Docket Number 50-346
License Number NPF-3
Serial Number 2572
Enclosure 1
Page 1

APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached are the requested changes 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 2572) concern:

Appendix A, Technical Specifications (TS):

Page XVI	Technical Specifications Index
1.8	Containment Integrity Definition
3/4.6.1.1	Containment Systems - Primary Containment - Containment Integrity
3/4.6.1.2	Containment Systems - Containment Leakage, and associated Bases
3/4.6.1.3	Containment Systems - Containment Air Locks, and associated Bases
6.16	Administrative Controls - Containment Leakage Rate Testing Program

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

By: Guy G. Campbell
Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me this 26th day of July, 1999.

Nora L. Flood
Notary Public, State of Ohio - Nora L. Flood
My commission expires September 4, 2002.

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 Specifications (TS), including changes to: TS Index Page XVI; TS Definition 1.8, Containment Integrity; TS 3/4.6.1.1, Containment Systems - Primary Containment - Containment Integrity; TS 3/4.6.1.2, Containment Systems - Containment Leakage, and associated Bases; and TS 3/4.6.1.3, Containment Systems - Containment Air Locks, and associated Bases. In addition, a new TS Section 6.16, Containment Leakage Rate Testing Program, is proposed to be added to the Administrative Controls Section of the Technical Specifications.

A. Time Required to Implement: In order to have the changes proposed by this amendment application in effect during the Twelfth Refueling Outage, which is presently scheduled to commence in April 2000, the DBNPS requests that this amendment application be approved by February 1, 2000, with implementation on or prior to April 1, 2000. The Containment Leakage Rate Testing Program, as required by Appendix J, Option B, Section V.B.4, and as referenced in Section 6.16 of the proposed Technical Specifications, will be effective prior to implementation of the amendment.

B. Reason for Change (License Amendment Request Number 96-0012):

The proposed changes would adopt the 10 CFR 50, Appendix J, Option B approach for Type B and C containment leakage rate testing. The Option B approach for Type A containment leakage rate testing has already been implemented via License Amendment No. 205, which was issued on February 22, 1996.

These changes are being submitted to the NRC as a Cost Beneficial Licensing Action (CBLA). The proposed changes, together with the previously approved changes implementing Option B for Type A testing, will provide a cost savings well in excess of the CBLA threshold of $100,000 over the DBNPS's remaining life, and will result in reduced personnel radiation exposure.

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

Docket Number 50-346
License Number NPF-3
Serial Number 2572
Attachment

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 96-0012

(35 pages follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 96-0012

TITLE:

License Amendment Application to Revise Technical Specification Definitions, Containment Systems Technical Specifications and Associated Bases, and Technical Specification Administrative Controls for Implementation of 10 CFR 50, Appendix J, Option B for Type B and C Containment Leakage Rate Testing.

DESCRIPTION:

The proposed changes would modify the Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1, Operating License NPF-3, Appendix A Technical Specification (TS) Definition 1.8, Containment Integrity; TS 3/4.6.1.1, Containment Systems - Primary Containment - Containment Integrity; TS 3/4.6.1.2, Containment Systems - Containment Leakage, and associated TS Bases; and TS 3/4.6.1.3, Containment Systems - Containment Air Locks, and associated TS Bases. In addition, a new TS Section 6.16, Containment Leakage Rate Testing Program, is proposed to be added to the Administrative Controls Section of the Technical Specifications. A related change to the TS Index is also proposed to reflect the new TS Section 6.16. These changes are being proposed in order to adopt the performance-based 10 CFR 50, Appendix J, "Option B" approach for Type B and C containment leakage rate testing. The Option B approach for Type A containment leakage rate testing has already been implemented via License Amendment No. 205, which was approved by the NRC on February 22, 1996.

The proposed Technical Specification changes were developed utilizing Appendix J Option B model Technical Specifications developed for the Boiling Water Reactor (BWR/4) Standard Technical Specifications (STS) as guidance. These model Technical Specifications were included as an attachment to an NRC letter to the Nuclear Energy Institute (NEI) dated November 2, 1995. Similar model Technical Specifications specific to the Babcock and Wilcox Owners Group (BWOG) STS are being processed as NEI Technical Specifications Task Force (TSTF) Change Traveler TSTF-52, Revision 1. Although this change traveler has not yet been approved by the NRC for incorporation into the BWOG STS, the DBNPS also used it as guidance. The BWR/4 and BWOG model STS for Appendix J Option B are hereinafter referred to as the "model TSs."

These changes are being submitted to the NRC as a Cost Beneficial Licensing Action (CBLA). As demonstrated below, these changes will not adversely impact nuclear safety. The proposed changes, together with the previously approved changes implementing Option B for Type A testing, will provide a cost savings well in excess of the CBLA threshold of

$100,000 over the DBNPS's remaining life, and will result in reduced personnel radiation exposure.

The proposed changes are described in detail below. Each of the proposed changes is also shown on the attached marked-up Operating License pages.

