooling Systems - ECCS Subsystems;

(3) TS Bases 3/4.6-2-1, Containment Systems - Depressurization and Cooling Systems - Containment Spray System

(4) Surveillance Requirement (SR) 4.5.2.d.2.b.

The supplemental letter of November 22, 1993, provided additional information and did not change the NRC staff's proposed no significant hazards determination.

## 2.0 EVALUATION

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 during normal plant shutdown.

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 fire, which could result in draining the BWST to the containment emergency sump and the loss of this water source for normal plant shutdown.

The power removal from valves DH-7A/7B during modes 1, 2, 3, and 4 is accomplished by locally opening the breakers for the valve operators at their respective motor control centers (MCCs). This requires manual action outside the control room. Action to close the breakers is necessary during the change from emergency core cooling system (ECCS) injection to recirculation following a postulated loss-of-coolant accident (LOCA) in order to manipulate the valves.

When the valve operators are energized (breakers closed), an interlock is provided between BWST outlet isolation valves DH -7A/7B and containment emergency sump recirculation valves DH - 9A/9B so that all valves cannot be opened at the same time. However, when the breakers are open, this interlock function will be maintained administratively because the control room operators are prevented by emergency procedure from shifting pump suction during a LOCA from the BWST to the containment emergency sump until the BWST low-low level safety features actuation system (SFAS) level 5 signal is received.

The BWST capacity is about 482,778 gallons which permits the cooldown operation up to 37 minutes in the case of a LOCA. It takes an operator only 3 minutes to get all four breakers closed once the BWST low-low level is received. Thus, the emergency core cooling function will be accomplished in the case of a LOCA and it is acceptable.

The interlock will maintain its function once the valve operator's breakers are closed. The operators will open valves DH-9A/9B. The interlock will automatically close valves DH-7A/7B thereby, realigning pump suction from the BWST to the containment emergency sump. The valves will take approximately 75 seconds to reach their new positions.

Current plant emergency procedures for a large break LOCA require the operators to close the breakers for valves DH-9A/9B. Closure of the breakers for valves DH-7A/7B will be added in the emergency procedure.

The NRC staff questioned the risk impact of the additional operator actions required for a LOCA and the operator training of the new procedure. The staff had a telecon on September 2, 1993, with the licensee to resolve these issues.

The licensee stated that LOCAs were not dominant contributors to core-damage for Davis-Besse. However, to assess the impact of this proposed change on the core-damage frequency, the operator actions associated with going to recirculation were revised to account for closing the breakers for DH-7A/7B. All operator actions associated with going to recirculation following a LOCA were modified. The results showed a slight increase in overall core-damage frequency (approximately a half percent). In addition, appropriate operator training will be provided prior to implementation of the license amendment. The licensee subsequently submitted this information in a letter dated November 22, 1993. The NRC staff has reviewed this information and finds it acceptable. Therefore, its concerns are resolved.

In the case of a fire, de-energization of valves DH-7A/7B and DH-9A/9B will ensure that water from the BWST is available for plant shutdown. A manual operator action will maintain the interlock function in the case of a LOCA. In addition, the licensee will revise the emergency procedure to include the closure of the breakers for valves DH-7A/7B in the case of a LOCA. Thus, the LOCA consequences remain acceptable based on the BWST capacity and the operator action to close the breakers.

Based on the above, the NRC staff finds that the proposed TS changes for Davis-Besse are acceptable.

## 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (58 FR 28061). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

Principal Contributor: K. Desai

Date: December 16, 1993