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Dominion®

February 6, 2006

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No. 05-896
KPS/LIC/GR: R2
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST - 219
ONE-TIME EXTENSION OF SURVEILLANCE REQUIREMENTS

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to Facility Operating License Number DPR-43 for Kewaunee Power Station (Kewaunee). The proposed amendment adds a license condition to extend certain Technical Specification (TS) surveillance test intervals on a one-time basis to account for the effects of an extended forced outage in the spring of 2005. The normal surveillance test intervals for the affected equipment are 18-months and are identified in Kewaunee TS as each refueling cycle, or operating cycle. Using the maximum TS allowed extension of 25 percent of the surveillance interval, each 18-month surveillance requirement must be completed within 22.5 months of its last performance.

Between February and July of 2005, Kewaunee was in a forced outage. In order to fully use the fuel loaded into the core during the 2004 refueling outage and yet maintain plant operation during the high summer electrical load period, the start of the next refueling outage will be delayed from April to September 2006 (5 months). As a consequence, some surveillance requirements will exceed their maximum allowed surveillance intervals prior to the scheduled September 2006 refueling outage, when the plant will be placed in a condition allowing their performance. Therefore, DEK requests an extension to the maximum allowed surveillance test intervals for the affected surveillances of up to a maximum of 23.9 months.

Additionally, DEK requests approval of relief request RR-G-4. Both Kewaunee TS 4.14 and ASME/ANSI OM, part 4, section 2.3.2 provide requirements for the examination of snubbers. Because the ASME OM section and TS 4.14 contain similar requirements, DEK is including this relief request with the license amendment request to avoid the confusion of separate submittals.

DEK requests approval of the proposed amendment and relief request by June 30, 2006 to facilitate scheduling of refueling activities. Once approved, DEK will implement this amendment within 60-days.

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Attachment 1 to this letter contains a description, a safety evaluation, a significant hazards determination and environmental considerations for the proposed changes. Attachment 2 contains the marked-up Operating License pages 4 and 4a. Attachment 3 contains the proposed Operating License, as revised. Attachment 4 contains relief request RR-G-4.

If you have any questions or require additional information, please contact Mr. Gerald Riste at 920-388-8424.

Very truly yours,

Leslie N. Hartz
Vice President-Nuclear Engineering

Attachments:

1. Discussion of Change, Safety Evaluation, Significant Hazards Determination and Environmental Considerations
2. Marked-Up Operating License Pages
3. Proposed Operating License Pages
4. 4th Ten Year Interval June 16, 2004-June 16, 2014 Request for Relief No. RR-G-4

Commitments made in this letter: None

cc: Regional Administrator
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is the Vice President - Engineering of Dominion Energy Kewaunee, Inc. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 6TH day of February, 2006.

My Commission Expires: May 31, 2006.

Vicki L. Hull

Notary Public

(SEAL)

Attachment 1

LICENSE AMENDMENT REQUEST - 219
ONE-TIME EXTENSION OF SURVEILLANCE REQUIREMENTS

DISCUSSION OF CHANGE, SAFETY EVALUATION, SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL CONSIDERATIONS

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

DISCUSSION OF CHANGES

INTRODUCTION

Dominion Energy Kewaunee, Inc. (DEK) is submitting a request to amend Operating License DPR-43 for the Kewaunee Power Station (Kewaunee). This license amendment request (LAR) provides for a one-time extension to specific surveillance requirement test intervals contained in Kewaunee Technical Specifications (TSs).

1.0 DESCRIPTION

The proposed amendment adds a license condition to extend certain TS surveillance test intervals, on a one-time basis, to compensate for the effects of a forced outage in the spring of 2005. The normal surveillance test intervals for the affected instruments or equipment are stated as each refueling cycle, operating cycle, or refueling outage as applicable. Each refueling cycle has been defined in Kewaunee TS as 18 months. Those surveillance test intervals required to be performed each operating cycle or each refueling outage have been reviewed to determine if the 18-month periodicity is applicable and if the surveillance test should be included in the amendment request.

Kewaunee TS 4.0.b provides a maximum allowable surveillance interval extension not to exceed 25% of the specified surveillance interval. Therefore, for an 18-month surveillance test interval (STI), the maximum allowable surveillance interval is 22.5 months. The maximum surveillance test interval that approval of this request would allow is 23.9 months.

2.0 PROPOSED CHANGE

DEK requests that the following License Condition be added to the Operating License DPR-43 for the Kewaunee Power Station.

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.

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 Heat 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

3.0 BACKGROUND

Kewaunee employs a pressurized water reactor Nuclear Steam Supply System furnished by Westinghouse Electric Corporation. The Nuclear Steam Supply System consists of a pressurized water reactor, Reactor Coolant System, and associated auxiliary fluid systems. The Reactor Coolant System is arranged as two-closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to one of the loops.

The reactor protection system (RPS) limits the range of various core and coolant parameters so that the departure from nucleate boiling ratio (DNBR) is not less than the safety limit value during anticipated operating transients. A block diagram of the RPS indicating various reactor trip functions and interlocks is shown in Kewaunee USAR Figure 7.2-3. The engineered safety features (ESF) actuation system (ESFAS) detects plant conditions that require automatic ESF equipment operation and actuates the appropriate ESF equipment when preset limits are reached. These systems, RPS and ESFAS, generally operate through a two-out-of-three or a two-out-of-four logic matrix. The trip logic channels for a typical two-out-of-three and a two-out-of-four trip function are represented in KPS USAR Figure 7.2-6.

In February of 2005, a design deficiency was discovered in the auxiliary feedwater (AFW) system at Kewaunee. This design deficiency resulted in a forced shutdown of the unit from February 20, 2005 until July 2, 2005. The forced outage resulted in a reduction of the operational time available this cycle for burnup of the reactor fuel. To allow full use of the fuel and to allow the plant to remain on-line during the high electrical load period of the summer months, a proposed license condition is requested to extend, on a one-time basis, the surveillance test interval of selected surveillance requirements that cannot be performed on-line.

The following table provides specific information on each affected surveillance requirement including:

1. The TS surveillance number.
2. A description of the protection feature.
3. Current surveillance due date (maximum allowable due date including the 25% extension).
4. Requested length of extension, in days, based on an October 7, 2006, surveillance completion date.
5. Plant condition where associated equipment is no longer required to be operable.
6. Number of days between the current surveillance due date and the scheduled date that the plant will be in a mode or condition where the equipment is not required to be operable.

Table 1					
Column 1	Column 2	Column 3	Column 4	Column 5	Column 6
Surveillance Requirement	Description	Due Date plus 25%	Requested Length of Extension (Days)	Required MODE if inoperable	Days between Columns 3 and Scheduled date for column 5
Table 4.1-1, Item 5	Reactor Coolant Flow - Calibration	9/01/2006	36	Hot Shutdown	1
Table 4.1-1, Item 6	Pressurizer Water Level - Calibration	8/28/2006	40	Hot Shutdown	4
Table 4.1-1, Item 7	Pressurizer Pressure - Calibration	8/30/2006	38	Cold Shutdown	3
Table 4.1-1, Item 11a	Steam Generator Low Level - Calibration	8/31/2006	37	Hot Shutdown	2
Table 4.1-1, Item 11b	Steam Generator High Level - Calibration	8/31/2006	37	Hot Shutdown	2
Table 4.1-1, Item 21	Containment Sump Level - Test	8/28/2006	40	< 2% Power	5
Table 4.1-1, Item 30	Fore Bay Water Level – Test	9/03/2006	34	RCS Tavg < 350°F	0
Table 4.1-1, Item 33	PORV Block Valve Position Indicator – Calibration	8/29/2006	41	Hot Shutdown	4
Table 4.1-1, Item 36	Reactor Coolant System Subcooling Monitor – Calibration and Test	8/31/2006	37	Hot Shutdown	2
Table 4.1-1, Item 42	Steam Generator Level (Wide Range) – Calibration	8/31/2006	37	Hot Shutdown	2
Table 4.1-3, Item 4	Containment Isolation Trip - Test	8/26/2006	42	Cold Shutdown	7
4.4.c.1.b	Shield Building Ventilation System Tests	8/26/2006	42	Hot Shutdown	7

Table 1					
Column 1	Column 2	Column 3	Column 4	Column 5	Column 6
Surveillance Requirement	Description	Due Date plus 25%	Requested Length of Extension (Days)	Required MODE if inoperable	Days between Columns 3 and Scheduled date for column 5
4.5.a.1	Safety Injection System Tests	8/26/2006	42	Cold Shutdown	7
4.5.a.2	Containment Vessel Internal Spray System	8/26/2006	42	Cold Shutdown	7
4.5.a.3	Containment Fancoil Units Tests	8/26/2006	42	Cold Shutdown	7
4.5.b.2.F	Residual Heat Removal System valve interlocks	8/31/2006	37	RCS Tavg < 350°F	2
4.6.a.2	Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment	8/26/2006	42	Hot Standby(A)	7(B)
4.6.a.3	Diesel Generator Inspection	A – 9/04/2006 B – 9/12/2006	A – 33 B – 25	Hot Standby	0(B)
4.6.a.4	Diesel Generator Load Rejection Test	8/26/2006	42	Hot Standby	7(B)
4.14	Testing And Surveillance Of Shock Suppressors (Snubbers)	8/31/2006	37	Hot Shutdown	2
4.17.a.2	Control Room Post Accident Recirculation System	8/26/2006	42	Hot Shutdown	7
6.12.b	System Integrity Program Integrated Leak Tests	8/25/2006	43	Not Applicable	NA

(A) Depending on equipment determined to be inoperable (e.g., diesel generator, safety injection system, internal containment spray) the Mode may vary.
(B) Operability also needed to support decay heat removal function.

Kewaunee TS 4.0.a states, in part;

> "Surveillance requirements shall be met during the operational MODES or other conditions specified for individual LIMITING CONDITION FOR OPERATION (LCO) unless otherwise stated in an individual surveillance requirement."

The 2006 refueling outage is currently scheduled to start on September 2, 2006. The current schedule places the plant in the shutdown condition (Hot Shutdown) at 0200 on September 2, 2006. Additionally, the reactor coolant system (RCS) temperature is scheduled to be < 350°F at approximately 1400 hours on September 2, 2006, and the plant is scheduled to be placed in the Cold Shutdown condition at approximately 2000 hours on September 2, 2006. Note this is Kewaunee's current schedule and as such, these dates are tentative and subject to change based on other factors associated with scheduling an outage.

Although this extension request would expire on October 7, 2006, the majority of the equipment, for which the extension is being sought, will not be required to be operable after approximately 2000 hours on September 2, 2006, see Table 1 Column 3. Although the majority of the equipment will not be required to be operable after September 2, 2006, the analysis associated with this extension request assumes the equipment is required to be operable until October 7, 2006.

