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October 31, 1994

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Additional Information on Third Ten-Year Interval Inservice Inspection Plan

References: 1) Letter dated December 16, 1993 from C. A. Schrock (WPSC) to Document Control Desk (NRC).

2) Letter dated August 31, 1994 from R. J. Laufer (NRC) to C. A. Schrock (WPSC).

By letter dated December 16, 1993 (reference 1), Wisconsin Public Service Corporation (WPSC) submitted the Third Ten-Year Inservice Inspection Program for the Kewaunee Nuclear Power Plant. In reference 2 you requested additional information to complete your review.

Please find attached our response to your questions as presented in reference 2. Also, per your request this document is being sent directly to the INEL Research Center. Should you have additional questions regarding the Third Ten-Year Interval Inservice Inspection Plan, please feel free to contact a member of my staff at your convenience.

Sincerely,

C. A. Schrock

C. A. Schrock
Manager - Nuclear Engineering

CAT/san
Attachment
cc: Boyd W. Brown - EG&G Idaho, Inc.
US NRC Region III
US NRC Senior Resident Inspector

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ATTACHMENT 1

to

Letter From C.A. Schrock (WPSC)

to

Document Control Desk (NRC)

Dated: October 31, 1994

Re: Additional Information on Third Ten-Year Interval Inservice Inspection Plan

A. NRC Request

Examination Category C-F-1 requires that all nonexempt Class 2 pressure-retaining austenitic stainless steel or high alloy welds be included in the total population for determining the 7.5% sample for examination. Included in this population are piping welds with wall thickness <3/8-inch, that are excluded from volumetric and surface examinations by the Code with the exception of the PWR high pressure safety injection system. Per the Code, approximately 90% of Kewaunee's Class 2, Examination Category C-F-1 pressure-retaining welds would be excluded from the Code volumetric and surface examination because the piping is less than 3/8-inch thick. The licensee recognizing that certain systems, critical to safety (i.e., RHR and ICS), would be excluded from examination, in the Program Plan proposed to "shift" a portion of the 7.5% sample to include thin wall pipe in its examination sample and perform surface examinations on the thin wall piping welds.

Service-induced degradation will generally occur from the inside of the pipe. As a result of the licensee's proposal to only perform surface examinations on thin wall piping welds, flaws in thin wall piping welds would not be detected until through wall leakage occurs. In essence, the licensee is reducing the quantity of Code-required volumetric examinations being performed, because the licensee only proposed to perform surface examinations on the thin walled piping. Therefore, the Code-required sample size is not being met and the same level of assurance of continued inservice integrity is not being provided. Based on that approximately 90% of Kewaunee's Class 2, Examination Category C-F-1 pressure-retaining welds would be excluded from volumetric and surface examination because the piping is less than 3/8-inch thick, the shifting of the sample to include thin-wall piping is considered technically prudent. The licensee's proposed shift in sampling and surface examinations are considered satisfactory with the exception that additional volumetric examinations should also be performed on the thin wall piping welds. Describe your plans with respect to performing volumetric examination on the thin-wall Class 2 piping welds.

WPSC Response

Footnote (2) of Table IWC-2500-1, Examination Category C-F-1 from the 1989 Edition of Section XI, provides guidance for the selection and distribution for examination of Class 2 piping welds.

> The welds selected for examination shall include 7.5%, but not less than 28 welds, of all austenitic stainless steel or high alloy welds not exempted by IWC-1220. (Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Category C-F-1. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.) The examinations shall be distributed as follows:

(a) the examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt austenitic stainless steel or high alloy welds in each system (i.e., if a system contains 30% of the nonexempt welds, then 30% of the nondestructive examinations required by Examination Category C-F-1 should be performed on that system);

(b) within a system, the examinations shall be distributed among terminal ends and structural discontinuities [See Note (3)] prorated, to the degree practicable, on the number of nonexempt terminal ends and structural discontinuities in that system; and

(c) within each system, examinations shall be distributed between line sizes prorated to the degree practicable.

As required by this footnote, WPSC applied a 7.5% sampling rate to the total weld count.

Neither footnote (2) nor other areas within the 1989 Edition of Section XI provide guidance regarding the examination method that could be applied to the thin-wall class 2 piping welds. To determine what examination method should be applied to the thin-wall RHR and ICS piping, WPSC referred to previous NRC endorsed editions and addenda of Section XI from 1980 to 1986. These previous editions and addenda of Section XI require a surface examination be imposed on a representative sample of the thin-wall class 2 piping welds.

