TABLE 4.1-3

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Equipment Tests***	Test	Frequency	Maximum Time Between Tests (Days)
1.	Control Rods	Rod drop times of all full length rods	Each refueling outage	N.A.
		Partial movement of all rods	Every 2 weeks	17
1a.	Reactor Trip Breakers	Open trip	Monthly	37
1b.	Reactor Coolant Pump Breakers-Open-Reactor Trip	Operability	Each refueling outage	N.A.
2.	Deleted			
3.	Deleted			25C
4.	Containment Isolation Trip	Operability	Each refueling outage	N.A.
5.	Refueling System Interlocks	Operability	Prior to each refueling outage	N.A.
6.	Ventilation System a. Shield Building b. Auxiliary Building SV Zone c. Spent Fuel Pool	Halide, DOP and Methyl Iodide Pressure Drop Test Visual Inspection	During each refueling outage except as specified in Note**	N.A.
7.	Fire Protection Pump and Power Supply	*Operability	Monthly	37
8.	Containment Leak Detect	Operability	Weekly	8
9.	Diesel Fuel Supply	*Fuel inventory	Weekly	8
10.	Turbine Stop and Governor Valves	Operability	Monthly(1)(2)	37(1)(2)
11.	Fuel Assemblies	Visual Inspection	Each refueling outage	N.A.
12.	Guard Pipes	Visual Inspection	Each refueling outage	N.A.

NOTES:

*See Specification 4.1.d

**Tests and frequency shall be in accordance with Specifications 4.4.d and 4.12.

***Following maintenance on the above equipment that could affect the operation of the equipment tests should be performed to verify operability.

(1) This test may be waived for end of cycle operations when boron concentrations are less than 150 ppm, due to operational limitations.

(2) A one-time extension of the frequency interval from monthly to a period of four months is granted, said extended period to end April 30, 1981. After April 30, 1981, monthly tests will again be required.

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4.2 REACTOR COOLANT SYSTEM IN-SERVICE INSPECTION

Applicability

Applies to in-service structural surveillance of the reactor coolant system boundary and functional testing of safety related pumps and valves associated with the reactor coolant system.

Objective

To assure the continued integrity of the reactor coolant system boundary and performance of safety related mechanical equipment associated with the reactor coolant system.

Specification

4.2.1 Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through the summer of 1975 as required by 10CFR50, Section 50.55a(g), except where such inspections are documented as being impractical and specific relief has been requested of the NRC pursuant to 10CFR50 Section 50.55a (g)(6)(i). Following formal response by the NRC to the relief request the inspection shall be in accordance with the relief as granted. Tables TS 4.2-1 through TS 4.2-3 specify the tests applicable to the Kewaunee Plant and the Code relief which has been granted.

4.2.2 Inservice testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through the summer of 1975 as required by 10CFR50 Section 50.55a(g), except where nuclear safety would be compromised by such tests, specific Technical Specification requirements address performance testing of

such equipment, or specific exemption has been requested of the NRC pursuant to 10CFR50 Section 50.55a(g)(6)(i). Following formal response by the NRC to the relief request the testing shall be in accordance with the relief as granted. Tables TS 4.2-4 through 4.2-5 specify the tests applicable to the Kewaunee Plant and Code relief which has been granted.

4.2.3 Steam Generator Tubes

Examinations of the steam generator tubes shall be in accordance with the in-service inspection program described herein. The following terms are defined to clarify the requirements of the inspection program.

Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

% Degradation is an estimated % of the tube wall thickness affected or removed by degradation.

Degraded Tube means a tube containing imperfections $\geq$ 20% of the nominal wall thickness caused by degradation.

Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

Tube Inspection means an inspection of the steam generator tube from the point of entry (e.g., hot leg side) completely around the U-bend to the top support of the opposite leg (cold leg).

1. Steam Generator Sample Selection and Inspection - The in-service inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in the single inspected generator, provided the previous inspections indicated that the two steam generators are performing in a like manner.

2. Steam Generator Tube Sample Selection and Inspection - The tubes selected for each in-service inspection shall:

a. Include at least 3% of the total number of tubes in all steam generators. The tubes selected for these inspections shall be selected on a random basis except as noted in 4.2.3.2.b.

b. Concentrate the inspection by selection of at least 50% of the tubes to be inspected from critical areas where experience in similar plants with similar water chemistry indicates higher potential for degradation.

c. Include the inspection of all non-plugged tubes which previous inspections revealed in excess of 20% degradation. The previously degraded tubes need only be inspected about the area of previous degradation indication if their inspection is not employed to satisfy 4.2.3.2.a and 4.2.3.2.b above.

d. The second and third sample inspections during each inservice inspection may be less than the full legnth of each tube by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

e. If a tube does not permit the passage of the eddy current inspection probe the entire length and through the U-band, this shall be recorded and an adjacent tube shall be inspected. The tube which did not allow passage of the eddy current probe shall be considered degraded.

The results of each sample inspection shall be classified into one of the following three categories.

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTE: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

3. Inspection Frequencies - The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

   a. Inservice inspections shall be performed at refueling intervals not more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

   b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.2-6 fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply

until a subsequent inspection meets the conditions specified in 4.2.3.3.a and the interval can be extended to a 90 month period.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.2-6 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.d and 3.4.a.5,
2. A seismic occurrence greater than the Operating Basis Earthquake,
3. A loss-of-coolant accident requiring actuation of the engineered safeguards, where the cooldown rate of the reactor coolant system exceeded 100°F/hr, or
4. A main steam line or feedwater line break, where the cooldown rate of the reactor coolant system exceeded 100°F/hr.

d. If the type of steam generator chemistry treatment is changed significantly, the steam generators shall be inspected at the next refueling following 3 months of power operation since the change.

4. Any tube which exhibits one or more of the following conditions shall be plugged prior to returning the steam generator to service:

a. Tube leak.

b. Tube wall degradation of 50% or more. If significant general tube thinning occurs this criteria will be reduced to 40% wall penetration.

NOTE: Second sentence was not included in P.A. but was added on in Amendment No. 18.

5. Reports

a. Following each inservice inspection of steam generator tubes, if there are any tubes requiring plugging, the number of tubes plugged shall be reported to the Commission within 15 days.

b. The results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.
2. Location and percent of wall-thickness penetration for each indication of a degradation.
3. Identification of tubes plugged.

c. Results of steam generator tube inspection which fall into Category C-3 and require a prompt notification of the Commission shall be reported pursuant to Specification 6.9.2.a. prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Basis

The inspection program will be in accordance with the requirements of Section XI of ASME Code 1974 Edition and Addenda through the summer of 1975 per the requirements of 10CFR50 Section 50.55a(g). The Examination requirements will be met to the extent practical through limitations on component configuration, accessibility and material composition.

The plant was not specifically designed to meet all the requirements of Section XI of the code; therefore, 100 percent compliance may not be feasible or practical.

However, access for in-service inspection was considered during the design, and modifications have been made where practical to make provision for maximum access within the limits of the current plant design.

The Reactor Coolant System was initially free of gross defects, and the system has been designed such that gross faults or defects should not occur throughout the plant lifetime. The ten-year surveillance program will reveal possible fault areas before any leak develops, should such problems actually occur.

The inservice testing requirements of Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through the summer of 1975 were developed following commercial operation of the Kewaunee Plant. The Technical Specifications address specific testing requirements of safety related equipment including pumps and valves, which were evaluated to be adequate by the NRC in granting the Operating License. The ASME Code does not take precedence over these established surveillance requirements of the Specifications and the methodology of those Technical Specification required tests.

Additional tests of equipment required to be listed by Section XI 1974 Edition and Addenda through the summer of 1975 and not addressed in the Technical Specification shall not compromise nuclear safety. Specific exemption will be provided by the NRC for those component tests which are impractical, compromise safety, or whose performance would require modification of the facility which degrades overall safety or is not cost-effective in light of 10CFR50 Section 50.109(g) considerations.

Tables TS 4.2-1 through TS 4.2-5 identify the specific tests which are applicable to the Kewaunee Plant. The tables clarify the requirements of the inservice inspection program and tabulate the exceptions to Section XI of the ASME Boiler

and Pressure Vessel Code. Each component to be inspected is addressed in the tables where the applicable safety class Code specifications and examination method is addressed. Where specific relief from the Code requirements is necessary, the justification for relief and alternate inspection method are included.

The following provide further clarification concerning the Class 1, 2, and 3 systems inspection programs.

(a) Articles IWC-3000 and IWD-3000 entitled, "Evaluation of Examination Results" are in the course of preparation by the Code Committee and, as yet, are not available for use. The rules of IWA-3000 will be utilized where applicable. The evaluation of any indications detected during any inservice examinations will be made using the acceptance standards for materials and welds specified in the code under which the specific component was constructed for those components not constructed in accordance with the rules of Section III.

(b) Articles IWC-4000 and IWD-4000 entitled, "Repair Procedures" state that the rules of IWB-4000 shall apply. It is considered that the repair procedures outlined in IWB-4000 are inappropriate for the Class 2 and 3 components in this program and the rules of IWA-4000 will be applied.

(c) Requirements for the visual examination of Class 1 systems and components for evidence of leakage during the performance of a system pressure test following each refueling are identified by IWB-5200. Exception is taken to the implementation of these requirements on those portions of Class I systems which are contained between two check valves or where pressure applied to the reactor coolant system will be retained at the first valve in the line. The portions of systems affected by this limitation are:

(i) Cold leg injection from accumulators between check valves SI22 A and B (8840A and B) and SI21 A and B (8841A and B) and test line to valves SI201A (8824A) and SI201B (8825B) and RHR return line valve RHR-11 (8703).

(ii) Cold leg high head injection between check valves SI 13 A and B (8842A and B) and SI 12 A and B (SI 118-1 and 118-3).

(iii) Reactor Vessel injection between check valves SI 304 A and B (8844A and B) and SI 303 A and B (8843A and B) and SI 16 A and B (SI 118-2 and 118-4).