The date of October 7, 2006, is requested to ensure an emergency diesel generator (EDG) is available to support decay heat removal via the residual heat removal system during the refueling outage. TS requirement 4.6.a.3 requires the EDGs be inspected during each major refueling outage. The basis for TS 4.6.a.3 states that the inspections are performed at refueling outage intervals in order to maintain the emergency diesel generator (EDG) in accordance with the manufacturer's recommendations. Thus, the October 7, 2006, completion date will allow scheduling flexibility to minimize shutdown risk while ensuring completion of the surveillances during the refueling outage.

In November of 2005, Kewaunee experienced another forced outage caused by service water leakage in the main electrical generator hydrogen cooling system. During this forced shutdown, the plant remained in the Hot Shutdown condition (RCS at approximately 547°F and 2235 psig). A review was performed of the surveillance requirements addressed in this submittal to determine if any could be performed during that outage. This review concluded that the only surveillance that could be performed in the Hot Shutdown condition was TS surveillance TS 4.14, "Testing and Surveillance of Shock Suppressors (Snubbers)." The other surveillances could not be performed in the Hot Shutdown condition.

Due to plant conditions during the November 2005 forced outage, one safety related snubber could not be inspected during the forced outage nor can it be inspected online

and still requires an extension of the surveillance test interval. For details, see section 4.0, item 4.19, "Testing and Surveillance of Shock Suppressors (Snubbers)."

4.0 TECHNICAL ANALYSIS

To determine if extending these surveillance test intervals would be acceptable, a review was performed of the past surveillances, the 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 results of the review are provided for each of the surveillances in Sections 4.1 through 4.20. Additionally, a calculation was performed to analyze the effect of drift on the instrument surveillance interval extensions.

The calculations used to provide justification for instrument calibration interval extensions from 18 to 24 months are based on iterations of Kewaunee Instrument Loop Uncertainty Calculations. These calculations provide an analysis of instrument loop uncertainties as 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, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants."

Using the methodology of NAD 4.6, uncertainty calculations have been performed using either a 24-month or 30-month drift period. The calculations use 30-month vendor provided drift information for the calculation's drift input if sufficient plant-specific drift data is not available. If sufficient plant-specific information is available, the calculations are performed using a plant-specific 18-month drift input.

Only one instrument, pressurizer water level, has sufficient plant-specific drift data that allowed its setpoint calculation to be performed using the 18-month drift information. 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%, and a 24-month instrument uncertainty calculation was performed with this new 24-month drift value.

The total loop uncertainty of the setpoint calculations (24-month and 30-month) were then reviewed to determine the available margin and ensure positive margin exists. Since the extension request requires the surveillance to be performed within 24 months, an acceptable 24-month or 30-month uncertainty calculation bounds the maximum surveillance test interval that this request would allow, if approved.

Example 1 provides information associated with the method used to determine if acceptable margin exists. This example uses 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 Setting (LSSS) for high

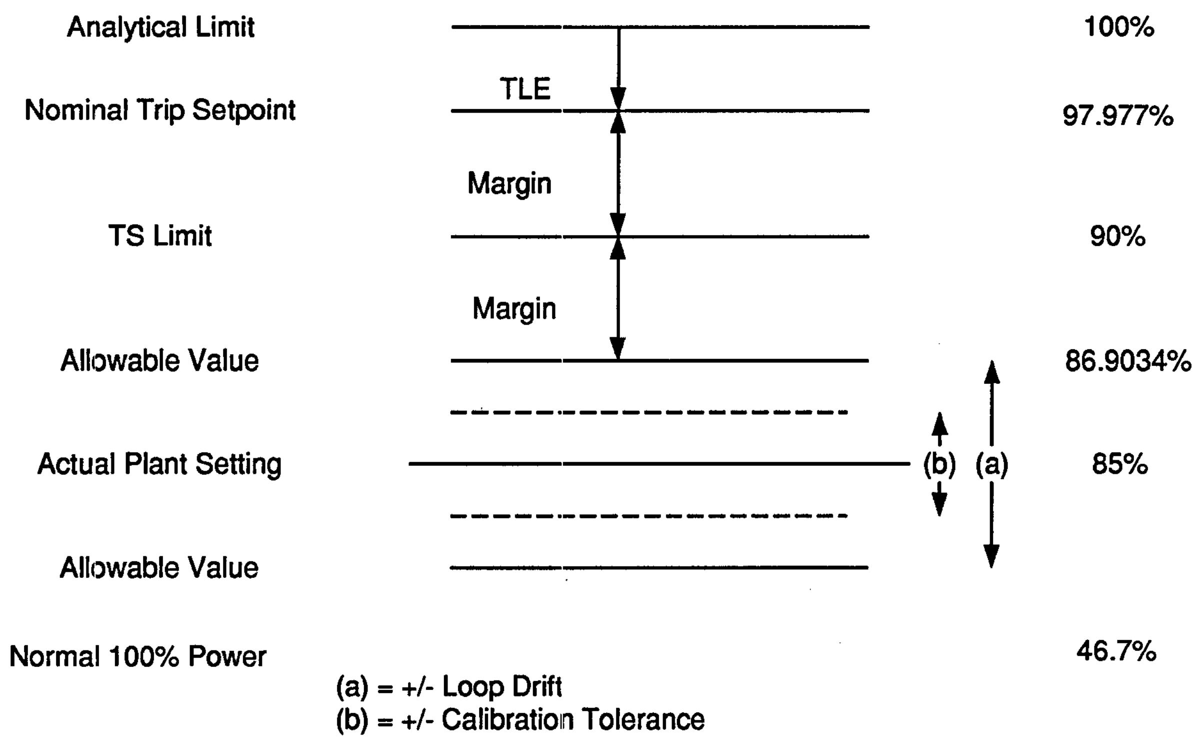
pressurizer water level be less than or equal to 90% of full scale. The analytical limit (AL) is 100% of full scale for this example.

To determine the nominal trip setpoint (minimum calculated value for actuation of the final setpoint device to initiate a protective action) the total loop error (TLE) is calculated per plant methodology. The TLE calculation consists of the measured and the unmeasured uncertainties and is where the effect of the change in drift values is shown. In the example, using the 24-month drift value the uncertainty calculation determined the TLE result as 2.023% of span or 2.023%, because the span is 100%. The nominal trip setpoint (NTSP) is determined by subtracting the TLE from the AL. In this case, the NTSP equals 97.977% (100% - 2.023% = 97.977%). The difference between the NTSP and the TS limit of 90% of full scale is considered margin.

To provide assurance that a TS LSSS value is not exceeded, additional margin can be found between the TS limit and the allowable value. From the actual plant setting (APS) a plus or minus band equal to the loop drift is established which determines the allowable value (AV) setting.

To achieve satisfactory results in Kewaunee's instrument calculations there must be no negative margin between the TS limit and the NTSP, and between the AV and the TS limit. As can be seen from Example 1, positive margin exists between the NTSP and the TS limit, and between the AV and the TS limit, therefore a 24-month surveillance test interval is justified.

Example 1
Pressurizer Level High Level Reactor Trip Instrument Calculation
For Illustrative Purposes Only - Not to Scale



Additionally, Kewaunee establishes a band around the APS that is typically equal to plus or minus the instrument accuracy, called calibration tolerance. If the as-found reading exceeds this band, the instrument is reset such that the as-left value is within the calibration tolerance.

Table 2, "Extension Request Summary," provides an overview of the bases for the extension for the surveillance test intervals. This table is found at the end of section 4.0, "Technical Analysis."

The following is a listing of the instrument/equipment surveillance test intervals that DEK is proposing to extend. For each the surveillance requirement, the current operability requirements, the basis for the operability requirement, and the basis for the surveillance test intervals extension is presented.

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

<u>Description of Current Requirement</u>

Kewaunee TS Table TS 3.5-2, "Instrument Operation Conditions for Reactor Trip," item 10 lists operability requirements for reactor coolant system (RCS) flow instrumentation

in one loop and in both loops. Each loop has three instrument channels that measure RCS flow in the loop and provide a reactor trip signal when two-of-three instrument channels indicate flow is less than the setpoint delineated in TS section 2.3, "Limiting Safety System Settings – Protective Instrumentation." TS Table 3.5-2 requires two channels per loop to be operable and if this condition cannot be met, the plant shall be placed in the Hot Shutdown condition. These instruments have a permissible bypass condition. When permissive (P) P-8 or P-7 meets the permissible bypass condition (reactor power less than 10% for P-8 and, for P-7, reactor power less than 10% and turbine impulse pressure less than 10% turbine power), the low flow single loop or both loop low flow trip, respectively, are bypassed.

Bases for Current Requirement

The low reactor coolant flow trip provides protection to prevent the reactor core from exceeding a departure from nucleate boiling (DNB) condition that can lead to fuel damage. The surveillance requirement ensures the trip circuitry remains within the setpoint and operational bounds established for this instrument to provide adequate protection.

Kewaunee USAR section 14.1.8, "Loss of Reactor Coolant Flow," describes the bases for the loss of flow reactor trip. The loss of reactor coolant flow events are categorized in the KPS USAR as either a flow coastdown accident or a locked-rotor accident.

The flow coastdown accident category includes partial and complete loss of reactor coolant flow events, and the reactor coolant pump under-frequency events. The locked-rotor category includes a hypothetical event that addresses an instantaneous seizure of a reactor coolant pump (RCP) rotor. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not promptly tripped.

The RCS low-flow reactor trip provides protection against a partial loss-of-coolant-flow accident, a complete loss-of-flow accident, and a locked-rotor accident. During these postulated events, flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low-flow signal.

Bases for the Proposed Change in Surveillance Interval

Reactor coolant flow instrumentation calibration is performed using Kewaunee surveillance procedure (SP) SP-36-014C, "Reactor Coolant Flow Transmitters Linearity and Hysteresis Test." A review, performed in the fall of 2005, of the previous three performances of surveillance procedure SP-36-014C and reviews of work orders, corrective action program issues, and operational experience over the last 54 months were conducted regarding the performance of these instruments. The instruments that are calibrated per SP-36-014C are Rosemount brand flow transmitters. These reviews

did not identify any failed surveillance procedure performances or significant instrument performance issues.

An instrument loop uncertainty calculation for the reactor coolant flow transmitters was performed. The calculation used the vendor (Rosemount) specified drift of 0.2% over 30 months. The calculation showed margin remained between the actual plant setting plus loop drift and the TS value over the 30-month period. Since this calculation shows satisfactory margin exists over a period of 30 months an extension of the interval from the current maximum of 22.5 months to 23.7 months is justified.

Therefore, based on the instrument performance reviews and the results of the calculation described above, it is acceptable to extend the calibration interval of the reactor coolant flow instrument channel described in TS Table 4.1-1, item 5 from the current maximum of 22.5 months to the proposed maximum of 23.7 months.

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

Description of Current Requirement

Kewaunee TS Table TS 3.5-2, "Instrument Operation Conditions for Reactor Trip," item 9 lists operability requirements for pressurizer high water level instrumentation. The pressurizer water level indication circuitry has three instrument channels that measure pressurizer water level. The instrumentation provides a reactor trip signal when two-of-three instrument channels indicate that pressurizer water level is greater than the setpoint delineated in TS section 2.3, "Limiting Safety System Settings – Protective Instrumentation." TS Table 3.5-2 requires two channels to be operable and if this condition cannot be met, that the plant be placed in the Hot Shutdown condition. These instruments have a permissible bypass condition, P-7. When P-7 meets the permissible bypass condition, the high pressurizer water level reactor trip is bypassed.