TS Definition 1.8, Containment Integrity:

It is proposed to revise TS 1.8.b to remove the words "and sealed" from this portion of the definition, which presently reads: "All equipment hatches are closed and sealed." The equipment hatch is utilized in Mode 5 (Cold Shutdown) and Mode 6 (Refueling). Following its use, prior to ascension into Mode 4 (Hot Shutdown), the equipment hatch is tested to verify sealing capability. Leakage requirements relative to the sealing of the equipment hatch are encompassed by TS Definition 1.8.d. The proposed revision would also clarify that there is only one equipment hatch. The revised TS 1.8.b would read: "The equipment hatch is closed."

It is also proposed to revise TS 1.8.d to reference the Containment Leakage Rate Testing Program for containment leakage rate limits, rather than Specification 3.6.1.2, in order to reflect changes being proposed for TS 3/4.6.1.2, as described below. The Containment Leakage Rate Testing Program is described in a new Administrative Controls Section, TS 6.16, which also includes specific leakage rate requirements.

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

Associated with the proposed change to TS 1.8.b, it is proposed to revise Surveillance Requirement (SR) 4.6.1.1.a.2 to clarify that there is only one equipment hatch and to remove the words "and sealed" from this SR, which presently requires verification at least once per 31 days that "All equipment hatches are closed and sealed." The revised SR would read: "The equipment hatch is closed."

It is also proposed to revise SR 4.6.1.1.c to reference the Containment Leakage Rate Testing Program for specific requirements. The new SR 4.6.1.1.c would read as follows:

Primary CONTAINMENT INTEGRITY shall be demonstrated:

c. By performing required visual examinations of the containment vessel and shield building in accordance with the Containment Leakage Rate Testing Program.

TS 3/4.6.1.2, Containment Systems - Containment Leakage:

It is proposed to revise Limiting Condition for Operation (LCO) 3.6.1.2 to reference the Containment Leakage Rate Testing Program for specific requirements. The new LCO 3.6.1.2 would read as follows:

> Containment leakage rates shall be in accordance with the Containment Leakage Rate Testing Program.

Present Action statement 3.6.1.2.a prevents entry into Mode 4 (Hot Shutdown) from Mode 5 (Cold Shutdown) if the containment leakage rate is not within limits. However, TS 3.0.4 applies to this LCO, and TS 3.0.4 already precludes entry into Mode 4 if the LCO is not met. In Modes 1 through 4, an unacceptable containment leakage rate would also violate CONTAINMENT INTEGRITY, thereby necessitating entry into LCO 3.6.1.1.

In addition, present Action statement 3.6.1.2.b provides requirements in the event of an unacceptable purge valve leakage rate. Such an occurrence would violate CONTAINMENT INTEGRITY, thereby necessitating entry into LCO 3.6.1.1, which has a more limiting Action statement.

Based on the above, Action statements 3.6.1.2.a and 3.6.1.2.b are proposed to be replaced with a single Action statement to read as follows:

> With containment leakage rate(s) not within limit(s), restore containment leakage rate(s) within limit(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

It is also proposed to revise SR 4.6.1.2 to reference the Containment Leakage Rate Testing Program for specific requirements. This reference would be contained in new (renumbered) SR 4.6.1.2.1. Accordingly, present SRs 4.6.1.2.a, d, e, f, and j would be deleted. It is noted that SRs 4.6.1.2.b, c, and i were deleted in a previous License Amendment. In addition, SRs 4.6.1.2.g and h, which describe special testing requirements for the containment purge and exhaust isolation valves, would be consolidated into new (renumbered) SR 4.6.1.2.2. The new SRs 4.6.1.2.1 and 4.6.1.2.2 would read as follows:

> 4.6.1.2.1 The containment leakage rates shall be determined in accordance with the Containment Leakage Rate Testing Program.
>
> 4.6.1.2.2 A special test shall be performed to verify that the containment purge and exhaust isolation valves leakage rate is within the limits specified in the Containment Leakage Rate Testing Program, by pressurizing the piping section including one valve inside and one valve outside the containment to a pressure greater than or equal to 20 psig:
>
> a. Each time the containment purge and exhaust isolation valves are opened, within 72 hours after valve closure, or prior to entering MODE 4 from MODE 5, whichever is later.
>
> b. Each time the plant has been in any combination of MODES 3, 4, 5 or 6 for more than 72 hours, if not performed in the previous 6 months.

Bases 3/4.6.1.2, Containment Leakage:

It is proposed to revise the first paragraph to add a cross-reference to the new Administrative Controls Section 6.16. It is also proposed to revise the first paragraph to clarify that the "peak accident pressure" referred to is the "peak design basis loss of coolant accident pressure." The revised terminology conforms to the definition in Appendix J Option B and conforms to the model TSs. The revised paragraph would read as follows:

> As described in Administrative Controls Section 6.16, the Containment Leakage Rate Testing Program is based on Option B of Appendix J of 10 CFR 50. The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak design basis loss of coolant accident pressure of 38 psig, P_a. As an added conservatism, the measured, overall, as-left integrated leakage rate is further limited to $\leq 0.75\ L_a$, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

TS 3/4.6.1.3, Containment Systems - Containment Air Locks:

It is proposed to revise LCO 3.6.1.3.b to reference the Containment Leakage Rate Testing Program for specific requirements. The new LCO 3.6.1.3.b would read as follows:

> Each containment air lock shall be OPERABLE with:
>
> b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

As described in further detail below, the new Administrative Controls Section 6.16 proposes an increased overall air lock leakage rate limit of $0.015\ L_a$.

