

May 16, 1994

Docket No. 50-346

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

Dear Mr. Shelton:

SUBJECT: CORRECTION TO AMENDMENT NOS. 181, 182, 183 AND 184 TO FACILITY
OPERATING LICENSE NO. NPF-3 (TAC NOS. M86933, M84912, M87339 AND
M86426)

On November 19, 1993, December 16, 1993, December 30, 1993, and December 30,
1993, the Commission issued in respective order, Amendment Nos. 181, 182, 183,
and 184. The amendments revised the Technical Specifications in response to your
applications dated June 23, 1993, November 9, 1992, August 30, 1993, and May 6,
1993, respectively.

Technical Specification pages transmitted with the subject amendments contained
typographical errors as follows: page 5-5 (Amendment No. 181), page B 3/4 5-2
(Amendment No. 182), page B 3/4 6-4 (Amendment No. 183), page 1-6a and page
3/4 4-12 (Amendment No. 184). Those errors have been corrected and the corrected
pages are enclosed.

Please accept our apologies for any inconvenience these administrative errors may
have caused you.

Sincerely,

ORIGINAL SIGNED BY

Garmon West, Jr., Acting Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosure:

TS pages 5-5, B 3/4 5-2, B 3/4 6-4,
1-6a and 3/4 4-12

cc w/enclosure:

See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

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Please accept our apologies for any inconvenience these administrative errors may have caused you.

Sincerely,

A handwritten signature in cursive script that reads "Garmon West, Jr.".

Garmon West, Jr., Acting Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosure:

TS pages 5-5, B 3/4 5-2, B 3/4 6-4,
1-6a and 3/4 4-12

cc w/enclosure:
See next page

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DESIGN FEATURES

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,110 ± 200 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

- a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 1% delta k/k for calculation uncertainty.
- b. A rectangular array of stainless steel cells spaced 12 31/32 inches on centers in one direction and 13 3/16 inches on centers in the other direction. Fuel assemblies stored in the spent fuel pool shall be placed in a stainless steel cell of 0.125 inches nominal thickness or in a failed fuel container.
- c. Fuel assemblies stored in the spent fuel pool in accordance with Technical Specification 3.9.13.

5.6.1.2 The new fuel storage racks are designed and shall be maintained with:

- a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 1% delta k/k for uncertainties as described in Section 9.1 of the USAR.
- b. A K_{eff} equivalent to less than or equal to 0.98 when immersed in a hydrogenous "mist" of such a density that provides optimum moderation (i.e., highest value of K_{eff}), which includes a conservative allowance of 1% delta k/k for uncertainties as described in Section 9.1 of the USAR.
- c. A nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks.
- d. Fuel assemblies having a maximum initial enrichment of 5.0 weight percent uranium-235.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below 9 feet above the top of the fuel storage racks.

DESIGN FEATURES

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 735 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limit of Table 5.7-1.

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 $> 280^{\circ}\text{F}$ 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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EMERGENCY CORE COOLING SYSTEMS

BASES

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

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

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.

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CONTAINMENT SYSTEM

BASES

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.

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CONTAINMENT SYSTEMS

BASES

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 in conjunction with 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.

As a backup to the CHD System and the Containment Hydrogen Purge System, the capability to install an external hydrogen recombination system has been provided.

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.

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DEFINITIONS

1.29 Deleted

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

1.31 Deleted

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.32 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specifications 6.9.1.10 and 6.9.1.11.

1.33 Deleted

1.34 Deleted

1.35 Deleted

1.36 Deleted

MEMBER(S) OF THE PUBLIC

1.37 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

SITE BOUNDARY

1.38 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

DEFINITIONS

UNRESTRICTED AREA

1.39 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes. The definition of UNRESTRICTED AREA used in implementing the Radiological Effluent Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the exclusion (fenced) area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the LIMITING CONDITIONS FOR OPERATION to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

1.40 Deleted

CORE OPERATING LIMITS REPORT

1.41 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7. Plant operation within these core operating limits is addressed in individual specifications.

TABLE 4.4-1

**MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION**

Preservice Inspection	No			Yes		
	Two	Three	Four	Two	Three	Four
No. of Steam Generators per Unit						
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One ¹			One ¹	One ²	One ³

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.
2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.
3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

**TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION**

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair by sleeving defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair by sleeving defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair by sleeving defective tubes
					C-3	Perform action for C-3 result of first sample
	C-3	Perform action for C-3 result of first sample	N/A	N/A		
	C-3	Inspect all tubes in this S.G., plug or repair by sleeving defective tubes and inspect 2S tubes in each other S.G. Report to the NRC prior to resumption of plant operation.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair by sleeving defective tubes. Report to the NRC prior to resumption of plant operation.	N/A	N/A

(1) $S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

DISTRIBUTION:

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PDIII-3 Reading File

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