May 27, 1997

Mr. Oliver D. Kingsley Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: ISSUANCE OF AMENDMENT - IMPLEMENTATION OF 10 CFR 50, APPENDIX J, OPTION B, PERFORMANCE-BASED CONTAINMENT LEAKAGE TESTING (TAC NO. M97698)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 5 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1. This amendment is in response to your application dated January 10, 1997, as supplemented May 2 and 15, 1997. The amendment modifies the Watts Bar Nuclear Plant (WBN) Unit 1 Technical Specifications (TS) in order to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, "Performance-Based Containment Leakage-Test Program." The revised Appendix J provided an Option B which allows performance based testing for containment leakage rate testing. The TS in Section 3.6 and associated Bases, TS Section 3.0.2 and TS Section 5.7 would be changed. Also, the schedular exemption for containment airlock testing specified in the facility License Condition 2.D(1) is no longer required and is deleted from the Watts Bar Unit 1 Operating License.

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

Sincerely,
ORIGINAL SIGNED BY
RONALD W. HERNAN FOR:

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures: 1. Amendment No. 5 to NPF-90
2. Safety Evaluation

cc w/enclosures: See next page

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

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Chief Nuclear Officer
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Sincerely,

Ronald W. Hernan for

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures: 1. Amendment No. 5 to NPF-90
2. Safety Evaluation

cc w/enclosures: See next page

Mr. Oliver D. Kingsley, Jr.
Tennessee Valley Authority

WATTS BAR NUCLEAR PLANT

cc:
Mr. O. J. Zeringue, Sr. Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Jack A. Bailey, Vice President
Engineering & Technical Services
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. J. A. Scalice, Site Vice President
Watts Bar Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Spring City, TN 37381

General Counsel
Tennessee Valley Authority
ET 10H
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. Raul R. Baron, General Manager
Nuclear Assurance and Licensing
Tennessee Valley Authority
4J Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Pedro Salas, Manager
Licensing and Industry Affairs
Tennessee Valley Authority
4J Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Paul L. Pace, Manager
Licensing and Industry Affairs
Watts Bar Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Spring City, TN 37381

Mr. Richard T. Purcell, Plant Manager
Watts Bar Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Spring City, TN 37381

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
61 Forsyth Street, SW., Suite 23T85
Atlanta, GA 30303-3415

Senior Resident Inspector
Watts Bar Nuclear Plant
U.S. Nuclear Regulatory Commission
1260 Nuclear Plant Road
Spring City, TN 37381

County Executive
Rhea County Courthouse
Dayton, TN 37321

County Executive
Meigs County Courthouse
Decatur, TN 37322

Mr. Michael H. Mobley, Director
Division of Radiological Health
3rd Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 5
License No. NPF-90

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated January 10, 1997, as supplemented May 2 and May 15, 1997 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 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 paragraphs 2.C.(2) and 2.D.(1) of Facility Operating License No. NPF-90 are hereby amended to read as follows:

2.C.(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 5 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Also, the license is amended by deleting Paragraph 2.D.(1) on Page 4 of Facility License NPF-90.

4. This license amendment is effective as of the date of its issuance, to be implemented no later than 30 days of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
1. Page 4 of License NPF-90*
2. Changes to the Technical Specifications

Date of Issuance: May 27, 1997

*Page 4 is attached, for convenience, for the composite license to reflect this change.

D. The following exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. Therefore, these exemptions are granted pursuant to 10 CFR 50.12.

(1) Deleted

(2) The facility was previously granted an exemption from the criticality monitoring requirements of 10 CFR 70.24 (see Special Nuclear Material License No. SNM-1861 dated September 5, 1979). The technical justification is contained in Section 9.1 of Supplement 5 to the Safety Evaluation Report, and the staff's environmental assessment was published on April 18, 1985 (50 FR 15516). The facilty is hereby exempted from the criticality alarm system provisions of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

(3) The facility requires an exemption from 10 CFR 73.55(c)(10). The justification for this exemption is contained in Section 13.6.9 of Supplement 15 and 20 to the Safety Evaluation Report. The staff's environmental assessment was published on April 25, 1995 (60 FR 20291). Pursuant to 10 CFR 73.5, the facility is exempted from the stated implementation schedule of the surface vehicle bomb rule, and may implement the same as late as February 17, 1996.

(4) The facility was previously granted an exemption from certain requirements of 10 CFR 73.55(d)(5) relating to the returning of picture badges upon exit from the protected areas, such that individuals not employed by TVA who are authorized unescorted access into protected areas can take their badges offsite (see 59 FR 66061, December 22, 1994). The granting of this exemption is hereby affirmed.