It is proposed to revise SR 4.6.1.3 to reference the Containment Leakage Rate Testing Program for specific requirements. Accordingly, SR 4.6.1.3.b would be deleted, and SR 4.6.1.3.a would be revised to read as follows:

> Each containment air lock shall be demonstrated OPERABLE:
>
> a. By performing required air lock leakage testing in accordance with the Containment Leakage Rate Testing Program.

As described in further detail below, the new Administrative Controls Section 6.16 proposes an increased air lock door seal leakage rate limit of $0.01\ L_a$ at ≥ 10 psig, and removes the provision relative to the performance of a door seal leakage rate test at P_a.

Related to the above changes, the current footnotes applicable to SR 4.6.1.3 would be deleted, and TS Action statement 3.6.1.3.a.3, which presently references SR 4.6.1.3.b, would be revised to instead reference "the overall air lock leakage rate test."

In addition, SR 4.6.1.3.a would be modified by an asterisked footnote, which would read:

> * One inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

<u>Bases 3/4.6.1.3, Containment Air Locks:</u>

Consistent with the changes proposed to TS 3/4.6.1.3, it is proposed to revise the first paragraph for clarification and to add a cross-reference to the new Administrative Controls Section 6.16. The revised paragraph would read as follows:

> The limitations on closure and leak rate for the containment air locks are required to ensure CONTAINMENT INTEGRITY and to meet the restrictions on overall containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests. Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program, which is described in Administrative Controls Section 6.16.

In addition, the following paragraphs would be added to the current Bases:

> One inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a design basis accident.
>
> The surveillance requirement which verifies that only one door in each air lock can be opened at a time is not part of the Containment Leakage Rate Testing Program. Therefore, its test frequency is subject to the provisions of Specification 4.0.2.

<u>TS Section 6.16, Administrative Controls - Containment Leakage Rate Testing Program:</u>

Consistent with the above changes, and consistent with the model TSs, a new TS 6.16, Containment Leakage Rate Testing Program, is proposed to be added, to read as follows:

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

1. A reduced duration Type A test may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

2. The fuel transfer tube blind flanges (containment penetrations 23 and 24) will not be eligible for extended test frequencies. Their Type B test frequency will remain at 30 months. However, As-found testing will not be required.

3. Following air lock door seal replacement, a door seal leakage rate test to demonstrate that the seal leakage rate is $\leq 0.01\ L_a$ with the volume between the door seals pressurized to ≥ 10 psig is an acceptable alternative to a door seal leakage test at P_a or an overall air lock leakage test at P_a.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 38 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $< 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.75\ L_a$ for Type A tests, $< 0.60\ L_a$ for all penetrations and valves subject to Type B and Type C tests, and $\leq 0.03\ L_a$ for all penetrations that are secondary containment bypass leakage paths;

2. A single penetration leakage rate of $\leq 0.15\ L_a$ for each containment purge penetration;

3. Air lock acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.015\ L_a$ when tested at $\geq P_a$,

b) For each door, seal leakage rate is $\leq 0.01\ L_a$ when the volume between the door seals is pressurized to ≥ 10 psig.

e. The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

f. The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

TS Index Page XVI:

It is proposed to revise the TS Index to reflect the addition of new TS Section 6.16.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The proposed changes involve Technical Specification surveillance test requirements, including 10 CFR 50 Appendix J Type A testing to measure the containment overall integrated leakage rate, and including testing to measure the leakage rate for penetrations and valves subject to 10 CFR 50 Appendix J Type B (e.g., containment penetrations and air lock door seals) and Type C (e.g., containment isolation valves) testing. There are no hardware modifications involved.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50), Section 50.54(o) specifies that primary reactor containments shall meet the containment leakage test requirements set forth in 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These test requirements provide for verification of the leak-tight integrity of the containment and of the systems and components which penetrate the containment, and establish the acceptance criteria for such tests. The purpose of the tests are to assure that (a) leakage through the containment, and systems and components penetrating the containment do not exceed allowable leakage rate values as specified in the Technical Specifications or associated Bases, and (b) surveillance of containment penetrations and isolation valves are performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

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 DBNPS 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. The containment vessel, including its penetrations, 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.

Access to the containment is provided by an equipment hatch, a personnel air lock, and an emergency air lock. Electrical and mechanical penetrations are provided for services to the containment. As mentioned above, periodic leakage rate tests of the containment vessel and leak tests of the testable penetrations are conducted to verify their continued leak-tight integrity.

The equipment hatch is a 10 CFR 50 Appendix J Type B penetration. It consists of a welded steel assembly, with a double-gasketed, flanged and bolted cover. Provision is made to pressurize the space between the double gaskets. The function of the equipment hatch is to allow for movement of equipment in and out of containment, e.g., during refueling outages.

The containment air locks are 10 CFR 50 Appendix J Type B penetrations. The personnel air lock is provided for routine transit into and out of containment when containment integrity is required. The emergency air lock is provided for emergency access and exit. Each air lock has two double-gasketed doors in series. Provision is made to pressurize the space between the gaskets for testing purposes. In addition, the air locks have nozzles installed that permit pressure testing of the air lock at any time.