After having established the number of welds for each item number, WPSC applied a sampling rate of 7.5% and distributed the welds as described in (a), (b), and (c) of footnote (2). The 7.5% sampling rate was applied to the weld count which included both ICS and RHR thin-wall piping. The selected examination method is based on wall thickness, NPS, material, and guidance from the 1989 Edition and previous NRC endorsed editions and addenda of Section XI.

Key elements of the Third Ten Year ISI Plan related to Examination Category C-F-1 are summarized in Table 1 on the next page.

Table 1
Wisconsin Public Service Corporation
Kewaunee Nuclear Power Plant
Third Ten Year Inservice Inspection Interval
Category C-F-1 Piping Weld Data

Item Number	System	Total No. of nonexempt welds in system boundary	Code Requirement				Total Scheduled for Examination
			% required	Subtotal Subject to Examination	Exam Method		
					V	S	
5.11 Circumferential Piping Welds Having Wall Thickness ≥ 3/8" and NPS>4"	Safety injection	10	7.5	1	X	X	1
	Residual Heat Removal	25	7.5	2	X	X	2
	Internal Containment Spray	2	7.5	1	X	X	1
5.12 Longitudinal Piping Welds Having Wall Thickness ≥ 3/8" and NSP >4"	Residual Heat Removal	5	7.5	1	X	X	2
5.13 Circumferential Piping Welds Having a Wall Thickness < 3/8" and NPS ≥ 4" Note (2)	Residual Heat Removal	276	7.5 Note (4)	21 Note (4)	NA	NA	21 (Surface)
	Internal Containment Spray	132	7.5 Note (4)	10 Note (4)	NA	NA	10 (Surface)
5.14 Circumferential Piping Welds Having a Wall Thickness <3/8" and NPS ≥ 4" Note (3)	Safety Injection	65	0	0	NA	NA	0
5.21 Circumferential Piping Welds Having a Wall Thickness > 1/5" and NPS ≥ 2" and ≤4"	Safety Injection	107	7.5	8	X	X	10
5.30 Socket Welds	Safety Injection	155	7.5	12		X	15
5.41 Circumferential Pipe Branch Connection Welds of Branch Piping ≥2"	Safety Injection	7	7.5	1		X	1
	Residual Heat Removal	6	7.5	1		X	1
	Internal Containment Spray	4	7.5	1		X	1

Notes:

1. There are a total of 320 C5.11, C5.21, C5.30 and C5.41 piping welds.
2. There are a total of 408 C5.13 piping welds. C5.13 represents an item number that was created by WPSC for RHR and ICS piping welds that are < 3/8" nominal wall thickness for piping that is > NPS 4". The majority of piping welds in the RHR and the ICS systems are not required to be examined in accordance with Table IWC-2500-1 since they are less than 3/8" thick. However, WPSC has scheduled a proper representative sample of these welds identified as item number C5.13.
3. There are a total of 65 C5.14 piping welds. C5.14 represents an item number created by WPSC for SI piping welds < 3/8" nominal wall thickness for piping that is > NPS 4". No C5.14 welds were scheduled for examination since they are not required to be examined in accordance with Table IWC-2500-1.
4. All C5.13 and C5.14 welds were included in the total weld count to which the 7.5% sampling rate was applied.

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There is a total of 794 nonexempt welds within this examination category. At a 7.5% sampling rate, sixty (60) welds are required for examination; WPSC has selected and scheduled 8.2% or sixty-five (65) welds. Of these sixty-five (65) welds sixteen (16) are scheduled for both volumetric and surface examinations while the remaining forty-nine (49) welds will receive a surface examination.

It should be pointed out that of these forty-nine (49) welds, fifteen (15) are socket welds and three (3) are branch connection welds. Only a surface examination is required by Code for Class 2 socket and branch connection welds of branch piping. The remaining thirty-one (31) of the scheduled sixty five (65) welds are located in the thin-wall portion of the Class 2 RHR and ICS systems and have been scheduled for surface examination based upon guidance from previous NRC endorsed editions and addenda of Section XI.

WPSC realizes that if service induced degradation were to occur, it could occur at other locations within the system regardless of Code classification because conditions with respect to pressure, temperature, water chemistry, and stresses are similar. For this reason, WPSC elected to take another look at the number of Class 1 and 2 austenitic pipe welds scheduled for volumetric examination versus the number required by Code. The selection and scheduling criteria used by WPSC is outlined in Section 6.0 of the Third Interval ISI Plan. The results are presented in Table 2.

