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Dominion®

March 29, 2006

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

Serial No. 06-170
KPS/LIC/GR: RO
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC. (DEK)
KEWAUNEE POWER STATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING INSERVICE INSPECTION THIRD 10-YEAR INTERVAL LIMITATION TO EXAMINATION RELIEF REQUESTS

By letter dated June 23, 2005, Kewaunee Power Station submitted Inservice Inspection Limitation to Examination Relief Requests. These requests for relief are for the third 10-year inservice inspection interval that ended June 16, 2004.

During the Nuclear Regulatory Commission (NRC) staff's review of these relief requests, the staff determined that additional information was required to complete its review. The questions that were raised by the NRC staff, as well as DEK's response to these questions, are provided in the attachment. These responses have been previously discussed with the NRC staff by teleconference on February 7, 2006; it is our understanding that the content of this submittal meets the needs of the NRC staff.

Based on DEK's review of the NRC staff's questions, DEK hereby withdraws the following relief requests:

RR-G-7-5	RR-G-7-6	RR-G-7-7
RR-G-7-11	RR-G-7-17	RR-G-7-18
RR-G-7-19	RR-G-7-24	RR-G-7-29
RR-G-7-38	RR-G-7-46	RR-G-7-58
RR-G-7-62	RR-G-7-63	RR-G-7-72
RR-G-7-73		

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

Very truly yours,

Michael G. Gaffney
Site Vice President – Kewaunee Power Station

A047

Attachment:

Response To Request For Additional Information Regarding Inservice Inspection Third 10-Year Interval Limitation To Examination Relief Requests

Commitments made in this letter: None

cc: Regional Administrator
U. S. Nuclear Regulatory Commission
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2443 Warrenville Road
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Mr. D. H. Jaffe
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Mr. S. C. Burton
NRC Senior Resident Inspector
Kewaunee Power Station

Attachment

<u>RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING INSERVICE INSPECTION THIRD 10-YEAR INTERVAL LIMITATION TO EXAMINATION RELIEF REQUESTS</u>

Dominion Energy Kewaunee, Inc.
Kewaunee Power Station

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING INSERVICE INSPECTION THIRD 10-YEAR INTERVAL LIMITATION TO EXAMINATION RELIEF REQUESTS

2.1 General Information Required:

2.1.1 The licensee has submitted multiple relief requests, based upon the argument that certain ASME Code examinations are impractical at their facility. However, the licensee has not presented adequate descriptive information and technical discussion to support a determination that the required examinations are impractical to perform to the extent required by the ASME Code. The licensee submitted many component design and some as-built drawings to support these requests, however, each request for relief must contain sufficient information to allow the staff to assess the validity of the licensee's determination of impracticality. This may include physical descriptions of the required examination areas (and drawings or sketches may supplement these), specific limitations that impact the required volumetric, surface, or visual examination coverage(s), weld cross-sectional coverage drawings, or other information necessary to support the request. In addition, the licensee should submit argument(s) as to why the use of other inspection methods would not reasonably increase the examination coverage(s).

For Requests for Relief RR-G-7-1 through RR-G-7-23, RR-G-7-64, and RR-G-7-71 through RR-G-7-73, please submit further information to address the following issues:

1) Provide a detailed description of examination limitations including access requirements for automated or manual ultrasonic techniques, and a description of the component geometry that restricts or limits examinations. The technical bases should include cross-sectional sketches of the weld indicating ultrasonic coverage(s) and details of the weld and base metal materials. The technical bases should also confirm, where applicable, that the examinations that were performed have been qualified under ASME Code, Section XI, Appendix VIII, and to which Supplement these methods were qualified. In addition, please list the Edition/Addenda of the ASME Code, Appendix VIII that was applied.

DEK Response:

Request for Relief RR-G-7-1:

In Reference 1 (see page 55), Relief Request RR-G-7-1 included the limitation to examination record, the as built drawing for the vertical residual heat exchanger AHRS1, the Inservice Inspection drawing M-1207 and a photograph of integrally welded attachment AHRS1-SW1. These documents identify that the weld area located at the bottom portion of integrally welded attachments AHRS1-SW1 and AHRS1-SW2 could not be examined due to the physical inability to perform proper surface cleaning and to apply liquid penetrant. These restrictions are related to component geometry. The limited access is due to the integrally welded attachment being bolted to the base support.

The residual heat exchanger AHRS1-1A shell material is A240 TP304 stainless steel, and the integrally welded attachment material is A285 Grade C carbon steel.

For residual heat exchanger AHRS1-1A and integrally welded attachments AHRS1-SW1 and AHRS1-SW2, ASME Code Section XI Appendix VIII, Performance Demonstration For Ultrasonic Examination Systems, did not apply as the examinations were performed by the liquid penetrant surface method.

Other surface examination methods (e.g., magnetic particle) would not be practical and could not reasonably increase the examination coverage, as the surface to be examined is stainless steel. VT-2 and VT-3 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

<u>Request for Relief RR-G-7-2:</u>

In Reference 1, Relief Request RR-G-7-2 included the limitation to examination records, the as built drawing XK-100-486 for the safety injection pumps, the Inservice Inspection drawing M-1707 and a photograph of the safety injection pumps. These documents identify that the weld area located at the bottom portion of integrally welded attachments APSI-1A-S1, APSI-1A-S3, APSI-1A-S4, APSI-1B-S1, APSI-1B-S2, and APSI-1B-S4 could not be examined due to the physical inability to perform proper surface cleaning and limited use of the magnetic particle Y-6 yoke for establishing a magnetic field and applying magnetic particle powder. These restrictions are related to component geometry. The limited access is due to the integrally welded attachment being bolted to the base support.

The safety injection pump casing is ASTM A266 Class 1 carbon steel, and the safety injection pump welded attachments are A216 WC A carbon steel.

The safety injection pumps are normally in standby condition and are normally operated during surveillance procedures. Since the integrally welded attachments are located on the external surface of the pump and are not subjected to known degradation mechanisms except during surveillance testing, inservice degradation is not expected.

For safety injection pump APSI-1A integrally welded attachments APSI-1A-S1, APSI-1A-S3 and APSI-1A-S4 and safety injection pump APSI-1B integrally welded attachments APSI-1B-S1, APSI-1B-S2 and APSI-1B-S4, ASME Code Section XI Appendix VIII, Performance Demonstration For Ultrasonic Examination Systems, did not apply as the examinations were performed by the magnetic particle surface method.

Other surface examination methods (liquid penetrant) would not be practical and could not reasonably increase the examination coverage, as the surface to be examined would still be limited by the integrally welded attachment being bolted to the base support. VT-2 and VT-3 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

Request for Relief RR-G-7-3:

In Reference 1, Relief Request RR-G-7-3 included the limitation to examination record, as built drawing XK-100-486 for safety injection pumps, Inservice Inspection drawing M-1707 and a photograph of safety injection pumps. These documents identify that the weld area located at the bottom portion of safety injection pump APSI-1A integrally welded attachment APSI-1A-S2 and safety injection pump APSI-1B integrally welded attachment APSI-1B-S3 could not be examined due to the physical inability to perform proper surface cleaning and limited use of the magnetic particle Y-6 yoke for establishing a magnetic field and applying magnetic particle powder. These restrictions are related to component geometry. The limited access is due to the integrally welded attachment being bolted to the base support.

The safety injection pump casing is ASTM A266 Class 1 carbon steel and the safety injection pump welded attachments are A216 WC A carbon steel.

The safety injection pumps are normally in standby condition and are normally operated during surveillance procedures. Since the integrally welded attachments are located on the external surface of the pump and are not subjected to known degradation mechanisms except during surveillance testing, inservice degradation is not expected.

Safety injection pump APSI-1A integrally welded attachment APSI-1A-S2 and safety injection pump APSI-1B integrally welded attachment APSI-1B-S3; ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems did not apply as the examinations were performed by the magnetic particle surface method.

Other current surface examination methods (liquid penetrant) would not be practical and could not reasonably increase the examination coverage, as the surface to be examined would still be limited by the integrally welded attachment being bolted to the base support. VT-2 and VT-3 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

Request for Relief RR-G-7-4:

In Reference 1, Relief Request RR-G-7-4 included limitation to examination records, Inservice Inspection drawing M-1198 Sheet 1 of 2 and a photograph of original reactor vessel closure head weld RV-W12. These documents identify that the integrally welded lifting lugs and the flange configuration limit portions of the required 0°, 45° shear (axial scan and circumferential scan) and 60° shear (axial scan and circumferential scan) manual ultrasonic examinations of RV-W12. Limitations occurred due to the inability to place the transducers (45° axial and 60° axial) underneath the integrally welded attachment and to place the transducers (0°, 45° and 60°) to scan axially and circumferentially to the weld. This was due to the flange being approximately 1 inch from the edge of weld RV-W12. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained of the areas not scanned.

The original reactor vessel closure head flange material is A508-64 Class 2 carbon steel and the reactor vessel closure head dome material is A533 Grade B Class 1 carbon steel.

ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems did not apply as the examinations were performed on the reactor vessel closure head flange weld RV-W12, which is exempt from the requirements of ASME Code Section XI Appendix VIII Supplements. Reactor vessel closure head flange weld RV-W12 was examined per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Article I-2000 Examination Requirements.

As noted in Relief Request RR-G-7-4, the original reactor vessel closure head was replaced during the fall 2004 refueling outage. The replacement reactor vessel closure head was manufactured as a one-piece forging eliminating the reactor vessel closure head flange weld.

Request for Relief RR-G-7-5: Charging pump pulsation dampener APD-1A head weld APD-1A-W1, charging pump pulsation dampener APD-1B head weld APD-1B-W4 and spare charging pump pulsation dampener top weld and bottom weld: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-6: Pressurizer head circumferential welds P-W3 and P-W5: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-7: Pressurizer head longitudinal weld P-W2: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-8:

In Reference 1, Relief Request RR-G-7-8 included the limitation to examination record, an as built drawing for vertical residual heat exchanger, Inservice Inspection drawing M-1207 and a photograph of residual heat exchanger shell circumferential weld AHRS1-W1. These documents identify that the weld area of AHRS1-W1, located at the intersection of integrally welded attachments AHRS1-SW1 and AHRS1-SW2, below the residual heat exchanger AHRS1-1A flange and above the 8 inch inlet and outlet nozzles, could not be examined. This weld area could not be examined due to the inability to place portions of the required 0°, 45°, and 60° shear (axial scan and circumferential scan) manual ultrasonic transducers on circumferential weld AHRS1-W1 and the shell base metal. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained of the areas not scanned.

The limited access to weld AHRS1-W1 is due to; 1) the integrally welded attachments being bolted to the base support and covering the circumferential weld for 0°, 45°, and 60° shear (axial scan and circumferential scan), 2) the insertion of the 8 inch inlet and outlet nozzles with reinforcing plates into the shell for 0°, 45°, and 60° shear (axial scan and circumferential scan) and 3) the bolted flange connection for 0°, 45°, and 60° shear (axial scan and circumferential scan).

Currently, ASME Boiler and Pressure Vessel Code Section XI Code Case N-706 "Alternative Examination Requirements of Table IWB-2500-1 and Table IWC-2500-1 for PWR Stainless Steel Residual and Regenerative Heat Exchangers Section XI, Division 1" is available for implementation with Nuclear Regulatory Commission approval. Use of Code Case N-706 would delete requirements for performing ultrasonic examination of residual heat exchanger AHRS1-1A shell circumferential weld AHRS1-W1 and preclude the need for a relief request due to limitations causing less than 90% coverage of the required weld volume. DEK does not intend to apply for relief per ASME Section XI Code Case N-706. However, it is DEK's position that verifying the integrity of shell circumferential weld AHRS1-W1 with a limited ultrasonic examination for residual heat exchanger AHRS1-1A shell circumferential weld AHRS1-W1 is preferable to deletion of the ultrasonic examination requirement.