Bases for Current Requirement

An allowable region of power and coolant temperature conditions is defined by the primary tripping functions, the overpower high delta temperature (ΔT) trip, over-temperature high ΔT trip, and the nuclear overpower trip. The operating region below these trip settings is defined so that no combination of power, temperature, and pressure could result in DNBR less than the DNBR correlation limit. Additional tripping functions such as a high pressurizer water level trip are provided to backup the primary tripping functions for specific accident conditions and mechanical failures. The pressurizer high water level reactor trip is provided as a backup to the pressurizer high-pressure reactor trip and may afford protection for the following:

- Uncontrolled Rod Control Cluster Assembly (RCCA) Withdrawal at Power
- Loss of External Electrical Load

Bases for the Proposed Change in Surveillance Interval

Pressurizer level instrumentation calibration is performed using Kewaunee surveillance procedure SP-36-017A, "Pressurizer Level Transmitter Calibration." A review, performed in the fall of 2005, 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 were conducted regarding the performance of these instruments. The instruments that are calibrated per SP-36-017A are ITT Barton brand level transmitters. These reviews did not identify any failed surveillance procedure performances or significant instrument performance issues.

The instrument loop uncertainty calculation for pressurizer level transmitters was analyzed. 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 is sufficient margin between the actual plant setting, with the 24-month loop drift added, and the TS limit to extend the calibration interval for the pressurizer level transmitters from the current maximum of 22.5 months to 23.8 months.

Therefore, based on the instrument performance reviews described above, it is acceptable to extend the calibration interval of the pressurizer water level instrumentation in TS Table 4.1-1, item 6 from the current maximum of 22.5 months to the proposed maximum of 23.8 months.

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

Description of Current Requirement

Pressurizer pressure instrument channels have operability requirements in support of the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS).

Kewaunee TS Table TS 3.5-2, "Instrument Operation Conditions for Reactor Trip," item 7 lists operability requirements for low pressurizer pressure instrumentation while item 8 lists operability requirements for high pressurizer pressure instrumentation. The pressurizer pressure indication circuitry has four instrument channels that measure pressurizer pressure. A reactor trip signal is generated when two-of-four instrument channels indicate pressurizer pressure is less than the associated setpoint delineated in TS section 2.3, "Limiting Safety System Settings – Protective Instrumentation." A reactor trip signal is also generated when two out of three pressurizer pressure instrument channels indicate pressure greater than the associated setpoint delineated in TS section 2.3, "Limiting Safety System Settings – Protective Instrumentation."

TS Table 3.5-2 requires two channels to be operable for the high pressurizer pressure trip and three channels to be operable for the low pressurizer pressure trip. If these operability requirements cannot be met, the plant shall be placed in the Hot Shutdown

condition. The low pressurizer pressure trip instruments have a permissible bypass condition, P-7. When P-7 meets the permissible bypass condition, the low pressurizer pressure reactor trip is bypassed.

Kewaunee TS Table TS 3.5-3, "Emergency Cooling," item 1.d, provides operability requirements associated with pressurizer instruments for low pressurizer pressure safety injection signal actuation. A safety injection (SI) emergency cooling signal is generated when two-of-three pressurizer pressure instrument channels indicate pressure less than the setpoint delineated in TS Table 3.5-1, "Engineered Safety Features Initiation Instrument Setting Limits." TS Table 3.5-3 requires two pressurizer pressure channels to be operable. If these operability requirements cannot be met, the plant is to be placed in the Hot Shutdown condition. If minimum conditions cannot be met within 24 hours, steps shall be taken to place the plant in the Cold Shutdown condition. The low pressurizer pressure SI actuation instruments have a permissible bypass condition. When reactor coolant system (RCS) pressure is less than 2000 psig, the low pressurizer pressure SI actuation signal can be bypassed.

Bases for Current Requirement

The pressurizer pressure instruments provide a reactor trip function and an engineered safety feature actuation function.

An allowable region of power and coolant temperature conditions is defined by the primary tripping functions, the overpower high ΔT trip, over-temperature high ΔT trip, and the nuclear overpower trip. The operating region below these trip settings is defined so that no combination of power, temperature, and pressure could result in DNBR less than the DNBR correlation limit. Additional trip functions such as high and low pressurizer pressure trip are provided to backup the primary tripping functions for specific accident conditions and mechanical failures.

Reference is made above to over power and over temperature ΔT variable reactor trip setpoints illustrated in KPS USAR Figure 14.0.2. This figure presents the allowable reactor coolant loop average temperature and ΔT for the design flow and power distribution, as a function of primary coolant pressure. The boundaries of operation defined by the over power ΔT trip and the over temperature ΔT trip are represented as "Protection Lines" on this diagram. The DNB lines represent the locus of conditions for which the DNBR equals the limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases if the applicable DNBR line at any point does not traverse the area enclosed with the maximum protection lines. The area of permissible operation (power, pressure, and temperature) is bounded by the high neutron flux (fixed setpoint), high pressurizer pressure (fixed setpoint), low pressurizer pressure (fixed setpoint), over power ΔT (variable setpoint) and over temperature ΔT (variable setpoint).

The primary purpose of the emergency core cooling system is to automatically deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits

the fuel-clad temperature and ensures that the core will remain substantially intact and in place, with its heat transfer geometry preserved. The initiation signal for emergency core cooling by the safety injection pumps and the residual heat removal pumps is the Safety Injection Signal, which is actuated by a low pressurizer pressure signal.

The pressurizer pressure reactor trips or safety injection initiation may afford protection during the following events:

- Uncontrolled RCCA Withdrawal At Power
- Excessive Heat Removal Due To Feedwater System Malfunctions
- Loss of External Electrical Load
- Steam Generator Tube Rupture
- Steam Line Break
- Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Bases for the Proposed Change in Surveillance Interval

Pressurizer pressure calibration is performed using Kewaunee surveillance procedure SP-36-020A, "Pressurizer Pressure Transmitters Calibration." A review, performed in the fall of 2005, 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 were conducted regarding the performance of the instruments that are calibrated per SP-36-020A. The instruments that are calibrated per SP-36-020A are Rosemount brand pressure transmitters.

The 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. Based on corrective action program reviews, there were not any significant instrument performance issues since the transmitter was replaced. The other reviews did not identify significant instrument performance issues.

Instrument loop uncertainty calculations regarding the pressurizer pressure transmitters were performed. These calculations were performed using vendor (Rosemount) specified drift of 0.2% over 30 months. The calculations showed margin remained between the actual plant setting, with loop drift added, and the TS 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 instrument performance reviews described above, 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, from the current maximum of 22.5 months to the proposed maximum of 23.8 months.

4.4 TS Table 4.1-1, Item 11a, "Steam Generator Low Level" - Calibration

Description of Current Requirement

Kewaunee TS Table TS 3.5-2, "Instrument Operation Conditions for Reactor Trip," item 11a lists operability requirements for steam generator low level instrumentation. Each steam generator low-level indication loop has three instrument channels that measure steam generator level. A reactor trip signal is generated when two-of-three instrument channels indicate steam generator water level is less than the associated setpoint delineated in TS section 2.3, "Limiting Safety System Settings – Protective Instrumentation." TS Table 3.5-2 requires two channels per loop to be operable for the steam generator low-level trip. If these operability requirements cannot be met, the plant shall be placed in the Hot Shutdown condition. A preemptive reactor trip is also generated when there is a steam flow / feedwater flow mismatch where steam generator level is decreasing below the normal range and reaches a level above the lo-lo level trip setpoint designated the lo setpoint.

Additionally, TS Table TS 3.5-3, "Emergency Cooling," item 4.a and item 5.a, lists operability requirements for the steam generator level instrumentation. This operability requirement is associated with the use of the steam generator water level to automatically start the auxiliary feedwater pumps. An automatic start signal for the motor driven auxiliary feedwater pumps is generated when two-of-three instruments for its associated steam generator indicate water level is less than the associated setpoint delineated in KPS TS section 2.3, "Limiting Safety System Settings – Protective Instrumentation." An automatic start signal for the turbine driven auxiliary feedwater pump is generated when two-of-three instruments in both steam generators indicate water level less than the associated setpoint delineated in TS section 2.3, "Limiting Safety System Settings – Protective Instrumentation." TS Table 3.5-3 requires two channels per loop to be operable for the steam generator lo-lo level trip. If these operability requirements cannot be met, the plant shall be placed in the Hot Shutdown condition.

Bases for Current Requirement

An allowable region of power and coolant temperature conditions is defined by the primary tripping functions, the overpower high ΔT trip, over-temperature high ΔT trip, and the nuclear overpower trip. The operating region below these trip settings is defined so that no combination of power, temperature and pressure could result in DNBR less than the DNBR correlation limit. Additional tripping functions such as steam generator lo-lo level trip are provided to backup the primary tripping functions for specific accident conditions and mechanical failures.

The Steam Generator Low Level reactor trip may afford protection for the following events:

- Loss of External Electrical Load

- Loss of Normal Feedwater

In the event of a large loss of load in which the steam dump valves fail to open or a complete loss of load with the steam dump valves operating, the main steam safety valves (MSSVs) may lift, and the reactor may be tripped by lo-lo steam generator water level.

Additional Protection

At Kewaunee the anticipated transient without scram (ATWS) mitigating system actuation circuitry (AMSAC) actuation on low steam generator water level design has been implemented, with AMSAC armed at all power levels. The logic of AMSAC uses four steam generator instrument channels (2 per steam generator). When three of the four steam generator level channels actuate for 25 seconds, the AMSAC circuit trips the turbine and starts all three auxiliary feedwater pumps.

A Diverse Scram System (DSS) is also actuated during an ATWS and provides an independent reactor trip. The DSS is initiated on a signal from the existing AMSAC system and de-energizes the Rod Drive MG Set exciter field. Removing the Rod Drive MG set exciter field will interrupt power to the control rod grippers, allowing the control rods to free fall into the core, ending the ATWS event.

Bases for the Proposed Change in Surveillance Interval

Steam generator low-level calibration is performed using Kewaunee surveillance procedure SP-05A-028A, "Steam Generator Level Transmitters Calibration." A review, performed in the fall of 2005, of the previous three performances of SP-05A-028A and reviews of work orders, corrective action program issues, and operational experience over the last 54 months were performed regarding the performance of the instruments that are calibrated per SP-05A-028A. The instruments that are calibrated per SP-05A-028A are Rosemount brand level transmitters. These reviews did not identify any failed surveillance procedure performances.

The operational experience reviews identified two Rosemount brand steam generator level transmitter problems/failures. However, these problems/failures are not applicable to the surveillance procedure extension discussion. 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. The other was associated with vibration on the instrument tubing causing electrical noise. The other reviews did not identify any significant performance issues.

