October 3, 2006

Mr. Randall K. Edington
Vice President-Nuclear and CNO
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: ADDITIONAL EXTENSION OF APPENDIX J, TYPE A, INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. MC9732)

Dear Mr. Edington:

The Commission has issued the enclosed Amendment No. 224 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications in response to your application dated January 30, 2006, as supplemented by letter dated May 17 and August 29, 2006.

The amendment would revise Cooper Nuclear Station Technical Specification Section 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow a one-time extension of no more than 5 years for the Type A, Integrated Leakage Rate Test (ILRT) interval. This revision is a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests as defined in Nuclear Energy Institute (NEI) document, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," pursuant to Part 50 of Title 10 of the *Code of Federal Regulations*, Appendix J, Option B. The requested exception is to allow the ILRT to be performed within 15 years from the last ILRT, last performed on December 7, 1998.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Brian Benney, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 224 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

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NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 224
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee), dated January 30, 2006, as supplemented by letter dated May 17 and August 29, 2006, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 224, are hereby incorporated in the license. The Nebraska Public Power District shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 3, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 224

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
5.0-16	5.0-16
5.0-17	5.0-17

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 224 TO

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated January 30, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML060330307), and as supplemented by letters dated May 17 (ADAMS Accession No. ML061420155) and August 29, 2006 (ADAMS Accession No. ML062480040), Nebraska Public Power District (the licensee) requested a Technical Specification (TS) change for Cooper Nuclear Station (CNS). Specifically, the change would allow a one-time change in their Appendix J, Type A test (containment Integrated Leakage Rate Test (ILRT)) interval from 10 years to a test interval of 15 years.

The supplements dated May 17 and August 29, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 25, 2006 (71 FR 23957).

The proposed change would revise CNS TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow a one-time extension of no more than 5 years for the Type A, ILRT interval. This revision is a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests as defined in Nuclear Energy Institute (NEI) document, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," pursuant to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), Appendix J, Option B. The requested exception is to allow the ILRT to be performed within 15 years from the last ILRT, last performed on December 7, 1998.

2.0 REGULATORY EVALUATION

Part 50 of 10 CFR, Appendix J, was revised, effective October 26, 1995, to allow licensees to perform containment leakage testing in accordance with the requirements of Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." License Amendment No. 180 to the CNS Operating License, dated March 3, 2000, permitted the implementation of 10 CFR Part 50, Appendix J, Option B. The amendment added TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program" that required Type A, B, and C testing to be performed in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," which specifies a method acceptable to the NRC staff for complying with Option B by approving the use of NEI 94-01 and American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-1994, subject to several regulatory positions in the guide. NEI 94-01 allows an extended interval of 10 years, based upon two consecutive successful tests. Part 50 of 10 CFR, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The two most recent Type A tests were performed satisfactorily at CNS in December 1995 and December 1998. With the approval of the requested extension of the ILRT interval, the next Type A Containment ILRT will be performed no later than December 7, 2013. As described in the January 30, 2006, submittal, the extended testing interval will not affect the existing Appendix J, Type B and Type C testing programs, any ASME Code requirements, or required alternate testing activities of relief requests already approved.

The licensee is requesting a change to TS Section 5.5.12, which would add an exception from the guidelines of RG 1.163 and NEI 94-01, Revision 0, regarding the Type A test interval. Specifically, the exception states that the first Type A test performed after the December 7, 1998; the subsequent Type A test shall be performed by December 7, 2013.

The local leakage rate tests (Type B and Type C tests), including their schedules, are not affected by this request.

3.0 PROPOSED CHANGE

CNS's TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," currently states:

> A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR [Part] 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. Exemption from Appendix J to 10 CFR Part 50 to allow reverse direction local leak rate testing of four containment isolation valves at Cooper Nuclear Station (TAC NO. M89769) (July 22, 1994).