Table 2
Wisconsin Public Service Corporation
Kewaunee Nuclear Power Plant
Third Ten Year Inservice Inspection Interval
Category B-J and C-F-1 Data

Code Class	Category and Item Number	Total Number of nonexempt welds in system boundary	Code Requirement		Total Scheduled for Examination	Number Scheduled Beyond Required
			% Required	Subtotal Subject to Examination		
1	B-J B9.11, B9.12, and B9.31	181	25	45	50	5
2	C-F-1 C5.11, C5.12, and C5.21	149	7.5	11	16	5

This data, generated from Section 7.0 of the Third Interval ISI Plan, shows that WPSC has scheduled ten (10) more volumetric examinations than required as documented in the Third Interval ISI Plan. Due to the higher number of examinations performed on the entire system, WPSC has a higher level of assurance of continued inservice integrity.

During fabrication of the Class 1 and 2 piping systems at KNPP, a surface examination was performed of the weld root, and the final weld was examined by both surface and volumetric examination methods. Many of these welds were re-examined over the last twenty years under the Section XI ISI program and in response to IE Bulletin 79-17, "Stagnant Borated Water Systems."

The ICS system is an emergency safeguards system which is not normally operated except for testing throughout the year. The function of the ICS system is to provide low pressure borated water into the containment building during a loss-of-coolant accident. Reviews of maintenance and repair activities conducted to develop the Third Interval ISI Plan indicate that no thru-wall degradation of the stainless steel ICS pressure boundary has occurred to date at KNPP and none is expected given that the system:

- has not been operated since construction, except for testing and inadvertent actuations;
- is subjected to low pressures and stresses during testing (design is 225 psig and 300°F on the suction side of the pumps, 500 psig and 300°F on the discharge side, and 180 psig and 300°F between the nozzles in containment and the first manual isolation valve); and
- contains a fluid that is periodically sampled to ensure water chemistry, e.g., Cl^- and F^-.

The Class 2 thin-wall portion of the RHR system is used to support normal plant operations on a limited basis throughout the year. The primary function of the RHR system is for the removal of decay heat from the reactor core and to reduce the temperature of the reactor coolant system (RCS) during startup and shutdown of the plant. This portion of the RHR system operates at approximately 400°F and 425 psig. This thin-wall portion of the RHR system can also be used for long term ECCS cooling by recirculating RCS fluid from containment sump B through the RHR heat exchangers to the suction of the SI and ICS pumps should a LOCA occur. To date, only isolated cases of degradation of the class 2 RHR stainless steel pressure boundary have occurred at KNPP (e.g., boric acid buildup on a RHR pump casing due to a casting defect during fabrication). Few or no new cases are expected in the future since this portion of the RHR system is operated for only a few days a year; is subjected to low pressures and stresses; and contains a fluid that is periodically sampled to ensure water chemistry, e.g., Cl^-, F^-, and O_2.

After having reviewed the ISI Plan, water chemistry controls, and pressure boundary performance, WPSC concludes that the proposed number of examinations and the selected examination methods are adequate and reasonable and in fact, meet the intent of the Code. Therefore, we have no current plans to increase the number of volumetric examinations on the thin-wall Class 2 piping. In addition, system integrity of the thin-wall class 2 piping is continuously monitored through the evaluation of industry information from INPO, NRC, utilities, vendors and inhouse activities such as periodic operator/HP/security tours, radiation area monitors, VT-2 examinations, and surface examinations.

B. NRC Request

Requests for Relief RR-1-1 and RR-1-2 state that certain restrictions inhibit access for 100% volumetric examination. The extent of the Code-required volumetric examination that will be performed must be evaluated. Specify the percentage of the Code-required volume that can and will be examined on each of the subject welds and explain how this examination, in addition to the surface examination, will provide a reasonable assurance of the continued inservice structural integrity of the systems.

WPSC Response

Due to access restrictions encountered during the first ISI interval, RR-1-1 and RR-1-2 are relief requests that were requested and granted by the NRC for the second ISI interval. In an effort to retain historical information relative to these welds WPSC again requested the relief requests for the third ISI interval. However, none of these welds were selected to be examined for the third ISI interval. All of the subject welds were volumetrically examined to the extent practical during the second ISI interval, except that the 3-inch diameter RTD weld received only a surface examination in accordance with the method of examination specified in the 1980 Edition through Winter 1981 Addenda of Section XI.