Each air lock door is designed such that with the other door in the same air lock open, the closed door can withstand and seal against the design pressure of the containment vessel. During Modes 1 through 4, when one containment air lock door is open, the other door in the same air lock is required to be closed in accordance with TS LCO 3.6.1.3.a. This requirement, in conjunction with the containment air lock leakage limits, ensures that containment integrity is maintained during entry and exit through the containment air locks.

Two fuel transfer penetrations (containment penetrations 23 and 24) are provided to transport fuel assemblies between the refueling canal and the spent fuel pool during refueling operations. Each penetration consists of a 30-inch diameter stainless steel pipe installed inside a 42-inch sleeve. The inner pipe acts as the transfer tube. Each fuel transfer tube has one blind flange with a double O-ring seal installed on the inside of the Containment Vessel. This provides a double barrier. Provisions have been made in the design to air-test the flanged end of the fuel transfer tube for leak-tightness after use.

The Containment Purge System is designed to provide clean fresh air to the containment vessel or to the shield building and penetration rooms. Normally, the Purge System is not in operation in the containment purge mode, and the associated containment purge supply and exhaust isolation valves are closed. The "special test" for the containment purge and exhaust isolation valves, as described in present TS SRs 4.6.1.2.g and h, is intended to detect gross degradation of seals on the valve seats. This special test is performed in addition to the Appendix J testing requirements.

EFFECTS ON SAFETY:

Background:

On October 26, 1995, a final NRC rule implementing a performance-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 Appendix J 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 parts 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 for complying with regulatory safety objectives. The final rule publication also states that based on specific, detailed analyses of data from the North Anna and Grand Gulf nuclear power plants, and data from twenty-two nuclear plants, as documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995, performance-based alternatives to current Local Leak Rate Testing (LLRT) methods are feasible with marginal impact on risk.

In accordance with proposed TS 6.16, a Containment Leakage Rate Testing Program will be established in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. 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 Part 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.

Using the guidance provided in NEI 94-01, test intervals will be established for each applicable component based on performance history. The maximum test interval will follow the guidance of RG 1.163. Only those components that have satisfactorily maintained actual leakage less than the allowable leakage will be tested less frequently than the present requirements. Based on the above-mentioned justification presented in the final rule publication, the changes in Type B and C test frequency will not have a significant adverse effect on nuclear safety. Moreover, the extension in Type B and C 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 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 TS 6.16, which requires establishment of a Containment Leakage Rate Testing Program in accordance with the guidelines contained in Regulatory Guide 1.163, and the proposed changes to TS 1.8.d, SR 4.6.1.1.c, LCO 3.6.1.2, SR 4.6.1.2, LCO 3.6.1.3.b, SR 4.6.1.3.a, Bases 3/4.6.1.2, and Bases 3/4.6.1.3 to refer to the Containment Leakage Rate Testing Program, would meet the above-mentioned requirements of 10 CFR 50 Appendix J regarding TS revisions. As discussed above, the conversion to Option B for Type B and C testing will have no significant adverse effect on nuclear safety.

SR 4.6.1.1.c:

The proposed change to SR 4.6.1.1.c, which involves visual examinations of the containment vessel and shield building, to reference the "Containment Leakage Rate Testing Program" in lieu of the present reference to "10 CFR 50 Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995" is in accordance with the model TSs. The requirement to establish a Containment Leakage Rate Testing Program is specified in the proposed new Administrative Controls TS 6.16, which includes references to 10 CFR 50 Appendix J, Option B, as modified by approved exemptions, and RG 1.163. Therefore, since the present SR 4.6.1.1.c references are, in effect, being relocated to a different section of the Technical Specifications, this is an administrative change that will have no adverse effect on nuclear safety.

LCO 3.6.1.2:

The proposed changes to LCO 3.6.1.2 to add a reference to the Containment Leakage Rate Testing Program and to delete specific containment leakage rate limits is in accordance with the model TSs. The requirement to establish a Containment Leakage Rate Testing Program is specified in the proposed new Administrative Controls TS 6.16, which includes the specific containment leakage rate limits presently included in LCO 3.6.1.2. Therefore, since the present LCO 3.6.1.2 requirements are, in effect, being relocated to a different section of the Technical Specifications, this is an administrative change that will have no adverse effect on nuclear safety.

Definition 1.8.d:

Related to the relocation of containment leakage rate limits from LCO 3.6.1.2, TS Definition 1.8.d is affected. As previously described, this definition, which presently refers to the "limits of Specification 3.6.1.2," is proposed to be changed to refer to the Containment Leakage Rate Testing Program. This is an administrative change that will have no effect on nuclear safety.

LCO 3.6.1.2 Action Statements a and b:

The proposed replacement of Action statements 3.6.1.2.a and 3.6.1.2.b with a single new Action statement 3.6.1.2 which requires application of the same shutdown requirements as the Action of Specification 3.6.1.1 in the event that the containment leakage rate is not within limits is a simplification for the following reasons:

The present Action statement 3.6.1.2.a includes a requirement to take action in the event that the measured overall as-left integrated containment leakage rate exceeds 0.75 L_a. This requirement is incorporated into the proposed new Administrative Controls TS 6.16, which includes a requirement that is technically the same but is formatted differently, in accordance with the model TSs. The TS 6.16 requirement states: "During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.75\ L_a$ for Type A tests..."