The other volumetric examination inspection method (i.e., radiography) would not be practical and could not reasonably increase the examination coverage as the surface area accessible for examination would still be limited by the integrally welded attachment bolted to the base support, the insertion of the 8 inch inlet and outlet nozzles with reinforcing plates into the shell, and the bolted flange connection. Additionally, performance of radiography would require residual heat exchanger AHRS1-1A to be removed from service, drained and disassembled, which DEK considers impractical unless the heat exchanger is required to be disassembled for other maintenance. VT-2 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

ASME Code Section XI Appendix VIII, "Performance Demonstration For Ultrasonic Examination Systems," does not apply to residual heat exchanger AHRS1-1A shell circumferential weld AHRS1-W1, as the examination was performed prior to the implementation date of May 22, 2000 for ASME Section XI Appendix VIII Supplement 2 Qualification

Requirements For Wrought Austenitic Piping Welds.

The residual heat exchanger AHRS1-1A shell material is A240 TP304 stainless steel, and the integrally welded attachment material is A285 Grade C carbon steel.

<u>Request for Relief RR-G-7-9:</u>

In Reference 1, Relief Request RR-G-7-9 included the limitation to examination record, an as built drawing for the vertical residual heat exchanger, Inservice Inspection drawing M-1207 and a photograph of AHRS1-W2. These documents identify that the weld area of residual heat exchanger 1A head circumferential weld AHRS1-W2, located at the intersection of integrally welded attachments AHRS1-SW1 and AHRS1-SW2, below the 8-inch inlet and outlet nozzles, could not be examined. This area could not be examined due to the inability to place portions of the required 0°, 45°, and 60° shear (axial scan and circumferential scan) manual ultrasonic transducers on circumferential weld AHRS1-W1 and the shell base metal. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained of the areas not scanned.

The limited access for 0°, 45°, and 60° shear (axial scan and circumferential scan) is due to the integrally welded attachments being bolted to the base support and covering the circumferential weld, and the insertion of the 8-inch inlet and outlet nozzles with reinforcing plates into the shell.

Currently ASME Boiler and Pressure Vessel Code Section XI Code Case N-706 "Alternative Examination Requirements of Table IWB-2500-1 and Table IWC-2500-1 for PWR Stainless Steel Residual and Regenerative Heat Exchanger Section XI, Division 1" is available for implementation with Nuclear Regulatory Commission approval. Use of Code Case N-706 would delete the requirements for performing ultrasonic examination of residual heat exchanger AHRS1-1A shell circumferential weld AHRS1-W2 and preclude the need for a relief request due to limitations causing less than 90% coverage of the required volume. DEK does not intend to apply for relief per ASME Section XI Code Case N-706. DEK's position is that verifying the integrity of shell circumferential weld AHRS1-W2 with a limited ultrasonic examination for residual heat exchanger AHRS1-1A shell circumferential weld AHRS1-W2 is preferable to the deletion of the ultrasonic examination requirement.

The other volumetric examination inspection method (i.e., radiography) would not be practical and could not reasonably increase the examination coverage, as the surface area accessible for examination would still be limited by the integrally welded attachment bolted to the base support and the 8-inch inlet and outlet nozzles with reinforcing plates into the shell. Additionally, performance of radiography would require residual heat exchanger AHRS1-1A to be removed from service, drained and disassembled, which DEK considers impractical unless the heat exchanger is disassembled for other maintenance. VT-2 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

The residual heat exchanger AHRS1-1A shell material is A240 TP304 stainless steel, and the integrally welded attachment material is A285 Grade C carbon steel.

ASME Code Section XI Appendix VIII, "Performance Demonstration For Ultrasonic Examination Systems," did not apply to the examination of the residual heat exchanger AHRS1-1A head circumferential weld as the examination was performed prior to the implementation date of ASME Section XI Appendix VIII Supplement 2 Qualification Requirements For Wrought Austenitic Piping Welds on May 22, 2000.

Request for Relief RR-G-7-10:

In Reference 1, Relief Request RR-G-7-10 included limitation to examination records and Inservice Inspection drawing M-1205, sheets 1 of 2 and 2 of 2. In order to achieve the necessary examination volume, three separate manual ultrasonic examinations were required: 1) 70° shear forward scan, 2) 70° shear reverse scan, and 3) 90° surface scan. These documents identify that the reactor coolant pump main flange bolting could not be examined due to the inability to place the manual ultrasonic 70° shear forward and 90° surface transducer probes completely through the in place bolting due to the bottom configuration, thus limiting the 90° surface examination and the 70° shear forward examination. There were no limitations for the third required scan of 70° shear reverse manual ultrasonic.

The reactor coolant pump main flange bolts are SA540 Grade B24 Class 4 carbon steel.

ASME Code Section XI Appendix VIII, "Performance Demonstration For

Ultrasonic Examination Systems," did not apply to the examinations of the reactor coolant pump 1A main flange bolting RCP-B1 through RCP-B8, RCP-B9, and RCP-B11 through RCP-B17 since the examinations were performed prior to the implementation date of ASME Section XI Appendix VIII Supplement 8, "Qualification Requirements For Bolts and Studs," on May 22, 2000.

Note: Examinations of the reactor coolant pump 1A main flange bolting RCP-B10 and RCP-B18 through RCP-B24 during the third period of the third interval, were performed using the 0° head shot method per the requirements of ASME Boiler and Pressure Vessel Code Section XI 1995 Edition 1996 Addenda Appendix VIII Supplement 8, "Qualification Requirements For Bolts and Studs." These examinations revealed no indications and no limitations. Examinations of the reactor coolant pump 1A main flange bolting RCP-B1 through RCP-B8 during the first period of the fourth interval were performed using the 0° head shot method per the requirements of ASME Boiler and Pressure Vessel Code Section XI 1998 Edition 2000 Addenda Appendix VIII Supplement 8, "Qualification Requirements For Bolts and Studs." These examinations also revealed no indications and no limitations.

The other volumetric examination inspection method (i.e., radiography) would not be practical and could not reasonably increase the examination coverage as the reactor coolant pump main flange bolts are installed in the reactor coolant pump casing. VT-2 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

<u>Request for Relief RR-G-7-11</u>: Regenerative heat exchanger head circumferential weld ARG-W9: Withdrawn per implementation of Code Case N-460.

<u>Request for Relief RR-G-7-12:</u>

In Reference 1, Relief Request RR-G-7-12 included the limitation to examination record and Inservice Inspection drawing M-1201 for SG-IR25, SG-IR26, SG-IR27 and SG-IR28. These documents identify that the steam generator 1A nozzle inside radius sections SG-IR25 and SG-IR26 and steam generator 1B nozzle inside radius sections SG-IR27 and SG-IR28 could not be examined due to the inability to place the 60° shear circumferentially (2 separate scans) manual ultrasonic transducer on the inner radius:

- of SG-IR25 due to integrally welded attachment SG-IA-23D
- of SG-IR26 due to integrally welded attachment SG-IA-23B
- of SG-IR27 due to integrally welded attachment SG-IB-23D
- of SG-IR28 due to integrally welded attachment SG-IB-23B

Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained of the areas not scanned. There were no limitations for the third ultrasonic examination performed with a 38° shear circumferential scan.

6.3% of each nozzle inner radius could not be examined. Although not covered by ASME Code Case N-460, which addresses Class 1 and Class 2 welds, the percentage examined exceeded the 90% required by ASME Code Case N-460.

The steam generator 1A and steam generator 1B nozzle inner radius are SA508 Class 3A carbon steel with stainless steel cladding.

ASME Code Section XI Appendix VIII, "Performance Demonstration For Ultrasonic Examination Systems," did not apply to the steam generator 1A nozzle inner radius SG-IR25 and SG-IR26 and steam generator 1B nozzle inner radius SG-IR27 and SG-IR28 examinations as the steam generator inner radius is exempt from the requirements of the ASME Code Section XI Appendix VIII Supplements.

Other volumetric examination methods (i.e., radiography) would not be practical and could not reasonably increase the examination coverage, as the surface to be examined would require access into the steam generator primary side. VT-2 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

<u>Request for Relief RR-G-7-13:</u>

In Reference 1, Relief Request RR-G-7-13 included a limitation to examination record, the as built drawing for the letdown heat exchanger, Inservice Inspection drawing M-1209, and a photograph of weld AHNR-W2. These documents identify that the weld area of AHNR-W2 located at the intersection of integrally welded attachments AHNR-SW1 and AHNR-SW2, below the 2-inch inlet and outlet nozzles, could not be examined. This weld area could not be examined due to the inability to place portions of the required 0°, 45°, and 60° shear (axial scan and circumferential scan) manual ultrasonic transducers on circumferential weld AHNR-W2 and the shell base metal. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained of the areas not scanned.

The limited access for the required 0°, 45°, and 60° shear (axial scan and circumferential scan) is due to the integrally welded attachments being bolted to the base support and covering the circumferential weld, as well as the insertion of the 2-inch inlet and outlet nozzles with reinforcing plates into the shell.

The letdown heat exchanger AHLD material is A240 TP304 stainless steel.

For examination of letdown heat exchanger head circumferential weld AHNR-W2, ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems Supplement 2 Qualification Requirements For Wrought Austenitic Piping Welds was performed to meet ASME Boiler and Pressure Vessel Code Section XI 1995 Edition 1996 Addenda.

The other volumetric examination inspection method (i.e., radiography) would not be practical and could not reasonably increase the examination coverage as the surface to be examined would still be limited by the integrally welded attachments bolted to the base support and the 2-inch inlet and outlet nozzles with reinforcing plates into the shell. Performance of radiography would require the letdown heat exchanger to be removed from service, drained and disassembled, which DEK considers impractical unless the heat exchanger is disassembled for other maintenance. VT-2 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

<u>Request for Relief RR-G-7-14:</u>

In Reference 1, Relief Request RR-G-7-14 submitted a limitation to examination record, Inservice Inspection drawing M-1212 for seal water injection filter AFSI-1A, and a photograph of AFSI-1A. These documents identify that the weld area of weld AFSI-W1, below the flange above the flange cover hinge plate, weld crown and above the nameplate, could not be examined. This weld area could not be examined due to the inability to place portions of the required 0°, 45° shear (axial scan and circumferential scan) and 60° shear (axial scan) manual ultrasonic transducers on circumferential weld AFSI-W1 and the shell base metal. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained of the areas not scanned.

The limited access is due to;

1) the flange cover hinge plate being welded to seal water injection filter AFSI-1A shell for 45° shear axial scan and circumferential scan,
2) the name plate being welded to the seal water injection filter AFSI-1A shell for 45° shear axial scan,
3) the weld crown for the 45° shear axial scan and circumferential scan and,
4) the bolted flange connection for 45° shear axial scan and circumferential scan.

The seal water injection filter AFSI-1A shell material is SA240 TP304 stainless steel.

For examination of seal water injection filter AFSI-1A shell circumferential weld AFSI-W1, ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems Supplement 2 Qualification Requirements For Wrought Austenitic Piping Welds was performed to meet ASME Boiler and Pressure Vessel Code Section XI 1995 Edition 1996 Addenda.

The other volumetric examination inspection method (i.e., radiography) would not be practical and would not substantially increase the examination coverage as the surface to be examined would still be limited by the flange cover hinge plate, name plate, and bolted flange connection. Performance of radiography would require seal water injection filter AFSI-1A to be removed from service, drained and disassembled, which DEK considers impractical unless the filter is disassembled for other maintenance. VT-2 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

Request for Relief RR-G-7-15:

In Reference 1, Relief Request RR-G-7-15 included a limitation to examination record, Inservice Inspection drawing M-1212 for seal water injection filter AFSI-1A and a photograph of AFSI-1A. These documents identify that the weld area of weld AFSI-W2 at the intersection of integrally welded attachments AFSI-SW1, AFSI-SW2 and AFSI-SW3, and below the 2-inch inlet nozzle, could not be examined. This weld area could not be examined due to the inability to place portions of the required 0°, 45° shear (axial scan and circumferential scan) and 60° shear (axial scan) manual ultrasonic transducers on circumferential weld AFSI-W2 and the shell base metal. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained of the areas not scanned.

The limited access is attributed to the integrally welded attachments being welded to the seal water injection filter 1A shell and covering the circumferential weld for the 0°, 45° shear (axial scan and circumferential scan) and 60° shear (axial scan). Additionally, the limited access is attributed to the insertion of the 2-inch inlet nozzle into the shell for the 45° shear (axial scan and circumferential scan) and 60° shear (axial scan).