10 CFR 50.55a(ii) allows plants with construction permits docketed prior to July 1, 1978 to use Tables IWB-2500 and IWB-2600, Category B-J of Section XI of the 1974 Edition and addenda through the Summer 1975 Addenda for determining the extent of examination for Code Class 1 pipe welds. WPSC adopted this practice for the selection of class 1 pipe welds under examination category B-J, except that the method, frequency, and applicable footnotes were extracted from the 1989 Edition of Section XI. Paragraph IWB-2420 (c) from the 1974 Edition and addenda through the Summer 1975 Addenda states the following:

> Where less than all the components are required to be inspected in the first inspection interval, a similar percentage of components not previously inspected (other than the preservice examinations) shall be required in each successive inspection interval.

Table IWB-2500 category B-J of the 1974 Edition through the Summer 1975 Addenda of Section XI requires that 25% of the welds be examined each inspection interval. The intent of the 1974 Edition through Summer 1975 Addenda of Section XI is to examine a different group of welds each inspection interval such that all of the welds would be examined once over the 40-year operating license of the plant. Accordingly, to the extent practical, WPSC selected a group of

25% of the welds not previously inspected for examination during the Third ISI Interval. Therefore, relief requests RR-1-1 and RR-1-2 are not needed for the Third ISI Interval and are hereby being withdrawn.

C. NRC Request

For Code Class 1 integral attachment welds to piping, pumps, and valves, the Code does not require examinations for the third and fourth intervals when implementing Inspection Program B. Examination of integral attachments in Code Class 2 and 3 systems is required in the third and fourth interval. The recently approved ASME Code Case N-509 (approved November 25, 1992, by the ASME) provides for continued inspection of Class 1 integral attachments for the life of the plant, as well as some readjustments in the sample inspection requirements for Code Class 2 and 3. Are you following the provisions of Code Case N-509? Describe your plans with respect to implementing this Code Case.

WPSC Response

As you know, Code Case N-509 has not been endorsed by the NRC thru Regulatory Guide 1.147, Revision 10. Furthermore, the Code Case invokes Section XI addenda thru 1990 which have not been approved in 10 CFR 50.55a.

Because of these restrictions, WPSC is not following the provisions of Code Case N-509. The third interval ISI plan is written to satisfy the requirements of the 1989 Edition of Section XI. The 1989 Edition as well as previous editions of Section XI requires examination of class 1 integral attachment welds to piping, pumps and valves prior to service and during the first and second intervals when implementing inspection program B. And the 1989 edition of Section XI and prior editions of Section XI requires examination of integral attachments in Code Class 2 and 3 systems each interval. At this time, WPSC does not have any current plans to implement this Code Case.

D. NRC Request

Code of Federal Regulations, Part 10, 50.55a(g)(6)(ii)(A), states that all licensees must augment their reactor vessel examinations by implementing once, during the inservice inspection interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B-A of the 1989 Code. For licensees, such as Kewaunee, with fewer than 40 months remaining in the interval on the effective date, deferral of the augmented examination is permissible with the conditions stated in the Regulations. Discuss how this augmented examination was (or will be) incorporated in the Kewaunee Nuclear Power Plant inspection program.

WPSC Response

In the past, although not required by earlier editions of Section XI, WPSC has scheduled all of the reactor vessel shell welds under Item No. B1.10 to be ultrasonically examined during each inspection interval. In keeping with this approach, WPSC has scheduled each of these welds to be examined during both the second and third inspection intervals.

In accordance with paragraph IWA-2400(c) of the 1980 Edition through Winter 1981 Addenda of Section XI, WPSC has decided to extend the second ten-year interval by one year (from June 16, 1984 to June 16, 1995) to complete the second ten-year interval reactor vessel examinations. WPSC will also perform third ten-year interval reactor vessel examinations at this time. However, the examinations will be performed to satisfy either the requirements of the remaining second ten-year interval or the first period, third ten-year interval. Individual examinations will not be credited toward both intervals. The 1995 refueling outage is tentatively scheduled from April 1, 1995 through May 19, 1995.

Details regarding the 1995 examination of the reactor vessel are defined in the KNPP second and third ten-year ISI plans and are summarized in Table 3.

