July 12, 2006

Mr. David A. Christian
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION - ISSUANCE OF AMENDMENT RE: SURVEILLANCE INTERVAL EXTENSION (TAC NO. MC9782)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 187 to Facility Operating License (FOL) No. DPR-43 for the Kewaunee Power Station. This amendment revises the FOL in response to your application dated February 6, 2006, as supplemented by letter dated May 5, 2006.

The proposed amendment adds a license condition to extend certain technical specification surveillance intervals on a one-time basis to account for the effects of an extended forced outage in the spring of 2005.

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

Sincerely,

/RA/

David H. Jaffe, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures:
1. Amendment No. 187 to
   License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

July 12, 2006

Mr. David A. Christian
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION - ISSUANCE OF AMENDMENT RE: SURVEILLANCE INTERVAL EXTENSION (TAC NO. MC9782)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 187 to Facility Operating License (FOL) No. DPR-43 for the Kewaunee Power Station. This amendment revises the FOL in response to your application dated February 6, 2006, as supplemented by letter dated May 5, 2006.

The proposed amendment adds a license condition to extend certain technical specification surveillance intervals on a one-time basis to account for the effects of an extended forced outage in the spring of 2005.

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

Sincerely,

/RA/

David H. Jaffe, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures:
1. Amendment No. 187 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

PUBLIC	LPL3-1 R/F	RidsNrrDorlLpl3-1	RidsNrrPMDJaffe
RidsNrrLATHarris	RidsOGCRp	RidsAcrscnwMailCenter	RidsNrrDirsItsb
G. Hill	RidsRgn3MailCenter		RidsNrrDorlDpr

ADAMS ACCESSION NUMBER: ML061640286 PACKAGE: ML061640302 TS:ML061940110

OFFICE	NRR/LPL3-1/PM	NRR/LPL3-1/LA	OGC	NRR/CPTB/BC(A)
NAME	DJaffe	THarris	APH	TLiu
DATE	7/7 /06	7/7/06	6 /29/06	6/29 /06*
OFFICE	NRR/CVB/BC	NRR/EEB/BC	NRR/ICB/BC	NRR/LPL3-1/BC
NAME	RDenning	GWilson	GHowe	LRaghavan
DATE	5 /31 /06*	6/8/06*	5/28 /06*	7/12/06

*Input via memo with minor changes.

OFFICIAL RECORD COPY

cc:

Resident Inspectors Office
U.S. Nuclear Regulatory Commission
N490 Highway 42
Kewaunee, WI 54216-9510

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Suite 210
2443 Warrenville Road
Lisle, IL 60532-4351

David Zellner
Chairman - Town of Carlton
N2164 County B
Kewaunee, WI 54216

Mr. Jeffery Kitsembel
Electric Division
Public Service Commission of Wisconsin
PO Box 7854
Madison, WI 53707-7854

Mr. Michael G. Gaffney
Dominion Energy Kewaunee, Inc.
Kewaunee Power Station
N490 Highway 42
Kewaunee, WI 54216

Mr. Chris L. Funderburk
Director, Nuclear Licensing and
Operations Support
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

Mr. Thomas L. Breene
Dominion Energy Kewaunee, Inc.
Kewaunee Power Station
N490 Highway 42
Kewaunee, WI 54216

Ms. Lillian M. Cuoco, Esq.
Senior Counsel
Dominion Resources Services, Inc.
Millstone Power Station
Building 475, 5th Floor
Rope Ferry Road
Waterford, CT 06385

Plant Manager
Kewaunee Power Station
N490 Highway 42
Kewaunee, WI 54216-9511

DOMINION ENERGY KEWAUNEE, INC.

DOCKET NO. 50-305

KEWAUNEE POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 187
License No. DPR-43

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

   A. The application for amendment by Dominion Energy Kewaunee, Inc. dated February 6, 2006, as supplemented by letter dated May 5, 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 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 license as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and
Technical Specifications

Date of Issuance: July 12, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 187

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Replace the following page of Facility Operating License No. DPR-43 with the attached revised page. The changed area is identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3

Replace the following pages of the Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
4	4
-	4a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 187 TO FACILITY OPERATING LICENSE NO. DPR-43

DOMINION ENERGY KEWAUNEE, INC.

KEWAUNEE POWER STATION

DOCKET NO. 50-305

1.0 INTRODUCTION

By application dated February 6, 2006, as supplemented by letter dated May 5, 2006, Dominion Energy Kewaunee, Inc. (the licensee), requested changes to the Facility Operating License (FOL) for Kewaunee Power Station (Kewaunee) . The supplement dated May 5, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 14, 2006 (71 FR 13172).

The proposed amendment adds a license condition to extend certain technical specification (TS) surveillance intervals on a one-time basis to account for the effects of an extended forced outage in the spring of 2005. Specifically, the following condition would be incorporated into the license:

2.C.(9) Surveillance Test Interval Relaxation

In lieu of the specified frequencies, Dominion Energy Kewaunee, Inc. may complete the surveillance requirements noted in Table 2.C.(9) on page 4a during the fall 2006 refueling outage, but not later than October 7, 2006.

