

Docket Number 50-346
License Number NPF-3
Serial Number 2337
Enclosure
Page 1


APPLICATION FOR AMENDMENT
TO
FACILITY OPERATING LICENSE NUMBER NPF-3
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NUMBER 1

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

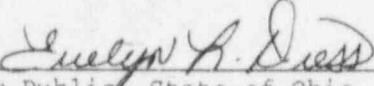
The proposed changes (submitted under cover letter Serial Number 2337) concern:

Appendix A, Technical Specifications (TS):

Page VI	Technical Specifications Index
3/4.6.1.1	Containment Systems - Primary Containment - Containment Integrity
3/4.6.1.2	Containment Systems - Containment Leakage, and associated Bases 3/4.6.1.2, Containment Leakage
3/4.6.1.6	Containment Systems - Containment Vessel Structural Integrity, and associated Bases 3/4.6.1.6, Containment Vessel Structural Integrity
3/4.6.5.3	Containment Systems - Shield Building Structural Integrity, and associated Bases 3/4.6.5.3, Shield Building Structural Integrity

By: 
J. P. Stetz, Vice President - Nuclear

Sworn to and subscribed before me this 12th day of December, 1995.


Notary Public, State of Ohio

Docket Number 50-346
License Number NPF-3
Serial Number 2337
Enclosure
Page 2

The following information is provided to support issuance of the requested changes to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specification (TS) 3/4.6.1.1, Containment Systems - Primary Containment - Containment Integrity, TS 3/4.6.1.2, Containment Systems - Containment Leakage, TS 3/4.6.1.6, Containment Systems - Containment Vessel Structural Integrity, TS 3/4.6.5.3, Containment Systems - Shield Building Structural Integrity, and associated TS Bases.

- A. Time Required to Implement: This change is to be implemented within 90 days after NRC issuance of the License Amendment.
- B. Reason for Change (License Amendment Request Number 95-0017):

The proposed changes would partially adopt the new 10 CFR 50 Appendix J "Option B" approach. The Option B approach would be limited at this time to Type A testing, which allows for completion of a Type A test (Integrated Leak Rate Test) once every 10 years, instead of three such tests required over a 10-year period.

Absent an exemption to the test frequency requirements of Option A of Appendix J to 10 CFR 50, NRC approval of this license amendment request is necessary in order to avoid the need to perform an unnecessary Type A test during the upcoming Tenth Refueling Outage (10RFO). The proposed changes constitute a Cost Beneficial Licensing Action which will provide a cost savings well in excess of \$100,000 over the DBNPS's remaining life.

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

Docket Number 50-346
License Number NPF-3
Serial Number 2337
Attachment

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 95-0017

(21 pages follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 95-0017

TITLE:

Revision of Technical Specification (TS) 3/4.6.1.1, Containment Systems - Primary Containment - Containment Integrity, TS 3/4.6.1.2, Containment Systems - Containment Leakage, TS 3/4.6.1.6, Containment Systems - Containment Vessel Structural Integrity, TS 3/4.6.5.3, Containment Systems - Shield Building Structural Integrity, and associated TS Bases.

DESCRIPTION:

The purpose of the proposed changes is to modify the Davis-Besse Nuclear Power Station (DBNPS) Operating License NPF-3, Appendix A Technical Specifications (TS) and associated Bases. These changes are being proposed in order to partially adopt the new 10 CFR 50 Appendix J "Option B" approach, which provides for completion of a Type A (Integrated Leak Rate Test) once every 10 years, instead of three such tests required over a 10 year period. The Option B approach adopted would be limited at this time to Type A testing. These changes are requested to be approved for use during the Tenth Refueling Outage, which is currently scheduled to commence on April 8, 1996. The proposed changes are described in detail below. Each of the proposed changes is also shown on the attachment.

TS 3/4.6.1.1, Containment Systems - Primary Containment - Containment Integrity:

In order to implement Option B for Type A testing, it is proposed to add a new Surveillance Requirement (SR) 4.6.1.1.c which would read as follows:

- c. By performing required visual examinations of the containment vessel and shield building in accordance with 10 CFR 50, Appendix J Option B, as modified by approved exemptions, and Regulatory Guide 1.163.