Table 3
Summary of 1995 Refueling Outage Reactor Vessel Examination

Edition of Section XI	Interval/Period	Parts Examined
1980 Edition thru Winter 1981 Addenda	2nd Interval - 3rd Period	All Shell Welds (B1.10) All Circ. Head Shell Welds (B1.21) Four Nozzle Welds (Inlet and SI)
1989 Edition	3rd Interval - 1st Period	Two Nozzle Welds (Outlet)

The WPSC practice of examining all of the reactor vessel shell welds once during each inspection interval and our plans to conduct these examinations during the upcoming refueling outage as scheduled in the second and third interval ISI plans satisfies the augmented reactor vessel examination requirements of 10 CFR 50.55a(g)(6)(ii)(A).

E. NRC Request

Relief Requests RR-1-3 and RR-1-4 apply to the ultrasonic examination of the pressurizer nozzle inner radius sections, and steam generator hot leg and intermediate leg nozzle inner radius sections, respectively. WPSC has generically described several obstacles associated with these volumetric examinations and has, in some cases, proposed visual examinations as alternatives. Ultrasonic methods and techniques have improved to the point that the bases cited by the licensee may no longer be obstacles to the performance of the required examinations at Kewaunee Nuclear Power Plant. Please provide sketches of the subject nozzle inner radius sections and discuss why a "best effort" volumetric examination should not be included as part of the proposed alternative examination.

WPSC Response

It has been WPSC practice to pursue the development of advanced NDE techniques and to perform examinations when and where possible for cases where relief is requested on the basis of technology limitations. Shortly after submitting the Third Interval ISI Plan to the NRC, WPSC contracted Westinghouse Electric Corporation to develop techniques and calibration blocks for ultrasonic examination of the pressurizer nozzle inner radius sections. The Westinghouse Electric Corporation ultrasonic examination procedure has been written to satisfy the code requirements. Therefore, WPSC has removed pressurizer safety nozzle inner radius P-IR8, pressurizer safety nozzle inner radius P-IR9, pressurizer relief nozzle inner radius P-IR10, and pressurizer spray nozzle inner radius P-IR11 from relief request No. RR-1-3. The revised relief request is submitted in Attachment 2. Relief is still requested for the pressurizer surge nozzle inner radius section due to access restrictions caused by the heater penetrations and wiring and other limitations discussed in the relief request.

WPSC continues to request relief (RR-1-4) from ultrasonic examination of the steam generator primary nozzle inner radius sections. A camera is usually installed inside the steam generator bowls each refueling outage to facilitate maintenance and inspections of the steam generator tubing. WPSC is proposing to use this camera to conduct visual examinations of the primary nozzle inside radius sections from inside the steam generators. We maintain that the combination of the proposed visual examination on the inside surfaces (each period) and the existing VT-2 examination on the outside surfaces (each refueling outage) are much more reliable for detection of cracks than an ultrasonic examination. For this reason, performance of a Code or "best effort" ultrasonic examination, beyond the proposed remote visual examination, will result in increased radiation exposure and cost without a corresponding increase in benefit.

WPSC is in a position to be able to continue conducting these superior examinations until we replace the steam generators. After the steam generators are replaced, WPSC desires to ultrasonically examine these inner radius sections. Continued visual examinations of the inner radius sections after steam generator replacement would result in unnecessary personnel radiation exposure because a camera would have to be manually installed inside the steam generator bowls for the sole purpose of this examination. Many of the technical difficulties associated with performing an ultrasonic examination of cast material will be eliminated with the use of forged nozzles.

Per your request, please find one copy of the following sketches contained in Attachment 3:

- Technique drawing of the pressurizer surge nozzle; and
- Technique drawing of a series 51 steam generator primary nozzle.

F. NRC Request

Verify that there are no additional requests for relief required at this time. If additional relief requests are required, submit them for staff review.

WPSC Response

WPSC initiated a review of the Third Interval ISI Plan and 1989 Edition of Section XI to identify whether additional relief requests are needed. As a result of this review, WPSC has identified the need for an additional relief request dealing with the VT-2 visual examination conducted during the Class 1 system pressure test. In accordance with 10 CFR 50.55a(g)(5) WPSC hereby submits relief request No. RR-G-2 for NRC staff review and approval. This new relief request is contained in Attachment 4.