The seal water injection filter AFSI-1A shell material is SA240 TP304 stainless steel.

ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems did not apply to the seal water injection filter AFSI-1A head circumferential weld AFSI-W2 as the examination was performed prior to the implementation of ASME Section XI Appendix VIII Supplement 2 Qualification Requirements For Wrought Austenitic Piping Welds on May 22, 2000.

The other volumetric examination inspection method (i.e., radiography) would not be practical and would not substantially increase the examination coverage as the surface to be examined would still be limited by the integrally welded attachments and the 2-inch inlet nozzle inserted into the shell. Performance of radiography would require seal water injection filter AFSI-1A to be removed from service, drained and disassembled, which DEK considers impractical unless the filter is disassembled for other maintenance. VT-2 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

Request for Relief RR-G-7-16:

In Reference 1, Relief Request RR-G-7-16 included limitation to examination records and Inservice Inspection drawing M-1206 for steam generator 1A. These documents identify that the weld area of SG-W2 could not be examined. This weld area could not be examined due to an inability to place the required 0°, 45°, and 60° shear (axial scan and circumferential scan) manual ultrasonic transducers on circumferential shell weld SG-W2, the shell base material and the weld crown. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained of the areas not scanned.

The limited access is due to the integrally welded pads welded to the steam generator 1A shell for 0°, 45° shear (axial scan and circumferential scan) and 60° shear (axial scan) and the blended weld crown configuration for the 45° and 60° shear (axial scan and circumferential).

The steam generator 1A circumferential shell is SA533 Grade A Class 1 carbon steel.

ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems did not apply to examination of the steam generator 1A shell circumferential weld SG-W2 as the examinations were performed on the steam generator shell circumferential weld, which is exempt from ASME Code Section XI Appendix VIII Supplements.

Other volumetric examination methods (e.g., radiography) would not be practical and could not reasonably increase the examination coverage, as the surface to be examined would require draining of, and access into, the steam generator. VT-2 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition. DEK periodically re-examines weld SG-W2 since it contains volumetric type reflectors, which have been shown to be acceptable per IWC-3500. Additionally, during steam generator replacement in 2001, the inside diameter (ID) surface of SG-W2 was examined by the magnetic particle method to ensure the volumetric type indications were not service induced.

Request for Relief RR-G-7-17: Steam generator 1B shell circumferential weld SG-W10: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-18: Steam generator 1B head circumferential weld SG-W9: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-19: Steam generator 1A and 1B tubesheet to shell welds SG-W25 and SG-W31: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-20:

In Reference 1, Relief Request RR-G-7-20 included the examination coverage record, a reactor vessel outlet nozzle to vessel weld exam area sketch and Inservice Inspection drawing M-1194. These documents identify that the weld area of RV-W7 could not be examined due to the inability to complete a tangential scan using an automatic ultrasonic examination system. Limited scan was due to the reactor vessel nozzle boss area as shown on the outlet nozzle weld exam area sketch.

Remote automated ultrasonic examinations of reactor vessel outlet nozzle to vessel weld RV-W7 included 0° longitudinal, 10° longitudinal, 30° longitudinal and 50° longitudinal from the nozzle bore and 45° shear, 60° shear, 0° longitudinal and 70° longitudinal from the reactor vessel shell.

The reactor vessel nozzle material is A508-64 Class 2 carbon steel with stainless steel cladding.

ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems did not apply to reactor vessel outlet nozzle to vessel weld RV-W7 in 1995, as the examination was performed prior to the implementation of ASME Section XI Appendix VIII Supplement 7 Qualification Requirements For Nozzle To Vessel Weld on November 22, 2002.

It should be noted that, while similar limitations continued to be encountered, the reactor vessel outlet nozzle to vessel weld RV-W7 was further examined in fall 2004 to meet KPS fourth10-year interval inservice inspection program (June 16, 2004 - June 16, 2014). Examinations were performed to satisfy the requirements of ASME Boiler and Pressure Vessel Code Section XI 1998 Edition 2000 Addenda including Appendix

VIII Supplement 7 Qualification Requirements For Nozzle To Vessel Weld.

Due to inaccessibility because of component geometry, no other current volumetric examinations could be performed from the ID to reasonably increase the percentage examined.

Request for Relief RR-G-7-21:

In Reference 1, Relief Request RR-G-7-21 included examination coverage record, a reactor vessel outlet nozzle exam area sketch and Inservice Inspection drawing M-1194. These documents identify that the weld area of RV-W10 could not be examined due to the inability to complete a tangential scan using an automatic ultrasonic examination system. Limited scan was due to the reactor vessel nozzle boss area as shown on the outlet nozzle exam area sketch.

Remote automated ultrasonic examinations of reactor vessel outlet nozzle to vessel weld RV-W10 included 0° longitudinal, 10° longitudinal, 30° longitudinal and 50° longitudinal from the nozzle bore and 45° shear, 60° shear, 0° longitudinal and 70° longitudinal from the reactor vessel shell.

The reactor vessel nozzle material is A508-64 Class 2 carbon steel with stainless steel cladding.

ASME Code Section XI Appendix VIII, "Performance Demonstration For Ultrasonic Examination Systems," did not apply to the reactor vessel outlet nozzle to vessel weld RV-W10 when examined in 1995, as the examination was performed prior to the implementation of ASME Section XI Appendix VIII Supplement 7, "Qualification Requirements For Nozzle To Vessel Weld," on November 22, 2002.

It should be noted that, while similar limitations continue to be encountered, the reactor vessel outlet nozzle to vessel weld RV-W10 was further examined in fall 2004 to meet KPS fourth 10-year interval inservice inspection program (June 16, 2004 - June 16, 2014). Examinations were performed to satisfy the requirements of ASME Boiler and Pressure Vessel Code Section XI 1998 Edition 2000 Addenda including Appendix VIII Supplement 7, "Qualification Requirements For Nozzle To Vessel Weld."

Due to inaccessibility, no other current volumetric examinations could be performed from the ID, to reasonably increase the percentage examined.

Request for Relief RR-G-7-22:

In Reference 1, Relief Request RR-G-7-22 included limitation to examination records, Inservice Inspection drawing ISIM-1703 and a photograph of RC-W76DM. These documents identify that portions of the weld area of steam generator 1A nozzle to safe end butt welds RC-W76DM and RC-W77DM could not be examined. These weld area portions could not be examined due to the inability to place the required 0°, 45°, and 60° longitudinal (axial and circumferential) manual ultrasonic transducers on the steam generator nozzle side or the safe end side due to the steam generator nozzle configuration. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained for the areas not scanned.

RC-W76DM and RC-W77DM consist of: 1) Alloy 600 Weld with alloy 690 ID cladding to 2) Safe End of SA 336 Class F 316LN stainless steel and 3) Alloy 690 Weld to 4) SA508 Class 3A carbon steel steam generator 1A nozzle.

ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems did not apply to RC-W76DM and RC-W77DM as these examinations were performed prior to the implementation of ASME Section XI Appendix VIII Supplement 10, "Qualification Requirements For Dissimilar Metal Welds," on November 22, 2002.

Another volumetric examination method (i.e., radiography) was performed as part of the KPS steam generator replacement during the 2001 refueling outage. VT-2 and surface examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

Request for Relief RR-G-7-23:

In Reference 1, Relief Request RR-G-7-23 included limitation to examination records and Inservice Inspection drawing ISIM-1704. These documents identify that portions of the weld area of steam generator 1B nozzle to safe end butt welds RC-W78DM and RC-W79DM could not be examined. These weld area portions could not be examined due to the inability to place the required 0°, 45°, and 60° longitudinal (axial and circumferential) manual ultrasonic transducers on the steam generator nozzle side or the safe end side due to the steam generator nozzle configuration. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained for the areas not scanned.

RC-W78DM and RC-W79DM consist of: 1) Alloy 600 Weld with Alloy 690 ID Cladding to 2) Safe End of SA 336 Class F 316LN stainless steel and 3) Alloy 690 Weld to 4) SA508 Class 3A carbon steel steam generator 1A nozzle.

ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems did not apply to RC-W78DM and RC-W79DM as these examinations were performed prior to the implementation of ASME Section XI Appendix VIII Supplement 10 Qualification Requirements For Dissimilar Metal Welds on November 22, 2002.

Another volumetric examination method (i.e., radiography) was performed as part of the KPS steam generator replacement during the 2001 refueling outage. VT-2 and surface examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

<u>Request for Relief RR-G-7-64:</u>

In Reference 1, Relief Request RR-G-7-64 included limitation to examination record, Inservice Inspection drawing M-1208 and a photograph of regenerative heat exchanger tubesheet to shell circumferential weld ARG-W11. These documents identify that the weld area of ARG-W11 at the location of rigid support bracket ARG-S2 could not be examined. This weld area could not be examined due to an inability to place the required 0° and 45° longitudinal (axial and circumferential) manual ultrasonic transducers on circumferential weld ARG-W11 and the shell base metal. Therefore, no cross sectional sketches of the weld indicating ultrasonic coverage could be obtained for the areas not scanned.

The limited access for the 45° longitudinal (axial and circumferential) is attributed to the rigid support bracket being bolted to the wall plate, welded stops (lug) around circumference of the regenerative heat exchanger and the difficulty in removing and replacing rigid support bracket in a 1–2 Rem per hour radiation field.

The regenerative heat exchanger material consists of; shell, A351 CF8; tubesheet, A182 F304 stainless steel; and head, A240 TP304 stainless steel.

Regenerative heat exchanger tubesheet to shell circumferential weld ARG-W11 ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems did not apply as the examination was performed on the regenerative heat exchanger tubesheet to shell circumferential cast material which is exempt from current required ASME Code Section XI Appendix VIII Supplements.

Currently ASME Boiler and Pressure Vessel Code Section XI Code Case N-706 "Alternative Examination Requirements of Table IWB-2500-1 and Table IWC-2500-1 for PWR Stainless Steel Residual and Regenerative Heat Exchanger Section XI, Division 1" is available for implementation with Nuclear Regulatory Commission approval. Use of Code Case N-706 would delete requirements for performing ultrasonic examination of regenerative heat exchanger tubesheet to shell circumferential weld ARG-W11 and preclude the need for a relief request due to limitations causing less than 90% coverage of required volume. DEK does not intend to apply for relief per ASME Section XI Code Case N-706. It is DEK's position that verifying the integrity of the regenerative heat exchanger tubesheet to shell circumferential weld ARG-W11 with a limited ultrasonic examination is preferable to the deletion of the ultrasonic examination requirement.

Other volumetric examination methods (e.g., radiography) would not be practical and could not reasonably increase the examination coverage, as the surface to be examined would still be limited by the rigid support bracket bolted to the wall plate, welded stops around circumference of the regenerative heat exchanger and the difficulty in removing and replacing a rigid support bracket in a 1– 2 Rem per hour radiation field. VT-2 examinations were performed as required by ASME Boiler and Pressure Vessel Code Section XI 1989 Edition.

Request for Relief RR-G-7-71:

In Reference 1, Relief Request RR-G-7-71 included a Reactor Vessel Coverage Estimate Breakdowns record, reactor vessel shell to flange weld RV-W1 Figure 1 and Figure 2 ultrasonic cross section coverage, and Inservice Inspection drawing M-1193. These documents identify that the weld area of RV-W1 could not be examined by remote automated ultrasonic due to reactor vessel flange configuration.

Remote automated ultrasonic examinations of RV-W1 included 45° longitudinal dual, 45° longitudinal and 45° shear from the reactor vessel ID, and manual ultrasonic examination of reactor vessel shell to flange weld RV-W1 included 0° longitudinal, 6° longitudinal, 12° longitudinal and 16° longitudinal from the reactor vessel flange surface.

The reactor vessel flange material is A508-64 Class 2 carbon steel with stainless steel cladding.

For reactor vessel shell to flange weld RV-W1, ASME Code Section XI Appendix VIII, "Performance Demonstration For Ultrasonic Examination Systems," Supplement 4, "Qualification Requirements For The Clad/Base Metal Interface of the Reactor Vessel," and Supplement 6, "Qualification Requirements for Reactor Vessel Welds Other Than Clad/Base Metal Interface," was performed to meet ASME Boiler and Pressure Vessel Code Section XI 1995 Edition, 1996 Addenda.