TS 3/4.6.1.2, Containment Systems - Containment Leakage:

It is proposed to revise Action statement 3.6.1.2.a to clarify that the acceptance criteria for measured overall integrated containment leakage rate is an "as-left" value.

In order to implement Option B for Type A testing, it is proposed to modify Surveillance Requirement (SR) 4.6.1.2 and SR 4.6.1.2.a to read:

4.6.1.2 The containment leakage rates shall be demonstrated as follows:

- a. Perform Type A tests in accordance with 10 CFR 50, Appendix J Option B, as modified by approved exemptions, and Regulatory Guide 1.163.

Current SRs 4.6.1.2.b and 4.6.1.2.c contain detailed requirements. The implementation of Option B for Type A testing, as specified in the proposed revised SR 4.6.1.2.a, obviates the need to include specific details in the TS. Therefore, it is proposed to delete these SRs.

It is proposed to modify SR 4.6.1.2.d to read as the following, in order to clearly state that Appendix J Option A will be followed for Type B and C tests:

- d. Perform Type B and C tests in accordance with 10 CFR 50, Appendix J Option A, as modified by approved exemptions.

It is also proposed to delete SR 4.6.1.2.i, which presently provides requirements regarding the calculation of test leakage rates, which are already requirements of Appendix J.

Bases 3/4.6.1.2, Containment Leakage:

It is proposed to revise the first paragraph to clarify that the acceptance criteria for measured overall integrated containment leakage rate is an "as-left" value. This change is consistent with the proposed change to Action statement 3.6.1.2.a.

It is proposed to delete the second paragraph, which discusses an exemption regarding the Type A test schedule. With the proposed adoption of 10 CFR 50 Appendix J Option B for Type A testing, this exemption is no longer applicable.

TS 3/4.6.1.6, Containment Systems - Containment Vessel Structural Integrity, and associated Bases 3/4.6.1.6, Containment Vessel Structural Integrity:

It is proposed to delete this TS and its associated Bases. The applicable requirements are contained in the proposed new SR 4.6.1.1.c, as discussed above.

TS 3/4.6.5.3, Containment Systems - Shield Building Structural Integrity, and associated Bases 3/4.6.5.3, Shield Building Structural Integrity:

It is proposed to delete this TS and its associated Bases. The applicable requirements are contained in the proposed new SR 4.6.1.1.c, as discussed above.

TS Index Page VI:

Consistent with the above changes, it is proposed to revise the TS Index to note the deletion of TS 3/4.6.1.6 and TS 3/4.6.5.3.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The activities affected involve Technical Specification surveillance testing, including 10 CFR 50 Appendix J Type A testing to measure the containment overall integrated leakage rate.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

The Containment Systems Technical Specifications Limiting Conditions for Operation (LCO) are based on requirements related to the control of offsite radiation doses resulting from major accidents. Compliance with these LCOs, as demonstrated by performance of the associated surveillance testing, ensures a containment configuration that will limit leakage to those paths and associated leakage rates assumed in the safety analysis.

The containment system is described in the Davis-Besse Updated Safety Analysis Report (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. 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 of 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.

EFFECTS ON SAFETY:

On October 26, 1995, a final rule implementing a performance-oriented and risk-based approach to containment leakage testing became effective. This approach is identified as "Option B" of Appendix J to 10 CFR 50. The rule redesignates the previous prescriptive requirements as "Option A" and allows licensees the option of following either set of requirements. The rule requires licensees to submit a license amendment request and to obtain NRC approval of the request prior to actual adoption of Option B or part thereof.

As stated in the Federal Register publication of the final rule, 60 FR 49495 dated September 26, 1995, the final rule improves the focus of the regulations by eliminating prescriptive requirements that are marginal to safety. Further, the final rule allows test intervals to be based on system and component performance and provides licensees greater flexibility for cost-effective implementation methods of regulatory safety objectives. The final rule publication also discusses the following specific findings documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," September, 1995, which justify the proposed change in frequency of Type A Integrated Leak Rate Testing (ILRT):

1. The fraction of leakages detected only by ILRT's is small, on the order of a few percent.
2. Reducing the frequency of ILRT testing from 3 every 10 years to one every 10 years leads to a marginal increase in risk.
3. At a frequency of one test every 10 years, industry-wide occupational exposure would be reduced by 0.087 person-sievert (8.7 person-rem) per year.