ATTACHMENT 2

to

Letter From C.A. Schrock (WPSC)

to

Document Control Desk (NRC)

Dated: October 31, 1994

Re: Relief Request No. RR-1-3

Relief Request No. RR-1-3

1. Components Affected

	Isometric
One Class 1 Nozzle: Pressurizer Surge Nozzle Inner Radius P-IR7	M-1200

2. Section XI Requirements

Volumetric examination of nozzle inner radius per the 1989 edition of Section XI, Table IWB-2500-1, Category B-D, Item B3.120.

3. Basis for Requesting Relief

Ultrasonic examination of the pressurizer surge nozzle inner radius section is undesirable for the following reasons:

a. Coarse grain found in castings causes sound to be attenuated.

b. Difficult to differentiate flaws from normal geometry (clad roll).

c. Access restrictions caused by the pressurizer heater penetrations and associated wiring. Due to the complexity of work on and around the heater penetrations, there is a possibility of damaging this equipment and a potential to adversely impact the outage duration due to scheduling conflicts.

d. Difficulty in removal and replacement of insulation around the heater penetrations and wiring.

e. Increased personnel exposure to radiation and high cost of the examination.

f. There is not a history of industry failures in this area.

4. Alternative Methods of Examination

The surge line (at the bottom of the pressurizer) is inaccessible for visual examination even when the manway (at the top of the pressurizer) is removed; therefore, no alternative examination on the pressurizer surge nozzle can be performed.

The integrity of this nozzle will be verified during the Class 1 system leakage test which is performed after each refueling outage during startup as required by Table IWB-2500-1, Category B-P, Item B15.20.

ATTACHMENT 3

to

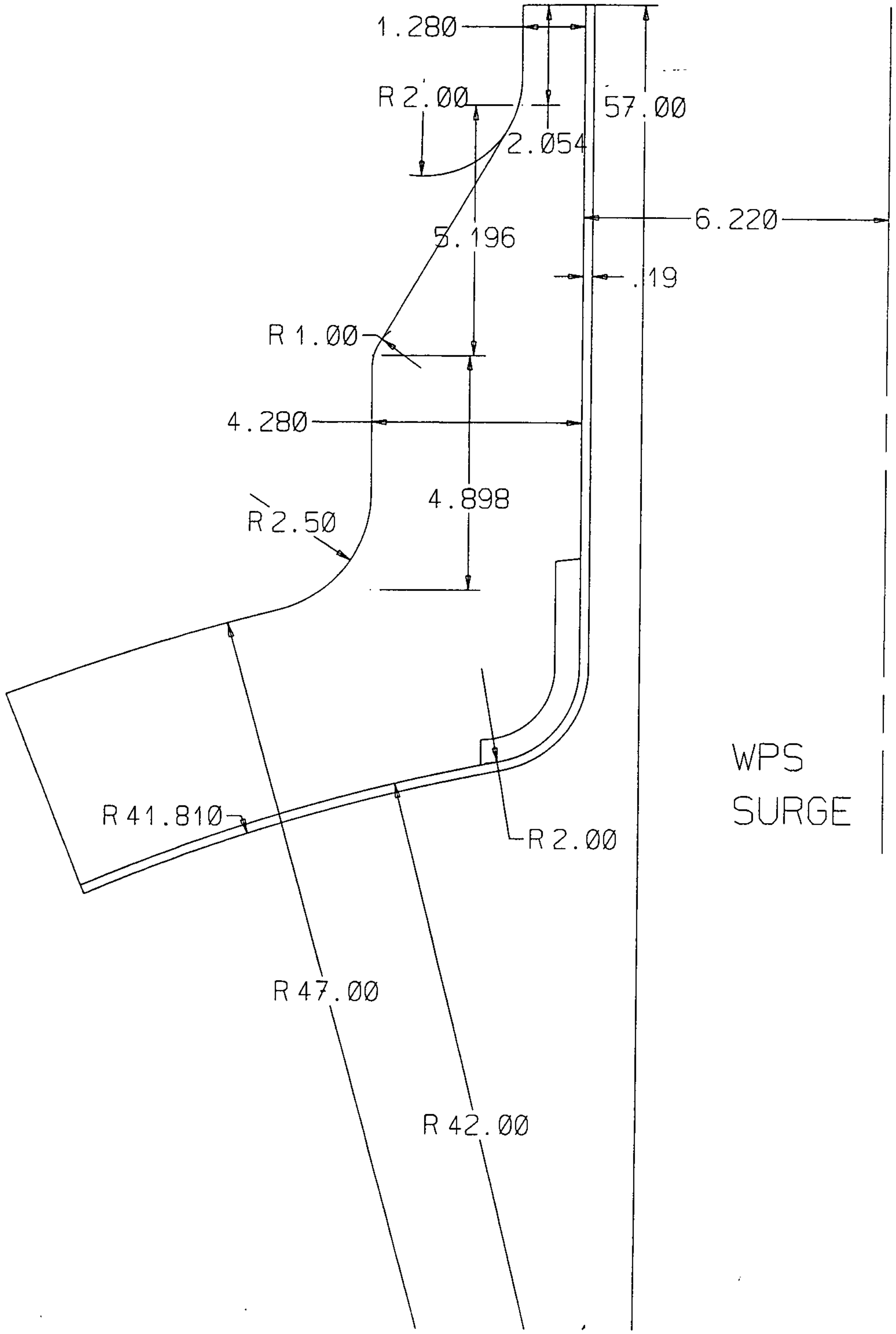
Letter From C.A. Schrock (WPSC)