Instrument loop uncertainty calculations regarding the steam generator narrow range level transmitters were reviewed. These calculations were performed using vendor (Rosemount) specified drift of 0.2% over 30 months. The calculation showed margin remained between the actual plant setting, plus the loop drift, and the TS limit over the 30-month period. The drift calculations would therefore envelop an extension of

calibration interval the current maximum of 22.5 months to the proposed maximum of 23.7 months.

Therefore, based on the instrument performance reviews described above, it is acceptable to extend the surveillance test interval of the steam generator low level instrumentation in Table 4.1-1, item 11a, from the current maximum of 22.5 months to the proposed maximum of 23.7 months.

4.5 TS Table 4.1-1, Item 11b, "Steam Generator High Level" - Calibration

Description of Current Requirement

Kewaunee TS Table TS 3.5-4, "Instrument Operating Conditions for Isolation Functions," item 4.a lists operability requirements for steam generator high level instrumentation. The steam generator high level indication circuitry has three instrument channels that measure the steam generator level. A main feedwater isolation signal is generated when two-of-three instrument channels indicate steam generator water level is greater than the associated setpoint. TS Table 3.5-4 requires two channels to be operable for the steam generator high level isolation function. If these operability requirements cannot be met, the plant shall be placed in the Hot Shutdown condition.

Bases for Current Requirement

The basis for the high steam generator water level isolation function is for protection against a feedwater system failure which results in excessive feedwater flow to one or both steam generators. Feedwater system failures, including the inadvertent opening of the feedwater regulating valves, have the potential of allowing increased feedwater flow to each steam generator resulting in excessive heat removal from the RCS.

The feedwater flow resulting from a fully open control valve is terminated by the steam generator hi-hi water level signal, which closes all main feedwater control and feedwater control-bypass valves, trips the main feedwater pumps, closes all feedwater pump discharge valves, and trips the turbine generator. Feedwater flow increase protection is also provided by a turbine trip on Hi-Hi steam generator water level.

Bases for the Proposed Change in Surveillance Interval

Steam generator high level instrument calibration is performed using Kewaunee surveillance procedure SP-05A-028A, "Steam Generator Level Transmitters Calibration." A review, performed in the fall of 2005, of the previous three performances of SP-05A-028A and reviews of work orders, corrective action program issues, and operational experience over the last 54 months were conducted regarding the performance of the instruments that are calibrated per SP-05A-028A. The instruments

that are calibrated per SP-05A-028A are Rosemount brand level transmitters. These reviews did not identify any failed surveillance procedure performances.

The operational experience reviews identified two Rosemount brand steam generator level transmitter problems/failures. However, these problems/failures are not applicable to the surveillance procedure extension discussion. 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. The other was associated with vibration on the instrument tubing causing electrical noise. The other reviews did not identify any other significant instrument performance issues.

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

Therefore, based on the instrument performance reviews described above, it is acceptable to extend the surveillance test interval of the steam generator high level instrumentation in Table 4.1-1, item 11b, from the current maximum of 22.5 months to the proposed maximum of 23.7 months.

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

<u>Description of Current Requirement</u>

Kewaunee TS 3.1.d.5 requires that when the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is operable. One of the systems that measure RCS leakage is the containment sump level indication system. If the sump level indication system is out-of-service, then other leak detection systems may be used to ensure this TS item is satisfied. If it is not satisfied, the reactor shall be brought to a condition where the reactor is producing less than 2% power.

<u>Bases for Current Requirement</u>

Kewaunee USAR section 4.1 states that one of the General Design Criteria (GDC) applicable to the Kewaunee Power Station is that means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary (GDC 16). Detection of leakage from the Reactor Coolant System to the Reactor Containment Vessel is provided by equipment, which continuously monitors containment air

radioactivity and humidity, and containment sump level. Positive indication from this equipment is provided in the control room.

Leakage detection equipment, such as the containment sump level instrumentation, monitors parameters in containment, which are indicative of a basic level of leakage from the reactor coolant pressure boundary. Any increase in the observed parameters is an indication of a change in RCS leakage. The basic design criterion is fulfilled by the detection of deviations from normal containment parameters including changes in air particulate radioactivity, radioactive gaseous activity, humidity, condensate runoff and in the case of gross leakage, the liquid inventory in the process systems and containment sump. Once a change in normal parameters is identified, a detailed review of RCS leakage can be performed to ensure RCS leakage is within allowable operational parameters or the plant is shutdown.

Additionally, a leak in any other system with piping located inside containment (such as the residual heat removal system, the service water headers or its associated fan coil unit) will be detected by increasing containment sump level.

<u>Bases for the Proposed Change in Surveillance Interval</u>

The containment sump level test is performed using Kewaunee surveillance procedure SP-30-052, "Containment Sump A and Reactor Cavity Sump C Level Test." A review, performed in the fall of 2005, 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 were conducted regarding the past performance of the instruments that are tested per SP-30-052. The instruments that are tested per SP-30-052 are Square D/Deleval turbine brand level switches/level alarms. These reviews did not identify any failed surveillance procedure performances or significant instrument performance issues.

Drift data for the associated level switches was reviewed 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 in the Sump Hi Level, Sump Hi-Hi Level, and Lead Pump Cutoff applications 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 in the Sump Hi Level, Sump Hi-Hi Level, and Lead Pump Cutoff applications were all within the acceptable tolerance band of the surveillance procedure.

Therefore, based on the instrument performance reviews described above, it is acceptable to extend TS Table 4.1-1, item 21, "Containment Sump Level Test," from the current maximum of 22.5 months to the proposed maximum of 23.8 months.

4.7 TS Table 4.1-1, Item 30, "Fore Bay Water Level" - Test

Description of Current Requirement

Kewaunee TS 3.3.e, Service Water," item 1, states:

> *The reactor shall not be made critical unless the following conditions are satisfied, except for LOW POWER PHYSICS TESTS and except as provided by TS 3.3.e.2.*

Under TS 3.3.e, item 1, is condition "B" which states:

> *The Forebay Water Level Trip System is OPERABLE.*

Kewaunee TS 3.3.e.2 states:

> *During power operation or recovery from an inadvertent trip, ONE service water train may be inoperable for a period of 72 hours. If OPERABILITY is not restored within 72 hours, then within 1 hour action shall be initiated to:*
>
> - *Achieve HOT STANDBY within the next 6 hours.*
> - *Achieve HOT SHUTDOWN within the following 6 hours.*
> - *Achieve and maintain Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods within an additional 36 hours.*

Therefore, if the forebay water level trip system is inoperable, the operability requirements of TS 3.3.e.1 are not met and the operators are directed to achieve and maintain Reactor Coolant System T_{avg} less than 350°F in accordance with TS 3.3.e.2.

Kewaunee TS Table TS 3.5-1, "Engineered Safety Features Initiation Instrument Setting Limits," item 7 is the forebay level. This table states that the purpose of the forebay level trip is to trip the circulating water pumps. TS Table 3.5-1 does not list any trip setting limit.

Bases for Current Requirement

The circulating water intake system is designed to provide a reliable supply of Lake Michigan water, regardless of weather or lake conditions, to the suction of two circulating water pumps, four service water pumps and two fire pumps. Normal operation is with one or two circulating water pumps operating, two to four service water pumps operating, with the fire pumps in standby. Following a DBA with loss of off-site power, the circulating water pumps would not be operating due to the loss of their power source, only the four-service water pumps would be running because they have an

emergency power source. The fire pumps are started either manually or automatically for fire protection.

The main intake line is a Class I structure designed to remain intact allowing flow during all natural phenomena. In the unlikely event that the main intake line becomes blocked, the circulating water pumps could lower the screenhouse forebay water level to below the service water pump inlet housing. Protection against this occurrence is afforded by tripping the circulating water pumps upon a redundant coincident forebay extreme low water level signal.

<u>Bases for the Proposed Change in Surveillance Interval</u>

The forebay water level test is performed using Kewaunee surveillance procedure SP-04-134, "Forebay Area Water Level Logic Test." A review, performed in the fall of 2005, 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 were conducted regarding the performance of the instrument channels that are tested per SP-04-134. The instruments that are tested per SP-04-134 are ITT Barton brand bubbler flow controllers/level control switches and the associated trip matrix relays and contacts. These reviews did not identify any significant instrument performance issues or failed surveillance procedure 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 acceptable instrument performance over the previous 54 months, it is acceptable that the surveillance test interval of the forebay water level test in TS Table 4.1-1, item 30, from the current maximum of 22.5 months to the proposed maximum of 23.6 months.

4.8 TS Table 4.1-1, Item 33, "PORV Block Valve Position Indicator" - Calibration

<u>Description of Current Requirement</u>

Kewaunee TS 3.1.a.5 requires two pressurizer power operated relief valves (PORVs) and their associated block valves to be operable during hot standby and operating modes. With one or both block valves inoperable, the operators are directed to restore the block valve(s) to an operable status within one hour or place its associated PORV in manual control. Subsequently, if the block valve(s) is not restored to an operable status within a prescribed period (72 hours for one inoperable block valve, one hour for two inoperable block valves), action shall be initiated to place the plant in the Hot Shutdown condition.

Kewaunee TS Table TS 3.5-6 requires two position indication channels per valve to be operable for the pressurizer power operated relief block valves. With one or both block valve(s) position indication channels inoperable, the operators are directed to restore the block valve position indication channel to an operable status within 14 days for one inoperable block valve position indication channel, or within one hour for two inoperable block valve(s). If the position indication channel(s) are not restored within the times above action shall be initiated to place the plant in the Hot Shutdown condition.

Bases for Current Requirement

The pressurizer vessel contains replaceable direct immersion heaters, multiple safety and relief valves, a spray nozzle and interconnecting piping, valves, and instrumentation. Block valves, provided with an emergency power supply, are located upstream of the pressurizer power operated relief valves (PORVs) to prevent a LOCA through an unseated relief valve. The block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break LOCA. As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

NRC letter dated July 2, 1980 (Reference 3), recommended certain TS changes be made to provide reasonable assurance that facility operation is maintained within acceptable limits following the implementation of theTMI-2 Lessons Learned Category "A" items. The surveillance requirement recommended in the July 2, 1980 letter for PORV Block valve indication was added to Kewaunee TS by License Amendment 38, dated November 6, 1981 (Reference 4).

Bases for the Proposed Change in Surveillance Interval

Pressurizer PORV block valve position indicator calibration is performed using Kewaunee surveillance procedures 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 surveillance procedures 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 review of the past surveillances, the position indicator has performed properly since 1993 (12 years), and extension of the surveillance test interval to a maximum to 23.8 months is acceptable.

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

Description of Current Requirement

Kewaunee TS Table TS 3.5-6, "Accident Monitoring Instrumentation Operating Conditions for Indication," item 2 lists operability requirements for Reactor Coolant System (RCS) subcooling instrumentation channels. The RCS subcooling indication circuitry has two instrument channels that provide control room indication of RCS subcooling. Table 3.5-6 requires two channels to be operable. If one channel is inoperable, it must be restored to an operable status within 14 days, or the plant shall be placed in the Hot Shutdown condition. Additionally, if no RCS subcooling instrument channels are operable, then within 72 hours one RCS subcooling instrument channel must be restored to an operable status or the plant is placed in the Hot Shutdown condition.