A new Table 2.C.(9) would be added to the FOL to indicate which TS surveillance intervals would be extended, as follows:

ENCLOSURE

Table 2.C.(9)		
Surveillance Requirement	Table Item Number	Title
Table 4.1-1	5	Reactor Coolant Flow - Calibration
Table 4.1-1	6	Pressurizer Water Level - Calibration
Table 4.1-1	7	Pressurizer Pressure - Calibration
Table 4.1-1	11a	Steam Generator Low Level - Calibration
Table 4.1-1	11b	Steam Generator High Level - Calibration
Table 4.1-1	21	Containment Sump Level - Test
Table 4.1-1	30	Fore Bay Water Level - Test
Table 4.1-1	33	PORV Block Valve Position Indicator - Calibration
Table 4.1-1	36	Reactor Coolant System Subcooling Monitor - Calibration and Test
Table 4.1-1	42	Steam Generator Level (Wide Range) - Calibration
Table 4.1-3	4	Containment Isolation Trip - Test
4.4.c.1.b		Shield Building Ventilation System Tests
4.5.a.1		Safety Injection System Tests
4.5.a.2		Containment Vessel Internal Spray System
4.5.a.3		Containment Fancoil Units Tests
4.5.b.2.F		Residual Heal Removal System valve interlocks
4.6.a.2		Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment
4.6.a.3		Diesel Generator Inspection
4.6.a.4		Diesel Generator Load Rejection Test
4.14		Testing And Surveillance Of Shock Suppressors (Snubbers)
4.17.a.2		Control Room Post Accident Recirculation System
6.12.b		System Integrity Program Integrated Leak Tests

The subject surveillances are to be conducted at least every 18 months (each refueling outage or operating cycle). Kewaunee TS 4.0.b provides a maximum allowable surveillance interval extension not to exceed 25 percent of the specified surveillance interval. Therefore, for an 18-month surveillance interval, the maximum allowable surveillance interval is 22.5 months. The maximum surveillance interval that would result from approval of the application, as supplemented, would be 23.9 months.

2.0 REGULATORY EVALUATION

The NRC staff considered the following requirements and guidance in the evaluation of the licensee's application:

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 sets forth the regulatory requirements for the content of the TS. This regulation requires, in part, that the TS contain limiting conditions of operation (LCO). Subsection (c)(2)(ii) of 10 CFR 50.36 gives four criteria to be used in determining whether an LCO is required to be included in the TS for a particular item. The third criterion requires a TS Limiting Condition for Operation for a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Additionally, 10 CFR 50.36(c)(3) states:

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

Part 50 of Appendix A, General Design Criterion (GDC) 19 requires, in part, that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Part 100, Section 100.11 specifies limits (as reference values) for offsite doses following a postulated fission product release. These limits are a whole body dose of 25 rem and a total radiation dose to the thyroid from iodine exposure of 300 rem.

Criterion 38 in the Kewaunee Updated Safety Analysis Report (USAR), "Reliability and Testability of Engineered Safety Features," requires that all engineered safety features (ESF) shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for the proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including ESF, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant. Criterion 38 is comparable to GDC 18. GDC 18, "Inspection and Testing of Electric Power Systems," of Appendix A to 10 CFR Part 50, requires that electric power systems that are important to safety be designed to permit appropriate periodic inspection and testing of such systems for operability and functional performance.

Criterion 39 in the Kewaunee USAR, “Emergency Power for Engineered Safety Features,” requires that alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the ESF. As a minimum, the on site power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system. Criterion 39 is comparable to GDC 17. GDC 17, “Electric Power Systems,” of Appendix A to 10 CFR Part 50, requires that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems and components that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The offsite power system must be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

ISA Standard ANSI/ISA 67.04.01-2000, “Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants” was used as a requirement for the evaluation of the application.

Pursuant to 10 CFR 50.55a(g)(4), American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b), 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable edition of Section XI of the ASME Code for the Kewaunee fourth 10-year inservice inspection ISI interval is the 1998 Edition up to and including the 2000 Addenda.

3.0 EVALUATION

3.1 Instrumentation and Controls

Instrumentation and control systems at Kewaunee provide input to the reactor trip system, ESF actuation and other important safety functions. The NRC staff has reviewed the licensee’s submittal with regard to extension of instrumentation and control surveillance intervals for Kewaunee. The following elements of the proposed TS change relate to instrumentation and control:

- TS Table 4.1-1, Item 5, Reactor Coolant Flow - Calibration
- TS Table 4.1-1, Item 6, Pressurizer Water Level - Calibration
- TS Table 4.1-1, Item 7, Pressurizer Pressure - Calibration
- TS Table 4.1-1, Item 11a, Steam Generator Low Level - Calibration

- TS Table 4.1-1, Item 11b, Steam Generator High Level - Calibration
- TS Table 4.1-1, Item 21, Containment Sump Level - Test
- TS Table 4.1-1, Item 30, Fore Bay Water Level - Test
- TS Table 4.1-1, Item 33, PORV Block Valve Position Indicator - Calibration
- TS Table 4.1-1, Item 36, Reactor Coolant System Subcooling Monitor - Calibration and Test
- TS Table 4.1-1, Item 42, Steam Generator Level (Wide Range) - Calibration
- TS 4.5.b.2.F, Residual Heat Removal System Valve Interlocks - Test

To determine the effects of extending these surveillance test intervals, the licensee performed a review of the past surveillances, corrective action program, work orders, and operating experience, as applicable. This review looked for failed surveillance procedure performances and significant performance issues (i.e., instrument found out of tolerance and cannot be returned to within tolerance and/or instrument failures). The licensee provided the results of the review for each of the surveillances in Sections 4.1 through 4.20 of the Attachment 1 of the application. Additionally, a calculation was performed to analyze the effect of drift on the instrument surveillance interval extensions.

The licensee performed calculations to justify the instrument calibration interval extensions from 18 to 24 months based on the procedures of Kewaunee Instrument Loop Uncertainty Calculations. These calculations included an analysis of instrument loop uncertainties using the methodology described in Kewaunee Power Station Nuclear Administrative Directive (NAD) NAD 4.6 “Plant Setpoint Accuracy,” which follows ISA Standard ANSI/ISA 67.04.01-2000.

Using the methodology of NAD 4.6, the licensee performed uncertainty calculations using either a 24-month or 30-month drift period. If sufficient plant-specific information was available, the calculations were performed using plant-specific, 18-month drift data. The calculations used 30-month vendor provided drift information if sufficient plant-specific drift data was not available.

Only one instrument, pressurizer water level, had sufficient plant-specific drift data that allowed its setpoint calculation to be performed using the 18-month drift data. In order to obtain 24-month values, the drift input for this instrument loop was multiplied by a factor of 24/18 months, or 133 percent, and a 24-month instrument uncertainty calculation was performed with this new 24-month drift value.