(d) Subsections IWB and IWC contain differing requirements for the hydrostatic testing of Class 1 and Class 2 systems and components. The implementation of these requirements is impractical when the only means of pressurizing the Class 2 system is through the Class 1 system or when the boundary between the two systems is a check valve arranged for flow from Class 2 to the Class 1 system. Exception is taken to the performance of the hydrostatic test requirements as required by Article IWC-2412 (a) on those portions of the Class 2 systems identified below. Visual examination for evidence of leakage will be conducted on these portions of the systems at the system nominal operating pressure in accordance with the requirements of IWB-5221 for the adjoining Class 1 system.

(i) R. C. pump seal bypass line from the orifice to CVC250 (AOV 8145).

(ii) R. C. pump seal leak off line to manually operated valves CVC 207 A and B (8148A and B).

(iii) R. C. pump seal injection line from check valve CVC205 A and B (CS 100-1 and 2) to manually operated valves CVC204 A and B (CS 7-1 and 2).

(iv) Charging line control valve by-pass line from check valve CVC 14 (CS 102-5) to manually operated valve CVC 13 (CS 101-24).

(v) Letdown line from valve LD3 (LCV 428) to orifice outlet valves LD4 A and B (8140 A and B) and LD4C (8141).

(vi) Pressurizer steam space sampling line from valve RC402 (9999A) to RC403 (SS13-5), pressurizer liquid space sampling line from valve RC412 (9999B) to RC413 (SS13-6) and loop sampling line from valve RC422 (9999C) to valve RC423 (SS13-7).

The potential for inadvertent overpressurization of the reactor coolant system causes additional concerns on the advisability of pressurizing Class 2 systems to considerably higher pressures than the adjacent Class 1 system and relief is requested from implementing the hydrostatic test requirements of IWC-2412(a) on the CVCS system where such potential exists. The chemical and volume control charging, seal injection and letdown systems are in continuous operation during normal plant operation and are continuously monitored to ensure continued integrity and performance.

(e) The examination requirements for Class 3 systems and components included in Table TS 4.2-3 are in accordance with IWD-2410(c) which specifies that 100 percent of the components be examined as required by IWA-5240 and IWD-2600 either during normal operation or during system inservice testing. An additional requirement of IWD-2410(b) is for the examination of Class 3 systems and components for evidence of leakage during the performance of a system pressure test in accordance with IWD-5000. The code does not stipulate that certain amounts of these examination requirements be completed within each 40-month period such that the system pressure test requirements may be deferred until the end of the ten year inspection interval. However, it should be noted, that these system pressure tests when required are impractical in those systems, such as component cooling, service water, spent fuel pit cooling, and boric acid transfer and recirculation, which are in continuous operation during all modes of plant operation. The continuous functional operation serves to demonstrate the structural and leak-tight integrity of these systems. Visual examinations of these systems will be performed at normal operation pressures to verify leak-tightness.

Tables TS 4.2-4 and TS 4.2-5 provide a listing of the ASME Code Class 1, 2, and 3 pumps and valves subject to the testing requirements of Subsections IWP and IWV of the ASME Boiler and Pressure Vessel Code, 1974 Edition, and Addenda through Summer, 1975.

The tabulation of pumps identifies the pumps to be tested, pump code classes, parameters to be measured and test intervals.

Similarly the tabulation of valves identifies the valves to be tested, Valve code classes, Section XI Category as defined by IWV-2000, and test frequencies. In both tabulations, relief from the testing requirements of Section XI is requested in cases where these requirements have been determined to be impractical. Where relief is requested, specific information is provided which identifies the applicable code requirement, justification for the relief request, and testing method to be used as an alternative. The pump and valve testing programs have been reviewed to ensure that testing the component at the specified interval will not place the plant in an unsafe condition.

The requirements for pump performance as currently specified in other sections of the technical specification is that they reach the required developed head at miniflow. Relief is requested from pump performance parameters of head and flow from having to meet the allowable ranges of test quantities as defined in Table IWP-3100-2 with the substitution of the tolerances of being no less than 10 percent of the performance curve as allowed by IWP-3210.

The Kewaunee plant systems as designed and installed do not contain provisions for the measurement of seat leakage as required by IWV-3420(d) during the performance of the valve leak rate test. Relief is requested from meeting this requirement. Integrated leakage of the containment isolation systems will be measured, as required by 4.4a of the technical specification.

Where valve actuation or leak rate tests are currently covered and documented by existing requirements of the plant technical specifications, relief is requested from maintaining separate additional records to meet the requirements of IWV-6000.

Included in the tabulation of valves are normally closed check valves for which operation will be verified at each refueling outage. Operation will be verified by establishing flow in each individual line. Flow will be verified as allowed by IWV-3520(b)(2), either by observation of installed flow instrumentation or by observing that flow is established into the reactor coolant system when opened up for refueling.

Normally closed category A and B valves and normally closed check valves which cannot be exercised at three month intervals are identified as being operated at each refueling outage in accordance with the requirements of IWV-3410(b)(1) and IWV-3520(b).

Manual valves or power operated valves that fail as is on loss of power and do not have to change position to perform their intended safety function are excluded from testing by IWV-1300 and are not included in Table 4.2-5. These valves are assured in their proper safeguard position by procedures, checklists, and/or periodic verification.

The inservice inspection and testing programs outlined in the attached tabulations have been developed as a result of a design review. Should certain ASME Section XI Code requirements be discovered to be impractical due to unforeseen reasons during the process of performing inspections or tests, relief will be requested from the specific Section XI Code requirement at that time.

Radiation levels in certain areas or of certain components may be found to prohibit the access for operators or inspectors to perform the inspections or tests described in this program. If source strengths cannot be reduced such that the personnel dose per weld inspected is less than 0.25 Man Rem for Class 2 and 3 inspections, the inspection will not be performed. Additional relief will be requested from the specific Section XI Code requirements if radiation doses are determined to be excessive.

The Suveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on the general guidance or Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

It is WPS policy to operate in a manner such that the secondary coolant will be maintained within those chemistry conditions found to result in negligible corrosion of the steam generator tubes. The allowable defects during plant operation would be limited by the limitation of steam generator tube leakage between the reactor coolant system and the secondary coolant system (reactor coolant-to-secondary leakage in excess of the limits of Specification 3.1.d). Steam generators having leakage less than these limits during operation are judged to have an adequate margin of safety between tube inspection intervals to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor coolant-to-secondary leakage of Specification 3.1.d can readily be detected by the secondary coolant radiation monitors. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exeeding the plugging limit. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect 20% tube degradation.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.2.

Table TS 4.2-1
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 1 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
B1.1	B-A	Reactor Vessel	III-A	Upper to Intermediate Shell Course Circumferential Weld	Volumetric	No
B1.1	B-A			Intermediate to Lower Shell Course Circumferential Weld	Volumetric	No
B1.1	B-A			Lower Head to Shell Circumferential Weld	Volumetric	No
B1.2	B-B			Lower head ring to disc circumferential weld	Volumetric	No
B1.3	B-C			Flange to Vessel Weld	Volumetric	No
B1.3	B-C			Closure Head to Flange Weld	Volumetric	No
B1.4	B-D			Outlet Nozzle to Shell Welds (2)	Volumetric	No
B1.4	B-D			Inlet Nozzle to Shell Welds (2)	Volumetric	No
B1.5	B-E			CRDM, Vent and In-Core Instrumentation penetrations and CRDM seal welds	Visual	No
B1.6	B-F			Primary Nozzle to Safe-End Welds	Volumetric & Surface	No
B1.7	B-G-1			Closure Studs (In Place)	NA	No - Note 1
B1.8	B-G-1			Closure Studs and Nuts and Surface	Volumetric	No

Table TS 4.2-1
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 1 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
B1.9	B-G-1			Vessel Flange Ligaments	Volumetric	No
B1.10	B-G-1			Closure Washers	Visual	No
B1.11	B-G-2			Conoseal Bolting	Visual	No
B1.12	B-H			Integrally Welded Supports	Volumetric	No - Note 2
B1.13	B-J-1			Closure Head Cladding	Volumetric	No - Note 3
B.1.14	B-J-1			Vessel Cladding	Visual	No
B1.15	B-N-1			Vessel Interior Surfaces and Internals	Visual	No
B1.16	B-N-2			Interior Attachments and Core Support Structures	NA	No - Note 4
B1.17	B-N-3			Core Support Structures	Visual	No
B1.18	B-O			Control Rod Drive Housings	Volumetric	No
B1.19	B-P			Exempted Components	Visual	No
B2.1	B-B	Pressurizer	III-A	Circumferential Shell Welds (3)	Volumetric	No
B2.1	B-B			Longitudinal Shell Welds (2)	Volumetric	No
B2.2	B-D			Nozzle to Vessel Welds (6)	NA	No - Note 5
B2.3	B-E			Heater Penetrations	Visual	No

Table TS 4.2-1
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 1 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
B2.4	B-F			Nozzle to Safe-End Welds (6)	Surface and Volumetric	No
B2.5	B-G-1			Pressure Retaining Bolting (In place)	NA	No - Note 6
B2.6	B-G-1			Pressure Retaining Bolting	NA	No - Note 6
B2.7	B-G-1			Pressure Retaining Bolting	NA	No - Note 6
B2.8	B-H			Integrally Welded Support	Volumetric	No
B2.9	B-I-2			Vessel Cladding	Visual	No
B2.10	B-P			Exempted Components	Visual	No
B2.11	B-G-2			Manway Bolting	Visual	No
B3.1	B-B	Steam Generators (2) Primary Side	III-A	Channel Head to Tubesheet Weld (2)	Volumetric	No
B3.2	B-D			Nozzle to Vessel Welds	NA	No - Note 7
B3.3	B-F			Nozzle to Safe End Welds (4)	Volumetric & Surface	Yes - Note 8
B3.4	B-G-1			Pressure Retaining Bolting (In Place)	NA	No - Note 6
B3.5	B-G-1			Pressure Retaining Bolting	NA	No - Note 6