Bases for Current Requirement

NRC letter dated July 2, 1980 (Reference 3), recommended certain TS changes be made to provide reasonable assurance that facility operation is maintained within acceptable limits following the implementation of theTMI-2 Lessons Learned Category "A" items.

The surveillance requirement recommended in the July 2, 1980 letter for Reactor Coolant System Subcooling Margin Monitor instrument channels was added to Kewaunee TS by License Amendment 38, dated November 6, 1981 (Reverence 4).

Bases for the Proposed Change in Surveillance Interval

RCS subcooling margin monitor calibration and testing is performed using Kewaunee surveillance procedures SP-36-162 and SP-36-163, "Reactor Coolant System Hot Leg Pressure Transmitter PT-419/PT-420 Calibrations." A review, performed in the fall of 2005, 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 were conducted regarding the performance of the instruments that are calibrated per SP-36-162/163. The instruments that are calibrated per SP-36-162/163 are Rosemount brand pressure transmitters. These reviews did not identify any failed surveillance procedure performances or significant instrument performance issues.

Instrument loop uncertainty calculations for the reactor coolant system hot leg pressure transmitters were reviewed. These calculations were performed using vendor (Rosemount) specified drift of 0.2% 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 envelops

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 instrument performance reviews described above, it is acceptable to extend the surveillance test interval of the RCS subcooling margin monitor instrumentation in TS Table 4.1-1, item 36, from the current maximum of 22.5 months to the proposed maximum of 23.7 months.

4.10 TS Table 4.1-1, Item 42, "Steam Generator Level (Wide Range)" - Calibration

Description of Current Requirement

Kewaunee TS Table TS 3.5-6, "Accident Monitoring Instrumentation Operating Conditions for Indication," item 11 lists operability requirements for Steam Generator Level (Wide Range) instrumentation channels. The Steam Generator Level (Wide Range) indication circuitry provides two instrument channels per steam generator that measure the Steam Generator Level (Wide Range). These channels provide control room indication of the Steam Generator Level (Wide Range). Table 3.5-6 requires two channels per steam generator to be operable. If one channel is inoperable, for either or both SGs, the channel(s) must be restored to operability within seven days or the plant shall be placed in the Hot Shutdown condition. Additionally, if no Steam Generator Level (Wide Range) instrument channels are operable for either steam generator, then within 72 hours one Steam Generator Level (Wide Range) instrument channel per steam generator must be returned to an operable status or a plant shutdown to Hot Shutdown conditions must be commenced.

Bases for Current Requirement

Steam generator water level (wide range) indication instrumentation was added to the Kewaunee TS in response to NRC Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The proposed amendment, which requested adding the steam generator wide range water level indication to the Kewaunee TS, stated that because this indication was considered a type A variable, TS requirements for this variable are appropriate.

The surveillance requirement for steam generator wide range instrument channels was added to Kewaunee TS by License Amendment 105, dated February 9, 1994.

Bases for the Proposed Change in Surveillance Interval

Steam generator water level (wide range) calibration is performed using Kewaunee surveillance procedure SP-05A-300, "Steam Generator A Wide Range Level Transmitter Calibrations," and SP-05A-301, "Steam Generator B Wide Range Level Transmitter Calibrations." A review of 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 were conducted regarding the performance of the instruments that are calibrated per SP-05A-300/301. The instruments that are calibrated per SP-05A-300/301 are Rosemount brand level transmitters. These reviews did not identify any failed surveillance procedure performances.

The operational experience reviews identified one Rosemount brand steam generator level transmitter problem/failure. However, this problem/failure is not applicable to the surveillance procedure extension discussion. This problem/failure was associated with transmitters that are used at Kewaunee on equipment associated with the turbine, turbine generator, and the feedwater system. See also TS Table 4.1-1 item 11a and 11b. The other reviews did not identify any other significant instrument performance issues.

Instrument loop uncertainty calculations regarding the steam generator wide range level transmitters were reviewed. These calculations were performed using vendor (Rosemount) specified drift of 0.2% over 30 months. The calculations 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 the surveillance test interval from the current maximum of 22.5 months to the proposed maximum of 23.7 months.

Therefore, based on the instrument performance reviews described above, it is acceptable to extend the surveillance test interval of the Steam Generator Level (Wide Range) instrumentation in TS Table 4.1-1, item 42, from the current maximum of 22.5 months to the proposed maximum of 23.7 months.

4.12 TS Table 4.1-3, Item 4, "Containment Isolation Trip" - Test

Description of Current Requirement

Kewaunee TS 1.0.g defines containment system integrity. Containment system integrity is defined to exist when, among other items, the required automatic Containment System isolation valves are operable, except as provided in TS 3.6.b. Kewaunee TS 3.6.a states that containment system integrity shall not be violated if there is fuel in the reactor which has been used for power operation, except when the reactor is in the cold shutdown condition with the reactor vessel head installed, or the reactor is in the refueling shutdown condition. The containment isolation trip test verifies that the automatic isolation valves required for containment isolation automatically shut when a containment isolation signal is received.

Additionally, Kewaunee TS Table TS 3.5-4, "Instrument Operating Conditions for Isolation Functions," provides operating requirements for the containment isolation function. This table identifies two initiation signals for the containment isolation trip

function, safety injection and manual. If the safety injection initiation circuit is inoperable, as determined by item 1 of TS Table TS 3.5-3, the operator action is to take action to place the plant in the cold shutdown condition if the condition of inoperability is not corrected within 24 hours. If the manual initiation circuit does not meet the minimum conditions of one of two channels operable then the plant shall be placed in the Hot Shutdown condition.

Therefore, the plant would be placed in either the Hot Shutdown condition or the Cold Shutdown condition depending on the cause of the inoperability.

Bases for Current Requirement

The principal function of the Containment Isolation System is to confine the fission products within the Primary Containment System boundary during accident conditions. 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 Engineered Safety Features functions. A containment isolation actuation signal is initiated by a Safety Injection Signal or by manual initiation. The containment isolation trip test verifies the automatic containment isolation valves close as required by manual initiation of the isolation actuation system.

Bases for the Proposed Change in Surveillance Interval

The containment isolation trip test is performed using Kewaunee surveillance procedure SP-56-078, "Containment Isolation Trip Test." A review was performed of the corrective action program database with only two issues found since 1998. One issue was due to a failed contact on a valve motor starter. The other issue dealt with a failure of redundant indication for containment isolation valve. The primary indication operated properly, but a redundant status panel light did not show the valve was in the proper position.

Additionally, a review was conducted on the surveillance performance since 1976. Minor issues, which did not cause a test failure, were identified including:

- Plant conditions prevent cycling of valves
- Redundant indication failures
- Improper limit switch adjustment
- Valves fail to return to post test position
- A valve failed to close due to a failed contact on its motor starter

One issue was identified during the review that indicated a valve required to close for containment isolation failed to close as required. This valve has a redundant containment isolation valve which would independently isolate the containment penetration and prevent a flow path for radioactive materials to escape containment.

Therefore, based on finding only minor issues with the performance of the containment isolation trip test, the risk of extending this surveillance interval from the current maximum of 22.5 months to the proposed maximum of 23.9 months poses minimal risk and is therefore acceptable.

4.13 TS 4.6.a.2, "Automatic Start of Each Diesel Generator, Load Shedding, and Restoration to Operation of Particular Vital Equipment" - Tests
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" - Tests
TS 4.5.a.3, "Containment Fancoil Units" – Tests
TS 4.17.a.2, "Control Room Post Accident Recirculation System." - Testing

Introduction

This item is different from the other surveillance test interval extensions in that six individual TS requirements are grouped together. The six are listed below with the specific requirement.

- TS 4.6.a.2, "Automatic Start of Each Diesel Generator, Load Shedding, and Restoration to Operation of Particular Vital Equipment"
    - Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment, all initiated by a simulated loss of all normal a-c station service power supplies together with a simulated safety injection signal. This test will be conducted at each REFUELING interval to assure that each diesel generator will start and assume required loads to the extent possible within 1 minute, and operate for *≥ 5 minutes while loaded with the emergency loads.*
- TS 4.4.c.1.b, "Shield Building Ventilation System" – Tests
    - Automatic initiation of each train of the system.
- TS 4.5.a.1, "Safety Injection System" – Tests
    - System tests shall be performed once per operating cycle or once every 18 months, whichever occurs first. With the Reactor Coolant System pressure ≤ 350 psig and temperature ≤ 350°F, a test safety injection signal will be applied to initiate operation of the system.
    - The test will be considered satisfactory if control board indication or visual observations indicate that all components have received the safety injection signal in the proper sequence and timing. That is, the appropriate pump motor breakers shall have opened and closed, and all valves shall have completed their travel.
- TS 4.5.a.2, "Containment Vessel Internal Spray System" – Tests

- System tests shall be performed once every operating cycle or once every 18 months, whichever occurs first. The test shall be performed with the isolation valves in the supply lines at the containment blocked closed.
- Verify a minimum of 76 spray nozzles per train are functioning properly by using an air or smoke test at a test interval not to exceed 10 years.
- The test will be considered satisfactory if control board indications or visual observations indicate all components have operated satisfactorily.

- TS 4.5.a.3, "Containment Fancoil Units" – Tests
  - Each fancoil unit shall be tested once every operating cycle or once every 18 months, whichever occurs first, to verify proper operation of the motor-operated service water outlet valves and the fancoil emergency discharge and associated backdraft dampers.
- TS 4.17.a.2, "Control Room Post Accident Recirculation System" – Testing
  - Automatic initiation of the system on a high radiation signal and a safety injection signal.

The surveillance test that Kewaunee is unable to perform within its surveillance test interval for each of these items is the automatic starting of the equipment on initiation of a safety injection signal with subsequent sequential loading of the ESF loads on the emergency diesel generators (TS 4.6.a.2). Because the surveillance procedure associated with this test satisfies the requirements for all of the above surveillance requirements, the TS requirements are grouped together.

An exception to TS 4.6.a.2 satisfying all of the above TS requirements is the containment vessel internal spray system test. This test verifies a minimum number of spray nozzles per train are functioning and is performed by a separate surveillance procedure. The spray nozzle surveillance will be performed within its required surveillance test interval, and therefore; surveillance test interval extension is not required for it.

Description of Current Requirement

Each of the above TS testing requirements involves a surveillance associated with the emergency diesel generators. TS 4.6.a.2 requires the automatic starting, load shedding, and restoration to operation of specific vital equipment. TSs 4.4.c.1.b, 4.5.a.1, TS 4.5.a.2, 4.5.a.3, and 4.17.a.2 are specific testing requirements, which demonstrate automatic initiation of specific equipment. The initiation of this equipment is accomplished during performance of surveillance requirement TS 4.6.a.2. Because these surveillance requirements are all performed during the conduct of TS 4.6.a.2 they have been grouped together.