The licensee then performed a review of the total loop uncertainty of the setpoint calculations (24-month and 30-month) to determine the available margin and to ensure that adequate margin exists. The licensee stated that the extension request requires the surveillance to be performed within 24 months. Thus, the 24-month or 30-month uncertainty calculation bounds the maximum surveillance test interval.

The licensee provided an example that provided information associated with the method used to determine if acceptable margin exists. This example used the pressurizer level, high level reactor trip, instrumentation to explain how surveillance test interval extensions were evaluated. TS 2.3.a.2.A requires that the Reactor Trip limiting safety system settings (LSSS) for high pressurizer water level be less than or equal to 90 percent of full scale. The analytical limit (AL) is 100 percent of full scale for this example.

To determine the nominal trip setpoint (NTSP) (minimum calculated value for actuation of the final setpoint device to initiate a protective action), the licensee calculated the total loop error (TLE) per plant methodology. The TLE calculation consisted of the measured and the unmeasured uncertainties to calculate the change in drift values. In this example, using the 24-month drift value, the uncertainty calculation determined the TLE result as 2.023 percent of span or 2.023 percent, because the span was 100 percent. The NTSP was determined by subtracting the TLE from the AL. In this case, the NTSP equaled 97.977 percent (100 percent - 2.023 percent = 97.977 percent). The difference between the NTSP and the trip setpoint limit of 90 percent of full scale is the margin. In this example AL is 100 percent, the NTSP (AL- TLE) is 97.977 percent, the trip setpoint limit (NTSP-margin) is 90 percent, the allowable value (AV) (trip setpoint limit - margin) is 86.9034 percent, and the actual plant setting (APS) is 85 percent demonstrating significant margin at different levels.

To provide assurance that an LSSS value is not exceeded, the licensee checked for additional margin between the trip setpoint limit and the AV. From the APS the licensee established a plus or minus band equal to the loop drift that determined the AV setting.

The licensee stated that it ensured that there was adequate margin between the trip setpoint limit and the NTSP, and between the AV and the trip setpoint limit. As can be seen from the example provided, adequate margin exists between the NTSP and the trip setpoint limit, and between the AV and the trip setpoint limit.

Additionally, Kewaunee established a band around the APS that is typically equal to plus or minus the instrument accuracy, called calibration tolerance. The licensee stated in the application that if the as-found reading exceeded this band, the instrument was reset such that the as-left value was within the calibration tolerance.

In response to the NRC staff's request for additional information, by letter dated May 5, 2006, the licensee provided AL, TLE, AL to trip setpoint margin, trip setpoint to APS Margin, AV, APS, loop drift, and calibration tolerance for TS Table 4.1-1, Items 5, 7, 11a, 11b, 36, and 42.

3.1.1 TS Table 4.1-1, Item 5, "Reactor Coolant Flow - Calibration"

The licensee stated that the reactor coolant flow instrumentation calibration is performed using Kewaunee surveillance procedure (SP) SP-36-014C, "Reactor Coolant Flow Transmitters Linearity and Hysteresis Test." In the fall of 2005, the licensee performed a review of the previous three performances of SP-36-014C and reviews of work orders, corrective action program issues, and operational experience over the last 54 months regarding the performance of these instruments. These reviews did not identify any failed surveillance procedure performances or significant instrument performance issues.

The licensee also performed an instrument loop uncertainty calculation for the reactor coolant flow transmitters. The calculation used the vendor specified drift of 0.2 percent over 30 months. The licensee stated that the calculation showed satisfactory margin between the actual plant setting plus loop drift, and the trip setpoint limit over the 30-month period.

Therefore, based on the above, the NRC staff concludes that it is acceptable to extend the calibration interval of the reactor coolant flow instrument channel described in TS Table 4.1-1, item 5, to the end of the fall 2006 refueling outage, but not later than October 7, 2006.

3.1.2 TS Table 4.1-1, Item 6, "Pressurizer Water Level - Calibration"

The licensee stated in the application that the pressurizer level instrumentation calibration is performed using Kewaunee SP-36-017A, "Pressurizer Level Transmitter Calibration." In the fall of 2005, the licensee performed a review of the previous three performances of SP-36-017A and reviews of work orders, corrective action program issues, and operational experience over the last 54 months regarding the performance of these instruments. These reviews did not identify any failed SP performances or significant instrument performance issues.

The licensee also analyzed the instrument loop uncertainty calculation for pressurizer level transmitters. An iteration of the instrument loop uncertainty calculation was performed using plant drift values extended from 18 months to 24 months. The calculation demonstrated that there was sufficient margin between the actual plant setting, with the 24-month loop drift added, and the trip setpoint limit to extend the calibration interval for the pressurizer level transmitters from the current maximum of 22.5 months to the proposed maximum of 23.8 months.

Therefore, the NRC staff concludes that based on the above, it is acceptable to extend the calibration interval of the pressurizer water level instrumentation in TS Table 4.1-1, Item 6, to the end of the fall 2006 refueling outage, but not later than October 7, 2006.

3.1.3 TS Table 4.1-1, Item 7, "Pressurizer Pressure - Calibration"

The licensee stated the pressurizer pressure calibration is performed using Kewaunee SP-36-020A, "Pressurizer Pressure Transmitters Calibration." In the fall of 2005, the licensee performed a review of the previous three performances of SP-36-020A and reviews of work orders, corrective action program issues, and operational experience over the last 54 months regarding the performance of the instruments.

The licensee's work order reviews identified a problem with one of the four pressurizer pressure transmitters that resulted in the replacement of the transmitter in October of 2004. Since replacement of this instrument, corrective action program reviews did not reveal any significant instrument performance issues.

The licensee also performed instrument loop uncertainty calculations regarding the pressurizer pressure transmitters. These calculations were performed using vendor specified drift of 0.2 percent over 30 months. The calculations showed margin remained between the actual plant setting with loop drift added, and the trip setpoint limit and would therefore envelop an extension of calibration interval from the current maximum of 22.5 months to the proposed maximum of 23.8 months.

