December 8, 1995

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50-346

Mr. John P. Stetz
Vice President- Nuclear
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449

SUBJECT: AMENDMENT NO. 204 TO FACILITY OPERATING LICENSE NO. NPF-3 - DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 (TAC NO. M93814)

Dear Mr. Stetz:

The Commission has issued the enclosed Amendment No. 204 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1. The amendment revises the Technical Specifications (TSs) in response to your application dated October 2, 1995.

This amendment revises TS Section 5.0, "Design Features," by adding a site location description, removing site area maps, removing containment and reactor coolant system design parameters, removing the description of the meteorological tower location, removing component cyclic or transient limits, and revising the fuel assembly description to include the use of ZIRLO clad fuel rods.

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:

Linda L. Gundrum, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. Amendment No. 204 to License No. NPF-3
2. Safety Evaluationn

cc w/encls: See next page

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

December 8, 1995

Mr. John P. Stetz
Vice President - Nuclear
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
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cc w/encls: See next page

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Davis-Besse Nuclear Power Station
Unit No. 1

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Port Clinton, Ohio 43452

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. 204
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 October 2, 1995, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 204, 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

Linda L. Gundrum

Linda L. Gundrum, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: December 8, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 204

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.

| Remove | Insert |
|---|---|
| XIV | XIV |
| 5-1 | 5-1 |
| 5-2 | 5-2 |
| 5-3 | -- |
| 5-4 | -- |
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| 5-6 | -- |
| 5-7 | -- |
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| 5-9 | -- |
| 5-10 | -- |

# INDEX

## DESIGN FEATURES

# 5.0 DESIGN FEATURES

## 5.1 Site Location

The Davis-Besse Nuclear Power Station, Unit Number 1, site is located on Lake Erie in Ottawa County, Ohio, approximately six miles northeast from Oak Harbor, Ohio and 21 miles east from Toledo, Ohio. The exclusion area boundary has a minimum radius of 2400 feet from the center of the plant.

## 5.2 (Deleted)

## 5.3 Reactor Core

### 5.3.1 Fuel Assemblies

The reactor core shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide ($UO_2$) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

### 5.3.2 Control Rods

The reactor core shall contain 53 safety and regulating control rod assemblies and 8 axial power shaping rod (APSR) assemblies. The nominal values of absorber material for the safety and regulating control rods shall be 80 percent silver, 15 percent indium and 5 percent cadmium. The absorber material for the APSRs shall be 100 percent Inconel.

## 5.4 (Deleted)

## 5.5 (Deleted)

## 5.6 Fuel Storage

### 5.6.1 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.

(continued)

5.6 Fuel Storage (continued)

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.

5.6.2 Drainage

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.

5.6.3 Capacity

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 (Deleted)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 204 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 October 2, 1995, the Toledo Edison Company, Centerior Service Company, and the Cleveland Electric Illuminating Company (the licensees), submitted a request for changes to the Davis-Besse Nuclear Power Station (DBNPS) Technical Specifications (TSs). The requested amendment would revise TS Section 5.0, "Design Features," by adding a site location description, remove site area maps, remove containment and reactor coolant system design parameters, remove the description of the meteorological tower location, remove component cyclic or transient limits, and revise the fuel assembly description to include the use of ZIRLO clad fuel rods.

## 2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. That regulation requires that the TSs include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

The Commission has provided guidance for the contents of TSs in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TSs to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TSs, as follows: (1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.[1]

As a result, existing TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents.

The policy statement also encouraged licensees to adopt the applicable improved Standard Technical Specifications (STSs) and provided some guidance for the conversion from the present plant specific TSs to the improved Standard TSs. However, specific guidance for converting the design features section of the TSs was not provided in the policy statement. DBNPS has a Babcock & Wilcox (B&W) designed nuclear steam supply system. The improved STS for B&W plants was published as NUREG-1430 in September 1992.