to

Document Control Desk (NRC)

Dated: October 31, 1994

Re: Sketches of Inner Radius Sections



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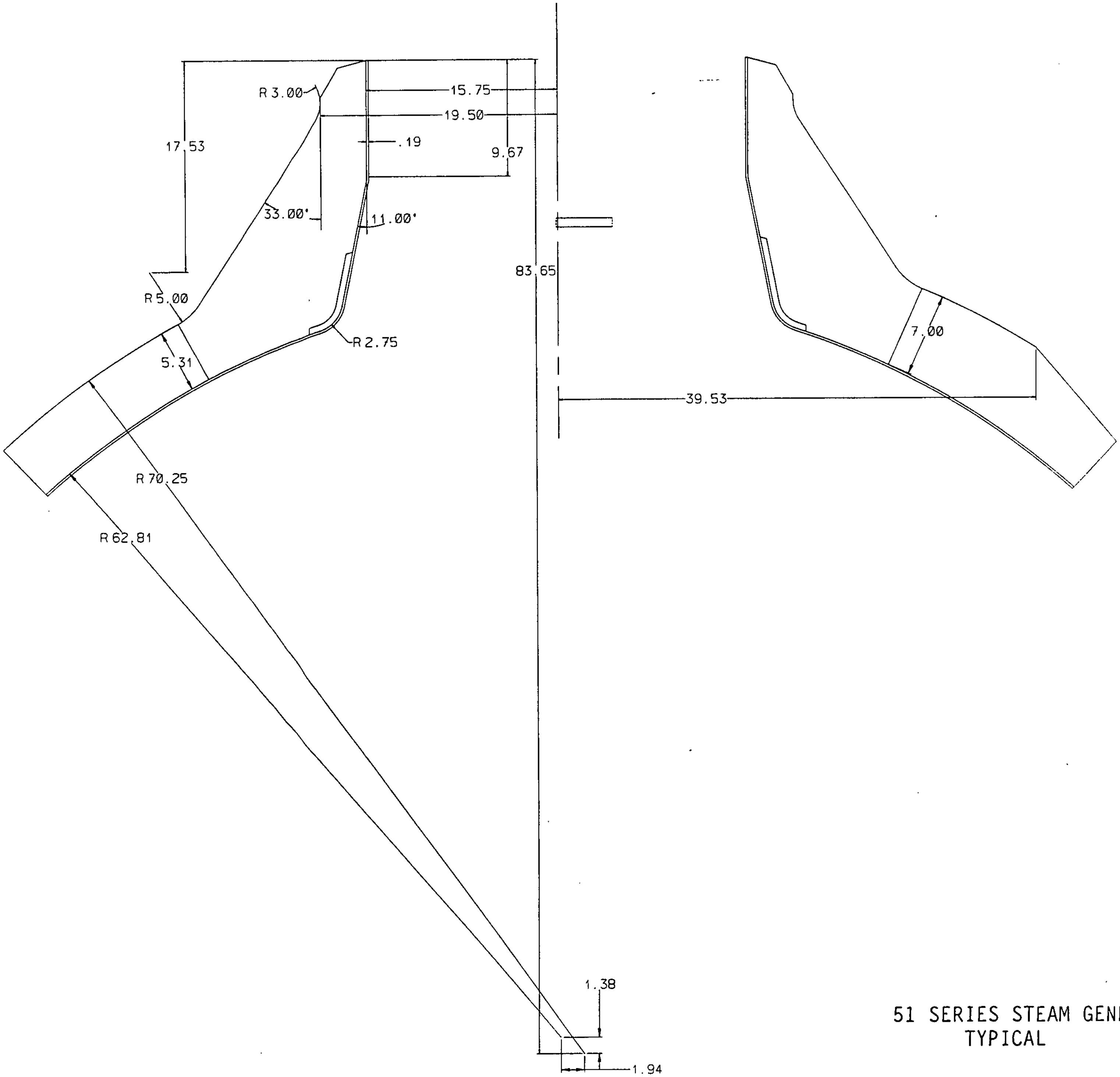
R 42.00