2. Exemption from Appendix J to 10 CFR Part 50 to allow MSIV testing at 29 psig and expansion bellows testing at 5 psig [pounds per square inch gauge] between the piles (Sept. 16, 1977).

The proposed change would add a third exception to TS 5.5.12 to specify the date of the next required Type A test. Specifically, the added exception would state:

3. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J", Section 9.2.3: The first Type A test performed after the December 7, 1998 Type A test shall be performed no later than December 7, 2013.

4.0 TECHNICAL EVALUATION

CNS is a General Electric (GE), Model 4 Boiling-Water Reactor (BWR-4) with a Mark I Primary Containment Pressure Suppression System. This system consists of the drywell, which houses the reactor vessel and reactor coolant recirculation loops, the pressure suppression chamber, which stores a large volume of water (known as the suppression pool), the connecting vent system between the drywell and pressure suppression chamber, isolation valves, vacuum relief system, and containment cooling systems. The drywell is a low-leakage steel pressure vessel designed to confine the reactor coolant that would be released during a postulated pipe rupture, and prevent the gross release of radioactive materials to the environment. The suppression chamber is a steel pressure vessel, toroidal in shape, designed to hold a large volume of water for use as a heat sink for any postulated transient or accident conditions in which the normal heat sink is unavailable. The suppression chamber is located below, and completely encircling the drywell. The drywell and the suppression chamber were designed and constructed to the requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The interior of the drywell and the suppression chamber were coated with a coating, which has been shown to satisfactorily withstand the temperatures and pressures of the steam environment postulated during a design-basis loss-of-coolant accident.

Plant Testing and Inspection Programs

The licensee stated in its amendment request that various inspections and tests are routinely performed in addition to periodic Type A testing, to assure primary containment integrity. These include Type B and C testing performed in accordance with Appendix J, Option B; inspection activities performed as part of the plant Inservice Inspection (ISI/IWE) program, which includes inspection of drywell and suppression chamber surfaces and structural elements; and containment isolation valve inservice testing. The results of these tests and inspections provide a high degree of assurance of continued primary containment integrity.

The CNS Appendix J, Type B and Type C testing program is described in Engineering Procedure 3.40, "Primary Containment Leakage Rate Testing Program," (PCLRT) and Surveillance Procedure 6.PC.501, "Primary Containment Local Leak Rate Tests." These procedures require testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the PCLRT Program as required by 10 CFR Part 50, Appendix J, Option B and RG 1.163. The Type B and C test program provides a means to detect or measure leakage across pressure containing or leakage limiting barriers of the primary reactor containment. The Type B and C test program provides a means to protect the health and safety of plant personnel and the public by maintaining the leakage from these components below appropriate limits. This amendment request does not affect the requirements of Type B or Type C tests scope, performance, or scheduling.

Part 50 of 10 CFR , Appendix J, Option B, Section III.A states: "A general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be conducted prior to each test, and at a periodic interval between tests based on the performance of the containment system." The licensee stated that since the interval for the Type A test has been previously extended to 10 years, this inspection is also conducted during two other refueling outages before the next Type A test in order to allow for early discovery of structural deterioration.

In the *Federal Register* dated August 8, 1996 (Vol. 61, No. 154; pages 41303-41312), the NRC revised 10 CFR 50.55a, "Codes and Standards," to incorporate by reference the 1992 Edition and Addenda of Subsections IWE and IWL of Section XI of the ASME Code. Subsections IWE and IWL specify the requirements for inservice inspection (ISI) of Class CC (concrete containments), and Class MC (metal containments) of light-water-cooled power plants. The amended rule became effective on September 9, 1996, and requires that licensees incorporate the new requirements into their ISI programs and complete the first containment inspection within 5 years, i.e., no later than September 9, 2001. Any repair or replacement activity to be performed on containments after the effective date of September 9, 1996, must be carried out in accordance with the respective requirements of Subsections IWE and IWL. CNS completed the first IWE-required containment inspection in November 1998, and the second during the Cycle 21 refueling outage in spring 2003. The third inspection is scheduled to be conducted during the Cycle 24 refueling outage in spring 2008. The three inspection periods during the first inspection interval are: First Period from September 9, 1996, to September 8, 2001; Second Period from September 9, 2001, to January 8, 2005; and Third Period from January 9, 2005, to May 8, 2008.