The overall containment leakage rate at the DBNPS has consistently remained well below the surveillance test acceptance criteria. During the five previous Type A tests conducted to date, the leakage has never exceeded 52.6% of L_a , the maximum allowable leakage rate at the peak containment internal pressure. In the last two Type A tests, conducted in October, 1988 and October, 1991, the leakage averaged only 12.5% of L_a . Based on this performance, under adoption of 10 CFR 50 Appendix J Option B, a 10-year Type A test interval is justified, with the next test due in October, 2001. Based on the above-mentioned justification presented in the final rule publication, this change in Type A frequency will not have a significant adverse effect on plant safety. Moreover, the extension in Type A test frequency will result in a significant reduction in personnel radiation exposure.

Section V.B of Option B of 10 CFR 50 Appendix J requires licensees who wish to voluntarily adopt Option B, or parts thereof, to submit to the NRC an implementation plan and a request for a revision to Technical Specifications, including a general reference in the plant Technical Specifications to the regulatory guide or other implementation document used by the licensee to develop a performance-based leakage-testing program. Section I of Option B of 10 CFR 50 Appendix J identifies Regulatory Guide 1.163 as a source of specific guidance concerning a performance-based leakage-testing program. Accordingly, the proposed addition of SR 4.6.1.1.c and the proposed changes to SR 4.6.1.2.a would require Type A testing, including visual examinations and leakage testing, to be performed in accordance with 10 CFR 50 Appendix J Option B, as modified by approved exemptions, and in accordance with Regulatory Guide 1.163. As discussed above, the conversion to Option B for Type A testing will have no significant adverse effect on plant safety.

Regulatory Guide 1.163 states that NEI 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," provides methods acceptable to the NRC Staff for complying with the provisions of Option B in Appendix J to 10 CFR 50, subject to certain listed provisions. The proposed change to Action statement 3.6.1.2.a to clarify that the acceptance criteria for measured overall integrated containment leakage rate is an "as-left" value is consistent with Section 9.2.5, "Surveillance Acceptance Criteria," of NEI 94-01, which states:

The As-found Type A test leakage rate must be less than the acceptance criterion of 1.0 L^a given in the plant Technical Specifications. Prior to entering a mode where containment integrity is required, the As-left Type A leakage rate shall not exceed 0.75 L^a. The As-left and As-found values are as determined by the appropriate testing methodology specifically described in ANSI/ANS 56.8-1994.

This is an administrative change and has no adverse effect on plant safety.

The proposed deletion of SRs 4.6.1.2.b and 4.6.1.2.c remove unnecessary details from the TS. Detailed requirements are contained in or referenced in Regulatory Guide 1.163, which is made a requirement by the new SR 4.6.1.2.a. Regulatory Guide 1.163 references NEI 94-01 for methods acceptable to the NRC Staff. NEI 94-01 Section 9.2.6, "Corrective Action," provides direction on corrective actions to be taken in the event of unacceptable performance for Type A testing. Section 9.2.6 states:

Once the cause determination and corrective actions have been completed, acceptable performance should be reestablished by performing a Type A test within 48 months following the unsuccessful Type A test. Following a successful Type A test, the surveillance frequency may be returned to once per 10 years.

NEI 94-01 Section 12.1, "Report Requirements," states that a post-outage report describing the containment leakage test results will be prepared and will be available for on-site NRC review. NEI 94-01 refers to ANSI/ANS 56.8-1994, "American National Standard for Containment System Leakage Requirements," for technical methods and techniques. Section 5.9 of ANSI/ANS 56.8-1994 provides details on the verification test requirements, similar to the supplemental test requirements presently described in SR 4.6.1.2.c. Since appropriate requirements will be maintained, deletion of these SRs 4.6.1.2.b and 4.6.1.2.c will have no adverse effect on plant safety.

The proposed change to SR 4.6.1.2.d clarifies that Type B and Type C testing will continue to be performed in accordance with 10 CFR 50 Appendix J Option A, as modified by approved exemptions. Detailed requirements regarding test pressure and test frequency are removed since these requirements are specified in 10 CFR 50 Appendix J Option A. These changes are administrative and will have no adverse effect on plant safety.