ATTACHMENT TO AMENDMENT NO. 5

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

Remove Pages	Insert Pages
3.6-2	3.6-2
3.6-7	3.6-7
3.6-14	3.6-14
5.0-28	5.0-28
-	5.0-28a
B 3.0-11	B 3.0-11
B 3.0-12	B 3.0-12
B 3.6-1	B 3.6-1
B 3.6-2	B 3.6-2
B 3.6-3	B 3.6-3
B 3.6-4	B 3.6-4
B 3.6-5	B 3.6-5
B 3.6-7	B 3.6-7
B 3.6-12	B 3.6-12
B 3.6-13	B 3.6-13
B 3.6-25	B 3.6-25
B 3.6-26	B 3.6-26
B 3.6-27	B 3.6-27

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	----------NOTES---------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program.
SR 3.6.2.2	----------NOTE---------- Only required to be performed upon entry or exit through the containment air lock. ---------- Verify only one door in the air lock can be opened at a time.	184 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.8	Verify the combined leakage rate for all shield building bypass leakage paths is $\leq 0.25\ L_a$ when pressurized to ≥ 15.0 psig.	In accordance with the Containment Leakage Rate Testing Program

5.7 Procedures, Programs, and Manuals

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.7.2.19 Containment Leakage Rate Testing Program

A program shall be established to implement 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 (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 15.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.19 Containment Leakage Rate Testing Program (continued)

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 6 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the surveillance requirement will include a note in the frequency stating, "SR 3.0.2 does not apply." An example of an exception when the test interval is not specified in the regulations, is the discussion in the Containment Leakage Rate Testing

(continued)

BASES

SR 3.0.2 (continued)

Program, that SR 3.0.2 does not apply. This exception is provided because the program alreadyincludes extension of test intervals.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain the radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a concrete base mat with steel membrane. It is completely enclosed by a reinforced concrete shield building. An annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to provide for the collection, mixing, holdup, and controlled release of containment out leakage. Ice condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment integrity. The shield building provides shielding and allows controlled filtered release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and Nuclear Steam Supply System.

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

(continued)

BASES

BACKGROUND (continued)

a. All penetrations required to be closed during accident conditions are either:

1. capable of being closed by an OPERABLE automatic containment isolation system, or

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves."

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks."

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break (SLB), and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) related to the design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25% per day in the safety analysis at P_a = 15.0 psig which bounds the calculated peak containment internal pressure resulting from the limiting design basis LOCA (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first start up after performing a required Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2), purge valves with resilient seals, and Shield Building containment bypass leakage (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J, Option B.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

(continued)

BASES (continued)

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock, Shield Building containment bypass leakage path, and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required leakage test is required to be < 0.6 L_a for combined Type B and C leakage and $\leq$ 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq$1.0 L_a. At $\leq$1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1 (continued)

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J, Option B, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."

2. Watts Bar FSAR, Section 15.0, "Accident Analysis."

3. Watts Bar FSAR, Section 6.2, "Containment Systems."

4. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a significant release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate (L_a) of 0.25% of containment air weight per day (Ref. 2), at the calculated peak containment pressure of 15.0 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an 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 DBA. Note 2 requires the results of the air lock leakage tests to be evaluated against the acceptance criteria of the Containment Leakage Rate Testing Program, 5.7.2.19. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when the containment air lock door is opened, this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every 184 days. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door status available to operations personnel and because the interlock is only disabled in MODES 5 and 6.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J, Option B, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."

2. Watts Bar FSAR, Section 15.0, "Accident Analysis."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4 (continued)

isolation time and Frequency of this SR are in accordance with the Inservice Testing Program or 92 days.

SR 3.6.3.5

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B (Ref. 4), is required to ensure OPERABILITY.

Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.6 (continued)

Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.7

Verifying that each 24 inch containment lower compartment purge valve is blocked to restrict opening to ≤ 50° is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

SR 3.6.3.8

This SR ensures that the combined leakage rate of all Shield Building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The as-left bypass leakage rate prior to the first startup after performing a leakage test, requires a calculation using maximum pathway leakage (leakage through the worse of the two isolation valves). If the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange, then the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. At all other times the leakage rate will be calculated using minimum pathway leakage.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.8 (continued)

The frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria. Although not a part of L_a, the Shield Building bypass leakage path combined leakage rate is determined using the 10 CFR 50, Appendix J, Option B, Type B and C leakage rates for the applicable barriers.

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis."

2. Watts Bar FSAR, Section 6.2.4.2, "Containment Isolation System Design," and Table 6.2.4-1, "Containment Penetrations and Barriers."

3. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."