The IWE containment inspection requirements are implemented at CNS through the "First Ten-Year Interval Containment Inspection Program for CNS." The general visual examination requirements specified in this containment inspection program satisfies the visual examination requirements specified in Option B. These are conducted at CNS in accordance with its Engineering Procedure 3.28.1.4, "General Visual Inspection of Containment Surfaces," or CNS-approved vendor procedures with similar requirements. A general visual inspection conducted by this procedure determines the structural integrity of the containment. The procedure includes inspections of the interior torus walls above the water level, the exterior of the torus above and below the water level, vent pipes and downcomers above the water level, eight drywell penetration interior surfaces, eight drywell-to-torus vent opening shield plates, interior and exterior of drywell head, and verification that the vent header support pins are properly secured in place.

Indications or relevant conditions reported during visual inspection of ASME Code Class MC components are evaluated by a certified Level II or Level III Inspector. The evaluation, when possible, includes a review of any applicable pre-service and ISI records, and fabrication records. The evaluations are reviewed by the ISI Engineer and any conditions exceeding the allowable standards of the ASME Code, Section XI, are entered into the Corrective Action Program. Containment inspections will continue to be performed during the proposed 5-year extension of the interval (December 2008 through December 2013) in accordance with the CNS PCLRT and ISI/IWE program as applicable.

During its review, the U.S. Nuclear Regulatory Commission (NRC) staff found that additional information was needed to complete its review. In a letter dated August 15, 2006, the NRC staff asked the licensee the following: "IWE-1240 requires licensees to identify the containment surface areas requiring augmented examinations. Provide the locations of the steel containment surfaces that have been identified as requiring augmented examination and a summary of the findings of the examinations performed." In its response dated August 29, 2006, the licensee stated that the CNS Containment Inspection Program identifies the wetted surface of the suppression pool (torus) as the only area requiring augmented examination. It is visually inspected in accordance with ASME Code, Section XI, 1992 Edition, 1992 Addenda. The examinations were performed during refueling outages in February 2001, and January 2005, where pitting was observed, primarily around the piping penetrations. Areas of pitting were recoated with an underwater epoxy coating to minimize subsequent pitting corrosion. Additional examinations of the remaining bays of the torus were performed in accordance with 10 CFR 50.55a(b)(2)(ix)(D) and the results were acceptable.

Based on the above discussion, the NRC staff finds that CNS has adequately identified the areas where augmented inspection is required, and where unacceptable results were found corrective measures were adequately taken.

<u>Approved Alternatives to Subsection IWE Requirements</u>

The following two relief requests, associated with the PCLRT Program, were approved by the NRC staff for application at CNS.

1. Relief Request No. RC-02

    IWE-2500, Table IWE-2500-1 requires seals and gaskets on airlocks, hatches, and other devices to be visually examined (VT-3) once each interval to assure containment leak-tight integrity.

    As an alternative, the leak tightness of seals and gaskets will be tested in accordance with 10 CFR [Part] 50, Appendix J, Type B testing. No additional alternatives to the visual examination, VT-3, of the seals and gaskets will be performed.

2. Relief Request No. RC-06, Revision 1

    IWE-5240 invokes the requirements of IWA-5240 as applicable following repair, replacement, or modification. IWA-5240 requires a VT-2 visual examination in conjunction with the pressure test.