The proposed deletion of SR 4.6.1.2.i, which presently provides requirements regarding the calculation of test leakage rates, removes unnecessary details from the TS. Detailed test methodology requirements applicable to 10 CFR 50 Appendix J Option B are provided in ANSI/ANS 56.8-1994, which is referenced by NEI 94-01, which is in turn referenced by Regulatory Guide 1.163. Since the proposed TS SR 4.6.1.2.a contains the requirement to perform Type A surveillance testing in accordance with 10 CFR 50 Appendix J Option B and Regulatory Guide 1.163, there is no need to include detailed requirements in the TS. This is an administrative change and has no adverse effect on plant safety.

The proposed deletion of TS 3/4.6.1.6 and TS 3/4.6.5.3, which presently contain detailed requirements for visual examinations of the containment vessel and shield building, are acceptable because the addition of new SR 4.6.1.1.c would require visual examinations of the containment vessel and shield building to be performed in accordance with 10 CFR 50 Appendix J Option B, as modified by approved exemptions, and in accordance with Regulatory Guide 1.163. Regulatory Guide 1.163 modifies the NEI 94-01 guidance for visual examination frequency by requiring that visual examinations be performed prior to performing a Type A test and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years. These inspections are on a schedule similar to that which currently exists in TS 3/4.6.1.6 and TS 3/4.6.5.3. Since detailed requirements for visual inspections are contained in 10 CFR 50 Appendix J Option B, and in NEI 94-01 as modified by Regulatory Guide 1.163, these changes will have no adverse effect on plant safety. Degradation of the containment vessel or shield building identified during these inspections would be evaluated under the DBNPS Potential Condition Adverse to Quality Program for reportability to the NRC in accordance with 10 CFR 50.72 and 50.73.

The proposed changes to the TS Bases and to the TS Index are administrative changes associated with the other proposed changes and have no adverse effect on plant safety.

SIGNIFICANT HAZARDS CONSIDERATION:

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

- 1a. Not involve a significant increase in the probability of an accident previously evaluated because accident initiators, conditions, or assumptions are not affected by the proposed changes.

The proposed changes to the Technical Specifications implement 10 CFR 50 Appendix J Option B for Type A testing, including visual examinations of the containment vessel and shield building, and make various administrative changes to the Technical Specifications and associated Technical Specification Bases. Therefore, as stated above, these proposed changes do not affect accident initiators, conditions, or assumptions.

- 1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not change the source term, containment isolation, or allowable releases.

The proposed changes involve containment leakage testing and test frequency. The allowable containment leakage rates presently specified in the Technical Specifications remain unchanged.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes.
3. Not involve a significant reduction in a margin of safety, for the reasons cited below.

The proposed changes involve containment leakage testing and test frequency. The allowable containment leakage rates presently specified in the Technical Specifications remain unchanged. The Technical Specifications, under the proposed changes, will continue to ensure containment system reliability by periodic testing performed in full compliance with 10 CFR 50 Appendix J.

As stated in the Federal Register publication of the final rule, 60 FR 49495 dated September 26, 1995, the final rule improves the focus of the regulations by eliminating prescriptive requirements that are marginal to safety. Further, the final rule allows test intervals to be based on system and component performance and provides licensees greater flexibility for cost-effective implementation methods of regulatory safety objectives. The final rule publication also discusses the following specific findings documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," September, 1995, which justify the proposed change in frequency of Type A Integrated Leak Rate Testing (ILRT):

1. The fraction of leakages detected only by ILRT's is small, on the order of a few percent.
2. Reducing the frequency of ILRT testing from 3 every 10 years to one every 10 years leads to a marginal increase in risk.

3. At a frequency of one test every 10 years, industry-wide occupational exposure would be reduced by 0.087 person-sievert (8.7 person-rem) per year.

Based on these considerations, it is concluded that the proposed changes do not involve a significant reduction in a margin of safety.

CONCLUSION:

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

ATTACHMENT:

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

REFERENCES:

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment 200.
2. DBNPS Updated Safety Analysis Report through Revision 19.
3. Final Rule, 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," 60 FR 49495, September 26, 1995.
4. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September, 1995.
5. NEI 94-01 Revision 0, "Nuclear Energy Institute Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J," July 26, 1995.
6. NUREG-1493, "Performance-Based Containment Leak-Test Program," September, 1995.
7. ANSI/ANS-56.8-1994, "American National Standard for Containment System Leakage Testing Requirements," August 4, 1994.