Table TS 4.2-1
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 1 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
B3.6	B-G-1			Pressure Retaining Bolting	NA	No - Note 6
B3.7	B-H			Integrally Welded Supports	NA	No - Note 6
B3.8	B-I-2			Vessel Cladding	Visual	No
B3.9	B-P			Exempted Components	Visual	No
B3.10	B-G-2			Manway Bolting	Visual	No
B4.1	B-F	Piping Pressure Boundary		Safe End to Pipe Welds	Volumetric and Surface	Yes - Note 8
B4.2	B-G-1			Pressure Retaining Bolting (In Place)	NA	No - Note 6
B4.3	B-G-1			Pressure Retaining Bolting	NA	No - Note 6
B4.4	B-G-1			Pressure Retaining Bolting	NA	No - Note 6
B4.5	B-J			Circumferential and Longitudinal Pipe Welds	Volumetric	Yes - Notes 9 and 10
B4.6	B-J			Branch Pipe Connection Welds exceeding 6-inch diameter	Volumetric	Yes - Note 11
B4.7	B-J			Branch Pipe Connection Welds 6-inch diameter and smaller	Surface	No
B4.8	B-J			Socket Welds	Surface	No
B4.9	B-K-1			Integrally Welded Supports	Volumetric	Yes - Note 12

Table TS 4.2-1
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 1 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
B4.10	B-K-2			Support Components	Visual	No
B4.11	B-P			Exempted Components	Visual	No
B4.12	B-G-2			Pressure Retaining Bolting	Visual	No
B5.1	B-G-1	Reactor Coolant Pumps		Main Flange Bolting	Volumetric	No
B5.1	B-G-1			Seal Housing Bolting	Volumetric	Yes - Note 13
B5.2	B-G-1			Main Flange and Seal Housing Bolting	Volumetric and Surface	No - Note 14
B5.3	B-G-1			Pressure Retaining Bolting	Visual	No
B5.4	B-K-1			Integrally Welded Supports	NA	No - Note 6
B5.5	B-K-2			Support Components	Visual	No
B5.6	B-L-1			Pump Casing Welds	Volumetric	No - Note 15
B5.7	B-L-2			Pump Casings	Visual	No
B5.8	B-P			Exempted Components	Visual	No
B5.9	B-G-2			Pressure Retaining Bolting	NA	No - Note 6
B6.1	B-G-1	Valve Pressure Boundary		Pressure Retaining Bolting (In Place)	NA	No - Note 6
B6.2	B-G-1			Pressure Retaining Bolting	NA	No - Note 6
B6.3	B-G-1			Pressure Retaining Bolting	NA	No - Note 6

Table TS 4.2-1
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 1 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
B6.4	B-K-1			Integrally Welded Supports	NA	No - Note 6
B6.5	B-K-2			Support Components	Visual	No
B6.6	B-M-1			Valve Body Welds	NA	No - Note 6
B6.7	B-M-2			Valve Bodies	Visual	No
B6.8	B-P			Exempted Components	Visual	No
B6.9	B-G-2			Pressure Retaining Bolting	Visual	No

ASME CODE CLASS 1 COMPONENTS

NOTES

1. The reactor vessel closure studs are removed during each refueling and there will be no need for examination in place as required by IWB-2600.

2. The reactor vessel has six supports, four of which are pads integral with the four primary nozzles and are excluded from examination requirements of IWB-2500 by Category B-H. The attachment welds and the vessel wall base metal beneath the two bracket supports will be examined from inside the vessel when the core barrel is removed at the end of the ten year interval.

3. Radiation levels beneath the closure head are 15 to 30 R/hr at 6 inches from the clad surface. The time required for access, cleaning and both visual and surface examination of two 36 square inch cladding patches could take a total of 15 minutes such that the examiner (or examiners) could be subjected to a total accumulated dose of up to 7.5 man Rem. The presence of the cladding material is not considered when performing the stress analyses of the vessel such that failure of the cladding would not effect the integrity of the vessel. The closure head cladding will be volumetrically examined concurrent with the examinations performed on the closure head to flange weld.

4. This requirement of IWB-2600 is applicable only to Boiling Water type reactors.

5. The pressurizer nozzles are integrally cast with the vessel heads and there are no welds requiring examination in accordance with the requirement of IWB-2600.

6. There are no items in this category on this component in the Kewaunee plant Class 1 systems.

7. The steam generator nozzles are integrally cast with the channel head and there are no welds in this category.

8. Examination of the steam generator primary nozzle to safe-end and safe-end to pipe weld is limited both by the nozzle geometry and surface condition and the limited surface preparation on the pipe side of the weld. The surface on the pipe side of the weld, which is a cast elbow, is machined for a distance of approximately 3 inches from the edge of the weld. Ultrasonic examination is limited to this distance from the edge of the weld. Examinations can be performed on the surface of the weld but are severely limited from the nozzle side by the rough, as cast surface. Surface examinations can be performed on 100 percent of the weld and the base metal on the pipe side.

9. Limitations may occur for the examination of piping system circumferential butt welds (Category B-J) when the welds occur at geometric discontinuities such as pipe to vessel welds, pipe to fitting welds or fitting to fitting welds. For pipe to fitting or pipe to vessel nozzle welds, examinations can be performed to the extent required by T-532 of Section V from the weld and pipe surfaces. Examination from the fitting side would be dependent upon the geometric configuration. Where elbows or tees are concerned, examination can be performed from the fitting side except where the intrados of the fitting prevents adequate ultrasonic coupling. No examinations can be performed from the fitting side when it is a valve or a flange. In all cases 100 percent of the weld material can be examined. In instances where welds occur at fitting to fitting, access restrictions as outlined above occur on both sides of the weld. In instances where ultrasonic examinations cannot be performed on 100 percent of the volume of the weld and heat affected zone, surface examinations will be performed to supplement the limited volumetric examination.

Welds in the Kewaunee Plant, Class 1 systems which due to limitations, would require surface examination are:

(a) Loop A Accumulator Discharge Line; Weld #5.
(b) Loop A RHR Take-off Line; Weld #11.
(c) Loop A High Head SIS Line; Weld #3.
(d) Loop A Pressurizer Spray Line; Weld #5.
(e) Loop A RTD Return Line; Weld #3.

(f) Loop B Accumulator Discharge Line; Weld #4.

(g) Loop B RHR Take-off Line; Weld #9.

(h) Loop B High Head SIS Line; Weld #3.

(i) Loop B Pressurizer Spray Line; Weld #5.

(j) Loop B RTD Return Line; Weld #2.

(k) Pressurizer Safety Valve PR3A (8010A) Line; Welds #4, 7 & 8.

(l) Pressurizer Safety Valve PR3B (8010B) Line; Welds #5, 8 & 9.

(m) Pressurizer Relief Line, Welds *9 & 13.

In instances where the locations of pipe supports or hangers restrict the access available for the examination of pipe welds as required by IWB-2600, examinations will be performed to the extent practical unless removal of the support is permissible without unduly stressing the system. Welds in the Kewaunee Plant, Class 1 systems which due to the location of supports, would require surface examination are:

(a) Loop A Accumulator Discharge Line; Weld #6

(b) Loop A RHR Take-off Line; Weld #8

(c) Loop A Pressurizer Spray Line; Weld #9.

(d) Loop B RHR Take off Line; Weld #7.

(e) Loop B Accumulator Discharge Line; Weld #12.

(f) Loop B RHR Return Line, Welds #6 & 7.

(g) Pressurizer Relief Line, Weld #12.

10. The 90 degree elbows in the crossover leg of the reactor coolant system are fabricated in two halves from austenitic stainless steel castings welded together by the electroslag process.

The structure and nature of the electroslag weld in the cast austenitic 90 degree elbows is such that the material is opaque to ultrasonic transmissions utilizing currently available techniques. Radiography is the only other available technique for volumetric examination. It is not possible to obtain code acceptable radiographs with double wall "shots" on these components which are approximately 38 inches diameter, 3 1/2 inches wall thickness, containing a 2 inch thick splitter plate and having radiation levels of up to 300 mr/hour on contact. Surface examination could be performed on these welds.

11. The configuration of the reactor coolant branch nozzle connection welds is such that ultrasonic examinations cannot be performed on the surface of the weld. Examinations will be performed to the extent practical from the pipe and nozzle surfaces adjacent to the weld. Surface examination of the weld will be performed to supplement the volumetric examination.

12. The piping system integrally welded supports are attached to the pipe by fillet welds. The configurations of such welds is such that examinations cannot be performed to the extent required by IWB-2600 and only the base material of the pipe wall can be examined by ultrasonic techniques. Surface examination will be performed on the integrally welded attachments to supplement the limited volumetric examination.

13. The reactor coolant pump seal housing bolts are of the socket head type and the configuration is such that ultrasonic examinations as required by IWB-2600 cannot be performed when the bolting is in place. Examinations will be performed to the extent required by IWB-2600 when the seal housing is disassembled for maintenance.

14. The reactor coolant pump main flange bolting is ultrasonically examined in place in accordance with the requirements of IWB-2600, Item B5.1. Both main flange and seal housing bolting will be examined as required by IWB-2600, Item B5.2 whenever a pump is disassembled for maintenance or at the end of the ten year interval when a pump will be disassembled for the performance of examinations required by Category B-L-1.

15. One of the reactor coolant pump casings in the Kewaunee plant is fabricated from two heavy wall austenitic steel casting joined together by a weld formed by the electroslag process. The structure and nature of this material is such that is opaque to ultrasonic transmission.

    Volumetric examinations as required by IWB-2600 will be attempted utilizing radiographic techniques. The success of these examinations will be dependent upon the availability of high energy gamma sources and the level of background radiation. Internal fittings in the pump may also provide restriction to the extent of examination that can be performed.