    Table IWE-2500-1, Category E-P, identifies the examination method as 10 CFR Part 50 Appendix J, and does not specifically require a VT-2 visual examination. Part 50 of 10 CFR, Appendix J, provides the requirements for testing, as well as the acceptable leakage criteria. These tests are performed by qualified test personnel using calibrated equipment to determine the leakage rate. In addition, 10 CFR 50.55a(b)(2)(x)(E) requires a general visual examination of the containment each inspection period. This inspection would identify any structural degradation that may contribute to leakage. A VT-2 visual examination will not provide additional assurance of safety beyond that of current Appendix J requirements.

Approval of the requested one-time extension of the ILRT interval will not affect the performance of the required alternate testing activities described in these relief requests.

During its review, the NRC staff found that additional information was needed to complete its review. In a letter dated August 15, 2006, the NRC staff asked the licensee the following: "For the examination of penetration seals and gaskets, and examination and testing of bolted connections associated with the primary containment pressure boundary, the licensee requested relief from the requirements of the Code. As an alternative, the licensee proposed to examine the above items during the leak-rate testing of the primary containment. Option B of Appendix J for Type B and Type C testing (per NEI 94-01 and RG 1.163), and the ILRT extension requested in this amendment for Type A testing, provide flexibility in the scheduling of these inspections. Discuss your schedule for examination and testing of seals, gaskets, and bolted connections that provide assurance regarding the integrity of the containment pressure boundary." In its response dated August 29, 2006, the licensee stated that the leak-tightness of the seals and gaskets are tested through local leakage rate tests (LLRT) pursuant to 10 CFR Part 50, Appendix J. For those penetrations that are routinely disassembled during refueling outages, a LLRT is required upon final assembly and prior to start-up (this includes seals, gaskets, and bolting, associated with containment equipment hatches, airlocks, and drywell head). For seals and gaskets that are not routinely disassembled, the frequency of LLRT is based on past performance and may be on a frequency up to once every 10 years. CNS performs LLRTs at intervals of no greater than once every 60 months. Containment bolted connections that are disassembled during the course of normally scheduled work activities receive a VT-1 visual examination. When conditions exceeding allowable standards are found, those are entered in a Corrective Action Program for resolution.

Based on the above discussion, the NRC staff finds that CNS has adequate programs to assure the integrity of the containment pressure boundary through inspection and testing of penetration seals, gaskets, and bolted connections.

Inservice Testing Program

CNS tests the containment isolation valves (CIVs) in accordance with the CNS Third Interval Inservice Testing Program, as required by 10 CFR 50.55a. This testing ensures operational readiness of the CIVs, such that they will perform their containment isolation function when called upon.

The CNS Painting and Coatings Program controls surface preparation, coating application, and coating inspection requirements for drywell and suppression chamber surfaces. CNS Procedures 3.41, "Painting and Coatings Program," and 7.0.15.1, "Service Level I Coating," provide the necessary controls for maintenance of safety-related coatings applied to the drywell and suppression chamber surfaces to ensure the coating protects these surfaces from corrosion, erosion, and mechanical damage or wear.

The licensee also stated in its license amendment that the CNS primary containment is inerted during power operation with nitrogen to maintain oxygen concentration within TS limits.

Maintaining the containment pressurized at power, and frequently monitoring the pressure, assures that gross containment leakage that may develop during power operation will be detected.

Information Notice 92-20

NRC Information Notice (IN) 92-20, "Inadequate Local Leak Rate Testing (LLRT)," was issued to alert licensees to problems with local leak rate testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem.

At CNS, the main steam and feedwater testable penetrations consist of a double-layered metal bellows. The inboard high-pressure side of the bellows is subject to drywell pressure. Therefore, the bellows are tested in its entirety during the performance of the Type A test. The bellows are tested for the integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage, and possible ruptures of the bellows.

CNS has performed numerous tests related to IN 92-20 since startup. The following is a summary of those tests:

> Plant startup to 1992: Eight sets of LLRTs conducted between the plies and five ILRTs conducted show the bellows to be acceptably leak-tight.
>
> 1993: One set of LLRTs between the plies and a "pass/fail" helium test applied from the direction of pressure in containment.
>
> 1994 to 1997: Two sets of LLRTs between the plies and an ILRT in 1995.
>
> 1998 to present: Performance of an ILRT in 1998 and performance of LLRTs between the plies in 1998, 2000, and 2005.