The remainder of the present Action statement 3.6.1.2.a is superfluous in that it repeats the limits presently specified in LCO 3.6.1.2.b and c. In addition, since present Action statement 3.6.1.2.a, in effect, prevents entry into MODE 4 in the event the specified limits are not met, the Action statement is in itself superfluous in that TS 3.0.4 applies to this LCO and thus already prevents entry into MODE 4 if the LCO is not met.

Present Action statement 3.6.1.2.b includes a requirement to take action in the event LCO 3.6.1.2.d, regarding single penetration leakage rate for the containment purge and exhaust valve special test, is not met. However, in the event any of the requirements of LCO 3.6.1.2 are not met, CONTAINMENT INTEGRITY would not be met, necessitating entry into Specification 3.6.1.1 which contains more restrictive requirements. This being the case, Action statement 3.6.1.2.b is also superfluous.

In summary, it is an administrative simplification to replace the present LCO 3.6.1.2 Action statements with the proposed new Action statement, and since this change is also conservative, the proposed change will have no adverse effect on nuclear safety.

<u>SR 4.6.1.2.a:</u>

The proposed change to SR 4.6.1.2.a, which involves Type A testing requirements, to renumber the SR as SR 4.6.1.2.1 and to reference the "Containment Leakage Rate Testing Program" in lieu of the present reference to "10 CFR 50 Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September, 1995" is in accordance with the model TSs. The requirement to establish a Containment Leakage Rate Testing Program is specified in the proposed new Administrative Controls TS 6.16, which includes references to 10 CFR 50 Appendix J, Option B, as modified by approved exemptions, and RG 1.163. Therefore, since the present SR 4.6.1.2.a references are, in effect, being relocated to a different section of the Technical Specifications, this is an administrative change that will have no adverse effect on nuclear safety.

<u>SR 4.6.1.2.d:</u>

The proposed deletion of SR 4.6.1.2.d, which involves Type B and C testing requirements in accordance with 10 CFR 50, Appendix J, Option A is acceptable since, as previously discussed, it is the intent of this license amendment request to adopt Option B requirements for Type B and C testing. Proposed new SR 4.6.1.2.1 encompasses all containment leakage rate testing, including Type B and C testing, and will require such testing to be performed in accordance with the Containment Leakage Rate Testing Program. As previously discussed, the changes in Type B and C test frequency which may result from the adoption of Option B will not have an adverse effect on nuclear safety.

SR 4.6.1.2.e:

The proposed deletion of SR 4.6.1.2.e, which regards combined bypass leakage rate testing, noting an exception for penetrations that are not individually testable, is acceptable since the quoted combined bypassed leakage rate limit is already included in the present LCO 3.6.1.2.c, which is proposed for relocation to TS 6.16, as previously discussed. In addition, the exception mentioned for "penetrations not individually testable" is not applicable since the DBNPS has no such penetrations. Further, such an exception is not included in the model TSs. In accordance with Appendix J, Option B, the frequency of testing will no longer be prescribed in the TS, but will be determined based upon performance history and the limits of Regulatory Guide 1.163. This change will, therefore, have no adverse effect on nuclear safety.

SR 4.6.1.2.f:

The proposed deletion of SR 4.6.1.2.f, which states that the air locks shall be in compliance with the requirements of Specification 3/4.6.1.3, is acceptable because TS 3/4.6.1.3 already addresses containment leakage aspects for the air locks. In addition, considering the proposed changes to TS 1.8.d, TS 3/4.6.1.3, and TS 6.16, containment leakage requirements relative to the air locks will continue to be maintained. Therefore, this change will have no adverse effect on nuclear safety.

SR 4.6.1.2.g and h:

The proposed consolidation of SR 4.6.1.2.g and SR 4.6.1.2.h, regarding special test requirements for the containment purge and exhaust isolation valves, into a new SR 4.6.1.2.2 is acceptable for the following reason. All the technical requirements presently included in the present two surveillance requirements are included in the proposed new SR with the exception that the new SR references the Containment Leakage Rate Testing Program for the leakage rate limit, in lieu of including the limit in the SR. Therefore, this is an administrative change that will have no adverse effect on nuclear safety.

SR 4.6.1.2.j:

The proposed deletion of SR 4.6.1.2.j, which states that the provisions of Specification 4.0.2 are not applicable, is acceptable since the proposed LCO 3.6.1.2 references the new TS 6.16 Containment Leakage Rate Testing Program which, in turn, includes a statement that the provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. Therefore, since the present SR 4.6.1.2.j requirement is, in effect, being relocated to a different section of the Technical Specifications, this is an administrative change that will have no adverse effect on nuclear safety.

LCO 3.6.1.3.b:

The proposed change to LCO 3.6.1.3.b to add a reference to the Containment Leakage Rate Testing Program and to delete the specific overall air lock leakage rate limit is in accordance with the model TSs. The requirement to establish a Containment Leakage Rate Testing Program is specified in the proposed new Administrative Controls TS 6.16, which includes the specific overall air lock leakage rate limit presently included in LCO 3.6.1.3.b. Therefore, since the present LCO 3.6.1.3.b requirements are, in effect, being relocated to a different section of the Technical Specifications, this is an administrative change that will have no adverse effect on nuclear safety.

The new Administrative Controls Section 6.16 proposes an increased overall air lock leakage rate limit of 0.015 L_a. The justification for this change is provided below.