No current other volumetric examinations could be performed to reasonably increase the percentage examined.

<u>Request for Relief RR-G-7-72:</u>

In Reference 1, Relief Request RR-G-7-72 included a Reactor Vessel Coverage Estimate Breakdowns record, reactor vessel safety injection nozzle to vessel weld RV-W11 Figure 1 Tan/Star coverage illustration of the safety injection nozzle protrusion limitation, and Inservice Inspection drawing M-1194. These documents identify that the weld area of RV-W11 could not be examined remote automated ultrasonically due to the reactor vessel safety injection nozzle to vessel protrusion configuration.

Remote automated ultrasonic examinations of RV-W11 included 45° longitudinal dual, 45° longitudinal and 45° shear from the reactor vessel ID for the star, and tangential scans and 0° longitudinal, 30° longitudinal and 45° longitudinal from the reactor vessel ID for the weld bore scans.

The reactor vessel safety injection nozzle to vessel material is A508-64 Class 2 carbon steel with stainless steel cladding.

For weld RV-W11, ASME Code Section XI Appendix VIII Performance Demonstration For Ultrasonic Examination Systems Supplement 7

Qualification Requirements For Nozzle To Vessel Weld was performed to meet ASME Boiler and Pressure Vessel Code Section XI 1995 Edition 1996 Addenda.

No other volumetric examinations from the ID could be performed to reasonably increase the percentage examined.

Review of the examination data indicates that 90.4% of weld RV-W11 has been examined. This percentage is based on combining the volumes from the double-sided and single-sided examinations. Since DEK is implementing ASME Code Case N-460, Relief Request RR-G-7-72 will be withdrawn.

<u>Request for Relief RR-G-7-73:</u>

In Reference 1, Relief Request RR-G-7-73 included a Reactor Vessel Coverage Estimate Breakdowns record, reactor vessel lower head circumferential weld RV-W4 Figure 1 ultrasonic cross section coverage and Inservice Inspection drawing M-1193. These documents identify that the weld area of RV-W4 could not be examined with remote automated ultrasonics due to four welded reactor vessel lower core supports.

Remote automated ultrasonic examinations of weld RV-W4 included 45° longitudinal dual, 45° longitudinal and 45° shear from the reactor vessel ID.

The reactor vessel lower head material is A508-64 Class 2 carbon steel with stainless steel cladding.

For reactor vessel lower head circumferential weld RV-W4, ASME Code Section XI Appendix VIII, "Performance Demonstration For Ultrasonic Examination Systems," Supplement 4, "Qualification Requirements For The Clad/Base Metal Interface of the Reactor Vessel," and Supplement 6, "Qualification requirements for Reactor Vessel Welds Other Than Clad/Base Metal Interface," was performed to meet ASME Boiler and Pressure Vessel Code Section XI 1995 Edition 1996 Addenda.

No other volumetric examinations from the ID could be performed to reasonably increase the percentage examined.

Review of the examination data indicates that 90.82% of weld RV-W4 has been examined. This percentage is based on combining the volumes from the double-sided and single-sided examinations. Since DEK is implementing ASME Code Case N-460, Relief Request RR-G-7-73 will be withdrawn.

2) *Indicate what degradation mechanism(s) are most likely to occur in the subject welds and a technical basis for whether the achieved examination coverage could reasonably be expected to detect this degradation.*

DEK Response:

Request for Relief RR-G-7-1: Residual heat exchanger AHRS1-1A integrally welded attachments AHRS1-SW1 and AHRS1-SW2.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding DEK Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-2: Safety injection pump APSI-1A integrally welded attachments APSI-1A-S1, APSI-1A-S3 and APSI-1A-S4 and safety injection pump APSI-1B integrally welded attachments APSI-1B-S1, APSI-1B-S2 and APSI-1B-S4.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding DEK Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-3: Safety injection pump APSI-1A integrally welded attachment APSI-1A-S2 and safety injection pump APSI-1B integrally welded attachment APSI-1B-S3.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding DEK Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-4: Reactor vessel closure head to flange weld RV-W12.

Degradation Mechanisms: No degradation mechanisms for the original reactor vessel closure head, as the original reactor vessel closure head was replaced during the fall 2004 refueling outage with the replacement reactor vessel closure head manufactured as a one-piece forging, thus eliminating the reactor vessel closure head flange weld.

Request for Relief RR-G-7-5: Charging pump pulsation dampener APD-1A head circumferential weld APD-1A-W1, charging pump pulsation dampener APD-1B head circumferential weld APD-1B-W4 and spare charging pump pulsation dampener top weld and bottom weld: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-6: Pressurizer head circumferential welds P-W3 and P-W5: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-7: Pressurizer head longitudinal weld P-W2: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-8: Residual heat exchanger AHRS1-1A shell circumferential weld AHRS1-W1.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding DEK Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-9: Residual heat exchanger AHRS1-1A head circumferential weld AHRS1-W2.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding DEK Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-10: Reactor coolant pump 1A main flange bolting RCP-B1 through RCP-B8, RCP-B9 and RCP-B11 through RCP-B17.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding DEK Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-11: Regenerative heat exchanger head circumferential weld ARG-W9: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-12: Steam generator 1A nozzle inside radius sections SG-IR25 and SG-IR26 and steam generator 1B nozzle inside radius sections SG-IR27 and SG-IR28.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding DEK Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-13: Letdown heat exchanger head circumferential weld AHNR-W2.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding DEK Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-14: Seal water injection filter AFSI-1A shell circumferential weld AFSI-W1.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding DEK Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-15: Seal water injection filter AFSI-1A head circumferential weld AFSI-W2.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-16: Steam generator 1A shell circumferential weld SG-W2.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-17: Steam generator 1B shell circumferential weld SG-W10: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-18: Steam generator 1B head circumferential weld SG-W9: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-19: Steam Generator 1A and 1B tubesheet to shell circumferential welds SG-W25 and SG-W31: Withdrawn per implementation of Code Case N-460.

Request for Relief RR-G-7-20: Reactor Vessel Outlet Nozzle To Vessel Weld RV-W7.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-21: Reactor Vessel Outlet Nozzle To Vessel Weld RV-W10.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-22: Steam Generator 1A Nozzle To Safe End Butt Welds RC-W76DM and RC-W77DM.

Degradation Mechanisms: For the KPS fourth 10-year interval (June 16, 2004 - June 16, 2014), KPS implemented a risk informed program following the guidelines of ASME Boiler and Pressure Vessel Code Section XI Code Case N-578 "Risk Informed Requirements For Class 1, 2 and 3 Piping Method B Section XI Division 1" and Electric Power Research Institute Risk Informed Program Topical Report TR-112657 Rev. B-A "Revised Risk-Informed Inservice Inspection Evaluation Procedure". The Nuclear Regulatory Commission approved the KPS risk informed program on September 23, 2005 under TAC Nos. MC2502, MC2508 and MC2537. Steam generator 1A nozzle to safe end butt weld RC-W76DM is currently classified as a Risk Characterization Category - 4, Risk Characterization Rank - Medium, Consequence Rank – High, Failure Potential Degradation Mechanisms – None, and Failure Potential Rank – Low. Steam generator 1A nozzle to safe end butt weld RC-W77DM is currently classified as a Risk Characterization Category - 4, Risk Characterization Rank - Medium, Consequence Rank – High, Failure Potential Degradation Mechanisms – None, and Failure Potential Rank – Low.

Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-23: Steam Generator 1B Nozzle To Safe End Butt Welds RC-W78DM and RC-W79DM.

Degradation Mechanisms: For the KPS fourth 10-year interval (June 16, 2004 - June 16, 2014) KPS implemented a risk informed program following the guidelines of ASME Boiler and Pressure Vessel Code Section XI Code Case N-578 "Risk Informed Requirements For Class 1, 2 and 3 Piping Method B Section XI Division 1" and Electric Power Research Institute Risk Informed Program Topical Report TR-112657 Rev. B-A "Revised Risk-Informed Inservice Inspection Evaluation Procedure". The Nuclear Regulatory Commission approved the KPS risk informed program on September 23, 2005 under TAC MC2502, MC2508 and MC2537. Steam generator 1B nozzle to safe end butt weld RC-W78DM is currently classified as a Risk Characterization Category - 4, Risk Characterization Rank - Medium, Consequence Rank – High, Failure Potential Degradation Mechanisms – None, and Failure Potential Rank –

Low. Steam generator 1B nozzle to safe end butt weld RC-W79DM is currently classified as a Risk Characterization Category - 4, Risk Characterization Rank - Medium, Consequence Rank – High, Failure Potential Degradation Mechanisms – None, and Failure Potential Rank – Low.

Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-64: Regenerative Heat Exchanger Tubesheet To Shell Circumferential Weld ARG-W11.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-71: Reactor Vessel Shell To Flange Weld RV-W1.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-72: Reactor Vessel Safety Injection Nozzle to Vessel Weld RV-W11.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

Request for Relief RR-G-7-73: Reactor Vessel Lower Head Circumferential Weld RV-W4.

Degradation Mechanisms: Reference Table 1: Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval.

2.1.2 Several of the licensee's requests for relief contain examination limitation reports. In general, these reports provide supplemental information to aid in evaluation of the requests, such as sketches of the components showing certain access features or restrictions, and completed percentages for each inspection method. However, it has been noted that in many cases, two different examination coverages have been reported. These differing values correspond to a procedure coverage percentage, and an ASME Code coverage percentage. Please explain why two limitation coverage values are listed, and describe how these relate to each of the subject requests.

DEK Response:

The site specific procedures used by KPS at the time of examinations for the third 10-year interval (June 16, 1994 - June 16, 2004) required that examinations be performed using the following ultrasonic examination techniques, which exceeded ASME Boiler and Pressure Vessel Code Section XI 1989 Edition requirements:

1. A 0°, 45° and 60° on austenitic and ferritic steel piping welds and austenitic vessel welds.

2. Examinations to be performed from BOTH sides of the weld to cover the required volume from ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Figures IWB-2500-8, IWC-2500-1, IWC-2500-2, and IWC-2500-7, as applicable.

3. If examinations could not be performed to satisfy items 1 and 2, the percentage of coverage not obtained was to be based on those criteria and listed as Procedure Percentage.

The requirements for compliance to ASME Boiler and Pressure Vessel Code Section XI 1989 Edition for the KPS third 10-year interval (June 16, 1994 – June 16, 2004) were ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Article III-4000 and Appendix III Supplement 4, which requires:

Article III-4000

III-4410 Beam Angle: The search unit and beam angle selected shall be capable of detecting the calibration reflectors, as described in III-3230(c), over the required angle beam path. A beam angle of 45 degrees in the material shall generally be used. Other angles may be used for evaluation of an indication, or where wall thickness or geometric configuration impedes effective use of the 45-degree angle beam for examination.

III-4420 Reflectors Parallel To The Weld Seam: The examination shall be performed using a sufficiently long examination beam path to provide coverage of the required examination volume in two-beam path directions. The examination shall be performed from two sides of the weld, where practicable, or from one side of the weld, as a minimum.

III-4430 Reflectors Transverse To The Weld Seam: The angle beam examination for reflectors transverse to the weld shall be performed on the weld crown on a single scan path to examine the weld root by one half V path in two directions along the weld.

Supplement 4 Austenitic and Dissimilar Metal Welds:

(1) III-4410 Beam Angle: The actual beam angle in the examination part shall be 40 degrees or greater for shear wave at the ID surface and 35 degrees or greater for refracted, longitudinal wave at the ID surface. The beam angle in the examination part shall be determined for each pipe size, schedule, and material to be examined for each plant. The beam angle measurements shall be used to assure coverage of the required examination volume by extending the calibration and examination distance, as required.

(2) III-4430 Reflectors Transverse To The Weld Seam: Substitute: The angle beam examination for reflectors transverse to the weld shall be performed in two directions covering the minimum area from ½ inch from one side of the weld crown to ½ inch from the other side of the weld crown including the crown.