Table TS 4.2-2
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 2 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
C1.1	C-A	Letdown Heat Exchanger (Tube Side)	III-C	Head to Shell Weld	Volumetric	No
C1.1	C-A			Shell to Flange Weld	Volumetric	No
C1.2	C-B			Nozzle to Vessel Weld	NA	No - Note 1
C1.3	C-C			Integrally Welded Supports	Surface	No
C1.4	C-D			Tubesheet Flange Bolting	NA	No - Note 1
C1.1	C-A	Regenerative Heat Exchngr	III-C	Head to Shell Welds (6)	Volumetric	Yes - Note 2
C1.1	C-A			Shell to Tubesheet Welds (6)	Volumetric	Yes - Note 2
C1.2	C-B			Nozzle to Vessel Welds (12)	NA	No - Note 1
C1.3	C-C			Integrally Welded Supports	NA	No - Note 1
C1.4	C-D			Pressure Retaining Bolting	NA	No - Note 1
C1.1	C-A	Residual Heat Exchngr (2) (Tube Side)	III-C	Head to Shell Welds	Volumetric	No
C1.1	C-A			Shell to Flange Welds	Volumetric	No
C1.2	C-B			Nozzle to Vessel Welds	Volumetric	Yes - Note 3

Table TS 4.2-2
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 2 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
C1.3	C-C			Integrally Welded Supports	Surface	No
C1.4	C-D			Tubesheet Flange Bolting	Visual and Volumetric	No
C1.1	C-A	Seal Water Return Filter	III-C	Cover Weldment to Shell Weld	Volumetric	Yes - Note 4
C1.1	C-A			Head to Shell Weld	Volumetric	Yes - Note 4
C1.2	C-B			Nozzle to Vessel Weld	NA	No - Note 1
C1.3	C-C			Integrally Welded Supports	Surface	No
C1.4	C-D			Pressure Retaining Bolting	NA	No - Note 1
C1.1	C-A	Volume Control Tank	III-C	Upper Head to Shell Weld	Volumetric	No
C1.1	C-A			Lower Head to Shell Weld	Volumetric	No
C1.2	C-B			Nozzle to Vessel Weld	NA	No - Note 1
C1.3	C-C			Integrally Welded Supports	Surface	No
C1.4	C-D			Manway Bolting	Visual and Volumetric	No
C1.1		Charging Pump Surge Vessel	VIII Div.I	Head to Shell Welds (2)	Volumetric	No

Table TS 4.2-2
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 2 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
C1.2				Nozzle to Vessel Welds	NA	No - Note 1
C1.3				Integrally Welded Supports	NA	No - Note 1
C1.4				Pressure Retaining Bolting	NA	No - Note 1
C1.1	C-A	Seal Water Injection Filter (2)	III-C	Shell to Flange Weld	Volumetric	No
				Head to Shell Weld	Volumetric	No
C1.2	C-B			Nozzle to Vessel Welds	NA	No - Note 1
C1.3	C-C			Integrally Welded Supports	Surface	No
C1.4	C-D			Tubesheet Flange Bolting	Visual and Volumetric	No
C1.1	C-A	Steam Generators (2) (Shell Side)	III-A	Upper Head to Shell Weld	Volumetric	No
C1.1	C-A			Upper Shell to Transition Weld	Volumetric	No
C1.1	C-A			Transition to Lower Shell Weld	Volumetric	No
C1.1	C-A			Lower Shell to Stub Barrel Weld	Volumetric	No
C1.1	C-A			Stub Barrel to Tubesheet Weld	Volumetric	No
C1.2	C-B			Steam Outlet Nozzle to Shell Weld	Volumetric	No

Table TS 4.2-2
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 2 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
C1.2	C-B			Feedwater Inlet Nozzle to Shell Weld	Volumetric	No
C1.3	C-C			Integrally Welded Supports	NA	No - Note 1
C1.4	C-D			Manway Bolting	Visual and Volumetric	No
C1.1	C-A	Reactor Coolant Filter	III-C	Cover Weldment to Shell Weld	Volumetric	Yes - Note 4
C1.1	C-A			Head to Shell Weld	Volumetric	Yes - Note 4
C1.2	C-B			Nozzle to Vessel Weld	NA	No - Note 1
C1.3	C-C			Integrally Welded Supports	Surface	No
C1.4	C-D			Pressure Retaining Bolting	NA	No - Note 1
C2.1	C-F; C-G	Piping Systems		Circumferential Butt Welds	Volumetric	Yes - Note 5
C2.2	C-F; C-G			Longitudinal Weld Joints in Fittings	Volumetric	No
C2.3	C-F; C-G			Branch Pipe to Pipe Welds	Volumetric	Yes - Note 6
C2.4	C-D			Pressure Retaining Bolting	Visual and Volumetric	No
C2.5	C-E-1			Integrally Welded Supports	Surface	No

Table TS 4.2-2
KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 2 COMPONENTS

TABLE IWB-2600 ITEM NO.	TABLE IWB-2500 EXAMINATION CATEGORY	SYSTEM OR COMPONENT	CODE APPLICABLE TO CONSTRUCTION	AREA TO BE EXAMINED	EXAMINATION REQUIREMENT	SECTION XI CODE RELIEF REQUESTED
C2.6	C-E-2			Support Components	Visual	No
C3.1	C-F	Residual Heat Removal Pumps (2)		Pump Casing Welds	NA	No - Note 1
C3.2	C-D			Pressure Retaining Bolting	Visual and Volumetric	No
C3.3	C-E-1			Integrally Welded Supports	NA	No - Note 1
C3.4	C-E-2			Support Components	Visual	No
C3.1	C-F	Charging Pumps (3)		Pump Casing Welds	NA	No - Note 1
C3.2	C-D			Pressure Retaining Bolting	Visual and Volumetric	No
C3.3	C-E-1	Charging Pumps (3)		Integrally Welded Supports	NA	No - Note 1
C3.4	C-E-2			Support Components	Visual	No
C4.1	C-F; C-G	Valves		Valve Body Welds	NA	No - Note 1
C4.2	C-D			Pressure Retaining Bolting	Visual and Volumetric	No
C4.3	C-E-1			Integrally Welded Supports	NA	No - Note 1
C4.4	C-E-2			Support Components	Visual	No

ASME CODE CLASS 2 COMPONENTS

NOTES

1. There are either no items in this category or items are excluded from Examination requirements of IWC-2600 by application of criteria given in IWC-1220. The specific details of exclusions are:
   (a) Letdown heat exchanger inlet and outlet nozzles are 2 inch diameter, tubesheet flange bolting is 7/8 inch diameter.
   (b) Regenerative heat exchanger inlet and outlet nozzles are 2 inch diameter, there are no integrally welded supports or pressure retaining bolting on this vessel.
   (c) Seal water return filter inlet and outlet nozzles are 2 inch diameter, cover flange bolting is 3/4 inch diameter.
   (d) Volume control tank inlet and outlet nozzles are 4, 3 and 2 inch diameter.
   (e) Charging pump surge vessel inlet nozzle is 3 inch diameter, there are no integrally welded supports or pressure retaining bolting on this vessel.
   (f) Seal water injection filter inlet and outlet nozzles are 2 inch diameter.
   (g) There are no integrally welded supports on the steam generators.
   (h) Reactor coolant filter inlet and outlet nozzles are 2 inch diameter and cover flange bolting is 3/4 inch diameter.
   (i) Residual heat removal pump casings have no pressure containing welds or integrally welded supports.
   (j) Charging pump casinsg have no pressure continaing welds or integrally welded supports.
   (k) There are no Class 2 valves with pressure containing welds or integrally welded supports.

   Portions of Class 2 systems and components that are exempt from the examination requirements of IWC-2520 by IWC-1220 are as follows:
   (a) CVCS Piping equal to or less than 4-inch nominal diameter is exempted by IWC-1220 (d).

(b) During plant operation, the contents of the SIS accumulators are normally sampled on a regular basis and this component and associated piping would be exempted by IWC-1220(c).

(c) The high head SIS piping is all equal to or less than 4-inch nominal diameter and exempted by IWC-1220(d).

(d) During plant operation, the high head SIS injection pumps are run on a periodic basis to recirculate flow to and from the RWST. Samples taken on a regular basis from the RWST would verify the chemistry of the system fluid and the pump and associated suction piping would be exempt from examination by IWC-1220(c).

(e) The containment spray system does not function during normal reactor operation and is exempted by IWC-1220(b).

2. The regenerative heat exchanger is a three pass vessel, having a total of six head to shell welds and six shell to tubesheet welds. Radiation levels adjacent to this heat exchanger are between 6 and 7 R/hr. The total time required for erection of scaffolding, removal of insulation, cleaning, performing examinations and restoration insulation could take a total of 3 to 4 hours for the examination of an 1/2 inch long portion of each of 12 welds in this category, such that personnel involved could be subjected to a total accumulated dose of up to 56 man Rem. It is felt that the potential personnel exposure to complete these examinations is excessive particularly when the examination is to establish the continued integrity of a vessel in system in which all the piping welds are exempt from examination by IWC-1220(d). Examination of this vessel for evidence of leakage during the performance of pressure tests will provide the same assurance of continued integrity as for the piping system with which it is associated.

3. The nozzle to vessel welds of the residual heat exchangers are covered by 12 inch diameter, one inch thick reinforcement pad. The weld is not accessible for examination by volumetric or surface methods. The area will be subject to visual examination for evidence of leakage during system pressure tests.

4. The thickness of the materials utilized for the construction of this component (0.165 to 0.185 inches) is such that meaningful results could not be expected with ultrasonic examination as required by IWC-2600. Surface and visual examination of these welds will be performed as an alternative method.

5. Examination of Class 2 piping system welds is limited to those occuring at geometric discontinuities such that some limitations may be expected at all locations. For pipe to fitting or pipe to vessel nozzle welds, examinations can be performed to the extent required by T-532 of Section V from the weld and pipe surfaces. Examination from the fitting side would be dependent upon the geometric configuration. Where elbows or tees are concerned, examination can be performed from the fitting side except where the intrados of the fitting prevents adequate ultrasonic coupling. No examinations can be performed from the fitting side when it is a valve or a flange. In all cases 100 percent of the weld material can be examined. In instances where welds occur at fitting to fitting, access restrictions as outlined above occur on both sides of the weld. In instances where ultrasonic examinations cannot be performed on 100 percent of the volume of the weld and heat affected zone, surface examinations will be performed to supplement the limited volumetric examination.

Preservice examinations were not conducted on the welds in Class 2 systems in the Kewaunee Plant and the following listing of welds are those where it is expected that supplemental surface examinations would be required.

(a) Loop A Feedwater, ISO WPS2-2101 Weld #19.
(b) Loop B Feedwater, ISO WPS2-2201 Welds #18 & 19.
(c) RHR System, ISO WPS2-2520, Welds #2, 22, 28, 29, 56, 57 and 58
(d) RHR System, ISO WPS2-2521, Welds #2, 5 and 11.
(e) RHR/Low head SIS System, ISO WPS2-2522, Weld #23.
(f) RHR System, ISO WPS2-2530, Welds #3, 20 & 32.
(g) RHR System, ISO WPS2-2531, Welds #3, 19, 34, 35, 42 and 48.
(h) Low Head SIS System, ISO WPS2-2540, Weld #12.
(i) Low Head SIS System, ISO WPS2-2541, Weld #16.
(j) Low Head SIS System, ISO WPS2-2542, Weld #31.