Therefore, based on the above, the NRC staff concludes that it is acceptable to extend the surveillance test interval for the calibration of the pressurizer pressure transmitter in TS Table 4.1-1, Item 7, to the end of the fall 2006 refueling outage, but not later than October 7, 2006.

3.1.4 TS Table 4.1-1, Item 11a, "Steam Generator Low Level - Calibration"
TS Table 4.1-1, Item 11b, "Steam Generator High Level - Calibration"
TS Table 4.1-1, Item 42, "Steam Generator Level (Wide Range) - Calibration"

The licensee stated that steam generator low-level and high-level calibration is performed using Kewaunee SP-05A-028A, "Steam Generator Level Transmitters Calibration." The licensee also stated in the application that the steam generator water level (wide range) calibration is performed using Kewaunee SP-05A-300, "Steam Generator A Wide Range Level Transmitters Calibration" and SP-05A-301, "Steam Generator B Wide Range Level Transmitters Calibration." In the fall of 2005, the licensee performed a review of the previous three performances of SP-05A-028A and the previous three performances of SP-05A-300/301, and reviews of work orders, corrective action program issues, and operational experience over the last 54 months regarding the performance of the instruments that are calibrated per SP-05A-028A, SP-050A-300, and SP-05A-301. These reviews did not identify any failed SP performances.

The operational experience reviews identified two Rosemount brand steam generator level transmitter problems/failures. One of the problems/failures was associated with transmitters that are used at Kewaunee on equipment associated with the turbine, turbine generator, and feedwater system. However, these problems/failures are not applicable to the SP extension discussion. The other was associated with vibration on the instrument tubing causing electrical noise. The other reviews did not identify any significant performance issues.

The licensee also reviewed the instrument loop uncertainty calculations regarding the steam generator narrow range level transmitters. These calculations were performed using vendor specified drift of 0.2 percent over 30 months. The calculation showed margin remained between the actual plant setting plus the loop drift, and the trip setpoint limit over the 30-month period. The drift calculations would therefore envelop an extension of calibration interval of the current maximum of 22.5 months to the proposed maximum of 23.7 months.

Therefore, based on the above, the NRC staff concludes that it is acceptable to extend the surveillance test interval of the steam generator instrumentation in Table 4.1-1, Items 11a, 11b, and 42 to the end of the fall 2006 refueling outage, but not later than October 7, 2006.

3.1.5 TS Table 4.1-1, Item 21, "Containment Sump Level - Test"

The licensee stated that the containment sump level test is performed using Kewaunee SP-30-052, "Containment Sump A and Reactor Cavity Sump C Level Test." In the fall of 2005, the licensee performed a review of the previous three performances of SP-30-052 and reviews of work orders, corrective action program issues, and operational experience over the last 54 months regarding the past performance of the instruments that are tested per SP-30-052. These reviews did not identify any failed SP performances or significant instrument performance issues.

The licensee also reviewed the drift data for the associated level switches from the three previous performances of SP-30-052. The greatest 18-month drift values (difference between the as-left values and as-found values of the next surveillance performance) of the level switch setpoints were extrapolated to 24 months. These greatest 24-month drift values were then added to the latest 18-month as-left values to get the projected worst-case 24-month level switch setpoint as-found values. The projected worst-case 24-month as-found values for the level switch setpoints were all within the acceptable tolerance band of the surveillance procedure.

Therefore, based on the above, the NRC staff concludes that it is acceptable to extend TS Table 4.1-1, Item 21, “Containment Sump Level Test,” to the end of the fall 2006 refueling outage but not later than October 7, 2006.

3.1.6 TS Table 4.1-1, Item 30, “Fore Bay Water Level - Test”

The licensee stated in the application that the forebay water level test is performed using Kewaunee SP-04-134, “Forebay Area Water Level Logic Test.” In the fall of 2005, the licensee performed a review of the previous three performances of SP-04-134 and reviews of work orders, corrective action program issues, and operational experience over the last 54 months regarding the performance of the instrument channels. These reviews did not identify any significant instrument performance issues or failed SP performances.

Instrument drift is addressed by the associated calibration procedure, SP-04-135, “Forebay Area Water Level Instruments Calibration.” SP-04-135 is being maintained at its 18-month surveillance test interval and is not part of this extension request since it is done on-line.

Therefore, based on the above, the NRC staff concludes that extension of surveillance test interval of the forebay water level test in TS Table 4.1-1, Item 30, to the end of the fall 2006 refueling outage, but not later than October 7, 2006, is acceptable.

3.1.7 TS Table 4.1-1, Item 33, “PORV [Power-Operated Relief Valve] Block Valve Position Indicator” - Calibration

The licensee stated pressurizer PORV block valve position indicator calibration is performed using Kewaunee SP-36-302A, “RC-PR-1A Pressurizer PORV Block Valve Position Indication Verification,” and SP-36-302B, “RC-PR-1B Pressurizer PORV Block Valve Position Indication Verification.” The intent of SP-36-302A and B is to reposition the PORV block valves, open and closed, and ensure the position indicator in the control room properly indicates the valves’ position. The completed SP-36-302A and SP-36-302B were reviewed to determine if issues had been uncovered because of the testing. Since 1993, the surveillances for the PORV block valves position indicators had been performed eleven times on each valve. Each time the indicators had performed satisfactorily and no problems were documented.

Therefore, based on the above, the NRC staff concludes that the position indicator has performed properly since 1993 (12 years), and extension of the surveillance test to the end of the fall 2006 refueling outage, but not later than October 7, 2006, is acceptable.