3.0 EVALUATION

The proposed changes described in the evaluation are considered by the licensee to be consistent with NUREG-1430, Revision 1, "Standard Technical Specifications for Babcock and Wilcox Plants," dated April, 1995, and the changes for the lead plant on this change, Calvert Cliffs Nuclear Power Plant, Units 1 and 2, as described in Amendments 204 and 182, issued March 14, 1995. Each subsection of 5.0, Design Feature, is described, the proposed changes described, the location and description provided in the Updated Safety

---

[1] The Commission recently adopted amendments to 10 CFR 50.36, pursuant to which the rule was revised to codify and incorporate these criteria. See Final Rule, "Technical Specifications," 60 FR 36953 (July 19, 1995). The Commission indicated that reactor core isolation cooling or isolation condenser, residual heat removal, standby liquid control, and recirculation pump trip systems are to be included in the TS under Criterion 4, although it recognized that other structures, systems and components could also meet this criterion (60 FR 36956).

Analysis Report (USAR), and the licensees' evaluation of the change. The changes are evaluated against the criteria contained in Section 182a of the Act which requires applicants for nuclear power plant operating licenses to state TSs to be included as part of the license. Section 182a of the Act states, in part:

> In connection with applications for licenses to operate production or utilization facilities, the applicant shall state such technical specifications, including information of the amount, kind, and source of special nuclear materials required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that utilization or production of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

The licensees' evaluation includes the Commission's requirements related to the content of TSs as set forth in 10 CFR 50.36. This regulation requires that the TSs include items in five specific categories, including: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. Specifically, 10 CFR 50.36(c)(4) states, in part, "Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of this section."

Additionally, 10 CFR 50.59, "Changes, tests and experiments," provides the criteria for determining if a proposed change to the features of a facility, as described in the safety analysis report, is an unreviewed safety question. The criteria in 10 CFR 50.59 for making this determination are: (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may increase; (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (3) if the margin of safety as defined in the basis for any TSs is reduced.

With the above citations as references, the following criteria is used to determine what information should be placed in the design features section of the TSs:

1. The amount, kind, and source of special nuclear material required;

2. The place of the use of the special nuclear materials; and

3. Those features of the facility, such as materials of construction and geometric arrangements, which if altered or modified would have a significant effect on safety and are not covered in the safety limits, limiting conditions for operation (LCO), or surveillance requirements of the TSs.

Based on the above criteria, the proposed changes to the DBNPS TS Section 5.0, Design Features, will remove some subsections and modify others to generally conform to the examples provided in the Design Features section of NUREG-1430 and the changes approved for the Calvert Cliffs Nuclear Power Plant.

3.1 TS 5.1, SITE

TS 5.1, Site, presently describes the Exclusion Area (Figure 5.1-1), the Low Population Zone (Figure 5.1-2), and the unrestricted area and site boundaries for radioactive liquid effluents (Figure 5.1-3) and for radioactive gaseous effluents (Figure 5.1-4). The licensees propose that these four figures be deleted, and that a site location description be added to TS 5.1 as follows:

> The Davis-Besse Nuclear Power Station, Unit Number 1, site is located on Lake Erie in Ottawa County, Ohio, approximately six miles northeast from Oak Harbor, Ohio and 21 miles east from Toledo, Ohio. The exclusion area boundary has a minimum radius of 2400 feet from the center of the plant.

The DBNPS site location is described in USAR Sections 1.2.1.1, Site, and 2.1.1, Site Location. The DBNPS occupies a site on the southwestern shore of Lake Erie and is located in Ottawa County in northwestern Ohio. As shown on USAR Figure 2.1-1, DBNPS Site Location & Geographic Features - 20 Mile Radius, the DBNPS site is located approximately six miles northeast from Oak Harbor, Ohio, and 21 miles east from Toledo, Ohio. The site boundary is depicted on USAR Figure 2.1-3, DBNPS Site Arrangement. As stated in USAR Section 2.1.2, Site Description, the site boundary shown on this figure is also the limit of the exclusion area. As stated in USAR Section 2.1.2.2, Boundaries for Establishing Effluent Release Limits, the site boundary is the boundary line which determines the limits of gaseous releases for purposes of the Offsite Dose Calculation Manual limits to fulfill the requirements of 10 CFR Part 20 pertaining to radioactive releases to unrestricted areas. As further stated in USAR Section 2.1.2.2, the distance from the vent stack for gaseous releases, located on the side of the shield building, to the nearest site boundary is 2,400 feet. The shield building is at approximately the center of the plant structures. USAR Section 2.1.2.2 also describes the discharge points for effluents from the station area for liquid discharges. USAR Section 2.1.3.3, Low Population Zone, describes the low population zone as the area outside of the site boundary within a radius of two miles from the center of the containment structure. This area is depicted on USAR Figure 2.1-7, DBNPS Low Population Zone.