(3) Table III-3430-1 Calibration Notches – Substitute: depth 10% of *t*

(4) Scanning from both sides of the weld is required where practical. Single side access limitations shall be noted in the examination data record. Cast items such as fittings, valve bodies, and pump casings may preclude meaningful examinations because of geometry and attenuation variables.

On data forms at KPS if examinations could not be performed to satisfy ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Article III and Appendix III Supplement 4 of the required volume from Figures IWB-2500-8, IWC-2500-1, IWC-2500-2, or IWC-2500-7, the percentage of coverage not obtained was to be based on those criteria and listed as Code Percentage.

2.2 Examination Coverage Obtained and Implementation of ASME Code Case N-460:

In the introductory portion of Enclosure 1 provided in the licensee's submittal, it is indicated that KNPP has adopted ASME Code Case N-460, which clarifies the ASME Code 100% coverage requirement to be coverage greater than 90% of the examination volume, or surface area, as applicable. The NRC has approved this ASME Code Case use on all Class 1 and 2 welds. However, greater than 90% examination coverage appears to have been obtained for specific welds in several requests for relief, as shown below. For each of the relief requests identified below, where greater than 90% examination volume coverage was achieved, please explain why relief is being requested.

2.2.1 Request for Relief RR-G-7-5, (TAC MC7901), Examination Category C-A, Item C1.20, head Welds APD-1A-W1 and APD-1B-W4 on Charging Pump Pulsation Dampeners

In section 4.0 of this request, the licensee stated that between 8.6% and 9.0% of the required ASME Code coverage could not be obtained. If greater than 90% of the welds were examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

DEK agrees that Examination Category C-A, Item C1.20, Head Welds APD-1A-W1 and APD-1B-W4 on the charging pump pulsation dampeners were examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-5 will be withdrawn.

2.2.2 Request for Relief RR-G-7-6 (TAC MC7902), Examination Category B-B, Item B2.11, Pressurizer head circumferential Welds P-W3 and P-W5

In section 4.0 of this request, the licensee stated that 2.0% of the required ASME Code coverage could not be obtained. If greater than 90% of the weld was examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

DEK agrees that Examination Category B-B, Item B2.11, Pressurizer Head Circumferential Welds P-W3 and P-W5 were examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-6 will be withdrawn.

2.2.3 Request for Relief RR-G-7-7 (TAC MC7903), Examination Category B-B, Item B2.11, Pressurizer head Longitudinal Weld P-W2

In section 4.0 of this request, the licensee stated that 3.0% of the required ASME Code coverage could not be obtained. If greater than 90% of the weld was examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

DEK agrees that Examination Category B-B, Item B2.11, Pressurizer Head Longitudinal Weld P-W2 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-7 will be withdrawn.

2.2.4 Request for Relief RR-G-7-11, Examination Category C-A, Item C1.20, Regenerative Heat Exchanger head circumferential Weld ARG-W9

In section 4.0 of this request, the licensee stated that 1.5% of the required ASME Code coverage could not be obtained. If greater than 90% of the weld was examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

DEK agrees that Examination Category C-A, Item C1.20, Regenerative Heat Exchanger head circumferential Weld ARG-W9 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-11 will be withdrawn.

2.2.5 Request for Relief RR-G-7-12 (TAC MC7908), Examination Category B-D, Item B3.140, Steam Generator 1A Nozzle Inside Radius Sections SG-IR25 and SG-IR26 and Steam Generator 1B Nozzle Inside Radius Section SG-IR27 and SG-IR28

In section 4.0 of this request, the licensee stated that 6.3% of the required ASME Code coverage could not be obtained. If greater than 90% of the inside radius sections was examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

DEK agrees that Examination Category B-D, Item B3.140, Steam Generator 1A Nozzle Inside Radius Sections SG-IR25 and SG-IR26, and Steam Generator 1B Nozzle Inside Radius Section SG-IR27 and SG-IR28 were examined greater than 90%. However, ASME Code Case N-460 Relief Request addresses Class 1 and Class 2 welds. SG-IR25, SG-IR25, SG-IR27 and SG-IR28 are Nozzle Inside Radius Sections (not welded) and a Relief Request would still be required. Although not covered by ASME Code Case N-460, which addresses Class 1 and Class 2 welds, the percentage examined exceeded the 90% required by ASME Code Case N-460.

2.2.6 Request for Relief RR-G-7-17 (TAC MC7913), Examination Category C-A, Item C1.10, Steam Generator 1B circumferential shell Weld SG-W10

In section 4.0 of this request, the licensee stated that 7.9% of the required ASME Code coverage could not be obtained. If greater than 90% of the inside radius sections was examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

DEK agrees that Examination Category C-A, Item C1.10, Steam Generator 1B Circumferential Shell Weld SG-W10 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-17 will be withdrawn.

2.2.7 Request for Relief RR-G-7-18 (TAC MC7914), Examination Category C-A, Item C1.20, Steam Generator 1B circumferential head Weld SG-W9

In section 4.0 of this request, the licensee stated that 0.113% of the required ASME Code coverage could not be obtained. If greater than 90% of the inside radius sections was examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

DEK agrees that Examination Category C-A, Item C1.20, Steam Generator 1B Circumferential Head Weld SG-W9 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-18 will be withdrawn.

2.2.8 <u>Request for Relief RR-G-7-19 (TAC MC7915), Examination Category C-A, Item C1.30, Steam Generator 1A and Steam Generator 1B tubesheet-to-shell Welds SG-W25 and SG-W31</u>

In section 4.0 of this request, the licensee stated that 9.0% of the required ASME Code coverage could not be obtained. If greater than 90% of the inside radius sections was examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

DEK agrees that Examination Category C-A, Item C1.30, Steam Generator 1A and Steam Generator 1B tubesheet to shell welds SG-W25 and SG-W31 were examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-19 will be withdrawn.

2.2.9 <u>Request for Relief RR-G-7-72 (TAC MC7968), Examination Category B-D, Item 3.90, Reactor Vessel Safety Injection nozzle-to-vessel Weld RV-W11</u>

In section 4.0 of this request, the licensee stated that 11.69% of Weld RV-W11 could not be examined because of access constraints. However, in section 6.0 the licensee stated that a coverage of 90.04% of the weld examination volume was achieved. It is unclear how much volumetric coverage was actually obtained during examination of this weld. If greater than 90% of the weld was examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

Review of the examination data including coverage estimates indicates that 90.4% is the appropriate coverage estimate. The double sided and single sided coverage results were considered in combination. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-72 will be withdrawn.

2.2.10 Request for Relief RR-G-7-73 (TAC MC7969), Examination Category B-A, Item B1.21, Reactor Vessel Lower Head Circumferential Weld RV-W4

In section 4.0 of this request, the licensee stated that 13.86% of Weld RV-W4 could not be examined because of access constraints. However, in section 6.0 the licensee stated that a coverage of 90.82% of the weld was achieved. It is unclear how much volumetric coverage was actually obtained during examination of this weld. If greater than 90% of the weld was examined, and KNPP is adopting ASME Code Case N-460, this request should be withdrawn. Otherwise please ensure that the request contains the general information required in Section 2.1 above.

DEK Response:

Review of the cover estimates and examination data indicates that it is appropriate to use the 90.82 % coverage estimate. The double sided and single sided coverage results were considered in combination. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-73 will be withdrawn

2.3 Request for Relief RR-G-7-4 (TAC MC7900), Examination Category B-A, Item B1.40, Reactor Vessel Closure Head-to-Flange Weld RV-W12

2.3.1 Please list the date that the original reactor pressure vessel (RPV) head was removed from service, and the date that the new RPV head was placed into service.

DEK Response:

The KPS original reactor vessel closure head was removed from service on October 15, 2004, when the reactor vessel closure head was lifted from the reactor vessel for the start of refueling activities and removal from the KPS site. The replacement reactor vessel closure head was placed into service on

December 4, 2004, following the fall 2004 refueling outage.

2.4 Request for Relief RR-G-7-10 (TAC MC7906), Examination Category B-G-1, Item B6.180, Reactor Coolant Pump 1A main closure flange bolts

2.4.1 Describe any indications noted during the examinations performed on these bolts. Also, state whether any of these bolts have been replaced during the interval, and if so, whether the request is for the replaced bolting or original bolting.

DEK Response:

There were no indications noted during the examinations performed on the reactor coolant pump 1A main closure flange bolts. The reactor coolant pump 1A main flange bolting examined for this relief request was the original bolting.

2.5 Requests for Relief Associated with the Steam Generators 1A and 1B

The licensee submitted several requests on shell and nozzle welds for these steam generators. However, it is unclear whether the requests are for examinations performed prior to replacement of the steam generators, or for new welds fabricated as part of the replacement project, which is stated to have occurred in 2001. The requests are:

RR-G-7-16 (TAC MC9712)	RR-G-7-17 (TAC MC9713)
RR-G-7-18 (TAC MC9714)	RR-G-7-19 (TAC MC9715)
RR-G-7-22 (TAC MC9718)	RR-G-7-23 (TAC MC9719)

DEK Response:

The steam generator replacement project occurred in September 2001. The replacement steam generators at Kewaunee included replacement of the lower shell and tubesheet and refurbishment of the original upper shell steam dome. The following is clarification:

RR-G-7-16 (TAC MC9712) for Steam Generator 1A Shell Circumferential Weld SG-W2: Manual ultrasonic examination was performed at the time of the steam generator replacement project. It is an original upper shell weld that remained with the upper shell and is currently inservice.

RR-G-7-17 (TAC MC9713) for Steam Generator 1B Shell Circumferential Weld SG-W10: Manual ultrasonic examination was performed at the time of the steam generator replacement project. It is an original upper shell weld that remained with the upper shell and is currently inservice.

RR-G-7-18 (TAC MC9714) for Steam Generator 1B Shell Circumferential Weld SG-W9: Manual ultrasonic examination was performed at the time of the steam generator replacement project. It is an original upper shell weld that remained with the upper shell and is currently inservice.

RR-G-7-19 (TAC MC9715) for Steam Generator 1A Tubesheet to Shell Circumferential Weld SG-W25 and Steam Generator 1B Tubesheet to Shell Circumferential Weld SG-W31: Manual ultrasonic examinations were performed at the time of the steam generator replacement project and are new fabrication welds currently inservice.

RR-G-7-22 (TAC MC9718) for Steam Generator 1A Nozzle to Safe End Butt Welds RC-W76DM and RC-W77DM: Manual ultrasonic examinations were performed at the time of the steam generator replacement project and are new fabrication welds currently inservice.

RR-G-7-23 (TAC MC9719) for Steam Generator 1B Nozzle to Safe End Butt Welds RC-W78DM and RC-W79DM: Manual ultrasonic examinations were performed at the time of the steam generator replacement project and are new fabrication welds currently inservice.

2.5.1 *All of the requests indicate that the examinations were performed in 2001, and it is assumed that the examinations occurred during an extended steam generator replacement outage. There are several questions and issues associated with the requests, depending on whether they are for welds on the old steam generators, original welds that remain inservice on the new steam generators, or newly fabricated welds. Please describe the replacement process, including which portions of the steam generators were actually replaced, and identify, in the context of component replacement, which of the original shell and nozzle welds remain and which new welds were fabricated as part of the replacement project. In addition, provide the following information:*

1) For applicable requests associated with original shell, nozzle and/or nozzle-to-pipe dissimilar metal welds, confirm that these welds continue to be in service as part of the new steam generators.

DEK Response:

The steam generator replacement project occurred in September 2001. The replacement steam generators at Kewaunee included replacement of the lower shell and tubesheet and refurbishment of the upper shell steam dome. The following is clarification:

SG-W2: Steam generator 1A shell circumferential weld is from the original steam generator 1A upper shell steam dome and continues to be inservice.

SG-W9: Steam generator 1B head circumferential weld is from the original steam generator 1B upper shell steam dome and continues to be inservice.

SG-W10: Steam generator 1B shell circumferential weld is from the original steam generator 1B upper shell steam dome and continues to be inservice.

SG-W25: Steam generator 1A tubesheet to shell circumferential weld is a new fabrication weld from the replacement steam generator lower shell and is currently inservice.

SG-W31: Steam generator 1B tubesheet to shell circumferential weld is a new fabrication Weld from the replacement steam generator lower shell and is currently inservice.