Additionally sections of the Loop B mainsteam and feedwater systems are encapsulated for pipe whip restraint and welds contained in these sections

are inaccessible for examination. Specific welds are:

(a) Loop B Mainsteam, Welds #7, 8, 9, and 10.

(b) Loop B Feedwater, Welds #1, 2, 3, 4, 5, 6, 9, 10, 11, 12, 13 and 14.

The code does not require that all welds in multiple streams be examined and consequently not all the welds listed above will be included in the inspection program during the life of the plant.

6. Where ultrasonic examinations cannot be performed on the surface of branch pipe to pipe welds, examinations will be performed to the extent practical from the pipe and nozzle surfaces adjacent to the weld. Surface examination of the weld will be performed to supplement the volumetric examination.

Table TS 4.2-3

KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 3 COMPONENTS

SYSTEM	COMPONENT DESCRIPTION/IDENTIFICATION		CODE APPLICABLE TO CONSTRUCTION	METHOD OF EXAMINATION	SECTION XI CODE RELIEF REQUESTED
Component Cooling	Pumps	APCC1 1A APCC2 1B		Visual/Operating Pressure	No
	Heat Exchangers (Shell Side)	AHCC1 1A AHCC2 1B	VIII	Visual/Operating Pressure	No
	Seal Water Heat Exchanger (Shell Side)	AHSW	VIII	Visual/Operating Pressure	No
	Letdown Heat Exchanger (Shell Side)	AHNR	VIII	Visual/Operating Pressure	No
	RHR Heat Exchangers (Shell Side)	AHRS1 (1A) AHRS2 (1B)	VIII	Visual/Operating Pressure	No
	Surge Tank	ATCS 1	VIII	Visual/Operating Pressure	No
	Excess Letdown Heat Exchangers (Shell Side)	AHEL 1A AHEL 1B	VIII	Visual/Operating Pressure	No
	R. C. Pump Oil Coolers	PCPC 1A PCPC 1B		Visual/Operating Pressure	No
	Containment Spray Pump Gland Coolers	1A 1B		Visual/Operating Pressure	No
	Safety Injection Pump Coolers	APSI 1A APSI 1B		Visual/Operating Pressure	No

Table TS 4.2-3

KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 3 COMPONENTS

SYSTEM	COMPONENT DESCRIPTION/IDENTIFICATION		CODE APPLICABLE TO CONSTRUCTION	METHOD OF EXAMINATION	SECTION XI CODE RELIEF REQUESTED
Component Cooling	RHR Pump Shaft Seal Heat Exchangers	APRH1 1A APRH2 1B		Visual/Operating Pressure	No
	Piping		B31.1	Visual/Operating Pressure	No
	Hangers and Supports			Visual	No
Service Water	Pumps	1A1 1A2 1B1 1B2		Visual/Operating Pressure	No
	Strainers	1A1 1A2 1B1 1B2		Visual/Operating Pressure	No
	Diesel Generator Heat Exchangers	1A 1B		Visual/Operating Pressure	No
	Component Cooling Heat Exchangers (Tube Side)	1A 1B		Visual/Operating Pressure	No
	Spent Fuel Pool Heat Exchanger (Shell Side)		VIII	Visual/Operating Pressure	No
	Safety Injection Pump Coolers	APSI 1A APSI 1B		Visual/Operating Pressure	No

Table TS 4.2-3

KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 3 COMPONENTS

SYSTEM	COMPONENT DESCRIPTION/IDENTIFICATION		CODE APPLI-CABLE TO CONSTRUCTION	METHOD OF EXAMINATION	SECTION XI CODE RELIEF REQUESTED
Service Water	Containment Fan Coil Units	1A 1B 1C 1D		Visual/Operating Pressure	No
	Piping			Visual/Operating Pressure	No
	Hangers and Supports			Visual	No
Auxiliary Feed Water	Motor Driven Pumps	1A 1B		Visual/Operating Pressure	No
	Turbine Driven Pump	1C		Visual/Operating Pressure	No
	Supports			Visual	No
Spent Fuel Pool Cooling	Pumps	1A 1B		Visual/Operating Pressure	No
	Heat Exchanger (Tube Side)		III C	Visual/Operating Pressure	No
	Piping		B31.1	Visual/Operating Pressure	No

Table TS 4.2-3

KEWAUNEE NUCLEAR PLANT
INSERVICE INSPECTION PROGRAM
ASME CODE CLASS 3 COMPONENTS

SYSTEM	COMPONENT DESCRIPTION/IDENTIFICATION		CODE APPLI-CABLE TO CONSTRUCTION	METHOD OF EXAMINATION	SECTION XI CODE RELIEF REQUESTED
Spent Fuel Pool Cooling	Hangers and Supports			Visual	No
Chemical and Volume Control	Boric Acid Tanks	ATBA 1A ATBA 1B		Visual/Operating Pressure	No
	Boric Acid Filter	FLBA	III C	Visual/Operating Pressure	No
	Boric Acid Transfer Pumps	APBA 1A APBA 1B		Visual/Operating Pressure	No
	Component Supports			Visual	No
Auxiliary Steam	Piping		B31.1	Visual/Operating Pressure	No

Table TS 4.2-4

KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 PUMPS

PUMP IDENTIFICATION	PUMP DESCRIPTION	ASME CODE CLASS	MEASURED PARAMETERS	TEST INTERVAL	SECTION XI CODE RELIEF REQUESTED
APSI 1A	High Head Safety	2	1. Speed (if variable)	NA	No
APSI 1B	Injection Pumps		2. Inlet Pressure (Pi)	Monthly	No
			3. Outlet Pressure (Po)	Monthly	No
			4. Differential Pressure (Pi-Po)	Monthly	No
			5. Flow Rate	NA	No - Note 1
			6. Vibration Amplitude	Monthly	No - Note 2
			7. Bearing Temperature	N/A	Yes - Note 3
			8. Lubricant Level or Pressure	Monthly	No
APRHI 1A	Residual Heat	2	1. Speed (if variable)	NA	No
APRHI 1B			2. Inlet Pressure (Pi)	Monthly	No
			3. Outlet Pressure (Po)	Monthly	No
			4. Differential Pressure (Pi-Po)	Monthly	No
			5. Flow Rate	NA	No - Note 1
			6. Vibration Amplitude	Monthly	No - Note 2
			7. Bearing Temperature	NA	Yes - Note 10
			8. Lubricant Level or Pressure	Monthly	No
1A	Auxiliary Feed	3	1. Speed (if variable)	NA	No
1B	Water (Motor Driven)		2. Inlet Pressure (Pi)	Monthly	No
			3. Outlet Pressure (Po)	Monthly	No
			4. Differential Pressure (Pi-Po)	Monthly	No
			5. Flow Rate	NA	No - Note 1

Table TS 4.2-4

KEWAUNEE NUCLEAR PLANT

INSERVICE TEST PROGRAM

ASME CODE CLASS 1, 2 AND 3 PUMPS

PUMP IDENTIFICATION	PUMP DESCRIPTION	ASME CODE CLASS	MEASURED PARAMETERS	TEST INTERVAL	SECTION XI CODE RELIEF REQUESTED
1A	Auxiliary Feed-	3	6. Vibration Amplitude	Monthly	No - Note 2
1B	Water (Motor Driven)		7. Bearing Temperature	Refueling	Yes - Note 4
			8. Lubricant Level or Pressure	Monthly	No
1A1	Service Water	3	1. Speed (if variable)	NA	No
1A2			2. Inlet Pressure (Pi)	NA	Yes - Note 6
1B1			3. Outlet Pressure (Po)	Every 3 Months	Yes - Note 7
1B2			4. Differential Pressure (Pi-Po)	NA	Yes - Note 6 & 7
			5. Flow Rate	NA	Yes - Note 8
			6. Vibration Amplitude	Monthly	No
			7. Bearing Temperature	NA	Yes - Note 9
			8. Lubricant Level or Pressure	NA	Yes - Note 9
APCC1 1A	Component Cooling	3	1. Speed (if variable)	NA	No
APCC2 1B			2. Inlet Pressure (Pi)	Monthly	No
			3. Outlet Pressure (Po)	Monthly	No
			4. Differential Pressure (Pi-Po)	Monthly	No
			5. Flow Rate	Monthly	No
			6. Vibration Amplitude	Monthly	No
			7. Bearing Temperature	NA	Yes - Note 10
			8. Lubricant Level or Pressure	Monthly	No

KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 PUMPS

PUMP IDENTIFICATION	PUMP DESCRIPTION	ASME CODE CLASS	MEASURED PARAMETERS	TEST INTERVAL	SECTION XI CODE RELIEF REQUESTED
1A	Containment Spray		1. Speed (if variable)	NA	No
1B			2. Inlet Pressure (Pi)	Monthly	No
			3. Outlet Pressure (Po)	Monthly	No
			4. Differential Pressure (Pi-Po)	Monthly	No
			5. Flow Rate	NA	No - Note 1
			6. Vibration Amplitude	Monthly	No - Note 2
			7. Bearing Temperature	NA	Yes - Note 11
			8. Lubricant Level or Pressure	Monthly	No
1C	Auxiliary Feedwater (Turbine Driven)		1. Speed (if variable)	Monthly	No
			2. Inlet Pressure (Pi)	Monthly	No
			3. Outlet Pressure (Po)	Monthly	No
			4. Differential Pressure (Pi-Po)	Monthly	No
			5. Flow Rate	NA	No - Note 1
			6. Vibration Amplitude	Monthly	No - Note 2
			7. Bearing Temperature	Refueling	Yes - Note 4
			8. Lubricant Level or Pressure	Monthly	No

INSERVICE TEST PROGRAM

ASME CODE CLASS 1, 2 & 3 PUMPS

NOTES

1. Under the operating mode in which these pumps will be tested the system will have fixed resistance. As allowed by footnote 1 of Table IWP-3100-1, this parameter will not be recorded.

2. The monthly performance test of this pump is run under recirculation "miniflow" conditions.

3. The bearing oil cooling system for this pump is cooled by the Service Water System. The system is not temperature stabilized, therefore, meaningful results from the recording of this temperature cannot be expected.