Successful LLRTs of the bellows, successful ILRTs, and an acceptable helium test demonstrates adequate testing and leak-tight integrity of the bellows at CNS. PSA-ES067, Risk Impact Assessment of Extending Containment Type A Test Interval for Cooper Nuclear Power Station considers the potential failure of containment bellows assemblies.

Based on the information provided by the licensee related to the inspection of the bellows, the NRC staff finds the bellows have been periodically tested (every 2 to 5 years), with acceptable results. It is the NRC staff's understanding that the licensee will continue testing the bellows periodically to demonstrate the leak-tight integrity of the bellows at CNS.

As additional information, the licensee stated that the issue of a through-wall torus shell crack discovered at the James A. Fitzpatrick Nuclear Power Plant (JAF) on June 27, 2005, was reviewed for applicability to CNS. Nebraska Public Power District (NPPD) concluded that the condition at JAF is not applicable to CNS due to differences in the internal and external structures, which include: (1) spargers are not installed on the high-pressure coolant injection and reactor core isolation coolant exhaust lines in the JAF torus; the spargers are installed on these lines in the CNS Torus; and (2) the supports for the exhaust lines at CNS are more robust than those at JAF.

Information Notice 88-82

NRC IN 88-82 related to degradation of torus coatings, recommends underwater inspection and repair. The licensee stated that NPPD began conducting inspections in response to IN 88-82 in 1989. In 1996 the inspections were incorporated into the ISI/IWE Program. IWE requires the performance of VT-3 visual examination of the interior submerged surfaces of the torus. The required examinations and their schedule are contained in the first 10-Year Interval Containment Inspection Program. IWE requirements include an inspection of the submerged portions of the torus. This is conducted at CNS by Engineering Procedure 3.28.1.5, "Visual Inspection of Containment Surfaces, VT-1/3," or CNS-approved vendor procedures with similar requirements. The inspections are looking for conditions such as cracks, wear, corrosion, erosion, or physical damage on the surface of the part or component.

Indications or relevant conditions reported during visual inspections of ASME Code, Class MC, components are evaluated by a certified Level II or Level III Inspector. The evaluation, when possible, includes a review of any applicable preservice and inservice inspection records, and fabrication records, and any condition exceeding the allowable standards of ASME Code, Section XI, are entered into the Corrective Action Program.

The licensee stated that the coatings on the submerged surfaces of the torus at CNS have degraded since original construction. The discrete areas of degraded coating have been repaired by the application of an underwater epoxy coating as necessary to meet the requirements and recommendations of IN 88-82. NPPD will continue to inspect the torus interior surfaces, including the submerged surfaces of the pressure boundary, and repair areas of degraded coating to ensure that these surfaces are adequately maintained throughout the life of the plant. These inspections are part of the ISI/IWE Program, "Successive Examinations." The first of these inspections was performed during the Cycle 22 refueling outage, conducted during January and February 2005. The next required IWE Code, "Successive Examination," is required during the Cycle 24 refueling outage, currently scheduled for spring 2008. CNS is continuing to monitor the industry and evaluate options for torus recoating in the future.