SR 4.6.1.3.a and b:

The proposed change to SR 4.6.1.3.a to add a reference to the Containment Leakage Rate Testing Program and to delete specific air lock door seal test requirements and leakage rate limits is in accordance with the model TSs. The requirement to establish a Containment Leakage Rate Testing Program is specified in the proposed new Administrative Controls TS 6.16, which includes the test requirements and leakage rate limits. Changes to the exceptions of the test frequency requirements currently specified in SR 4.6.1.3.a.1 and SR 4.6.1.3.a.2, and to the leakage rate limit for the air lock seals, when the volume between the door seals is pressurized, are addressed separately below. Since the present SR 4.6.1.3.a requirements are, in effect, being relocated to a different section of the Technical Specifications, this is an administrative change that will have no adverse effect on nuclear safety.

Relative to the test frequency requirements currently specified in SR 4.6.1.3.a.1 and SR 4.6.1.3.a.2, as noted above, test intervals will be established using the guidance provided in NEI 94-01, which is endorsed by RG 1.163. Section 10.2.2 of NEI 94-01 requires an air lock test frequency of at least once per 30 months, an air lock door seal test within seven days after each containment entry when containment integrity is required, and an air lock door seal test at least once per 30 days during periods of multiple containment entries where the air lock doors are routinely used for access more frequently than once per seven days. As previously discussed, the changes in test frequency which may result from the adoption of Option B will not have a significant adverse effect on nuclear safety.

Related to the proposed changes to SR 4.6.1.3.a, a new footnote is proposed stating that one inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. Since either door in an air lock is alone capable of withstanding and sealing against the design pressure of the containment vessel, the inoperability of one door, assuming the operability of the remaining door, does not make the air lock penetration itself inoperable. This proposed footnote is consistent with the model TSs. Accordingly, this proposed change is a clarification that will have no adverse effect on nuclear safety.

The proposed deletion of SR 4.6.1.3.b, regarding overall air lock leakage test requirements, is acceptable since the requirements for overall air lock leakage testing are enveloped by the proposed SR 4.6.1.3.a, which would read: "By performing required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program." The above discussion of changes in test frequency for SR 4.6.1.3.a also applies to SR 4.6.1.3.b.

Related to the proposed changes to SR 4.6.1.3.a and SR 4.6.1.3.b, two footnotes that previously applied to these SRs are no longer applicable under the proposed changes, and are, therefore, proposed to be deleted. The present footnote which reads: "Exemption to Appendix 'J' of 10 CFR 50," refers to an exemption to 10 CFR 50 Appendix J, Section III.D.2(b)(ii), which was approved by the NRC on November 1, 1994 (Toledo Edison Log Number 4434). This exemption is no longer required or applicable upon adoption of the 10 CFR 50 Appendix J Option B requirements for air lock testing. The present footnote which reads: "The provisions of Specification 4.0.2 are not applicable," is, in effect, being relocated to the proposed new Administrative Controls TS 6.16, which includes a statement that the provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. These are administrative changes that will have no adverse effect on nuclear safety.

The new Administrative Controls Section 6.16 proposes an increased air lock door seal leakage rate limit from "no detectable leakage" to $\leq 0.01\ L_a$ at ≥ 10 psig, and removes the provision relative to the performance of a door seal leakage rate test to demonstrate a seal leakage rate of $\leq 0.0015\ L_a$ at P_a. The justification for these changes is provided below under the TS 6.16 discussion.

<u>LCO 3.6.1.3 Action Statement a.3:</u>

Related to the proposed change to SR 4.6.1.3.b, TS Action statement 3.6.1.3.a.3, which presently references SR 4.6.1.3.b, would be revised to reference "the overall air lock leakage rate test." This is an administrative change and will have no adverse effect on nuclear safety.

<u>TS 6.16:</u>

The proposed addition of TS 6.16, Containment Leakage Rate Testing Program, is consistent with other changes proposed in this license amendment request, and is consistent with the model TSs. This section contains the overall integrated leakage rate limit and the individual leakage rate limits for Type A, B and C tests prior to plant startup following the testing. This change will have no adverse effect on nuclear safety.

It is noted that whereas the overall integrated leakage rate limit is specified in the current LCO 3.6.1.2 as "$\leq L_a$", the same limit, as relocated to the new TS 6.16.d.1, is "$< 1.0\ L_a$". Similarly, whereas the combined leakage rate limit for Type B and C tests is specified in the current LCO 3.6.1.2 as "$\leq 0.60\ L_a$", the same limit, as relocated to the new TS 6.16.d.1, is "$< 0.60\ L_a$". The proposed limits are conservative compared to the current limits, and are consistent with the guidance provided by NEI 94-01, which has been endorsed by the NRC. Hence, these changes will have no adverse effect on nuclear safety.

As noted above, air lock leakage rate limits have been relocated from TS 3/4.6.1.3 to TS 6.16. Limiting Condition for Operation 3.6.1.3.b presently specifies "an overall air lock leakage rate of $\leq$ 0.002 L_a at P_a, 38 psig." Surveillance Requirement 4.6.1.3.a presently specifies "either no detectable seal leakage when the volume between the door seals is pressurized to 10 psig, or by verifying a seal leakage rate of $\leq$ 0.0015 L_a when the volume between the door seals is pressurized to P_a, 38 psig." In each of these cases, the required test pressure is presently given as a fixed value. Since it is conservative to test at a pressure greater than this fixed value, the associated TS 6.16 air lock acceptance criteria specify "$\geq$" when referring to the required test pressure. This change is consistent with the model TSs and will have no adverse effect on nuclear safety.