RC-W76DM: Steam generator 1A nozzle to safe end butt weld is a new fabrication weld from the replacement steam generator tubesheet and is currently inservice.

RC-W77DM: Steam generator 1A nozzle to safe end butt weld is a new fabrication weld from the replacement steam generator tubesheet and is currently inservice.

RC-W78DM: Steam generator 1B nozzle to safe end butt weld is a new fabrication weld from the replacement steam generator tubesheet and is currently inservice.

RC-W79DM: Steam generator 1B nozzle to safe end butt weld is a new fabrication weld from the replacement steam generator tubesheet and is currently inservice.

2) For any Class 1 and 2 welds fabricated as part of the replacement, describe all preservice examinations, including results and limitations, and confirm that any limitations to ASME Code requirements have been submitted as proposed alternatives under 10 CFR 50.55a(a)(3)(i) or (ii). Note that impracticality is not a viable basis for limited examination on new welds.

DEK Response:

SG-W25: Steam generator 1A tubesheet to shell circumferential weld is a fabrication weld on the replacement steam generator and is currently inservice. DEK agrees that Examination Category C-A, Item C1.30, steam generator 1A tubesheet to shell weld SG-W25 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-19 will be withdrawn. Preservice examination was performed on SG-W25 to meet ASME Boiler and Pressure Vessel Code Section XI 1989 Edition. There were no indications recorded.

SG-W31: Steam generator 1B tubesheet to shell circumferential weld is a fabrication weld on the replacement steam generator and is currently inservice. DEK agrees that Examination Category C-A, Item C1.30, steam generator 1B tubesheet to shell weld SG-W31 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-19 will be withdrawn. Preservice examination was performed on SG-W31 to meet ASME Boiler and Pressure Vessel Code Section XI 1989 Edition. There were no indications recorded that exceeded ASME Boiler and Pressure Vessel Code Section XI Acceptance Standards.

Steam generator 1A nozzle to safe end butt welds RC-W76DM and RC-W77DM and steam generator 1B nozzle to safe end butt welds RC-W78DM and RC-W79DM were classified as fabrication welds. This classification is as defined in ASME Boiler and Pressure Vessel Code Section XI Code Case N-416-2 Alternate Pressure Test Requirement for Welded Repairs, Fabrication Welds for Replacement Parts and Piping Subassemblies, or Installation of Replacement Items by Welding Class 1, Class 2 and Class 3 Section XI, Division 1. Use of ASME Code Case N-416-2 was approved for use at KPS by the Nuclear Regulatory Commission on February 9, 2001 TAC NO. MB0307. The fabrication of RC-W76DM, RC-W77DM, RC-W78DM and RC-W79DM was performed on site at KPS following the shipment of the steam generators from the manufacturer (Ansaldo-Energia). KPS, working with it's nondestructive examination vendor LMT Inc. developed an ultrasonic procedure using the latest available techniques in summer 2001 to examine the four nozzle to safe end butt welds to meet ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Examination Volume Figure IWB-2500-8. The examination was only able to examine 61.9% of RC-W76DM and RC-W77DM and 59.75% of RC-W78DM and RC-W79DM. The main limitation was the inability to scan from the steam generator nozzle side due to the nozzle configuration, thus limiting the examination to a one side examination using 0°, 45° longitudinal and 60° longitudinal transducers. The required examination volumes when examined from the one side were 82.6% for welds RC-W76DM and RC-W77DM and 79.7% for welds RC-W78DM and RC-W79DM. There were no indications noted by ultrasonic examinations. Relief requests for RC-W76DM, RC-W77DM, RC-W78DM and RC-790DM to satisfy ASME Code requirements under 10 CFR 50.55a(a)(3)(i) or (ii) were included as part of the third 10-year limitation to examination relief requests.

Since the performance of these examinations in June 2001, ASME Boiler and Pressure Vessel Code Section XI has implemented requirements to perform Appendix VIII Supplement 10 Qualification Requirements For Dissimilar Metal Piping Welds. DEK, with their NDE vendor LMT Inc. and Electric Power and Research Institute (EPRI), developed a site specific procedure and specimen (NEP-15.45 Ultrasonic Examination of Steam Generator Primary Side Nozzle To Safe Ends For Inservice Inspection) to examine the KPS replacement steam generator nozzle to safe end butt welds RC-W76DM, RC-W77DM, RC-W78DM and RC-W79DM to meet Appendix VIII Supplement 10 Qualification Requirements For Dissimilar Metal Piping Welds. During the KPS 2004 refueling outage, an ultrasonic examination was performed on RC-W76DM to meet the KPS fourth 10-year interval program developed to ASME Boiler and Pressure Vessel Code Section XI 1998 Edition 2000 Addenda. Examination on RC-W76DM was performed to meet Appendix VIII Supplement 10 Qualification Requirements For Dissimilar Metal Piping Welds using the qualified procedure, specimen and personnel developed through the EPRI Performance Demonstration Initiative Program by DEK and LMT Inc. The examination of RC-W76DM using the newly developed technique examined 100% of the required ASME Boiler and Pressure Vessel Code Section XI IWB-2500-8 Volume. No indications were recorded. Since the configuration of the remaining three nozzle to safe end butt welds RC-W77DM, RC-W78DM and RC-W79DM are basically the same, DEK expects that performance of ultrasonic examination of these three remaining welds during the fourth 10-year interval (June 16, 2004 - June 16, 2014) will also be 100% examined.

2.6 <u>Request for Relief RR-G-7-20 (TAC MC7916), Examination Category B-D, Item 3.90, Reactor Vessel Outlet Nozzle-to-Shell Weld RV-W7</u>

2.6.1 Describe the access requirements necessary for the remote examination tool and why coverage could not be obtained for the tangential scan.

DEK Response:

The nozzle boss or protrusion is a limiting factor for the tangential (Clockwise and Counterclockwise) exams parallel to the nozzle weld conducted from the reactor vessel ID surface.

Description of Limitations: The nozzle boss or protrusion typically obstructs some portion of the examination volume and therefore coverage of that volume by tangential scanning beams. (Reference attached Cross Section View, Figure 1, RR-G-7-20 Supplemental). The examination is conducted with the transducers scanning in a circular motion from a starting point of the greatest sized circle, decreasing in increment size by one half inch for each increment down to the stopping point, which is the smallest circle. The stopping point (Figure 1) is just above the protrusion. At this point, no further scans parallel to the examination volume are possible. Scanning on top of the protrusion is not possible because of the limited size and uneven scan surface of the protrusion. This is a limiting factor in all tangential scans of Westinghouse PWR outlet nozzles.

The examination volume effectively examined in the tangential scan was correctly reported as 43.24%. This volume included the weld nugget, where indications were recorded in reactor vessel outlet nozzle RV-W7. The indications were assessed in terms of the applicable criteria in IWB-3000 and were found to be within the allowable limits. Indication locations were consistent with prior examination data. The unexamined portion of the exam volume is nozzle barrel forging base material, which had no indications in prior examinations or in the current examination performed from the nozzle bore.

Because the tangential examination comprises one half of the required examinations of the reactor vessel nozzle to shell weld, it must be combined with the perpendicular scans (bore and vessel ID) with equal weight. Averaged appropriately as one number, coverage was 70.2% for the reactor vessel outlet nozzle to shell weld RV-W7.

Note: Reactor vessel outlet nozzle to vessel weld RV-W7 was examined in fall 2004 to meet KPS fourth 10-year interval inservice inspection program (June 16, 2004 - June 16, 2014). Examinations were performed to satisfy the requirements of ASME Boiler and Pressure Vessel Code Section XI 1998 Edition 2000 Addenda including Appendix VIII Supplement 7 Qualification Requirements For Nozzle To Vessel Weld.

2.7 <u>Request for Relief RR-G-7-20 (TAC MC7916), Examination Category B-D, Item</u>

3.90, Reactor Vessel Outlet Nozzle-to-Shell Weld RV-W10

2.7.1 Describe the access requirements necessary for the remote examination tool and why coverage could not obtained for the tangential scan.

DEK Response:

The nozzle boss or protrusion is a limiting factor for the tangential (Clockwise and Counterclockwise) exams parallel to the nozzle weld conducted from the reactor vessel ID surface.

Description of Limitations: The nozzle boss or protrusion typically obstructs some portion of the examination volume and therefore coverage of that volume by tangential scanning beams. (Reference attached Cross Section View, Figure 1, RR-G-7-20 Supplemental). The examination is conducted with the transducers scanning in a circular motion from a starting point of the greatest sized circle, decreasing in increment size by one half inch for each increment down to the stopping point which is the smallest circle. The stopping point (Figure 1) is just above the protrusion. At this point, no further scans parallel to the examination volume are possible. Scanning on top of the protrusion is not possible because of the limited size and uneven scan surface of the protrusion. This is a limiting factor in all tangential scans of Westinghouse PWR outlet nozzles.

The examination volume effectively examined in the tangential scan was correctly reported as 43.24%. This volume included the weld nugget, where indications were recorded in reactor vessel outlet nozzle RV-W10. The indications were assessed in terms of the applicable criteria in IWB-3000 and were found to be within the allowable limits. Indication locations were consistent with prior examination data. The unexamined portion of the exam volume is nozzle barrel forging base material, which had no indications in prior examinations or in the current examination performed from the nozzle bore.

Because the tangential examination comprises one half of the required examinations of the reactor vessel nozzle to shell weld, it must be combined with the perpendicular scans (bore and vessel ID) with equal weight. Averaged appropriately as one number, coverage was 70.2% for the reactor vessel outlet nozzle to shell weld RV-W10.

Note: Reactor vessel outlet nozzle to vessel weld RV-W10 was examined in fall 2004 to meet KPS fourth 10-year interval inservice inspection program (June 16, 2004 - June 16, 2014). Examinations were performed to satisfy the requirements of ASME Boiler and Pressure Vessel Code Section XI 1998 Edition 2000

Addenda including Appendix VIII Supplement 7 Qualification Requirements For Nozzle To Vessel Weld.

2.8 <u>Request for Relief RR-G-7-71 (TAC 7967), Category B-A, Item B1.30 Reactor Vessel to Flange Weld RV-W1</u>

2.8.1 Provide clarification as to why a full-vee path examination or other technology such as phased array ultrasonics, could not be performed, and whether this full-vee path examination or phased array technology would increase volumetric coverage.

<u>DEK Response:</u>

The flange to shell weld RV-W1 was examined using ASME Boiler and Pressure Vessel Code Section XI Appendix VIII approved techniques to the extent practical considering the component geometry. Access is provided from the reactor vessel ID surface for automated examinations, and from the flange seal surface for manual examinations. The combination of automated and manual examinations produced coverage of 74.4% of the examination volume. The limiting factor is the taper transition on the ID surface at the junction of the weld joint and the flange (Reference Cross-Section view, Figure 2, RR-G-7-71 Supplemental).

Other exam methodologies such as Phase Array and modified standard techniques such as using a single element full-vee path arrangement were evaluated and determined to be ineffective in providing any meaningful additional coverage. The problem is that the taper transition is curved with a 3-inch radius. This is a somewhat unique design for Westinghouse PWR vessel flange. The curved taper transition does not allow for proper contact of the transducers in either perpendicular or parallel scans. For parallel scans, scanning about the taper would result in a severe miss-orientation of the examination beam with respect to the volume (Figure 2). This is the biggest contributor to the coverage limitation. In perpendicular scans, a phase array probe would be limited and even shallow angle approaches would not reach the volumes of forging base metal surrounding the curved taper transition. Full-vee path techniques are not qualified to ASME Section XI Appendix VIII by any PWR vendor and may only increase coverage 3%-5%. The efficacy of this particular approach has not been proven by qualification, and it is the preference of DEK to apply only qualified automated exam techniques for the remote mechanized scans from the ID surface.

2.9 ASME Code Case N-460 (N-460) was approved for use under Regulatory Guide 1.147, Revision 14. This Code Case provides an alternative requirement for

Class 1 and 2 welds when the examination volume or area cannot be examined due to interferences. Approximately eight of your relief requests meet the criteria of N-460. Please explain why relief is still necessary for these eight components.