51 SERIES STEAM GENERATOR
TYPICAL

ATTACHMENT 4

to

Letter From C.A. Schrock (WPSC)

to

Document Control Desk (NRC)

Dated: October 31, 1994

Re: Relief Request No. RR-G-2

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Relief Request No. RR-G-2

1. Components Affected

   All Class 1 pressure retaining bolted connections that are insulated.

2. Section XI Requirements

   VT-2 visual examination per the 1989 Edition of Section XI, Table IWB-2500-1, Examination Category B-P and paragraph IWA-5242 which states:

   > (a) For systems borated for the purpose of controlling reactivity, insulation shall be removed from pressure retaining bolted connections for visual examination VT-2. For other components, visual examination VT-2 may be conducted without the removal of insulation by examining the accessible and exposed surfaces and joints of the insulation. Essentially vertical surfaces of insulation need only be examined at the lowest elevation where leakage may be detectable. Essentially horizontal surfaces of insulation shall be examined at each insulation joint.

3. Basis for Requesting Relief

   Satisfying the Code requirement of removing insulation from pressure retaining bolted connections for visual examination of borated systems will require significant planning and scheduling due to operational concerns, personnel radiation, and personnel safety. VT-2 examinations of the Class 1 System at the Kewaunee Nuclear Power Plant are performed at a system operating pressure of 2235 psig and a system temperature of 547°F. Area radiation levels range from 5 mr/hr to 100 mr/hr. Reinsulating and the removal of access equipment after the VT-2 examination will require additional staff to be exposed to higher system pressure, system temperature, and radiation levels than would be experienced during cold shutdown or refueling shutdown.

   Additionally, the time required to replace insulation and remove the access equipment after the VT-2 examination may delay plant startup for an anticipated short time duration between performance of the Class 1 system pressure test and placing the reactor into critical operation. This relief request is intended to cover all pressure retaining bolted connections that are insulated and require VT-2 visual examination under Table IWB-2500-1. Representative components listed below are insulated, are part of or connected to the reactor coolant system, contain pressure retaining bolting, and are pressurized during the Class 1 system pressure test.

Pressure Retaining Components With Bolted Connections That Are Insulated	
reactor vessel flange	8" Valve RHR-1A
pressurizer manway	8" Valve RHR-1B
steam generator primary side manways	6" Valve SI-13A
2" Valve LD-2	6" Valve SI-13B
2" Valve LD-3	12" Valve SI-22A
3" Valve PS-1A	12" Valve SI-22B
3" Valve PS-1B	6" Valve SI-304A
3" Valve RC-103A	6" Valve SI-304B
3" Valve RC-103B	

4. Alternative Method of Examination

A. Perform the VT-2 visual examinations required by Table IWB-2500-1 without removal of insulation.

B. For pressure retaining bolted connections in valves and the pressurizer manway perform a supplemental VT-3 visual examination once every inspection period (3⅓ years) without disassembly and without the system under operating pressure and temperature, during cold shutdown or refueling shutdown. No supplemental examinations are required to ensure integrity of the pressure retaining studs in the reactor vessel flange since they are removed and cleaned to facilitate refueling of the reactor vessel each outage. The steam generator primary side manway bolting is also disassembled and cleaned each refueling outage so no supplemental examinations are needed to ensure their integrity.

Performing the VT-2 visual examinations during cold shutdown or refueling shutdown will significantly reduce the plant operational concerns, personnel radiation and personnel safety. Since borated water leaves a crystalline residue, the proposed supplemental VT-3 visual examination (in addition to the Class 1 system pressure test, area radiation monitors, and RCS leakage detective system) provides reasonable assurance that leakage at pressure retaining bolted connections will be detected and corrected. The proposed VT-3 visual examination at cold or refueling shutdown will permit a more thorough examination than during the Class 1 system pressure test due to better accessibility.