4. Bearing temperatures are recorded at refueling since that situation provides reference conditions in which the bearing temperatures have had adequate time stabilized.

5. Deleted

6. The service water pumps are vertical design with no means of direct inlet pressure measurement as required by IWP-4200. Inlet pressure to these pumps is a function of the level of water in the forebay. Relief is required from meeting pressure tap construction and location requirements of IWP-4211 and IWP-4212. Minimum forebay level is already a Technical Specification requirement for operation of the plant.

7. The outlet pressure of the individual Service Water pumps cannot be recorded because there is no individual discharge pressure indication. Instrumentation is available to monitor system pressure at the com-

mon outlet header of the four service water pumps. The design heat load is such that multiple pumps must be in operation at all times. There is no plant operating conditions when a single pump could be run to determine performance parameters as required by IWP-3400(a). By operating various combinations of pumps and comparing common header discharge pressure, the individual performance of each pump can be monitored.

8. There is no instrumentation to monitor this operating parameter in the plant system as designed and constructed. As indicated in Note 7 above, the system heat load is such that multiple pumps must be in operation at all times and flow rate will vary with plant power levels and equipment in operation.

9. The service water pumps are open line shaft pumps that depend primarily of the liquid being pumped for the lubrication of the pump and lineshaft bearings. The bearing lubricating water flow can be verified by sight glass but pressure cannot be monitored. All pump bearings are submerged and lubricant is allowed to leak off into the sump and is not piped back, such that temperatures cannot be monitored.

10. There is no installed instrumentation to monitor this parameter. The design of the pump is such that article IWP-3500 cannot be met.

11. No remote bearing temperature instruments exist and measurements would not be meaningful since these pumps cannot be run under these conditions long enough to stabilize.

12. Deleted.

Generic Note:

Table IWP-3100-1, Note 2, implies measuring static suction pressure prior to testing. This measurement has no significane with reference to the Kewaunee Plant. Sources of supply have alarms and/or indicators which indicate low level; therefore, these measurements will not be taken.

Table TS 4.2-5

KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
SI-22A SI-22B	8840 A 8840 B	C	1	12-inch, Check, Accumulator discharge to Cold legs	Each Refueling	N/A	Notes 1, 1A
SI-21A SI-21B	8841 A 8841 B	C	1	12-inch, Check, Accumulator discharge to Cold legs	Each Refueling	N/A	Notes 1, 1A
SI-20A SI-20B	8800 A 8800 B	E	2	12-inch, MOV, norm. open Accumulator Discharge Stop Valves	N/A	N/A	Note 9
SI-13A SI-13B	8842 A 8842 B	C	1	6-inch, Check, HPSI to Cold Legs	Each Refueling	N/A	Note 1
SI-12A SI-12B	SI-118-1 SI-118-3	C	1	2-inch, Check, HPSI to Cold Legs	Each Refueling	N/A N/A	Note 1
SI-11A SI-11B	8801 A 8801 B	E	2	2-inch, MOV, norm. open HPSI to Cold Legs	N/A	N/A	Note 6, 6B
SI-16A SI-16B	SI-118-2 SI-118-4	C	1	2-inch, check, HPSI to Reactor Vessel Core Flood	Each Refueling	N/A	Note 1
SI-15A SI-15B	8802 A 8802 B	B	2	2-inch, MOV, norm. closed HPSI to Reactor Vessel Core Flood	Every 3 Months	N/A	Note 6, 6B
SI-303A SI-303B	8843 A 8843 B	C	1	6-inch, check, LPSI to Reactor Vessel	Each Refueling	N/A	Note 1
SI-106A SI-106B	SI-113-1 SI-113-2	C	2	1-inch, Safety Valve Accmumulator Safety Valve	Note 3	N/A	

Table TS 4.2-5

KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
SI-304A SI-304B	8844 A 8844 B	C	1	6-inch, Check, HPSI and LPSI to Reactor Vessel	Each Refueling	N/A	Note 1
SI-312	8831	C	2	3/4-inch, safety valve LPSI relief to PRT	Note 3	N/A	
SI-208 SI-209	8826 B 8826 A	B	2	2-inch, MOV, norm. open, Test Line to Refuel. Water Stor. Tank	Every 3 Months	N/A	
SI-201A SI-201B SI-202A SI-202B	8824 A 8825 B 8825 A 8824 B	A	2	3/4-inch, AOV, norm. closed SI test line	N/A	Each Refueling	Note 2
SI-204	SI 1-7	A	2	3/4-inch manual valve, norm closed Safety Injection Test Line	N/A	Each Refueling	Note 2
SI-9A	8806 A	E	2	3-inch, MOV, norm. open HPSI line to Cold Leg Injection	N/A	N/A	
SI-5A SI-5B	SI 9-1 SI 9-2	E	2	6-inch, MOV, norm. open HPSI Pump Suction	Each Refueling	N/A	
SI-6A SI-6B	8812 B 8812 B	C	2	4-inch, Check, HPSI pump discharge	Each Refueling	N/A	Note 1
NG-107	8820	A	2	1-inch, AOV, norm. open N_2 Supply to Accumulators	Every 3 Months	Each Refueling	Note 4
SI-4A SI-4B	8808 A 8808 B	B	2	12-inch, MOV, norm. closed RWST Supply to HPSI	Every 3 Months	N/A	
SI-2A SI-2B	8809 A 8809 B	B	2	8-inch, MOV, norm. closed BAT Supply to HPSI	Every 3 Months	N/A	
SI-3	8809 C	E	2	8-inch, MOV, norm. open BAT Supply to HPSI	N/A	N/A	

Table TS 4.2-5

KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
SI-351A SI-351B	8804 A 8804 B	A	2	12-inch, MOV, norm. closed Containment Sump Recirc to RHR	Every 3 Months	Each Refueling	Note 6, 6A
SI-350A SI-350B	8805 A 8805 B	A	2	12-inch, MOV, norm. closed Containment Sump Recirc to RHR	Every 3 Months	Each Refueling	Note 6, 6A
RHR-300A RHR-300B	8816 A 8816 B	B	2	6-inch, MOV, norm. closed HPSI Pump Suction from RHR	Each Refueling	N/A	Note 7
SI-300A SI-300B	8810 A 8810 B	B	2	10-inch, MOV, norm. closed RWST supply to RHR pumps	Every 3 Months	N/A	
SI-301A SI-301B	8811 A 8811 B	C	2	10-inch, Check, RWST supply to RHR Pumps	Each Refueling	N/A	Note 1, 1A
RHR-1A RHR-1B	8702 A 8702 B	B	1	8-inch, MOV, norm. closed RHR Take-off from Hot Legs	Cold Shutdown	N/A	Note 8,8A
RHR-2A RHR-2B	8701 A 8701 B	B	1	8-inch, MOV, norm. closed RHR Take-off from Hot Legs	Cold Shutdown	N/A	Note 8, 8A
RHR-3A RHR-3B	8710 A 8710 B	C	2	8-inch, Check, RHR pump suction from Hot Legs	Cold Shutdown	N/A	Note 8, 8A
RHR-5A RHR-5B	8712 A 8712 B	C	2	8-inch, Check, RHR pump discharge	Cold Shutdown	N/A	Note 8, 8A
RHR-33	8705	C	2	2-inch, Relief Valve RHR Suction relief valve	Note 3	N/A	
RHR-33-1	AC-115-001	C	2	6-inch, Safety Relief, RHR to RC Hot Leg, Low Temp. overpressure protection	Note 3	N/A	

Table TS 4.2-5

KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
RHR-400A RHR-400B	ICS 6-5 ICS 6-6	B	2	6-inch, MOV, norm, closed RHR Supply to ICS pumps	Every 3 Months	N/A	
RHR-401A RHR-401B	ICS 4-1 ICS 4-2	C	2	6-inch, Check RHR supply to ICS pumps	Each Refueling	N/A	Note 15
PR-3A PR-3B	8010 A 8010 B	C	1	6-inch, Safety Valve, PZR Safety Relief to PRT	Note 3	N/A	
MG-(R)-513 MG-(R)-512	8025 8026	A	2	3/8-inch, AOV, norm. closed PRT vent to gas	Every 3 Months	Each Refueling	Note 4
NG-302	8028	A	2	3/4-inch, AOV, norm. open N-2 supply to PRT	Every 3 Months	Each Refueling	Note 4
NG-304	RC 1-1	C	2	3/4-inch, Check N-2 supply to PRT	N/A	Each Refueling	Note 2
MU-1010-1	8029	A	2	2-inch, AOV, norm. open Make-up H_2O to PRT	Every 3 Months	Each Refueling	Note 4
MU-1011	RC 2-1	C	2	2-inch, Check, Make-up H_2O to PRT	N/A	Each Refueling	Note 2
MG-(R)-503 MG-(R)-504	9159 A 9159 B	A	2	3/8-inch, AOV, norm. closed RCDT Vent to Gas Analyzer	Every 3 Months	Each Refueling	Note 4

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KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
MG-(R)-509 MG-(R)-510	9160 A 9160 B	A	2	1-inch, AOV, norm. open RCDT to Vent Header	Every 3 Months	Each Refueling	Note 4
RC-507 RC-508	9170 A 9170 B	A	2	3-inch, AOV, norm. open RCDT pump discharge	Every 3 Months	Each Refueling	Note 4
MD-(R)-134 MD-(R)-135	9182 A 9182 B	A	2	3-inch, AOV, norm. closed Containment Sump Pump Discharge	Every 3 Months	Each Refueling	Note 4
RC-402 RC-403	SS13-1 SS13-5	A	1	3/8-inch, AOV, norm. open PZR Steam Space Sample	Every 3 Months	Each Refueling	Note 4
RC-412 RC-413	SS13-2 SS13-6	A	1	3/8-inch, AOV, norm. open PZR Liquid Space Sample	Every 3 Months	Each Refueling	Note 4
RC-422 RC-423	SS13-3 SS13-7	A	1	3/8-inch, AOV, norm. open RC Hot Leg Sample	Every 3 Months	Each Refueling	Note 4
LD-4A LD-4B LD-4C	8140 A 8140 B 8141	A	2	2-inch, AOV, norm. open Outlet from Letdown Orifices	Every 3 Months	Each Refueling	Note 4
LD-6	8147	A	2	2-inch, AOV, norm. open Letdown to heat exchanger	Each Refueling	Each Refueling	Notes 4, 8
CVC-212 CVC-211	8100 A 8100 B	A	2	3-inch, MOV, norm. open RC pump Seal Return	Each Refueling	Each Refueling	Notes 8, 8B