Section 182a of the Act requires that the place of the use of the special nuclear material be specified. Presently, TS 5.1.1 meets this requirement by reference to TS Figure 5.1-1, which is a map of the site. In the proposed change to TS 5.1, the place of use would be specified by a description rather than a map. This is consistent with NUREG-1430, which proposes a text description of the site location. In addition, the proposed change would delete TS Figures 5.1-1, 5.1-2, 5.1-3, and 5.1-4, however, as described previously, this information is presently included in the USAR. In accordance with 10 CFR Part 100, the site description includes a minimum distance to the Exclusion Area Boundary to ensure that the area, for which the licensee has the authority to determine all activities including the exclusion or removal of personnel and property from the area, is clearly associated with the "place of use" referred to in Section 182a of the Act. The inclusion of the maps in the USAR ensures that any change to either the boundaries or the zone will have to be evaluated using the 10 CFR 50.59 process. Sufficient regulatory controls exist under 10 CFR 50.59 to assure continued protection of the public health and safety. The changes to TS 5.1 are, therefore, acceptable.

3.2 TS 5.2 CONTAINMENT

TS 5.2, Containment, presently describes the containment vessel and shield building. It is proposed that TS 5.2 be deleted. The containment system is described in USAR Sections 1.2.10, Containment Systems, 3.8.2.1, Containment Vessel, 3.8.2.2, Shield Building and 6.2.1, Containment Vessel Functional Design. The containment is composed of a steel containment vessel and a reinforced concrete shield building. The containment vessel is a low leakage cylindrical steel pressure vessel with a hemispherical dome and ellipsoidal bottom. It is designed to withstand a postulated loss-of-coolant accident (LOCA) and to confine a postulated release of radioactive material. As described in USAR Section 1.2.10.1, Principal Design Criteria, the containment vessel design maximum internal pressure is 40 psig at a coincident temperature of 264 °F. The shield building is a reinforced concrete structure having a cylindrical shape with a shallow dome roof. It completely surrounds the containment vessel and is designed to provide biological shielding during normal operation and from hypothetical accident conditions. An annular space is provided between the shield building and the containment vessel. The shield building provides a means for collection and filtration of fission product leakage from the containment vessel following a hypothetical accident. In addition, the building provides environmental protection for the containment vessel from adverse atmospheric conditions and external missiles.

The change proposes that TS 5.2 be removed consistent with NUREG-1430. Although certain modifications or alterations to containment could have a significant impact on plant safety, adequate control of the containment systems limiting conditions for operations are included in TS Section 3/4.6. Therefore, this information need not be specified in the Design Features section. The requirements of TS 5.2 are not required to be in the TS under 10 CFR 50.36 or § 182a of the Act and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement discussed above. The containment vessel and shield building are described in USAR Chapters 1, 3, and 6 and changes to this information would require evaluation using the 10 CFR 50.59 process. Sufficient regulatory controls exist under 10 CFR 50.59 to assure continued protection of the public health and safety.

3.3 TS 5.3 REACTOR CORE

TS 5.3, Reactor Core, presently describes fuel assemblies and control rods. The proposed change would revise the fuel assembly description in TS 5.3.1 to allow the use of ZIRLO clad fuel rods in addition to zircaloy clad fuel rods.