TS 4.4.c.1.b requires the automatic initiation of each train of the shield building ventilation system.

TS 4.5.a.1 requires a safety injection system test where a safety injection signal is applied to initiate operation of the system. This safety injection system test is considered satisfactory if control board indication or visual observations indicate that all components have received the safety injection signal in the proper sequence and timing. That is, the appropriate pump motor breakers shall have opened and closed, and all valves shall have completed their travel.

TS 4.5.a.3 requires each containment fancoil unit be tested to verify proper operation of the motor-operated service water outlet valves and the fancoil emergency discharge and associated backdraft dampers.

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

All these required tests are performed during one surveillance test, which is conducted by initiation of a safety injection signal and tripping the engineered safety feature (ESF) 4160 volt bus undervoltage relays. This surveillance tests the automatic start of the emergency diesel generators, the load shedding, the restoration to operation of particular equipment, and the automatic initiation of the SBV system, the safety injection system, the containment fan coil units, and the control room post-accident recirculation system by a safety injection signal.

Bases for Current Requirement

The Engineered Safety Features (ESF) at Kewaunee are:

- Containment System,
- Safety Injection System,
- Containment Cooling System;
- Containment Spray System
- Auxiliary Feedwater System,
- Special Zone Ventilation Systems,
- Diesel Generators, and
- Station Batteries.

During a design basis accident, the release of fission products from the reactor fuel is limited by the Safety Injection System which, by cooling the core, keeps the fuel in place and substantially intact and limits metal water reaction to an insignificant amount.

Kewaunee GDC 46 states that design provisions shall be made so that components of the Emergency Core Cooling System can be tested periodically for operability and functional performance.

On-line functional testing of the safety injection system provides information, which confirms the proper automatic sequencing of load addition to the diesel generator. The

on-line functional test verifies the ability to trip source breakers, restore power to ESF buses, load shed, and sequentially start ESF equipment. This test is started by simulating a blackout on the ESF bus concurrent with initiating safety injection. Safety injection pumps use their normal recirculation path and will not inject into the RCS.

The emergency diesel generators overall system test demonstrates that the emergency power system and the control system for the ESF equipment function automatically in the event of loss of all other sources of a-c power, and that the diesel generators start automatically in the event of a loss-of-coolant accident. 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 diesel generators. This test is conducted by simultaneously unblocking the safety injection signal and simulating a loss-of-voltage signal.

Thus, the ESF systems are tested for proper starting and sequencing in one surveillance test. Therefore, all of these tests are included in this one request.

<u>Bases for the Proposed Change in Surveillance Interval</u>

A review of the past performances of SP-33-110, "Diesel Generator Automatic Test," and the corrective action program database were performed to determine if any issues have been encountered with this surveillance. Ten issues were identified dating back to 1996.

Of the issues identified, several were minor issues.

1. A test setup problem that did not affect the outcome of the surveillance.
2. A delay in a status light illuminating during the test.
3. Both control room post accident recirculation fans started during the testing of one train. The reason the other train started was a parallel start signal generated by a radiation monitor for the control room ventilation system.
4. A valve opened unexpectedly during the test, but the unexplained opening could not be repeated during troubleshooting.
5. A valve that had a history of opening during perturbations in its power supply opened during the test due to the same type of perturbations. This issue was resolved by modification to the valve power supply.

Four were relay problems.

6. One was caused by a lifted lead not being re-landed. The lead was lifted in response to an issue from a previous surveillance where the lead required lifting but because of relay problems, could not be landed. When this surveillance was performed the un-landed wire prevented the relay from actuating, this was later corrected.

7/8/9. Three past cases of relays which either failed to latch or reset. These issues did not cause a failure of the test but caused problems when returning the plant to the post-test condition.

One additional issue occurred in 1996 when the emergency diesel generator failed to remain attached to the vital bus.

10. The output breaker for emergency diesel generator (EDG) "A" initially closed but immediately reopened. The root cause of this event was an inadequate design change installation procedure. During the 1996 refueling outage, the EDG "A" output breaker was replaced. The installation procedure did not adequately adjust an actuating linkage shaft after an adjustment was made to the actuator arm pivot and the retest procedure did not verify proper operation of all contacts on the associated stationary switch. This condition led to the EDG "A" output breaker closing and immediately opening and remaining open due to an anti-pumping feature of the breaker. The actuating linkage shaft was subsequently adjusted and the test satisfactorily completed.

Therefore, based on the past performance of the surveillance procedure having produced acceptable results, except for where pre-test maintenance had been performed (i.e., new EDG output breaker or wire removed), extending this surveillance poses minimal risk and is acceptable from the current maximum of 22.5 months to the proposed maximum of 23.9 months.

4.16 TS 4.5.b.2.F, "Residual Heat Removal System Valve Interlocks" - Test

Description of Current Requirement

TS 3.3.b, "Emergency Core Cooling System," requires that the reactor not be made critical unless two SI/RHR trains are operable with each train comprised of one operable safety injection pump, one operable residual heat removal pump, one operable residual heat removal heat exchanger, and an operable flow path consisting of all valves, piping and interlocks associated with the above train of components and required to function during accident conditions.

TS 3.1.a.2, "Decay Heat Removal Capability," requires at least two of four heat sinks to be operable whenever the average reactor coolant temperature is ≤ 350°F but > 200°F. The four heat sinks include, Steam Generator 1A, Steam Generator 1B, Residual Heat Removal Train A, and Residual Heat Removal Train B. If less than the number of required heat sinks are operable, corrective action shall be taken immediately to restore the minimum number to an operable status.

TS 3.1.a.2 also states the two residual heat removal trains are required to be operable whenever the average reactor coolant temperature is ≤ 200°F and irradiated fuel is in the reactor, except when in the refueling mode with the minimum water level above the top of the vessel flange ≥ 23 feet, one train may be inoperable for maintenance. If one

residual heat removal train is inoperable, then corrective action shall be taken immediately to return it to the operable status.

To verify operability of the RHR system, residual heat removal system valve interlock tests are performed by SP-34-145D, "Residual Heat Removal Valve RHR-11 Reactor Coolant System Interlock Test."

Bases for Current Requirement

Verifying that the RHR valve auto-closure interlocks are operable ensures that RCS pressure will not pressurize the RHR system beyond its design pressure of 600 psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < 450 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift.

Bases for the Proposed Change in Surveillance Interval

The RHR system valve interlock test is performed using Kewaunee surveillance procedure 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. The instruments that are tested per SP-34-145D are a Foxboro brand controller and associated relays and contacts. These reviews did not identify any failed surveillance procedure performances or significant instrument performance associated issues.

Instrument drift regarding the Rosemount pressure transmitter is addressed per SP-36-163, "Reactor Coolant System Hot Leg Pressure Transmitter PT-420 Calibration." See the discussion associated with TS Table 4.1-1, Item 36, in Section 4.9.

Instrument drift regarding the remaining instruments within the associated instrumentation loop is addressed per 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 instrument performance reviews described above, it is acceptable to extend the surveillance test interval for the RHR system valve interlocks in TS 4.5.b.2.F from the present maximum of 22.5 months to the proposed maximum of 23.7 months.

4.17 TS 4.6.a.3, "Diesel Generator Inspection"

Description of Current Requirement

Kewaunee TS 3.7, "Auxiliary Electrical Systems," requires that the reactor not be made critical unless both diesel generators are operable. Two underground storage tanks combine to supply at least 35,000 gallons of fuel oil for either diesel generator and the day tanks for each diesel generator contain at least 1,000 gallons of fuel oil.

During power operation or recovery from an inadvertent trip, one diesel generator may be inoperable for a period not exceeding 7 days provided the other diesel generator is tested daily to ensure operability and the engineered safety features associated with the operable diesel generator are operable. If operability is not restored within the time specified, then within 1 hour action shall be initiated to achieve Hot Standby within the next 6 hours.

To verify that the emergency power sources and equipment are operable, each diesel generator is inspected at each major refueling outage.

Bases for Current Requirement

Diesel Generator inspections are performed at refueling outage intervals in order to maintain the diesel generators in accordance with the manufacturer's recommendations.

Bases for the Proposed Change in Surveillance Interval

Inspection of the electrical components associated with the EDGs is performed during each refueling outage under SP-10-111-1, "Inspection of Diesel Generator A (Electrical) (QA-1)," and SP-10-211-1, "Inspection of Diesel Generator B (Electrical) (QA-1)." Mechanical maintenance is also performed during the refueling outage using procedures SP-10-111-2, "Inspection of Diesel Generator A (Mechanical) (QA-1)," and SP-10-211-2, "Inspection of Diesel Generator B (Mechanical) (QA-1)." After the completion of the electrical and mechanical maintenance each diesel generator is tested under SP-10-111-3, "Inspection of Diesel Generator A (Component Retest)," and SP-10-211-3, "Inspection of Diesel Generator B (Component Retest)."

A review of the previous three performances of the diesel generator inspection procedures was performed 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 review identified procedure deficiencies as well as failures of equipment (replaced during the inspection procedure) to pass initial post-maintenance testing. However, the review did not identify any issues that could have been prevented by performing the surveillance at a shorter frequency.

Maintenance on the EDGs during the next refueling outage is limited to the items that are performed on an 18-month frequency since the 6-year and 12-year maintenance is

not due during the 2006 refueling outage. The 12-year maintenance was last performed during the 2001 refueling outage, while the 6-year maintenance was last performed during the 2004 refueling outage. 18-month mechanical maintenance includes various cleaning, lubrication and inspection tasks as well as replacement of filter media. 18-month electrical maintenance is limited to cleaning and inspection of the equipment in the EDG electrical cabinets as well as resistance measurements of select relays within these cabinets.

Based on the review of items identified during previous EDG work and the limited scope of the inspection and maintenance scheduled for the next refueling outage, there is minimal additional risk in extending the frequency of the emergency diesel generator inspections required by TS 4.6.a.3 from the current maximum of 22.5 months to the proposed maximum of 23.6 months.

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

Description of Current Requirement

Kewaunee TS 3.7, "Auxiliary Electrical Systems," requires that the reactor not be made critical unless both diesel generators are operable. Two underground storage tanks combine to supply at least 35,000 gallons of fuel oil for either diesel generator and the day tanks for each diesel generator contain at least 1,000 gallons of fuel oil.

During power operation or recovery from an inadvertent trip, one diesel generator may be inoperable for a period not exceeding 7 days provided the other diesel generator is tested daily to ensure operability and the engineered safety features associated with the operable diesel generator are operable. If operability is not restored within the time specified, then within 1 hour action shall be initiated to achieve Hot Standby within the next 6 hours.

To verify that the emergency power sources and equipment are operable, the diesel generator load rejection test is required to be performed at least once per 18 months, in accordance with IEEE 387-1977, Section 6.4.5.

Bases for Current Requirement

The load rejection test demonstrates the capability of rejecting the maximum rated load without overspeeding or attaining voltages which would cause the diesel generator to trip, mechanical damage, or harmful overstresses.