3.1.8 TS Table 4.1-1, Item 36, "Reactor Coolant System Subcooling Monitor - Calibration and Test"

The licensee stated in the application that reactor coolant system (RCS) subcooling margin monitor calibration and testing is performed using Kewaunee SP-36-162 and SP-36-163, "Reactor Coolant System Hot Leg Pressure Transmitter PT-419/PT-420 Calibrations." In the fall of 2005, the licensee performed a review in the application that, of the previous three performances of SP-36-162/163 and reviews of work orders, corrective action program issues, and operational experience over the last 54 months regarding the performance of the instruments. These reviews did not identify any failed SP performances or significant instrument performance issues.

The licensee also reviewed instrument loop uncertainty calculations for the RCS hot-leg pressure transmitters. These calculations were performed using vendor (Rosemount) specified drift of 0.2 percent over 30 months. For the subcooling margin monitor, the calculation determined the plus and minus total loop error under normal and post-accident conditions. These TLE values are inputs to determine the indicated values that operating procedures use as decision points for procedure actions. The current calculations use the drift value over the 30-month period, and therefore, envelop an extension of surveillance test interval from the current maximum of 22.5 months to the proposed maximum of 23.7 months.

Therefore, based on the above, the NRC staff concludes that it is acceptable to extend the surveillance test interval of the RCS subcooling margin monitor instrumentation in TS Table 4.1-1, Item 36, to the end of the fall 2006 refueling outage but not later than October 7, 2006.

3.1.9 TS 4.5.b.2.F, "Residual Heat Removal (RHR) System Valve Interlocks"

The licensee stated in the application that the RHR system valve interlock test is performed using Kewaunee SP-34-145D, "Residual Heat Removal Valve RHR-11 Reactor Coolant System Interlock Test." A review performed in the fall of 2005 of the previous three performances of SP-34-145D and reviews of work orders, corrective action program issues, and operational experience over the last 54 months were conducted regarding performance of the instrumentation that is tested per SP-34-145D. These reviews did not identify any failed SP performances or significant instrument performance associated issues.

The RHR system valve interlocks utilize Rosemount pressure transmitters. Instrument drift regarding the Rosemount pressure transmitter is addressed in SP-36-163, "Reactor Coolant System Hot Leg Pressure Transmitter PT-420 Calibration," and the discussion associated with TS Table 4.1-1, Item 36, Section 3.2.8 of this safety evaluation.

Instrument drift regarding the remaining instruments within the associated instrumentation loop is addressed in SP-36-198, "Reactor Coolant System Hot Leg Pressure Loop 420 Calibration." This procedure is being performed within its 18-month surveillance test interval.

Therefore, based on the above, the NRC staff concludes that it is acceptable to extend the surveillance test interval for the RHR system valve interlocks in TS 4.5.b.2.F to the end of the fall 2006 refueling outage, but not later than October 7, 2006.

3.2 Emergency Power System

The emergency power systems at Kewaunee provide electrical power to safety-related systems in the event of loss of offsite power. The NRC staff has reviewed the application with regard to extension of emergency power system surveillance intervals for Kewaunee. The following elements of the proposed TS change relate to the emergency power system:

- TS 4.6.a.2 - "Automatic Start of each Diesel Generator (DG), Load Shedding, and Restoration to Operation of particular Vital Equipment"
- TS 4.6.a.3 - "Diesel Generator Inspection"
- TS 4.6.a.4 - "Diesel Generator Load Rejection Test"

The licensee submitted a supplemental response dated May 5, 2006, in response to NRC staff's request for additional information. The NRC staff was interested in whether there were any other surveillances that will be performed during the extended surveillance test interval (STI), that will help ensure that the equipment will function as intended. The licensee stated that normally scheduled preventive maintenance and surveillance testing will continue to be performed on the DGs until the refueling outage plant shutdown. The normal maintenance and surveillance testing includes the following tests, monitoring and sampling:

- DG Availability Test (monthly and quarterly)
- Auto Load Sequencer Test (monthly)
- Vibration Monitoring (monthly)
- Sampling of the Jacket Water (monthly)
- Sampling of the Lube Oil (monthly)

The licensee also clarified that during the first week of the upcoming refueling outage the extended surveillances for the DG "B" will be performed. Immediately following completion of the work on DG "B," the DG "A" surveillances will be performed. No additional maintenance is considered necessary during the limited period associated with the extended STI because the normal maintenance will have recently been completed, the extended surveillances will be completed shortly after the refueling outage starts, and the testing/maintenance of the opposite train diesel is not allowed when one diesel is inoperable.

3.2.1 TS 4.6.a.2, "Automatic Start of each Diesel Generator, Load Shedding, and Restoration to Operation of Particular Vital Equipment"

TS 4.6.a.2 requires the automatic starting, load shedding, and restoration to operation of specific vital equipment. In the application, the licensee indicated that the automatic start of each DG, load shedding, and restoration to operation of particular vital equipment, are initiated by a simulated loss of all normal alternating current station service power supplies together with a simulated safety injection signal. This test is normally conducted at each refueling interval to assure that each DG will start and assume required loads to the extent possible within one minute, and operate for $5 minutes while loaded with the emergency loads.

This surveillance test is unique in that five other surveillance tests rely on it to meet their requirements. TS 4.4.c.1.b - Shield Building Ventilation System, TS 4.5.a.1 - Safety Injection System, TS 4.5.a.2 - Containment Vessel Internal Spray System, TS 4.5.a.3 - Containment Fancoil Units, and TS 4.17.a.2 - Control Room Post Accident Recirculation System, require the automatic starting of the respective equipment on initiation of a safety injection signal with subsequent sequential loading of the ESF loads on the DG. This test demonstrates proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, and sequential starting of essential equipment, to the extent possible, as well as the operability of the DG. This test is conducted by simultaneously unblocking the safety injection signal and simulating a loss-of-voltage signal.

The application states that testing of the DGs is performed using SP-33-110, "Diesel Generator Automatic Test." A search of the corrective action program data base has identified 10 issues related to performance of SP-33-110 dating back to 1996. Five of the issues identified were determined by the licensee to be of a minor nature and had no impact on the performance or outcome of the test. Four of the issues were related to problems with the relays and did not cause a failure of the test. The licensee did identify one issue that occurred in 1996, when the DG failed to remain attached to the vital bus. The licensee stated that the output breaker for the DG "A" initially closed, but immediately reopened. The root cause of this event was an inadequate design change installation procedure that was performed during the 1996 refueling outage.