The use of ZIRLO as fuel rod cladding has been previously accepted by the Commission in 10 CFR 50.46 and its inclusion in TS 5.3 would facilitate potential future use by the DBNPS. The proposed change would also remove from TS 5.3.2 the description of the control rods and the discussions of the length of the control rod absorber material, the cladding material for the control rods and axial power shaping rod (APSR) assemblies, and the length of the APSR absorber material.

The reactor core is described in USAR Section 4.2.1, Fuel and USAR Appendix 4.B, Reload Report. The function of the reactor core is to generate power for an analyzed period. The complete core consists of one-hundred and seventy-seven (177) fuel assemblies arranged in a square lattice to approximate a cylinder. A fuel assembly is normally composed of two hundred and eight (208) fuel rods, sixteen (16) control rod guide tubes, one (1) instrument tube assembly, eight (8) spacer grids, and two (2) end fittings. The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15 array.

The fuel rods consist of zircaloy cladding containing fuel pellets in a columnar arrangement. Zircaloy is a zirconium-based alloy. Each fuel pellet consists of a high density, low enrichment, uranium dioxide material. The safety function of the fuel rod cladding is to provide a primary barrier to prevent the release of fission products. The control rod assemblies, including extended life control rod assemblies, are described in USAR Section 4.2.3.1.1.1, Control Rod Assembly (CRA) and Extended Life Control Rod Assembly (ELCRA). There are fifty-three (53) CRAs in the reactor core. The safety function of the CRAs is to shut down the reactor. In addition, the control rods control fast reactivity changes, and, in conjunction with the APSRs, burnable poison rods, and soluble boron, control lifetime reactively and power distribution changes.

The APSRs are described in USAR Section 4.2.3.1.1.2, APSRA Black and Gray Type. There are 8 APSRs in the reactor core. The function of the APSRs is to maintain an acceptable axial distribution of power.

Since the Act requires that the amount, type, and source of special nuclear material be specified in the TSs, TS 5.3 will be retained, however, several changes are proposed.

As previously described, a change is proposed to revise the fuel assembly description to allow the use of ZIRLO clad fuel rods, consistent with NUREG-1430 and 10 CFR 50.46. ZIRLO is a zirconium-based alloy. As described in USAR Section 4.2.1 and USAR Appendix 4.B, Reload Report, zircaloy is presently used for fuel rod cladding, hence a change to the USAR (and a 10 CFR 50.59 evaluation) would be required to use ZIRLO cladding. In addition, the present TS 5.3.1 requirement that fuel design be "analyzed to comply with all fuel safety design bases" will ensure that any future use of ZIRLO cladding will not have an adverse effect on safety. Sufficient regulatory controls exist under 10 CFR 50.59 to assure continued protection of the public health and safety.

Also as previously described, a change is proposed to remove much of the detail in TS 5.3.2 regarding the description of control rods and APSRs, including the length of absorber material and cladding material. The information being removed is presently contained in USAR Sections 4.2.3.1.1.1 (Table 4.1-6) and 4.2.3.1.1.2 (Table 4.2-7). Based on the above, the proposed changes to TS 5.3 are acceptable.

3.4 TS 5.4 REACTOR COOLANT SYSTEM

TS 5.4, Reactor Coolant System (RCS), presently describes the RCS including operating pressures and temperatures, and total water and steam volume. It is proposed that TS 5.4 be deleted.

The RCS is described in detail in USAR Section 5.0. The RCS consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping. The system, located entirely within the Containment Vessel (CV), is arranged in two heat transport loops, each with two reactor coolant (RC) pumps and one steam generator. RC is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes transferring heat to the steam and water on the shell side of the steam generator. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector and as a solvent for the soluble poison (boron in the form of boric acid) utilized in chemical shim reactivity control. Design pressure and temperatures for RC pressure boundary (RCPB) components are provided in USAR Table 5.1-1a, Design Conditions for RCPB, and USAR Table 5.1-1b, Components Within RCPB. Volumes of the various RCS components are shown on USAR Figure 5.1-1, DBNPS RCS Flow Diagram at Full Power Steady State Conditions.