During its review, the NRC staff found that additional information was needed to complete its review. In a letter dated August 15, 2006, the NRC staff asked the licensee the following: "Inspections of some steel containment structures have identified degradation of uninspectable areas (the gap between the shield wall and drywell). These degradations cannot be found by visual (i.e., VT-1 or VT-3) examinations unless they are through the thickness of the shell or when 100 percent of the uninspectable surfaces are periodically examined by ultrasonic testing. Discuss what approach is used at CNS to identify degradation of uninspectable areas of the containment." In its August 29, 2006, response, the licensee stated that based on CNS experience and that of its peers, the drywell shell would be susceptible to degradation only if (1) there were significant chronic leakage, and (2) the sand cushion drains were plugged. One area of the drywell identified by the licensee as being inaccessible for examination is the air gap between the steel shell and the concrete shield wall. The only source of water into the air gap is the reactor cavity when flooded during refueling outages. Sand cushion drain lines route any leakage into the air gap away from the drywell shell. These drain lines are monitored for leakage whenever the reactor cavity is flooded. CNS has not experienced any leakage of the refueling bellows that could introduce water to the exterior of the containment. The sand cushion drain lines have been previously confirmed not to be plugged. The licensee also stated that the containment is continuously pressurized by the containment inerting system, and is periodically monitored, which provides assurance that the gross containment leakage that may develop during power operation as a result of significant degradation of the containment would be detected.

Based on the above discussion, the NRC staff finds that the licensee has adequately identified the areas that are inaccessible for examination and provided a good explanation of how they monitor for leakage in the drywell area. The NRC staff concludes that the licensee adequately monitors the drywell for possible degradation.

The NRC staff also asked the licensee the following question: "In response to GL 87-05, CNS committed to implement new surveillance procedures for inspection of sand cushion drains during refueling outages. Discuss the results obtained from the inspection." In its August 29, 2006, response, the licensee stated that procedures to monitor the sand cushion drain lines for leakage were implemented at CNS, the first during April and May of 1989. The sand cushion drain lines are monitored for water leakage daily when the reactor cavity is flooded during outages and no leakage has been identified. The NRC staff finds this response acceptable.

On the basis of its review of the information provided by the licensee in its amendment and request for additional information responses, the NRC staff finds that CNS's IWE and IWL programs will maintain the structural integrity of the CNS containment.

<u>Risk Assessment</u>

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the January 30, 2006, application for license amendment. Additional analysis and information was provided by the licensee in its letter dated May 17, 2006. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) Topical Report (TR)-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, industry undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for CNS early in the plant's life required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak, that was detectable only by a Type A test, goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the pressurized-water reactor and BWR representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person-rem per year in increased public dose.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem per year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak was present. Since the Option B rulemaking was completed in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per year and increases in large early release frequency (LERF) less than 10^{-7} per year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original frequency of three tests in a 10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided analyses, as discussed below. The following comparisons of risk are based on a change in test frequency from three tests in 10 years (the test frequency under Appendix J, Option A) to one test in 15 years. This bounds the impact of extending the test frequency from one test in 10 years to one test in 15 years. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. Given the change from a three in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk is estimated to be less than 0.01 person-rem per year. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original three in 10 years to one in 15 years is estimated to be about

4.1 x 10^{-7} per year based on consideration of internal events and external events (i.e., fire and seismic events.) There is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in ASME Code, Section XI, Subsections IWE/IWL). Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is estimated to be less than 1 x 10^{-8} per year.

When the calculated increase in LERF is in the range of 10^{-7} per year to 10^{-6} per year, applications are considered if the total LERF is less than 10^{-5} per year. The licensee estimates that the total LERF for internal and external events, including the requested change, is about 4.7 x 10^{-6} per year, which meets the total LERF criteria. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of less than 0.1 percentage point for the cumulative change of going from a test frequency of three in 10 years to one in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy, of RG 1.174, therefore, is acceptable.

5.0 STATE CONSULTATION

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

6.0 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. 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 published April 25, 2006 (71 FR 23957). 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.

7.0 CONCLUSION

On the basis of the above discussion, the NRC staff concludes that the licensee has adequate procedures to examine and monitor the structural integrity of the containment of the CNS. The NRC staff finds the revised TS 5.5.12 acceptable. Therefore, granting of a one-time 5-year extension to the current 10-year test interval for the containment ILRT as proposed by the licensee in TS Section 5.5.12 of the proposed TS amendment request is acceptable. 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.

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