4. Title 10, Code of Federal Regulations, Part 50, Appendix J, Option B, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 5 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B "Performance-Based Requirements" to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By letter dated January 10, 1997, as supplemented May 2 and May 15, 1997, the Tennessee Valley Authority (the licensee) submitted a request for changes to the Watts Bar Nuclear Plant (WBN), Unit 1, Technical Specifications (TS). The requested changes would permit implementation of 10 CFR Part 50, Appendix J, Option B. The licensee has established a "Containment Leakage Rate Testing Program" and proposed adding this program to the TS. The program references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," which specifies a method acceptable to the NRC for complying with Option B dated September 1995. The May 2 and May 15, 1997 letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous

ENCLOSURE

performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

RG 1.163, was developed as a method acceptable to the NRC staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the Watts Bar TS 5.7.2.19.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's January 10, 1997, letter, as supplemented May 2 and May 15, 1997 to the NRC proposes to establish a "Containment Leakage Rate Testing Program" and proposes to add this program to the TS. The program references RG 1.163, which specifies a method acceptable to the NRC for complying with Option B. This requires a change to existing TS surveillance requirements 3.6.1.1, 3.6.2.1, 3.6.3.8 and the addition of the "Containment Leakage Rate Testing Program" to Section 5.7.2.19. Corresponding bases were also modified.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B and C testing on a performance basis.

TS Changes

Surveillance Requirements 3.6.1.1 for visual examination and leakage rate testing, except for air locks; 3.6.2.1 for air lock testing; 3.6.3.8 for shield building bypass leakage paths and the program description in TS 5.7.2.19 "Containment Leakage Rate Testing Program" are consistent with the staff's model TS guidance of November 2, 1995 and are acceptable.

The Bases for SR 3.0.2 (pages B 3.0-11 and 12) and for TS 3.6.2 (pages B 3.6-7, 12 and 13) are consistent with the staff's model TS guidance of November 2, 1995. Throughout the Watts BAR TS pages related to containment leakage testing, wherever the term "Appendix J" is used it has been augmented to read "Appendix J, Option B." This is appropriate since Watts Bar's containment leakage program has been revised to be based on compliance with 10 CFR Part 50, Appendix J, Option B.

The Bases for TS 3.6.1 (pages B 3.6-1, 2, 3, 4 and B 3.6-5) have been revised consistent with the staff's generic guidance with the following notation. The Bases for SR 3.6.1.1 includes the term "$< 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage." These terms are consistent with Appendix J, Option B, and are acceptable. Also, the licensee's references include RG 1.163 and do not include the NEI 94-01 and ANSI/ANS-56.8-1994 documents referenced in the staff's model TS guidance. This is acceptable since the referenced RG 1.163 incorporates the NEI and ANSI/ANS documents by reference.

The Bases for SR 3.6.3 (pages B 3.6-25, 26 and 27) have been revised consistent with the staff's generic guidance with the following notation. SR 3.6.3.8 has been revised to provide that as-left leakage from shield building bypass leakage paths shall be calculated using maximum pathway leakage and as-found leakage will be calculated using minimum pathway leakage, rather than using maximum pathway leakage at all times. The licensee's justification is that, using maximum pathway methodology for both as-found and as-left calculations would be unnecessarily restrictive and not consistent with the intent of NEI 94-01, which allows the proposed appproach when summing up the leakage rates for all Type B and C components for comparison to the 0.6 L_a limit. The staff concurs with TVA's assessment for these penetrations and finds TVA's proposed changes to be acceptable.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and the Model TS of the November 2, 1995, letter and are, therefore, acceptable to the staff.

License Condition 2.D(1)

TVA also proposes to delete the schedular exemption granted by license condition 2.D(1) from the license. The exemption was discussed in part in Supplement 19 to the Watts Bar Safety Evaluation Report as follows:

> In SSER 4, the staff stated that, as a result of the applicant's request of December 3, 1994, a partial exemption from paragraph III.D.2(b)(ii) of Appendix J to 10 CFR Part 50 would be granted. This will have the effect of permitting substitution of the seal leakage test of paragraph III.D.2(b)(iii) for the full-pressure test of paragraph III.D.2(b)(ii). Paragraph III.D.2(b)(ii) requires that if an air lock is opened during Modes 5 and 6, an overall air lock leakage test at not less than P_a be performed before plant heatup and startup (i.e., entering Mode 4). The exemption will permit that if no maintenance that could affect sealing capability has been performed on an air lock, then no full-pressure test need be performed.
>
> Subsequently, Appendix J was revised so that it now provides two options. Option A contains requirements identical to those before the revision. Option B permits use of performance-based technical specifications. The choice of Option A or B is voluntary on the licensee's part.

The schedular exemption is applicable to portions of the Commission's regulations known as "Option A" of 10 CFR 50, Appendix J. With licensee's adoption and the NRC staff's approval of testing in accordance with "Option B" of 10 CFR 50, Appendix J, testing of the containment air locks will be performed in accordance with "Option B". Accordingly, the exemption is no longer applicable and may be deleted from the license.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The 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 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards

consideration, and there has been no public comment on such finding (62 FR 4356 dated January 29, 1997). 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 Commission 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Pulsipher and R. Martin

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