As described in USAR Section 1.2.2.3, Safety Considerations, the RCS is designed to maintain its integrity under all operating conditions, thereby minimizing the release to the containment vessel of fission products that escape the primary barrier (the fuel rod cladding).

The proposed change will remove TS 5.4, consistent with NUREG-1430. Other sections of the TSs (such as TS Section 3/4.4, Reactor Coolant System) adequately control the RCS parameters such as temperature, pressure, and boundary degradation, which could have a significant impact on safety. The requirements of TS 5.4 are not required to be in the TS under 10 CFR 50.36 or § 182a of the Act and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement discussed above. Further, the information being removed from TS 5.4 is presently contained in USAR Chapter 5 and will be adequately controlled using the 10 CFR 50.59 process. Therefore, the proposed change is acceptable.

3.5 TS 5.5 METEOROLOGICAL TOWER LOCATION

TS 5.5, Meteorological Tower Location, presently contains a description for the location of the meteorological tower (Figure 5.1-1). It is proposed that TS 5.5 be deleted.

The location of the meteorological towers is described in USAR Section 2.3.3, On-Site Meteorological Measurements Program, and depicted on USAR Figure 1.2-12, Site Plan, and USAR Figure 2.1-3, DBNPS Site Arrangement. As described in USAR Section 2.3.3, the two towers, a 100 meter tower and a nearly 10 meter tower, are located within a fenced compound in the southwest corner of the site. The on-site Meteorological Monitoring System (MMS) complies with the requirements of Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants," proposed Revision 1, dated September 1980. As described in Section 7.5.5 of the DBNPS Emergency Plan, meteorological instrumentation readouts are located in the DBNPS Control Room. The Emergency Control Center and the Technical Support Center can also obtain this instrumentation data through the Data Acquisition and Display System. The data can be used to estimate potential radiation doses to the public resulting from actual routine or accidental releases of radioactive materials to the atmosphere, or to evaluate the potential dose to the public as a result of hypothetical reactor accidents.

The proposed change will eliminate TS 5.5, consistent with NUREG-1430. The requirements of TS 5.5 are not required to be in the TS under 10 CFR 50.36 or § 182a of the Act and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement discussed above. The meteorological tower location is presently contained in USAR Section 2.3.3 and will be adequately controlled using the 10 CFR 50.59 process. This change is, therefore, acceptable.

3.6 TS 5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

TS 5.7, Component Cyclic or Transient Limit, presently contains component cyclic or transient limits (Table 5.7-1). It is proposed that TS 5.7 and Table 5.7-1 be deleted.

The component cyclic or transient limits for the RCS components are described in USAR Sections 5.1.3, Reaction Loads, and 5.1.4, Service. Each component of the RCS is designed to withstand the effects of cyclic loads due to temperature and pressure changes in the RCS. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shut down operation. The design service life is 40 years. Design cycles are shown in USAR Table 5.1-8, Transient Cycles - 40 Year Design Life. The licensees propose to review and update the table as necessary under the provisions of 10 CFR 50.59.

The proposed change will eliminate TS 5.7 consistent with NUREG-1430. The requirements of TS 5.7 are not required to be in the TS under 10 CFR 50.36 or § 182a of the Act and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement discussed above. Further, the information contained in TS 5.7 is either already in the USAR or will be added to the next update of the USAR, and will be controlled by the requirements of 10 CFR 50.59 and 10 CFR 50.55a. This change is, therefore, acceptable.

3.7 MISCELLANEOUS CHANGES

The licensees propose various editorial changes to make the format of TS 5.0 generally consistent with the format of NUREG-1430. These changes are administrative and will have no adverse effect on safety. These changes are, therefore, acceptable.

2.8 EVALUATION OF PROPOSED CHANGES

The staff has evaluated each of the proposed changes and has determined that the rationale for eliminating or revising each section is consistent with Commission guidance and regulations.

4.0 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.

5.0 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 effluent 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 (60 FR 56371). 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.

6.0 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.

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