Regarding the overall air lock leakage rate limit, an increased limit of 0.015 L_a is proposed, compared to the current LCO 3.6.1.3.b limit of 0.002 L_a. The current limit is much more restrictive than the limit of 0.05 L_a specified in NUREG-0103, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," Revision 0, dated June 1, 1976, upon which the original DBNPS Technical Specifications were modeled. Containment leakage is comprised of two components: filtered air and unfiltered air or bypass leakage. The USAR Chapter 15 accident analyses account for the bypass leakage. The proposed TS 6.16 includes a limit on secondary containment bypass leakage, 0.03 L_a, which is unchanged from the current LCO 3.6.1.2.c value. Air lock leakage is a component of bypass leakage. NUREG-0103 specified limits of 0.05 L_a for overall air lock leakage, and 0.10 L_a for bypass leakage, thus the specified air lock leakage limit comprised 50% of the bypass limit. An overall air lock leakage limit of 0.015 L_a is proposed, corresponding to 50% of the bypass limit, similar to NUREG-0103. This proposed change has the potential to alleviate future maintenance burden, and any associated personnel radiation exposures, due to the overly restrictive current limit. Since the current bypass leakage limit will remain the same, the accident analyses are not affected by the proposed increase in the overall air lock leakage limit. Hence, this change will have no adverse effect on nuclear safety.

Regarding the air lock door seal leakage rate limit, an increased limit of 0.01 L_a at $\geq$ 10 psig is proposed, compared to the current SR 4.6.1.3.a limit of "no detectable leakage when the volume between the door seals is pressurized to 10 psig." As stated previously, air lock leakage is a component of bypass leakage, and the current bypass leakage limit will remain the same, therefore the accident analyses are not affected by the proposed increase in the air lock door seal leakage rate limit. Further, this proposed change has the potential to alleviate future maintenance burden and any associated personnel radiation exposures, due to the overly restrictive current limit. It is also proposed to remove the provision allowing performance of the door seal leakage rate test at P_a, with a corresponding seal leakage rate limit of 0.0015 L_a, in lieu of the 10 psig test. Door seal leakage rate testing at the DBNPS has historically been conducted at the lower pressure. With the revised acceptance criteria for the lower pressure test, a provision for a test at P_a is unnecessary. These proposed changes are consistent with the model TSs. Based on the above, these changes will have no adverse effect on nuclear safety.

Included in the proposed TS 6.16 are three exceptions to Regulatory Guide (RG) 1.163. Each of these exceptions is discussed in further detail below. In addition, it is noted that errata applicable to Nuclear Energy Institute (NEI) 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J" have been issued by NRC letter dated March 6, 1996. NEI 94-01 is referenced by Regulatory Guide 1.163. The DBNPS does not consider the use of the errata to be an "exception" to Regulatory Guide 1.163.

Exception Regarding Type A Test Methodology

This exception states that a reduced duration Type A test may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1. Use of this topical report was approved by the Atomic Energy Commission (AEC) in February 1973. Since this test methodology has been used at the DBNPS and is anticipated to be used in the future, the DBNPS prefers to list this as an exception so as to avoid any future confusion as to its acceptability. Continued use of this test methodology will have no adverse effect on nuclear safety.

Exception Regarding Fuel Transfer Tube Penetrations

This exception addresses the surveillance test requirements for the fuel transfer tube blind flanges (containment penetrations 23 and 24). It is proposed to retain their Type B test frequency at 30 months, while eliminating the requirement to perform As-found testing. As-left testing will continue to be performed following reinstallation of the blind flanges after fuel transfer operations are completed during a refueling outage.

It is desirable to minimize testing since the blind flanges are located in the tip-ender area of the refueling canal inside containment, which is a high radiation/high contamination area during the test evolution. In addition to the As-Low-As-Reasonably-Achievable (ALARA) concerns, there are also industrial safety concerns. Personnel access to the area is via a vertical ladder, requiring fall protection. In addition, the floor at this elevation is typically wet and slippery, which is a potential hazard to the test personnel. Finally, the high temperature in the area requires test personnel to observe heat stress precautions.

A review of the surveillance test history from September 1991 through May 1998, which includes the Seventh Refueling Outage through the most recent Eleventh Refueling Outage, shows no test failures.

Based on the aforementioned ALARA and industrial safety concerns, and given the excellent surveillance test performance history for the fuel transfer tube penetrations, it is considered that the proposed exception is justified. Given the retention of a 30-month test frequency and the continued performance of As-left testing, there will be no adverse effect on nuclear safety.

Exception Regarding Air Lock Door Seal Testing Following Replacement

This exception addresses air lock door seal testing requirements. Under the guidance of NEI 94-01 Section 10.2.2.2, either an overall air lock test at P_a, or a leak rate test of the affected component, at P_a, is required following maintenance on an air lock pressure retaining boundary. For the case of maintenance involving replacement of a door seal, the proposed exception would allow the additional option of performing a reduced pressure test. Since a reduced pressure test is acceptable for routine air lock door seal leakage rate testing, there will be no adverse effect on nuclear safety.