In response to the NRC staff's request for additional information, the licensee submitted supplemental information in a letter dated May 5, 2006. In response to the NRC staff's concern about any plans for pre-test maintenance of the DGs similar to the event in 1996, the licensee stated that the 2006 refueling outage maintenance on the DGs will not begin until the testing required by TS 4.6.a.2 has been completed. The licensee also clarified the corrective actions taken in response to the event when the DG failed to remain attached to the vital bus in 1996. The corrective actions included adjusting the actuating linkage shaft of the DG "A" output breaker in order to satisfactorily complete the surveillance test and a review of the other 17 breakers replaced during the same outage.

The NRC staff has evaluated the licensee's circumstance and the justifications provided. On the basis of the favorable history of the DG during testing, the fact that there will be no pre-test maintenance performed, and the short duration of the proposed extension, the NRC staff concludes that the proposed change to extend the STI for TS 4.6.a.2 would have no significant adverse impact upon the safety of plant operations and is, therefore, acceptable. The tests themselves are not being changed and will be conducted in compliance with NRC regulations. Even with the extended STI, the requirements of 10 CFR 50.36(c)(3) are still being met. The history of the DG testing and the fact that no pre-test maintenance is being performed will continue to assure the necessary quality of systems and components set forth in the requirement as well as those specified in the GDC.

The NRC staff concludes, based upon the above, that the extension of the surveillance interval until the end of the fall 2006 refueling outage, but not later than October 7, 2006, is acceptable.

3.2.2 TS 4.6.a.3, “Diesel Generator Inspection”

TS 4.6.a.3 requires that each diesel generator be inspected at each major refueling outage. In its request, the licensee indicated that TS 3.7, “Auxiliary Electrical Systems,” requires that the reactor not be made critical unless both diesel DGs are operable. To verify that the emergency power sources and equipment are operable, each DG is inspected at each refueling outage. DG inspections are performed at refueling outage intervals in order to maintain the DGs in accordance with the manufacturer’s recommendations.

The licensee reviewed the previous three performances of the DG inspection procedures to identify issues that may have been influenced by the amount of time that had passed since the last performance of the maintenance and post-maintenance testing. The licensee stated that the review did identify procedure deficiencies as well as failures of equipment (replaced during the inspection procedure) to pass initial post-maintenance testing, but that none of these issues could have been prevented by performing the surveillance at a shorter interval.

In response to the NRC staff’s request for additional information, the licensee submitted supplemental information in a letter dated May 5, 2006. In response to the NRC staff’s concern about the procedure deficiencies and equipment failures identified in the performance review and any corrective actions taken for these deficiencies and failures, the licensee identified 15 issues with procedure deficiencies and equipment failures. Three issues were related to procedure deficiencies/incorrect performance of procedure. In each case, the procedure was revised to correct either the deficiency or appropriate corrective actions were taken to address operator performance error. Seven issues were related to equipment failure. In each case, the component(s) had been either worked on or replaced during the procedure. Upon failure, they were replaced and successfully tested. Upon the review of these 15 issues, the NRC staff concludes that none of these issues could have been prevented by performing the surveillance at a shorter interval.

The license also stated that this refueling outage does not include the 6-year or 12-year maintenance of the DGs. The 6-year and 12-year maintenance were last performed during the 2001 and 2004 refueling outages, respectively. The 18-month mechanical maintenance includes various cleaning, lubrication and inspection tasks as well as replacement of filter media. The electrical maintenance of the 18-month maintenance is limited to cleaning and inspection of the equipment in the DG electrical cabinets as well as resistance measurements of select relays within these cabinets.

On the basis of the favorable history of the performance of the DGs during inspection, the limited scope of the inspection and maintenance for this refueling outage, and the short duration of the proposed extension, the NRC staff concludes that the proposed change to extend the STI for TS 4.6.a.3 would have no significant adverse impact upon the safety of plant operations and is, therefore, acceptable. The tests themselves are not being changed and will be conducted in compliance with NRC regulations. Even with the extended STI, the requirements of 10 CFR 50.30(c)(3) are still being met. The limited scope of work to be performed during this outage, the history of the DG inspections and the fact that none of the previous procedure deficiencies or equipment failures could not have been prevented by performing the surveillance at a shorter interval will continue to assure the necessary quality of systems and components set forth in the requirement as well as those specified in the GDC.

The NRC staff concludes, based upon the above, that the extension of the surveillance interval until the end of the fall 2006 refueling outage, but not later than October 7, 2006, is acceptable.

3.2.3 TS 4.6.a.4, "Diesel Generator Load Rejection Test"

TS 4.6.a.4 requires that the DG load rejection test be performed at least once every 18 months in accordance with Institute of Electrical and Electronics Engineers (IEEE) 387-1977, Section 6.4.5. In the application, the licensee indicated that TS 3.7, "Auxiliary Electrical Systems," requires that the reactor not be made critical unless both DGs are operable. To verify that the emergency power sources and equipment are operable, the DG load rejection test is required to be performed in accordance with IEEE 387-1977, Section 6.4.5, at least once every 18 months. The load rejection test demonstrates the capability of rejecting the maximum rated load without over-speeding or attaining voltage that could cause the DG to trip, mechanical damage, or harmful overstresses.

The licensee indicated in the application that they reviewed the previous three performances of the DG elevated load and load reject tests, as well as additional corrective action program issues or work orders initiated during the performance of these tests to identify issues that may have been influenced by the amount of time that had passed since the last performance of the testing. The licensee stated in the application that the review did identify procedure deficiencies, as well as failures of equipment to pass initial post-maintenance testing, but that none of these issues could have been prevented by performing the surveillance at a shorter interval.