Bases for the Proposed Change in Surveillance Interval

A review of the previous three performances of the EDG elevated load and load reject tests was performed to identify issues that may have been influenced by the amount of time that had passed since the last performance of the test. This review included the

completed procedure for these tests (SP-42-047A, "Diesel Generator A Operational Test" and SP-42-047B, "Diesel Generator B Operational Test") as well as additional corrective action program issues or work orders that were initiated during the performance of these tests.

The review identified procedure deficiencies as well as failures of equipment to pass their initial post-maintenance test, but did not identify any issues that could have been prevented by performing the test at a shorter frequency. A review of operating experience specific to the Electro-Motive Division (EMD) EDGs was also performed by searching the EMD owners group web site for problems experienced while performing elevated load and load reject tests.

The EMD owner's group web-site review did not identify any problems that were induced by the frequency of the test but did identify that the testing is typically performed after maintenance or replacement of the governor. Various documents found on the EMD owners group web site indicate that the elevated load test provides assurance that the internals on the governor are operating correctly in addition to ensuring the governors 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 load and that the governor controls engine speed to prevent an over-speed trip should the output breaker open during full load operation.

Since any maintenance that may involve the EDG governor will not be performed until the refueling outage, and no frequency related dependency was identified in operating experience, there is minimal additional risk in extending the frequency of the elevated load and load reject tests from the current maximum of 22.5 months to the proposed maximum of 23.9 months.

4.19 TS 4.14, "Testing and Surveillance of Shock Suppressors (Snubbers)."

Description of Current Requirement

TS 3.14, "Shock Suppressors (Snubbers)," state that the reactor shall not be made critical unless all safety-related shock suppressors are operable. If any safety-related shock suppressor is found inoperable, then within 72 hours the inoperable shock suppressor shall be restored to an operable condition or replaced with a spare shock suppressor of similar specifications. If the inoperable shock suppressor cannot be made operable or replaced then the fluid line restrained by the inoperable shock suppressor shall, if feasible, be isolated from other safety-related systems if otherwise permitted by the TS. Thereafter, operation may continue subject to any limitations by the TS for that fluid line. If the shock suppressor cannot be made operable, replaced, or the system isolated; actions shall be initiated to shut down the reactor.

Bases for Current Requirement

Shock suppressors (snubbers) are designed to prevent unrestrained pipe motion under dynamic loads, as 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 shock suppressors.

Bases for the Proposed Change in Surveillance Interval

There are 99 installed safety-related snubbers in the Kewaunee Power Station, 2 large-bore snubbers and 97 small-bore snubbers.

During the November forced outage the two large-bore and 57 small-bore snubbers were inspected in containment in accordance with 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 hydraulic snubbers performed in accordance with SP-55-180 have identified recordable indications, these indications did not result in the unsatisfactory operational readiness of the hydraulic snubber.

Based on the snubber inspections performed during the November 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. DEK 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). A code relief request is provided in Attachment 4 of this document. This relief request provides additional details regarding the location of this snubber and justification for extending its surveillance interval.

Therefore, the extension of the visual examination from the current maximum of 22.5 months to the proposed maximum of 23.7 months, poses an insignificant 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.

4.20 TS 6.12.b, "System Integrity Program Integrated Leak Tests."

Description of Current Requirement

Kewaunee TS 6.12 states that the licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include provisions establishing preventive maintenance and periodic visual inspection requirements and integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

Bases for Current Requirement

In July of 1980, the NRC issued a letter providing model specifications based on the NRC staff's evaluation of the actions taken at Kewaunee to satisfy the Category "A" items of the NRC staffs recommendations resulting from TMI-2 lessons learned (Reference 3). In this letter, the NRC indicated that the Kewaunee license should be amended by adding license conditions related to a system integrity measurements program and improved iodine measurement capability.

KPS License Amendment 38 was issued November 6, 1981, and included the requirement for the system integrity program, TS 6.12. In the "Notice of Issuance of Amendment to Facility Operating License" section of License Amendment 38, the NRC stated that the safety evaluation was included in their letter transmitted to the licensee dated April 18, 1980. The NRC stated that the purpose of this technical specification was to develop and implement a leakage reduction program for Kewaunee.

Bases for the Proposed Change in Surveillance Interval

The affected surveillance procedures are:

- SP-34-091, "RHR Hydrostatic Test,"
- SP-23-193, "Containment Spray System Leakage Test,"
- SP-33-195, "Safety Injection System Leakage Test,"
- SP-23-080, "ICS and SI Valve Leakage Test," and
- SP-33-325, "Measurement of RCS/RHR Leak-By of RHR-299A and RHR-299B," and;
- SP-56A-090, "Containment Local Leak Rate Type B & C Test."

Surveillance procedures SP-34-091, SP-23-193, and SP-33-195 measures external leakage from the residual heat removal, internal containment spray, and safety injection systems, respectively. Surveillance procedures SP-23-080 and SP-33-325 measure leakage from the SI, ICS, and RHR systems back to the RWST. Surveillance procedure SP-56A-090 measures leakage from the hydrogen analyzers and the post-accident vent re-routes.

Test results for the last five refueling outages were reviewed. It was found that all results are stable and well within acceptance criteria. Extrapolation of the as-found values from the previous tests determined that an extension of the test interval from 22.5 months to 23.9 months would not result in exceeding the tests acceptance criteria.

Therefore, an extension of test intervals from 22.5 months to 23.9 months is not expected to result in any challenges to the relevant system integrity program acceptance criteria.

Table 2

Extension Request Summary

Surveillance Requirement	Table Item	Title	Requested Extension Surveillance Interval (Months)	Determined/Calculated Acceptable Surveillance Interval (Months)
Table 4.1-1	5	Reactor Coolant Flow - Calibration	23.7	30(A)
Table 4.1-1	6	Pressurizer Water Level - Calibration	23.8	24(A)
Table 4.1-1	7	Pressurizer Pressure - Calibration	23.8	30(A)
Table 4.1-1	11a	Steam Generator Low Level - Calibration	23.7	30(A)
Table 4.1-1	11b	Steam Generator High Level - Calibration	23.7	30(A)
Table 4.1-1	21	Containment Sump Level - Test	23.8	24(B)
Table 4.1-1	30	Fore Bay Water Level - Test	23.6	23.6(C)
Table 4.1-1	33	PORV Block Valve Position Indicator - Calibration	23.8	23.8(C)
Table 4.1-1	36	Reactor Coolant System Subcooling Monitor – Calibration and Test	23.7	30(A)
Table 4.1-1	42	Steam Generator Level (Wide Range) - Calibration	23.7	30(A)
Table 4.1-3	4	Containment Isolation Trip - Test	23.9	23.9(D)
4.4.c.1.b		Shield Building Ventilation System Tests	23.9	23.9 (D)
4.5.a.1		Safety Injection System Tests	23.9	23.9 (D)
4.5.a.2		Containment Vessel Internal Spray System	23.9	23.9 (D)
4.5.a.3		Containment Fancoil Units Tests	23.9	23.9 (D)
4.5.b.2.F		Residual Heat Removal System Valve Interlocks	23.7	23.7 (C)
4.6.a.2		Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment	23.9	23.9 (D)
4.6.a.3		Each diesel generator shall be inspected	23.6	23.6 (E)
4.6.a.4		Diesel Generator Load Rejection Test	23.9	23.9 (F)

Table 2

Extension Request Summary

Surveillance Requirement	Table Item	Title	Requested Extension Surveillance Interval (Months)	Determined/Calculated Acceptable Surveillance Interval (Months)
4.14		Testing And Surveillance Of Shock Suppressors (Snubbers)	23.7	23.7 (D)
4.17.a.2		Control Room Post Accident Recirculation System	23.9	23.9 (D)
6.12.b		System Integrity Program Integrated Leak Tests	23.9	23.9 (D)

(A) Extension acceptability is based on acceptable setpoint calculations, past performance of surveillance procedures and reviews of work order history, corrective action program issues, and operational experience.
(B) Extension acceptability is based on past performance of surveillance procedures, a calculation of acceptable 24 month instrument drift using an extrapolation of the 18 month drift results, and reviews of work order history, corrective action program issues, and operational experience.
(C) Extension acceptability is based on past performance of surveillance procedures and reviews of work order history, corrective action program issues, and operational experience
(D) Extension acceptability is based on acceptable past performance of the surveillance procedures.
(E) Extension acceptability is based on scope of inspection and routine surveillance test performed during power operation that demonstrate operability.
(F) Extension acceptability is based on this surveillance being used to demonstrate EDG operability post-maintenance.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Dominion Energy Kewaunee, Inc (DEK) is requesting a one-time, temporary deferral in the performance of a limited number of technical specification (TS) surveillance requirements for the Kewaunee Power Station (Kewaunee). This request is needed to avoid a premature shutdown of Kewaunee solely to perform the surveillance requirements.

The proposed amendment would add a condition to the Operating License to extend certain Technical Specification surveillance requirement intervals, on a one-time basis. The surveillance requirement intervals would be extended up to 43 days, but no later than October 7, 2006, to permit performance during the next refueling outage. The 2006 refueling outage has been re-scheduled from April 2006 to September 2006 because the plant was in a forced outage for approximately 4 months in the spring of 2005. The affected surveillance requirements are those that cannot reasonably be performed until the next refueling outage or during a future forced outage prior to the next refueling outage. The affected surveillance requirements are:

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	Table 4.1-3	Item	4	Containment Isolation Trip - Test
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	4.5.b.2.F			Residual Heat Removal System Valve Interlocks
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.3			Diesel Generator Inspection
TS	4.6.a.4			Diesel Generator Load Rejection Test
TS	4.14			Testing And Surveillance of Shock Suppressors (Snubbers)

TS 4.17.a.2 Control Room Post Accident Recirculation System
TS 6.12.b System Integrity Program Integrated Leak Tests

10 CFR 50.91(a)(1) requires that licensee requests for operating license amendments be accompanied by an evaluation of significant hazard is posed by issuance of an amendment. DEK has evaluated whether or not a significant hazards consideration is involved with the proposed amendment using the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The requested action is a one-time extension to the performance interval of a limited number of TS surveillance requirements. The performance of these surveillances, or the failure to perform these surveillances, is not a precursor to an accident. Performing these surveillances or failing to perform these surveillances does not affect the probability of an accident. Therefore, the proposed delay in performance of the surveillance requirements in this amendment request does not increase the probability of an accident previously evaluated.

A delay in performing these surveillances does not result in a system being unable to perform its required function. In the case of this one-time extension request, the relatively short period of additional time that the systems and components will be in service before the next performance of the surveillance will not affect the ability of those systems to operate as designed. Therefore, the systems required to mitigate accidents will remain capable of performing their required function. No new failure modes have been introduced because of this action and the consequences remain consistent with previously evaluated accidents. Therefore, the proposed delay in performance of the surveillance requirements in this amendment request does not involve a significant increase in the consequences of an accident.

Therefore, operation of the facility in accordance with the proposed license amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment does not involve a physical alteration of any system, structure, or component (SSC) or a change in the way any SSC is operated. The

proposed amendment does not involve operation of any SSCs in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the one-time surveillance requirement deferrals being requested.

Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment is a one-time extension of the performance interval of a limited number of TS surveillance requirements. Extending these surveillance requirements does not involve a modification of any TS Limiting Conditions for Operation. Extending these surveillance requirements does not involve a change to any limit on accident consequences specified in the license or regulations. Extending these surveillance requirements does not involve a change to how accidents are mitigated or a significant increase in the consequences of an accident. Extending these surveillance requirements does not involve a change in a methodology used to evaluate consequences of an accident. Extending these surveillance requirements does not involve a change in any operating procedure or process.

The instrumentation and components involved in this request have exhibited reliable operation based on the results of the most recent performance of their 18-month surveillance requirements.

Based on the limited additional period of time that the systems and components will be in service before the surveillances are next performed, as well as the operating experience that these surveillances are typically successful when performed, it is reasonable to conclude that the margins of safety associated with these surveillance requirements will not be affected by the requested extension.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, Dominion Energy Kewaunee, Inc. concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The US Atomic Energy Commission (AEC) issued their Safety Evaluation (SE) of the Kewaunee Power Station (KPS) on July 24, 1972 with supplements dated December 18, 1972 and May 10, 1973. In the AEC's SE, section 3.1, "Conformance with AEC General Design Criteria," described the conclusions the AEC reached associated with the General Design Criteria in effect at the time. The AEC stated:

> *The Kewaunee plant was designed and constructed to meet the intent of the AEC's General Design Criteria, as originally proposed in July 1967. Construction of the plant was about 50% complete and the Final Safety Analysis Report (Amendment No. 7) had been filed with the Commission before publication of the revised General Design Criteria in February 1971 and the present version of the criteria in July 1971. As a result, we did not require the applicant to reanalyze the plant or resubmit the FSAR. However, our technical review did assess the plant against the General Design Criteria now in effect and we are satisfied that the plant design generally conforms to the intent of these criteria.*

As such the appropriate criteria KPS is licensed to from the Final Safety Analysis (Amendment 7), which has been updated and now titled the Updated Safety Analysis Report (USAR) is listed below.

<u>Criterion 19 - Protection Systems Reliability</u>

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

<u>Criterion 25 - Demonstration of Functional Operability of Protection Systems</u>

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

<u>Criterion 38 – Reliability and Testability of Engineered Safety Features</u>

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, 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 39 – Emergency Power for Engineered Safety Features

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Criterion 46 – Testing of Emergency Core Cooling System Components

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 47 – Testing of Emergency Core Cooling Systems

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Criterion 48 – Testing of Operational Sequence of Emergency Core Cooling Systems

A capability shall be provided to test, under conditions as close to design as practical, the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

Criterion 57 – Provisions for Testing of Isolation Valves

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Criterion 59 – Testing of Containment Pressure Reducing Systems Components

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 60 – Testing of Containment Spray Systems

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

Criterion 61 – Testing of Operational Sequence of Containment Pressure Reducing Systems

A capability shall be provided to test, under conditions as close to the design as practical, the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Criterion 63 – Testing of Air Cleanup Systems Components

Design provisions shall be made so that active components of the air cleanup systems, such as fans and damper, can be tested periodically for operability and required functional performance.

Criterion 65 – Testing of Operational Sequence of Air Cleanup Systems

A capability shall be provided to test, under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the designing air flow delivery capability.

This LAR proposes to extend the interval for testing of some of the surveillances required by Kewaunee TS. The tests themselves are not being changed and therefore the acceptance criteria listed will still be met.

In conclusion, based on the considerations discussed above, (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.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENT

A search of NRC actions on license amendments revealed several applicable precedents concerning extensions of surveillance requirement intervals. These include:

- Cooper Power Station, Docket NO. 50-298, License NO. DPR-46, License Amendment 205. The NRC staff approved the surveillance interval extensions by letter dated July 14, 2004 (Adams Accession NO. ML041960078).
- Palisades Nuclear Plant, Docket NO. 50-255, License NO. DPR-20, Licensed Amendment 206. The NRC staff approved the surveillance interval extensions by letter dated December 19, 2001 (Adams Accession NO. ML013540433).

8.0 REFERENCES

1. Generic Letter 83-37, "NUREG-0737 Technical Specifications," dated November 1, 1983.
2. Letter from Morton B. Fairtile (NRC) to D.C. Hintz (WPSC), Kewaunee License Amendment 59, dated January 9, 1985.
3. Letter from Darrell G. Eisenhut (NRC) to all PWR Licensees, dated July 2, 1980, TMI-2 Lessons Learned Category A Model Technical Specifications.
4. Letter from Steven A. Varga (NRC) to Eugene R. Mathews (WPSC), Kewaunee License Amendment 38, dated November 6, 1981.

Attachment 2

LICENSE AMENDMENT REQUEST - 219
ONE-TIME EXTENSION OF SURVEILLANCE REQUIREMENTS

MARKED UP OPERATING LICENSE PAGES

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

(6) Steam Generator Upper Lateral Supports

The design of the steam generator upper lateral supports may be modified by reducing the number of snubbers from four (4) to one (1) per steam generator.

(7) Deleted

(8) Operator Actions

The auxiliary feedwater system local manual operator actions as described in the License Amendment Request submitted May 5, 2005, and supplemented on June 9, 2005, shall be eliminated no later than completion of Kewaunee refueling outage R-29.

(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.

D. The licensee shall comply with applicable effluent limitations and other limitations and monitoring requirements, if any, specified pursuant to Section 401(d) of the Federal Water Pollution Control Act Amendments of 1972.

E. This license is effective as of the date of issuance, and shall expire at midnight on December 21, 2013.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:

Appendices A and B - Technical Specifications

Date of Issuance: December 21, 1973

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 Heat 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 Postaccident Recirculation System
6.12.b		System Integrity Program Integrated Leak Tests

Attachment 3

LICENSE AMENDMENT REQUEST - 219
ONE-TIME EXTENSION OF SURVEILLANCE REQUIREMENTS

PROPOSED OPERATING LICENSE PAGES

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

(6) Steam Generator Upper Lateral Supports

The design of the steam generator upper lateral supports may be modified by reducing the number of snubbers from four (4) to one (1) per steam generator.

(7) Deleted

(8) Operator Actions

The auxiliary feedwater system local manual operator actions as described in the License Amendment Request submitted May 5, 2005, and supplemented on June 9, 2005, shall be eliminated no later than completion of Kewaunee refueling outage R-29.

(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.

D. The licensee shall comply with applicable effluent limitations and other limitations and monitoring requirements, if any, specified pursuant to Section 401(d) of the Federal Water Pollution Control Act Amendments of 1972.

E. This license is effective as of the date of issuance, and shall expire at midnight on December 21, 2013.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

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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 Heat 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 Postaccident Recirculation System
6.12.b		System Integrity Program Integrated Leak Tests

Attachment 4

LICENSE AMENDMENT REQUEST - 219
ONE-TIME EXTENSION OF SURVEILLANCE REQUIREMENTS

4TH TEN YEAR INTERVAL. JUNE 16, 2004 – JUNE 16, 2014
REQUEST FOR RELIEF NO. RR-G-4

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

4TH TEN YEAR INTERVAL JUNE 16, 2004 – JUNE 16, 2014
REQUEST FOR RELIEF NO. RR-G-4

1. COMPONENT AFFECTED

- **Class 1 Small Bore Grinnel Hydraulic Snubber (RC-H72) located in the Containment Pressurizer Vault.**

2. ASME SECTION XI REQUIREMENTS

ASME Boiler and Pressure Vessel Code Section XI 1998 Edition through the 2000 Addenda IWF-5000 referencing ASME/ANSI OM, Part 4, 1987 Edition with OMa-1988 using the VT-3 visual examination method described in IWA-2213. ASME/ANSI OM, Part 4, Section 2.3.2 Inservice Examination Frequency, Section 2.3.2.2, Examination Intervals, requires a VT-3 visual examination to be performed at 18 month Intervals. Per Note 1: This examination period may vary in time by + or – 25% to coincide with planned outages.

3. BASIS FOR REQUESTING RELIEF

Kewaunee Power Station small bore hydraulic snubber VT-3 visual examinations were performed during the scheduled fall 2004 refueling outage. The next required VT-3 visual examinations were scheduled for performance during the refueling outage to commence on April 1, 2006. Kewaunee Power Station entered a forced shutdown on February 20, 2005, which was completed July 2, 2005. Due to the extended forced outage, the start of the next refueling outage was re-scheduled from April 1, 2006, to September 2, 2006. Based on the current requirements of ASME/ANSI OM, Part 4, Section 2.3.2.2, the next VT-3 visual examination of Kewaunee Power Station small bore hydraulic snubbers would require performance by August 30, 2006. The August 30, 2006 date will be slightly prior to current scheduled shutdown date of the Kewaunee Power Station of September 2, 2006. Note: This is Kewaunee's current schedule and as such, these dates are tentative and subject to change based other factors associated with scheduling an outage.

Performance of the VT-3 visual examination during power operation is not practical due to the inaccessibility of the small bore hydraulic snubber (RC-H72). This snubber is located in the containment pressurizer vault on a 3/4" relief vent line. Performance, in conjunction with other Class 1, Class 2 and Class 3 small bore hydraulic snubber VT-3 visual examinations located in containment during the Kewaunee Power Station forced shutdown in November 2005, was not practical for the small bore Grinnel hydraulic snubber (RC-H72) located in the containment pressurizer vault due to safety concerns of high temperature and access when at Hot Shutdown conditions.

4TH TEN YEAR INTERVAL. JUNE 16, 2004 – JUNE 16, 2014
REQUEST FOR RELIEF NO. RR-G-4

4. ALTERNATIVE METHODS OF EXAMINATION

Pursuant to 10CFR50.55a(a)(3)(i) Kewaunee Power Station requests use of the proposed alternative for implementation of ASME Boiler and Pressure Vessel Code Section XI 1998 Edition through the 2000 Addenda, IWF-5000 referencing ASME/ANSI OM, Part 4 1987 Edition with OMa-1988 using the VT-3 Visual Examination method described in IWA-2213.

Perform VT-3 Visual Examination of Kewaunee Power Station small bore Grinnel hydraulic snubber (RC-H72) located in the containment pressurizer vault during the fall 2006 refueling outage scheduled to commence September 2, 2006. The fall 2006 refueling outage will ensure radiation levels, access and temperatures will be at acceptable levels when VT-3 examinations are scheduled and performed.

Compliance with the proposed alternative will provide an adequate level of quality and safety for the VT-3 Visual Examination of Kewaunee Power Station small bore Grinnel hydraulic snubber (RC-H72). Adequate level of quality and safety is based on the short time frame (less than 30 days) between required VT-3 Visual Examinations per ASME/ANSI OM, Part 4 and actual VT-3 Visual Examinations.

5. IMPLEMENTATION SCHEDULE

Kewaunee Power Station Fourth Ten Year Interval June 16, 2004 – June 16, 2014.