DEK Response:

DEK agrees that Code Case N-460 was approved for use under Regulatory Guide 1.147 Revsion14 and thus wishes to withdraw the following relief requests:

Relief Request RR-G-7-5: DEK agrees that Examination Category C-A, Item C1.20, Head Circumferential Welds APD-1A-W1 and APD-1B-W4 on charging pump pulsation dampeners APD-1A and APD-1B were examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-5 will be withdrawn.

Relief Request RR-G-7-6: DEK agrees that Examination Category B-B, Item B2.11, Pressurizer Head Circumferential Welds P-W3 and P-W5 were examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-6 will be withdrawn.

Relief Request RR-G-7-7: DEK agrees that Examination Category B-B, Item B2.11, Pressurizer Head Longitudinal Weld P-W2 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-7 will be withdrawn.

Relief Request RR-G-7-11: DEK agrees that Examination Category C-A, Item C1.20, Regenerative Heat Exchanger Head Circumferential Weld ARG-W9 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-11 will be withdrawn.

Relief Request RR-G-7-17: DEK agrees that Examination Category C-A, Item C1.10, Steam Generator 1A Shell Circumferential Weld SG-W10 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-17 will be withdrawn.

Relief Request RR-G-7-18: DEK agrees that Examination Category C-A, Item C1.20, Steam Generator 1B Head Circumferential Weld SG-W9 was examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-18 will be withdrawn.

Relief Request RR-G-7-19: DEK agrees that Examination Category C-A, Item C1.30, Steam Generator 1A and Steam Generator 1B Tubesheet To Shell Circumferential Welds SG-W25 and SG-W31 were examined greater than 90%. With DEK implementing ASME Code Case N-460, Relief Request RR-G-7-19 will be withdrawn.

Additionally DEK would also like to withdraw the following relief requests per ASME Code Case N-460 approved for use under Regulatory Guide 1.147 Revsion14:

Relief Request RR-G-7-29: 29-inch ID Reactor Coolant Pipe Circumferential Weld RC-W5 due to examination volume of 95.4% being achieved.

Relief Request RR-G-7-38: 6-inch Safety Injection Circumferential Weld SI-W14 due to examination volume of 93.5% being achieved.

Relief Request RR-G-7-46: 3-inch Pressurizer Spray Circumferential Weld PS-W4 due to examination volume of 91.2% being achieved.

Relief Request RR-G-7-58: 30-inch Main Steam Circumferential Weld MS-W50 due to examination volume of 92.0% being achieved.

Relief Request RR-G-7-62: 16-inch Feedwater Circumferential Weld FW-W24 due to examination volume of 92.0% being achieved.

Relief Request RR-G-7-63: 16-inch Feedwater Circumferential Weld FW-W52 due to examination volume of 92.0% being achieved.

Relief Request RR-G-7-72: Reactor Vessel Safety Injection Nozzle to Vessel Weld RV-W11 due to the reevaluation of the appropriate coverage estimate of 90.4%.

Relief Request RR-G-7-73: Reactor Vessel Lower Head Circumferential Weld RV-W4 due to the reevaluation of the appropriate coverage estimate of 90.82%. Additionally DEK would also like to withdraw the following relief request per WCAP-15666.

Relief Request RR-G-7-24: Reactor Coolant Pump 1A Flywheel. NDE Procedure NEP-15.12 Ultrasonic Examination of Reactor Coolant Pump Flywheels For Inservice Inspection requires that the manual ultrasonic bore hole probe be inserted into each gage hole plug (either 4 or 6 for a typical flywheel), calibrate using flywheel bolt holes and then extending the metal path distance to a maximum of 35 inches. The gage hole plugs are located at a distance of 14.5 inches from the center of the flywheel on a circumference of 29 inches. The manual ultrasonic probe examination is performed by placing the probe into each of the 4 or 6 gage holes and rotating for 360°, thus scanning the flywheel metal for a distance of 49.5 inches from the center of the keyway bore. This is determined by scanning from the gage hole towards the keyway bore for the inward distance of 14.5 inches and towards the outer circumference for a distance of 35 inches. KPS flywheel upper plate circumference is 75.0 inches and the flywheel lower plate circumference is 65.0 inches. Since the volume from the inner bore of the flywheel to the circle of one half the outer radius has been scanned, DEK will withdraw Relief Request RR-G-7-24.

2.10 *In your relief requests where partial examination coverage was obtained, no mention was made on whether there were any recordable or reportable indications. If there were indications noted, please annotate the relief requests as such and indicate if there was any change in the indications since the previous inservice examination.*

DEK Response:

A search of Relief Requests RR-G-7-1 through RR-G-7-73 examination data that contained Recordable Indications revealed that these indications were evaluated based on ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards. Due to improved examination techniques since the previous examinations and implementation of Appendix VIII criteria, comparison of the indications from previously recorded data is difficult to substantiate. However, no discernible changes were recorded:

Relief Request RR-G-7-3: Safety Injection Pump APSI-1A Integrally Welded Attachment APSI-1A-S2, Surface examination curvilinear indication was recorded that was acceptable per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards.

Relief Request RR-G-7-4: Original Reactor Vessel Closure Head To Flange Weld RV-W12 Surface Magnetic Particle Linear Indication was recorded that was acceptable per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards.

The original reactor vessel closure head was replaced during the fall 2004 refueling outage with the replacement reactor vessel closure manufactured as a one-piece forging eliminating the reactor vessel closure head flange weld.

Relief Request RR-G-7-14: Seal Water Injection Filter AFSI-1A Shell Circumferential Weld AFSI-W1: Manual ultrasonic laminar indications were recorded that were consistent with data recorded in 1982, 1986, 1988,1992 and 1998. Indications recorded in 2001 were acceptable per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards.

Relief Request RR-G-7-15: Seal Water Injection Filter AFSI-1A Head Circumferential Weld AFSI-W2: Manual ultrasonic laminar indications were recorded that were consistent with data recorded in 1982, 1986, 1988 and 1992. Indications recorded in 1998 were acceptable per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards.

Relief Request RR-G-7-16: Steam Generator 1A Shell Circumferential Weld SG-W2. Manual ultrasonic planar indications were recorded that were consistent with data recorded in 1991. Indications recorded in 2001 exceeded ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards but were accepted by fracture analysis as referenced in Westinghouse WCAP 11476 Rev. 3, "Handbook on Flaw Evaluation, Kewaunee Unit No.1 Steam Generators Upper Shell To Cone Weld," and DEK and LMT Inc. evaluations. Additionally, during steam generator replacement in 2001 the ID surface of SG-W2 was examined by the magnetic particle method to insure indications were not service induced ID connected. Magnetic particle examinations revealed no indications of 100% of the ASME Boiler and Pressure Vessel Code Section XI IWC-2500-1 required volume.

Relief Request RR-G-7-17: Steam Generator 1B Shell Circumferential Weld SG-W10. Manual ultrasonic planar indications were recorded that were consistent with data recorded in 1996. Indications recorded in 2001 exceeded ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards but were accepted by fracture analysis as referenced in Westinghouse WCAP 11476 Rev. 3, "Handbook on Flaw Evaluation, Kewaunee Unit No.1 Steam Generators Upper Shell To Cone Weld," and DEK and LMT Inc. evaluations. Additionally, during steam generator replacement in 2001 the ID surface of SG-W10 was examined by the magnetic particle method to insure indications were not service induced ID connected. Magnetic particle examinations revealed no indications of 100% of the ASME Boiler and Pressure Vessel Code Section XI IWC-2500-1 required volume.

Relief Request RR-G-7-18: Steam Generator 1B Circumferential Head Weld SG-W9. A manual ultrasonic spot indication was recorded that was acceptable per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards.

Relief Request RR-G-7-19: Steam Generator 1B Tubesheet To Shell Circumferential Weld SG-W31. A manual ultrasonic planar indication was recorded that was acceptable per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards.

Relief Request RR-G-7-20: Reactor Vessel Outlet Nozzle To Vessel Weld RV-W7. Remote automated ultrasonic planar indications recorded were acceptable per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards.

Relief Request RR-G-7-21: Reactor Vessel Outlet Nozzle To Vessel Weld RV-W10. Remote automated ultrasonic planar indications recorded were acceptable per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards.

Relief request RR-G-7-25: 16-inch Feedwater Integrally Welded Attachment FDW-H170. Visual examination recorded three loose nuts on the anchor studs. The three loose nuts were corrected, reexamined and accepted.

Relief Request RR-G-7-72: Reactor Vessel Safety Injection Nozzle To Vessel Weld RV-W11. Remote automated ultrasonic planar indications recorded were acceptable per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Acceptance Standards.

2.11 The majority of your relief requests indicate that it is impractical to redesign the component in order to obtain the necessary examination coverage. This rationale meets the staff's criteria as an acceptable basis for impracticality under 10 CFR 50.55a(g)(5)(iii). However, relief requests RR-G-7-25, -30, -44, and -57 explain that the burden caused by compliance is due to removal of the interference(s). These examples are not supported by an explanation of the burden in man-rem, hours, expense or schedule. Please provide an explanation of the burden for these items. If these items require redesign of the component to obtain coverage, please state as such.

DEK Response:

In responding to this question, DEK estimated the number of hours required to remove the associated interference. These estimations were necessary because of the inability to accurately scope the interference removal task. To accurately scope the interference removal task a walkdown of the site is needed, which, due to weld inaccessibility during power operation, could not be performed. DEK believes the hour estimations under-estimate the actual hours required to complete the removal of the interference.

Relief Request RR-G-7-25: 16-inch Feedwater Integrally Welded Attachment FDW-H170

Burden:

Radiation Levels: Less than 1 millirem (mr) per hour

Maintenance: Removal of Encapsulation and Reinstallation: 120 man-hours for cutting welded stitches, cutting encapsulation sleeve, removal of encapsulation sleeve, reinstallation, rewelding encapsulation sleeve and rewelding stitches. Man-hours are a minimum based on review of available photographs and as built drawings. Actual man-hours would be dependent on the work site evaluation of the encapsulation sleeve and rigging requirements.

Engineering Time: Man-hours: None

Note: Loop B 16-inch Feedwater integrally welded attachment FDW-H170 was scheduled for examination per ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Table IWC-2500-1 Examination Category C-C Item No. C3.20, which required 100% of Piping Supports ¾ inch or greater be surface examined over the third 10-year interval. Similar non-encapsulated Loop A 16-inch Feedwater integrally welded attachment FDW-H169 was examined during the third 10-year interval by surface method and no indications and no limitations were recorded. During the KPS fourth 10-year interval prepared to ASME Boiler and Pressure Vessel Code Section XI 1998 Edition 2000 Addenda, Table IWC-2500-1, Examination Category C-C, Item No. C3.20, only 10% of the Class 2 Piping welded integrally attachments are required to be examined. Since Loop B 16-inch feedwater integrally welded attachment FDW-H170 is restricted by encapsulation, no examinations were scheduled on this item for the fourth 10-year interval.

For the KPS fourth 10-year interval (June 16, 2004 - June 16, 2014), KPS implemented a risk informed program following the guidelines of ASME Boiler and Pressure Vessel Code Section XI Code Case N-578 "Risk Informed Requirements For Class 1,2 and 3 Piping Method B Section XI Division 1" and Electric Power Research Institute Risk Informed Program Topical Report TR-112657 Rev. B-A "Revised Risk-Informed Inservice Inspection Evaluation Procedure". The Nuclear Regulatory Commission approved the KPS risk informed program on September 23, 2005 under TAC Nos. MC2502, MC2508 and MC2537.

The two 16-inch feedwater circumferential welds FW-W41 and FW-W42 located on either side of the elbow where FDW-H170 is located are currently classified as: FW-W41 - Risk Characterization Category – 6a, Risk Characterization Rank - Low, Consequence Rank – Medium, Failure Potential Degradation Mechanisms – None, and Failure Potential Rank – Low; and FW-W42 - Risk Characterization Category – 6a, Risk Characterization Rank - Low, Consequence Rank – Medium, Failure Potential Degradation Mechanisms – None, and Failure Potential Rank – Low.

<u>Relief Request RR-G-7-30</u>: 3-inch Pressurizer Relief Circumferential Weld PR-W12.