In response to the NRC staff's request for additional information, the licensee submitted supplemental information in a letter dated May 5, 2006. In response to the NRC staff's concern about the procedure deficiencies and equipment failures identified in the performance review and any corrective actions taken for these deficiencies and failures, the licensee identified five issues with procedure deficiencies. Three issues were related to identifiable procedure deficiencies. In each case, the procedure was revised to correct the deficiency. In the other two issues, the tests were performed again and found satisfactory. However, the problems could not be repeated during later tests. Upon the review of these five issues, the NRC staff concludes that none of these issues could have been prevented by performing the surveillance at a shorter interval.

The licensee also stated that a review of operating experience specific to the Electro-Motive Division (EMD) DGs was also performed by searching the EMD owners group Internet web site for problems experienced while performing elevated load and load reject tests. Various documents found on the EMD owners group web site indicate that the elevated load test provides assurance that the governor is operating correctly in addition to ensuring the governor's ability to control the engine under elevated load conditions. Similar documents indicate that the purpose of the full load reject test is to verify that the governor is able to respond to large changes in loads, and that the governor controls engine speed to prevent an over-speed trip should the output breaker open during full load operation. The elevated load and load reject tests are typically performed after maintenance or replacement of the governor. The review did not identify any problems that were induced by the frequency of the test, and the licensee states in the application that no maintenance involving the DG governor will be performed before the refueling outage.

On the basis of the favorable history of the performance of the DGs during testing, favorable industry operating experience specific to the EMD DGs, and the short duration of the proposed extension, the NRC staff concludes that the proposed change to extend the STI for TS 4.6.a.4 would have no significant adverse impact upon the safety of plant operations and is, therefore, acceptable. The tests themselves are not being changed and will be conducted in compliance with NRC regulations. Even with the extended STI, the requirements of 10 CFR 50.30(c)(3) are still being met. The industry operating experience of the EMD DGs showed no frequency related dependency, the fact that any maintenance that may involve the DG governor will not be performed until the refueling outage, the history of the DG testing, and since none of the previous procedure deficiencies could not have been prevented by performing the surveillance at a shorter interval will continue to assure the necessary quality of systems and components set forth in the requirement as well as those specified in the GDC.

The NRC staff concludes, based upon the above, that the extension of the surveillance interval until the end of the fall 2006 refueling outage but not later than October 7, 2006, is acceptable.

3.3 Engineered Safety Features and Containment Systems

Engineered Safety Features and Containment Systems provide for control of post-accident conditions (e.g. core cooling) and containment of radioactive releases, respectively. The NRC staff has reviewed the application with regard to extension of ESF and containment system STI for Kewaunee as follows:

- TS Table 4.1-3 (Item 4) verifies that the automatic isolation valves required for containment isolation automatically shut when a containment isolation signal is received.

- TS 4.4.c.1.b verifies automatic initiation of each train of the Shield Building Ventilation System.

- TS 4.5.a.1 verifies that with the RCS pressure # 350 psig and temperature # 350 $^{\circ}$F, a test safety injection signal will be applied to initiate operation of the safety injection system.

- TS 4.5.a.2 verifies that with the containment isolation valves in the supply lines at the containment blocked closed, a minimum of 76 spray nozzles per train are functioning properly by using an air or smoke test.

- TS 4.5.a.3 verifies proper operation of the motor-operated service water outlet valves and the fancoil emergency discharge and associated backdraft dampers.

- TS 4.17.a.2 verifies automatic initiation of the control room post accident recirculation system on a high radiation signal and a safety injection signal.

- TS 6.12.b perform required periodic visual inspection and integrated leak testing in accordance with the system integrity program.

3.3.1 TS Table 4.1-3 (Item 4) "Verification of Automatic Isolation Valves - Containment Isolation"

An isolation actuation system is provided to close those automatically operated containment isolation valves in fluid line penetrations used during normal operation, but not required for ESF functions. A containment isolation actuation signal is initiated by a Safety Injection signal or by a manual initiation. TS Table 4.1-3, (Item 4) is performed to verify that the automatic isolation valves required for containment isolation automatically shut when a containment isolation signal is received. This requirement is performed using Kewaunee SP-56-078, "Containment Isolation Trip Test."

In order to determine if the requested extension could be granted and reasonable assurance of safety maintained, the NRC staff examined the surveillance performance history and the corrective action program database for the "Containment Isolation Trip Test," which was provided to the staff via the licensee's submittal. Dating back to 1976, there were several minor performance issues identified, none of which caused a test failure.

The NRC staff reviewed this information with particular attention on the historical performance and the length of the STI extension. The past performance of this SP has produced acceptable results. The NRC staff concludes that the historical performance (reliability) indicates that failure is not likely during the requested STI extension, and that reasonable assurance of safety will be maintained..

The NRC staff concludes, based upon the above, that the extension of the surveillance interval until the end of the fall 2006 refueling outage, but not later than October 7, 2006, is acceptable.

3.3.2 TS 4.4.c.1.b, "Shield Building Ventilation System Tests"
TS 4.5.a.1, "Safety Injection System Tests"
TS 4.5.a.2, "Containment Vessel Internal Spray System"
TS 4.5.a.3, "Containment Fancoil Units Tests"
TS 4.17.a.2, "Control Room Post Accident Recirculation System"

TS 4.6.a.2 requires the automatic starting, load shedding, and restoration to operation of specific vital equipment. TS 4.4.c.1.b, TS 4.5.a.1, TS 4.5.a.2, TS 4.5.a.3, and TS 4.17.a.2 are specific testing requirements, which demonstrate automatic initiation of specific equipment, which are accomplished during performance of TS 4.6.a.2. These TSs are all performed during the conduct of TS 4.6.a.2, by initiation of a safety injection signal and tripping the ESF 4160 volt bus undervoltage relays. This surveillance tests the automatic start of the DGs, the load shedding, the restoration to operation of particular equipment, and the automatic initiation of the shield building ventilation system, the safety injection system, the containment fan coil units, and the control room post-accident recirculation system by a safety injection signal.