Bases 3/4.6.1.2 and 3/4.6.1.3, and Index:

The proposed changes to TS Bases 3/4.6.1.2 and 3/4.6.1.3, and to the TS Index, are administrative changes associated with the other proposed changes and have no adverse effect on nuclear safety.

Definition 1.8.b and SR 4.6.1.1.a.2:

The proposed change to TS 1.8.b would remove the words "and sealed" from this portion of the definition, which presently reads: "All equipment hatches are closed and sealed." The equipment hatch is utilized in Mode 5 (Cold Shutdown) and Mode 6 (Refueling). Following its use, prior to ascension into Mode 4 (Hot Shutdown), the equipment hatch is tested to verify sealing capability. Leakage requirements relative to the sealing of the equipment hatch are encompassed by TS Definition 1.8.d; therefore, the words "and sealed" are unnecessary. In addition, removal of these words is consistent with Bases B3.6.1, "Containment Systems - Containment," of NUREG-1430, "Improved Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," Revision 1, April 1995. The proposed change would also clarify that there is only one equipment hatch.

The proposed change to Surveillance Requirement (SR) 4.6.1.1.a.2 would remove the words "and sealed" from this SR, which presently requires verification at least once per 31 days that "All equipment hatches are closed and sealed." This change is associated with the proposed change to TS 1.8.b. Removal of the words "and sealed" would eliminate the possibility that the present wording of the SR could be misinterpreted to require that the Appendix J Type B Local Leak Rate Test for the equipment hatch be performed on a 31 day frequency. The proposed change would also clarify that there is only one equipment hatch.

Based on the above, these proposed changes are clarifications and have no adverse effect on nuclear 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 consequences 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. The Davis-Besse Nuclear Power Station 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 and Bases implement 10 CFR 50 Appendix J Option B for Type B and C Local Leak Rate Testing, based on the guidance of Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." Provided that components have performed satisfactorily on a historical basis, this guidance permits the use of extended testing frequencies. 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 or total allowable releases. With the exception of the proposed increase in the containment air lock leakage limits, the proposed changes do not affect the total allowable containment leakage rates presently specified in the Technical Specifications. Although the air lock leakage limits are proposed to be increased, the accident analyses are based on the current TS allowable maximum bypass leakage, which is not proposed to be changed. Therefore, increases in leakage limits for individual components, such as the air locks and their door seals, which are constituents of bypass leakage, will have no effect on the radiological consequences described in the accident analyses.

The proposed TS changes relating to implementation of 10 CFR 50 Appendix J Option B may result in a small, but acceptable increase in post-accident containment leakage, due to the increased probability that due to generally increased intervals between tests, an unacceptable leakage rate could go undetected for a longer length of time. NUREG-1493, "Performance-Based Containment Leak-Test Program," September, 1995, which provided the technical basis for the 10 CFR 50 Appendix J Option B rulemaking, provides a detailed evaluation of the expected leakage and its consequences and concludes that increased test frequencies are workable without significant risk impacts.

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. The proposed changes do not affect the methodology used in conducting containment leak rate testing. The proposed changes do not involve a change to the plant design or operation and, therefore, will not introduce any new or different failure modes or initiators.

3. Not involve a significant reduction in a margin of safety.

   The proposed changes relating to implementation of 10 CFR 50, Appendix J, Option B do not significantly affect the allowable containment leakage rates presently specified in the Technical Specifications. The Technical Specifications, under the proposed changes, will continue to ensure containment reliability by periodic testing performed in full compliance with 10 CFR 50, Appendix J.

CONCLUSION:

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not constitute a significant hazards consideration. Furthermore, 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 unreviewed 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 231.

2. DBNPS Updated Safety Analysis Report through Revision 21.

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. NUREG-1430, "Improved Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," Revision 1, April 1995.

8. Model Technical Specifications, "BWR/4 STS Option B Model, October 31, 1995," as attached to the NRC letter from Christopher I. Grimes, Chief, Technical Specifications Branch, to David J. Modeen, Director, Operations and Management, Nuclear Energy Institute, dated November 2, 1995.

9. Nuclear Energy Institute (NEI) Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-52, Revision 1.

10. NRC License Amendment No. 205 to Facility Operating License No. NPF-3, dated February 22, 1996 (Toledo Edison Log Number 4793).

11. Exemption from the Requirements of 10 CFR Part 50, Appendix J - Davis-Besse Nuclear Power Station, Unit No. 1 (TAC No. M90649), dated November 1, 1994 (Toledo Edison Log Number 4434).

12. NRC letter from David L. Morrison, Director, Office of Nuclear Regulatory Research, to Regulatory Guide Distribution List for Division 1, "Changes to NEI 94-01, Revision 0," dated March 6, 1996.

13. Bechtel Topical Report BN-TOP-1, Revision 1, "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants," dated November 1, 1972.

14. AEC letter from R. C. DeYoung, Assistant Director for Pressurized Water Reactors, Directorate of Licensing, to R. D. Allen, Vice President, Bechtel Corporation, dated February 1, 1973.

15. NUREG-0103, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," Revision 0, dated June 1, 1976.