Table TS 4.2-5

KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
CVC-205B CVC-205A	CS100-1 CS100-2	A/C	2	2-inch, Check, RC pump seal injection	N/A	Each Refueling	Note 6D
CC-400A CC-400B	9411 A 9411 B	B	3	10-inch, MOV, norm. closed CC water to RHR heat exchangers	Every 3 Months	N/A	Note 6E
CC-651	AC111-1	C	3	3-inch, Check, CC water to excess letdown heat exchangers	N/A	Each Refueling	Note 1A
CC-653	AC15-1	E	3	3-inch, MOV, norm. closed, CC water from excess letdown heat exchanger	Every 3 Months	N/A	
SW-903A SW-903B SW-903C SW-903D	SW10-1 SW10-2 SW10-3 SW10-4	B	2	8-inch, MOV, norm. open Service Water return from RCFC units	Every 3 Months	N/A	
SW-6010	SW66-1	A	2	2-inch, manual, norm. closed Service water to Containment Hose Stations	N/A	Each Refueling	Note 2
SW-601A SW-601B SW-502	SW54-1 SW54-2 SW52-3	B	3	3-inch, MOV, norm. closed SW to Aux. Feed Pumps	Every 3 Months	N/A	
SW-901A SW-901B SW-901C SW-901D	SW7-1 SW7-2 SW7-3 SW8-4	C	3	8-inch, Check, SW to RCFC units	N/A	Each Refueling	Notes 11, 6 & 6F
SW-501A SW-501B	SW50-2 SW50-1	C	3	3-inch, Check SW to Aux. Feed Pumps	At Cold Shutdown	N/A	Note 1A

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KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
SW-1300A SW-1300B	SW4-1 SW4-2	B	3	10-inch, MOV, norm. closed SW to CC heat Exchangers	Refueling	N/A	Note 17
SW-301A SW-301B	SW48-2 SW48-2	B	3	4-inch, AOV, norm. closed SW return from D/G coolers	Note 16	N/A	
SW-3A SW-3B	SW-17-1 SW-17-2	E	3	24-inch, AOV, norm. open SW pump disc cross connect	Every 3 Months	N/A	
SW-4A SW-4B	SW-18-1 SW-18-2	B	3	20-inch, AOV, norm. open SW supply to Turbine Buidling	Every 3 Months	N/A	
SW-1A1 SW-1A2 SW-1B1 SW-1B2	SW-52-1 SW-52-2 SW-52-3 SW-52-4	C	3	14-inch Check SW pump discharge	Every 3 Months	N/A	Note 18
RBV 14-1 RBV 14-2	RBV 14-1 RBV 14-2	A	2	2-inch, MOV, norm. closed Air supply to containment	Every 3 Months	Each Refueling	
LOCA-2A LOCA-2B	RBV 14-3 RBV 14-4	A	2	2-inch, MOV, norm. closed H_2 control post LOCA sample	Every 3 Months	Each Refueling	
LOCA-3A LOCA-3B	RBV 15-1 RBV 15-2	A	2	1-inch, AOV, norm. closed H_2 Control post LOCA Containment Air Sample	Every 3 Months	Each Refueling	
LOCA-10A LOCA-10B	RBV 16-1 RBV 16-2	A	2	1-inch, AOV, norm. closed H_2 Control post LOCA air sample to gas analyzer	Every 3 Months	Each Refueling	
RBV-1 RBV-2	RBV 4-3 RBV 4-4	A	2	36-inch, AOV, norm. closed Containment Purge Supply	At Cold Shutdown	Each Refueling	Note 4 & 8D

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KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
RBV-4 RBV-3	RBV 4-1 RBV 4-2	A	2	36-inch. AOV, norm. closed Containment Purge Exhaust	At Cold Shutdown	Each Refueling	Note 4 & 8D
VB-10A VB-10B	RBV 2-1 RBV 2-2	A	2	18-inch, AOV, norm. closed Containment Vacuum Breaker	At Cold Shutdown	Each Refueling	Note 4 & 8D
AS-1 AS-2	RBV 20-1 RBV 20-2	A	2	1-inch, AOV, norm. open Containment air sample to radiation monitors	Every 3 Months	Each Refueling	Note 4
AS-32	RBV 20-3	A	2	1-inch, AOV, norm. open Containment air sample return	Every 3 Months	Each Refueling	Note 4
BT-2A BT-3A BT-2B BT-3B	SD 27-1 SD 27-2 SD 27-3 SD 27-4	A	2	2-inch, MOV, norm. open Steam Generator Blowdown isolation valves	Every 3 Months	Each Refueling	
MS-1A MS-1B	SD 24-1 SD 24-2	B	2	30-inch, AOV, norm. open Main Steam Isolation Valve	At Cold Shutdown	N/A	Notes 8, & 8C
MS-100A MS-100B	SD 26-1 SD 26-2	B	2	3-inch, MOV, norm. open MS to turbine driven AFW pump	Every 3 Months	N/A	
MS-102	SD 26-3	B	3	3-inch, MOV, norm. closed MS to turbine driven AFW pump	Every 3 Months	N/A	
MS-101 A MS-101 B	SD 28-1 SD 28-2	C	3	3-inch, Check MS to turbine driven AFW pump	Every 3 Months	N/A	

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KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
SD-1A1 SD-1A2 SD-1A3 SD-1A4 SD-1A5 SD-1B1 SD-1B2 SD-1B3 SD-1B4 SD-1B5	SD 23-1 SD 23-2 SD 23-3 SD 23-4 SD 23-5 SD 23-6 SD 23-7 SD 23-8 SD 23-9 SD 23-10	C	2	6-inch, Safety relief Mainsteam Line	Note 3	N/A	
BT-31 B BT-32 B BT-31 A BT-32 A	SS 14-1 SS 14-2 SS 14-3 SS 14-4	B	2	3/8-inch, AOV, norm. closed S/G Blowdown Sample	Every 3 Months	N/A	Note 4
FW-12A FW-12B	F 7-1 F 7-2	B	2	16-inch, MOV, norm. open Main FW to S/G isolation valves	At Cold Shutdown	N/A	Notes 8 & 8C
AFW-4A AFW-4B	F 15-1 F 15-2	C	2	4-inch, Check Valves, AFW to Steam Generators	At Cold Shutdown	N/A	Note 10
AFW-1A AFW-1B AFW-1C	F 16-3 F 16-6 F 16-1	C	3	3-inch, Check AFW pumps discharge	At Cold Shutdown	N/A	Note 10
AFW 10A AFW 10B	F 33-1 F 33-2	B	3	3-inch, MOV, norm. open Turbine driven AFW pump discharge	Every 3 Months	N/A	
IA 102 IA 103	SA 31-1 SA 31-2	A/C	2	1-inch, Check, Instrument air containment supply	N/A	Each Refueling	Note 12

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KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM
ASME CODE CLASS 1, 2 AND 3 VALVES

VALVE IDENTIFICATION		SECTION XI CATEGORY PER IWV-2000	ASME CODE CLASS	VALVE DESCRIPTION/FUNCTION	TEST FREQUENCY		SECTION XI CODE RELIEF REQUESTED
OPS NO.	FSAR NO.				EXERCISE	LEAKAGE	
ICS-3A ICS-3B	ICS 3-1 ICS 3-2	C	2	6-inch, Check, Containment Spray Pump Suction	Every 3 Months	N/A	Note 13
ICS-4A ICS-4B	ICS 4-3 ICS 4-4	C	2	6-inch, Check Containment Spray Pump Discharge	Every 3 Months	N/A	Note 13
ICS-5A ICS-6A ICS-5B ICS-6B	ICS 6-1 ICS 6-2 ICS 6-3 ICS 6-4	B	2	6-inch, MOV, norm. closed Containment Spray Pump Discharge	Every 3 Months	N/A	
ICS-8A ICS-8B	ICS 4-5 ICS 4-6	C	2	6-inch, Check ICS discharge line spray header	Note 14	N/A	Notes 14 & 6C

KEWAUNEE NUCLEAR PLANT
INSERVICE TEST PROGRAM NOTES
ASME CODE CLASS 1, 2, AND 3 VALVES

1. Operation of these normally closed check valves will be verified each refueling outage. Operation will be verified by establishing and observing flow through the individual line. Establishing flow through these check valves would introduce safety injection flow into the reactor coolant system.

   IWV-3410(b) states that normally closed valves that cannot be operated during normal plant operation shall be specifically identified by the Owner and exercised during each cold shutdown. The term "cold shutdown" covers a wide variety of plant conditions which can exist with the reactor coolant temperature equal to, or less than, 200°F. Under cold shutdown conditions, the system may be "solid" such that introducing flow into the system to exercise check valves would have a high probability of overpressurizing the system. These tests can only be performed when the reactor coolant system is vented to the containment atmosphere and partially drained such as for a refueling. Similarly, under cold shutdown conditions reactor coolant pumps may be in operation and to close normally open valves in systems serving these components would jeopardize their safe operation.

   A. Full open or full stroke position cannot be identified on certain check valves. Operability will be verified by flow measurements or by observing a tank level change, as in the case of the Accumulator discharge check valves.

2. These valves provide no safety function except as a containment pressure boundary in case of line rupture. It is normally closed and stays closed to perform this function, thus requires no exercise testing.

3. Testing of safety relief valves will be in accordance with the requirements of IWV-3500.

4. An exception is taken to the requirement to determine and measure the stroke time of air operated valves. This time is variable depending on the air pressure at the valves and is a meaningless value. The only meaningful measurement is the closure time for air operated isolation valves such as these. Acceptance criteria will be established and valve closure time will be measured to ensure acceptable operation.

5. This valve provides no safety function except to prevent backflow of Accumulator discharge into RHR system. This valve is normally closed and remains closed to provide this function.

6. Valves associated with safeguard systems which will be operated during and following an accident are required to open or remain open for the systems to fulfill the safeguard function. These systems will be used to cool the containment and the reactor core post accident. These systems are designed to remain intact post accident and in effect are an extension of the containment. The following surveillance will assure leak tightness of these systems.