With regard to the vessel internal spray system test (TS 4.5.a.2), this test verifies that a minimum number of spray nozzles per train are functioning and is performed by a separate SP in TS 4.5.a.2.B. As indicated in the application, the spray nozzle surveillance will be performed within its required surveillance test interval, and therefore, surveillance test interval extension is not required for it. The other requirements of the vessel internal spray system test, as detailed in TS 4.5.a.2.A and C, still need to be satisfied which will require a surveillance interval extension.

The NRC staff concludes, based upon the above, and the discussion in Section 3.2.1. here in, that the extension of the surveillance interval until the end of the fall 2006 refueling outage, but not later than October 7, 2006, is acceptable.

3.3.3 TS 6.12.b, "System Integrity Program"

The requirements of TS 6.12.b are intended to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a transient or accident to as low as practical levels. These requirements include provisions for establishing preventive maintenance and periodic visual inspections and integrated leak tests for each system.

The affected SPs associated with TS 6.12.b are:

- SP-34-091, "RHR Hydrostatic Test," which measures external leakage from the RHR.

- SP-23-193, "Containment Spray System Leakage Test," which measures external leakage from the internal containment spray system (ICS).

- SP-33-195, "Safety Injection System Leakage Test," which measures external leakage from the safety injection system (SI).

- SP-23-080, "ICS and SI Valve Leakage Test," which measures leakage from the SI and ICS systems back to the refueling water storage tank (RWST).

- SP-33-325, "Measurement of RCS/RHR Leak-By of RHR-299A and RHR-299B," which measures leakage from the RHR system back to the RWST.

- SP-56A-090, "Containment Local Leak Rate Type B & C Test," which measures leakage from the hydrogen analyzers and the post-accident vent re-routes.

The licensee reviewed these test results for the last five refueling outages and states in the application that all results are stable and well within the acceptance criteria. The licensee additionally performed an analysis to calculate the effect of drift on the instrument surveillance interval extension by extrapolating the as-found values from the previous tests.

The NRC staff reviewed this information with particular attention to the historical performance and the length of the requested extension. The past performance of these SP have produced acceptable results. The NRC staff's concludes that the historical performance (reliability) indicates that failure is not likely in the requested extension period of 42 days.

The NRC staff concludes, based upon the above, that the extension of the surveillance interval until the end of the fall 2006 refueling outage, but not later than October 7, 2006, is acceptable.

3.4 TS 4.14, "Testing and Surveillance of Shock Suppressors"

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads, which might occur during seismic activity or severe plant transients, while allowing normal thermal motion during startup or shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic event or other events initiating dynamic loads. It is therefore required that all snubbers designed to protect the reactor coolant and other safety-related systems or components be operable during reactor operation. The intent of this TS is to prohibit startup or continued operation with defective safety-related snubbers. TS 4.14, "Testing and Surveillance of Shock Suppressors (Snubbers)," states that if the number of snubbers found inoperative during inspection or during the inspection interval is zero, than the next required inspection interval shall be within 18 months ± 25 percent.

There are 99 installed safety-related snubbers at Kewaunee, two large bore snubbers and 97 small-bore snubbers. During the November 2005 forced outage, the two large-bore and 57 small-bore snubbers in containment were inspected in accordance with procedures SP-55-313, "Steam Generator Hydraulic Snubber Testing" and SP-55-180, "Hydraulic Shock Suppressor (Snubber) Testing (QA-1)." Two of the 57 small-bore snubbers inspected were not safety-related snubbers; thus, 55 safety-related snubbers were inspected. One additional small-bore safety-related snubber, RC-H72, in containment could not be inspected due to the environmental conditions present. The remaining 41 small-bore safety-related snubbers in the auxiliary and turbine building are being inspected on-line. While past visual examination of the small-bore safety-related snubbers performed in accordance with procedure SP-55-180 have identified recordable indications, these indications did not result in the unsatisfactory operational readiness of the snubber.

Based on the snubber inspections performed during the November 2005 forced outage and the continued on-line inspections, one small-bore snubber will exceed its maximum surveillance interval by the start of the 2006 refueling outage. Kewaunee therefore, requests an extension to the surveillance test interval for this snubber to allow for the delayed inspection of small-bore safety-related snubber RC-H72. Therefore, the extension of the visual examination from the current maximum of 22.5 months (18 months + 25 percent) to the proposed maximum of 23.7 months, does not pose a significant increase in risk and provides an adequate level of quality and safety with respect to the operational readiness of small-bore safety-related hydraulic snubber RC-H72.

Generic Letter (GL) 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Action," provides alternative visual inspection requirements that licensees may propose in lieu of the existing TS requirements. GL 90-09, Table 4.7-2, "Snubber Visual Inspection Interval," Note 3 states that if the number of unacceptable snubbers is equal to or less than the number in Column A of Table 4.7-2, the next inspection interval may be twice the previous interval but not greater than 48 months (for 24 month refueling outage intervals). In the May 6, 2006, supplement, the licensee stated that during the 2004 refueling outage, three indications were identified, none of which rendered the affected snubber inoperable. The number of unacceptable snubbers was zero. Therefore, based on GL 90-09, Table 4.7-2, Note 3, the next visual inspection interval at Kewaunee could be extended to the twice the previous interval i.e. 36 months (2 x 18 months). The licensee proposed to perform the visual inspection of small bore snubber RC-H72 during the next scheduled refueling outage in fall 2006, in lieu of the TS frequency. This delay will change the TS required visual inspection from 22.5 months to approximately 23 months, which is still less than 36 months and is thus acceptable.

The NRC staff concludes, based upon the above, that the extension of the surveillance interval until the end of the fall 2006 refueling outage, but not later than October 7, 2006, is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (71 FR 13172). Accordingly, this 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 this amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: K. Corp
B. Lee
H. Walker
S. Mazumdar
G. Bedi

Date: July 12, 2006