   A. Residual Heat Removal

      (i) Those portions of the Residual Heat Removal System external to the isolation valves at the Reactor Coolant System shall be hydrostatically tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through the summer of 1975.

   B. Safety Injection System (High Head)

      (i) Those portions of the Safety Injection System in-service post-accident shall be hydrostatically tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through the summer of 1975.

C. Internal Containment Spray System

(i) Those portions of the Internal Containment Spray System in service post accident shall be hydrostatically tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through the summer of 1975.

D. Chemical and Volume Control Charging System

(i) The Chemical and Volume Control Charging System piping from the charging pump discharge to the Reactor Coolant System shall be inspected for leakage in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda throught the summer of 1975.

(ii) No testing is required as allowed by IVW-3410(d).

E. Component Cooling System

(i) The Component Cooling System piping shall be inspected for leakage in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through the summer of 1975.

F. Service Water - Fan Coil Cooling System

(i) The Service Water System piping for the fan coil coolers which are located within containment shall be inspected for leakage in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through the summer of 1975.

7. Failure of either of these valves in an open position during operational testing would provide an open connection between the RHR system and the suction side of the SI pumps. The operating pressure of the RHR system under normal conditions is in excess of the design pressure of the SI system suction piping and an open cross connection would prevent operation of the RHR system for it's function of normal cool down of the reactor system.

8. These valves will be tested in accordance with the exception to category A and B valves allowed by IWV-3410(b)(1).

A. RHR system operating pressure limits prohibit operation of these valves above 400 psig.

B. These CVCS system valves must remain open as long as the Reactor Coolant pumps are running.

C. The main steam and feedwater isolation valves occasionally require valve stem packing adjustments. These adjustments are performed during plant operation and do not allow full stroke testing.

D. The containment purge system valves are normally closed during operation.

9. These normally open control valves remain open to perform their safeguard function. Exercise testing during normal plant operations is not necessary to ensure performance of their safeguard function.

10. Operation of these normally closed check valves will be verified by establishing and observing flow through the individual line prior to startup of the plant.

11. These check valves remain in an open position during all phases of plant operation and remain open to fulfill their function following an incident. Exception is taken to the performance of exercising tests as required by IWV-3520.

12. The only safeguard function required for these check valves is to provide containment isolation. Operability tests are not performed during plant operation or shutdown since these lines are required to operate. No testing is required as allowed by IWV-3410(d).

13. The operation of these check valves will be partially verified during performance testing of the pumps. These valves are of a split disc design and the low flow at pump test conditions will only require limited actuation of the valves.

14. Introducing flow through these valves would result in the fluid being sprayed into the containment. Due to the split disc design of these valves, utilizing a compressible gas would provide assurance of only minimal movement of the valve discs. These check valves will be physically inspected to observe freedom of disc movement every five years.

15. The operation of these check valves will be partially verified during refueling by flow testing of these lines. Normal plant operation prohibits testing of these lines.

16. Proper operation of these control valves is verified on the monthly 4-hour surveillance tests of the Diesel Generators.

17. Cooling of the CC heat exchangers is normally accomplished by 4-inch bypass valves. Opening of these 10 inch valves could cause thermal shock to the heat exchangers and thereby the Reactor Coolant Pumps.

18. There is no instrumentation to measure flowrate through individual valves. Valve operation (open) will be continually verified by total system operation. Valve operation (closure) will be verified by the quarterly pump test.

TABLE 4.2-6

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in the other S. G. Prompt notification to NRC pursuant to specification 6.9.2	The other S.G.'s are C-1	None	N/A	N/A
			Some S.G.'s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.2.	N/A	N/A

S = 6%/n Where n is the number of steam generators inspected during an inspection.

from the integrated leak rate test result, yield a leak rate value not greater than the periodic leak rate L_{tm}.

If the leak rate determined by any test exceeds the maximum allowable leak rate, the test schedule applicable to subsequent integrated leak tests shall be subject to review and approval by the Commission. If the leak rate determined by two consecutive periodic tests exceeds the maximum allowable leak rate, subsequent tests shall be performed at each major refueling outage until two consecutive tests have been performed for which the leak rate does not exceed the maximum allowable. The initial retest schedule specified above shall then be resumed on the basis of starting a new ten-year service period.

b. Isolation Valves and Local Leak Rate Tests

1. Deleted 25C

2. The following containment penetrations shall periodically be leak-tested at a pressure not less than 46 psig, using pressure decay, soap bubble, halogen detection, or equivalent methods.

   A. Containment penetrations which rely on resilient seals, gaskets, or sealing compounds.

   B. The personnel air locks.

   C. The equipment hatch.

   D. The flange of the fuel transfer tube.

   E. Other components which develop leaks and require repairs to meet allowable integrated leakage limits.

to this value.

c. Deleted.

25C

d. Shield Building Ventilation System

1. At least once per operating cycle or once every 18 months whichever occurs first, the following conditions shall be demonstrated:

   A. Pressure drop across the combined HEPA filters and charcoal absorber banks is less than 10 inces of water and the pressure drop across any HEPA filter bank is less than 4 inches of water at the system design flow rate (+10%).

   B. Automatic initiation of each train of the system.

   C. Operability of heaters at rating and the absence of defects by visual inspection.

2. A. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

   B. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and

that for a leak rate, have been investigated thoroughly. A summary of the items of conservatism involved in the reference calculation and the magnitude of their effect upon off-site dose demonstrates the collective effect of conservatism in these assumptions. (Refer to Appendix H, FSAR)

The reference initial leak rate in this analysis is 0.5 weight percent of air per 24 hours at the peak pressure of the Design Basis Accident. The resulting two-hour doses at the nearest site boundary are significantly less than the guidelines presented in 10 CFR 100.

The pre-operational integrated leak rate tests are specified at both full design pressure and at reduced pressure, with later periodic tests performed only at reduced pressure, as suggested in the relevant AEC guide [3], and at the frequency indicated in the guide for the design and leak rate test pressures. The operational limit on leak rate $L_{tm} = 0.75\ L_t$, provides a 25 percent allowance for possible leakage deterioration between integrated leak rate tests. The six-month allowance on test schedule provides flexibility necessary to permit tests to be performed at times of scheduled or unscheduled plant outage.

The frequent leak-testing of isolation valves and other penetrations, (areas which may reasonably be expected to be responsible for any excess leakage, rather than the containment shell itself) will provide reassurance, approximately annually, that the allowable leak rate limit is met. These tests will also indicate specific areas of deterioration that may warrant repair before their leakage is excessive.

Containment Isolation valves will be tested for operability in accordance with the In-Service-Inspection program provided in Section 4.2.

25C

The operability of the Residual Heat Removal System shall be

functionally tested in accordance with the In-Service-Inspection program provided in Section 12. 25C

The Shield Building Ventilation System consists of two independent systems that have only a discharge point in common, the Containment System Vent. Both systems are normally activated and one alone must be capable of accomplishing the design function of the system. The periodic tests will demonstrate the capability of both the separate and combined systems.

Reliable simulation of the transient effects of accident-related heat flow from the Reactor Containment Vessel to the annulus appears to be difficult as well as inconvenient, and the necessary differences between any test conditions and predicted accident conditions would still require supporting analysis. Only the heat input to the annulus could be test-simulated, and not the heat transfer which determines the heat input. However, analysis supported by the results of actual tests without heat addition will provide reliable means of determining system performance with heat addition. The major uncertainties in system performance relate to such "as-built" considerations as Shield Building in-leakage, actual system losses, and overall transient response. These areas can be directly refined in the analysis model from the results of the tests specified. The effects of heat addition are readily incorporated, in a conservative manner where necessary, by considering extreme variations of heat transfer coefficients and transient containment temperature conditions. Such analysis performed during final design has demonstrated, for example, that a slight increase in the capacity of the fans was sufficient to accomodate more severe assumptions regarding heat transfer through the shell. It is expected that nearly any deviation in system behavior discovered during initial testing can be similarly offset by increases in the capacity of these fans, which have minimal power requirements (12 hp and 1 hp for the recirculation and discharge fans, respectively).

2. Containment Vessel Internal Spray System

A. System tests shall be performed during each major refueling outage. The test shall be performed with the isolation valves in the supply lines at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

B. The spray nozzles shall be checked for proper functioning at least every five years using either air with telltales or smoke tests to determine that all nozzles are clear.

C. The test will be considered satisfactory if control board indications or visual observations indicate all components have operated satisfactorily.

3. Deleted

25C

b. Component Tests

1. Pumps

A. The safety injection pumps, residual heat removal pumps, and containment spray pumps shall be started and operated on recirculation flow at intervals not greater than once every month of power operation.

B. Deleted

25C

2. Valves

A. Deleted

B. Deleted

25C

C. Deleted

D. Spray additive tank valves shall be tested during each major refueling outage.

E. Closing of the boric acid tank isolation valves and concurrent opening of refueling water storage tank valves upon receipt of simulated Lo Lo boric acid tank level signal shall be tested at intervals not to exceed once every month during power operation.

F. Residual Heat Removal System valve interlocks shall be tested during each major refueling outage.

Basis

The Safety Injection System and the Containment Vessel Internal Spray System are principal plant safety systems that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a Containment Vessel Internal Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine system tests to be performed during refueling shutdowns with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Vessel Internal Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action and verification is made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1]

During reactor operation, the instrumentation which is depended upon to initiate safety injection and containment spray is checked daily and the initiating and logic circuits are tested monthly (in accordance with Specification 4.1). In addition, the active components (pumps and valves) are tested to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. Test intervals are established in accordance with the In-Service-Inspection program provided by Section 4.2. 25C 25C

Testing of the closure of the boric acid tank isolation valves with concurrent opening of the refueling water storage tank valves upon receipt of simulated lo-lo boric acid tank level signal is performed to verify proper operation to prevent inadvertent spillage of refueling water storage tank water through the boric acid tank should the isolation valves fail to close.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System, and the containment fan-coil units. The accumulators are a passive safety feature. In accordance with Specification 4.1, the water volume and pressure in the accumulators are checked each shift. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

Reference: (1) FSAR Section 6.2

4.7 Deleted

4.8 Deleted

Deleted