Burden:

Radiation Levels: 20 mr – 30 mr per hour

Removal of Rigid Restraint RRRC-39 and Reinstallation: Maintenance man-hours: 8 hours. Estimate based on review of photograph and as built drawings determined that the removal of RRRC-39 will be dependent on work site evaluation to access the ability to slide RRRC-39 away from 3-inch pressurizer relief circumferential weld PR-W12. Evaluation will be performed under the Engineering Time man-hours. A minimum estimate would be at least 8 man-hours, which is standard for a typical SUPPORT removal and replacement.

Engineering Time: Man-hours: 40 Hours Rigging Analysis
The removal of RRRC-39 would provide access to the remaining inaccessible area of 3-inch pressurizer relief circumferential weld PR-W12. An evaluation of the limitation to examination record and the attached photograph taken later appears to show that the area not examined by liquid penetrant is 26%. The area examined by liquid penetrant examination recorded no indications. Exposure (240 mr) and man-hours dedicated to access the remaining 26% for liquid penetrant examination would be a burden to DEK with limited benefit.

Note: Class 1 3-inch pressurizer relief circumferential weld PR-W12 was selected for examination per the requirements of ASME Boiler and Pressure Vessel Code Section XI 1974 Edition Summer 1975 Addenda in effect for the KPS third 10-year interval as permitted by 10CFR 50.55a(b)(2)(ii). Examination requirements were to examine 25% of all Class 1 welds each 10-year interval and therefore 100% over the life of the plant i.e. 40 years. For the KPS fourth 10-year interval (June 16, 2004 - June 16, 2014) KPS implemented a risk informed program following the guidelines of ASME Boiler and Pressure Vessel Code Section XI Code Case N-578 "Risk Informed Requirements For Class 1,2 and 3 Piping Method B Section XI Division 1" and Electric Power Research Institute Risk Informed Program Topical Report TR-112657 Rev. B-A "Revised Risk-Informed Inservice Inspection Evaluation Procedure". The Nuclear Regulatory Commission approved the KPS risk informed program on September 23, 2005 under TAC Nos. MC2502, MC2508 and MC2537.

The 3-inch pressurizer relief circumferential weld PR-W12 is currently classified as a Risk Characterization Category - 4, Risk Characterization Rank - Medium, Consequence Rank – High, Failure Potential Degradation Mechanisms – None, and Failure Potential Rank - Low and was not selected for examination during the fourth 10-year interval due to known limitations. Other welds on this piping section with identical Risk Characterization Category, Risk Characterization Rank, Consequence Rank, Failure Potential Degradation Mechanisms and Failure Potential Rank were selected for examination due to accessibility.

<u>Relief Request RR-G-7-44</u>: 10-inch Pressurizer Surge Circumferential Weld RC-W64

Burden:

Radiation Levels: 40 mr – 95 mr per hour

Removal of Rigid Restraint RR134-6 and Reinstallation: Maintenance man-hours; 8 hours for removal of RR134-6. Man-hours are a minimum based on review of available photographs and as built drawings. Actual man-hours would be dependent on work site evaluation of the rigid restraint for rigging requirements, personnel accessibility, removal and reinstallation.

Engineering Time: Man-hours; RR134-6 was removed from service under previous design change request for the pressurizer surge line, which removed the shims to RR134-6. 40 man-hours would be required to prepare work order, revise drawings and develop a removal plan of RR-134-6. The removal of RR134-6 would provide access to the remaining inaccessible area. An evaluation of the limitation to examination record shows that the area for 10-inch pressurizer surge circumferential weld RC-W64 not examined by ultrasonic examination is 21.6% and liquid penetrant is 12.7%. Ultrasonic examination included 0°, 45° shear (axial scan and circumferential scan) and 60° shear (axial scan). There were no indications noted by ultrasonic examination and liquid penetrant examination of 10-inch pressurizer surge circumferential weld RC-W64. Examination Exposure (480 mr estimate) and man-hours dedicated to access the remaining 21.6% for ultrasonic examination and 12.7% for liquid penetrant examination would be a burden to DEK with limited benefit.

Note: Class 1 10-inch pressurizer surge circumferential weld RC-W64 was selected for examination per the requirements of ASME Boiler and Pressure Vessel Code Section XI 1974 Edition Summer 1975 Addenda in effect for the KPS third 10-year interval as permitted by 10CFR 50.55a(b)(2)(ii). Examination requirements were to examine 25% of all Class 1 welds each 10-year interval and therefore 100% over the life of the plant i.e. 40 years. For the KPS fourth 10-year interval (June 16, 2004 - June 16, 2014), KPS implemented a risk informed program following the guidelines of ASME Boiler and Pressure Vessel Code Section XI Code Case N-578 “Risk Informed Requirements For Class 1,2 and 3 Piping Method B Section XI Division 1” and Electric Power Research Institute Risk Informed Program Topical Report TR-112657 Rev. B-A “Revised Risk-Informed Inservice Inspection Evaluation Procedure”. The Nuclear Regulatory Commission approved the KPS risk informed program on September 23, 2005 under TAC Nos. MC2502, MC2508 and MC2537.

The 10-inch pressurizer surge circumferential weld RC-W64 is currently classified as a Risk Characterization Category - 2, Risk Characterization Rank – High, Consequence Rank – High, Failure Potential Degradation Mechanisms – TASCS (Thermal Stratification Cycling and Striping) and TT (Thermal Transients), and Failure Potential Rank - Medium but was not selected for examination during the fourth 10-year interval due to known limitations. Other welds on this piping section with identical Risk Characterization Category, Risk Characterization Rank, Consequence Rank, Failure Potential Degradation Mechanisms and Failure Potential Rank were selected for examination due to accessibility.

Relief Request RR-G-7-57: 3-inch Pressurizer Spray Circumferential Weld PS-W10.

Burden:

Radiation Levels: 100 mr per hour post shielding

Removal of Rigid Restraint RRRC-19 and Reinstallation: Maintenance man-hours: 8 hours. Man-hours are a minimum based on review of available photographs and as built drawings. Actual man-hours would be dependent on the work site evaluation of the rigid restraint for rigging requirements, personnel accessibility, removal and reinstallation.

Engineering Time: Man-hours; 8 hours Rigging Analysis

The removal of RRRC-19 would provide access to the remaining inaccessible area of the 3-inch pressurizer spray circumferential weld PS-W10. An evaluation of the limitation to examination record shows that the area not examined by liquid penetrant is 36%. The area examined by liquid penetrant examination recorded no indications. Exposure (800 mr) and man-hours dedicated to access the remaining 36% (Bottom 4") for liquid penetrant examination would be a burden to DEK with limited benefit.

Note: Class 1 3-inch pressurizer spray circumferential weld PS-W10 was selected for examination per the requirements of ASME Boiler and Pressure Vessel Code Section XI 1974 Edition Summer 1975 Addenda in effect for the KPS third 10-year interval as permitted by 10CFR 50.55a(b)(2)(ii). Examination requirements were to examine 25% of all Class 1 welds each 10-year interval, and therefore 100% over the life of the plant, i.e. 40 years. For the KPS fourth 10-year interval (June 16, 2004 - June 16, 2014), KPS implemented a risk informed program following the guidelines of ASME Boiler and Pressure Vessel Code Section XI Code Case N-578 "Risk Informed Requirements For Class 1,2 and 3 Piping Method B Section XI Division 1" and Electric Power Research Institute Risk Informed Program Topical Report TR-112657 Rev. B-A "Revised Risk-Informed Inservice Inspection Evaluation Procedure". The Nuclear Regulatory Commission approved the KPS risk informed program on September 23, 2005 under TAC MC2502, MC2508 and MC2537.

The 3-inch pressurizer spray circumferential weld PS-W10 is currently classified as a Risk Characterization Category - 4, Risk Characterization Rank – Medium, Consequence Rank – High, Failure Potential Degradation Mechanism – None, and Failure Potential Rank - Low but was not selected for examination during the fourth 10-year interval due to known limitations. Other welds on this piping section with identical Risk Characterization Category, Risk Characterization Rank, Consequence Rank, Failure Potential Degradation Mechanisms and Failure Potential Rank were selected for examination due to accessibility.

2.12. In reference to Relief Request No. RR-G-7-24, the required volume for ultrasonic examination of an in-place flywheel stated in the staff evaluation report to WCAP-15666 is the volume from the inner bore of the flywheel to the circle of one half the outer radius. Please provide information as to whether the required volume was examined in spite of the existing interference from anti-rotation paws. If so, a relief may not be necessary.

<u>DEK Response:</u>

Relief Request RR-G-7-24: Reactor Coolant Pump 1A Flywheel RCP-1A-FLY. NDE Procedure NEP-15.12 Ultrasonic Examination of Reactor Coolant Pump Flywheels For Inservice Inspection requires that the manual ultrasonic bore hole probe be inserted into each gage hole plug (either 4 or 6 for a typical flywheel), calibrate using flywheel bolt holes and then extending the metal path distance to a maximum of 35 inches. The gage hole plugs are located at a distance of 14.5 inches from the center of the flywheel on a circumference of 29 inches. The manual ultrasonic probe examination is performed by placing the probe into each of the 4 or 6 gage holes and rotating for 360°, thus scanning the flywheel metal for a distance of 49.5 inches from the center of the keyway bore. This is determined by scanning from the gage hole towards the Keyway bore for the inward distance of 14.5 inches and towards the outer circumference for a distance of 35 inches. KPS flywheel upper plate circumference is 75.0 inches and the flywheel lower plate circumference is 65.0 inches. Since the volume from the inner bore of the flywheel to the circle of one half the outer radius has been scanned, DEK will withdraw Relief Request RR-G-7-24.

2.13 On several relief requests for piping examinations, you have listed percentages of volumetric coverage in accordance with the ASME Code and your ultrasonic examination procedures, which are not the same. Please explain the difference in examination coverage in regard to the Code percentage and the procedure percentage.

DEK Response:

The site specific procedures used by DEK at the time of examinations for the third 10-year interval (June 16, 1994 - June 16, 2004) required that examinations be performed using the following ultrasonic examination techniques which exceeded ASME Boiler and Pressure Vessel Code Section XI 1989 Edition:

1. A 0°, 45° and 60° on austenitic and ferritic steel piping welds and austenitic vessel welds.

2. 0°, 45° and 60° examinations to be performed from BOTH sides of the weld to cover the required volume from ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Figures IWB-2500-8, IWC-2500-1, IWC-2500-2, and IWC-2500-7, as applicable.

3. If examinations could not be performed to satisfy items 1 and 2, the percentage of coverage not obtained was to be based on those criteria and listed as Procedure Percentage.

The requirements for compliance to ASME Boiler and Pressure Vessel Code Section XI 1989 Edition for the KPS third 10-year interval (June 16, 1994 – June 16, 2004) were ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Article III-4000 and Appendix III Supplement 4, which requires:

Article III-4000

III-4410 Beam Angle: The search unit and beam angle selected shall be capable of detecting the calibration reflectors, as described in III-3230(c), over the required angle beam path. A beam angle of 45 degrees in the material shall generally be used. Other angles may be used for evaluation of an indication, or where wall thickness or geometric configuration impedes effective use of the 45 degrees angle beam for examination.

III-4420 Reflectors Parallel To The Weld Seam: The examination shall be performed using a sufficiently long examination beam path to provide coverage of the required examination volume in two-beam path directions. The examination shall be performed from two sides of the weld, where practicable, or from one side of the weld, as a minimum.

III-4430 Reflectors Transverse To The Weld Seam: The angle beam examination for reflectors transverse to the weld shall be performed on the weld crown on a single scan path to examine the weld root by one half V path in two directions along the weld.

Supplement 4 Austenitic and Dissimilar Metal Welds:

(1) III-4410 Beam Angle: The actual beam angle in the examination part shall be 40 degrees or greater for shear wave at the ID surface and 35 degrees or greater for refracted, longitudinal wave at the ID surface. The beam angle in the examination part shall be determined for each pipe size, schedule, and material to be examined for each plant. The beam angle measurements shall be used to assure coverage of the required examination volume by extending the calibration and examination distance, as required.

(2) III-4430 Reflectors Transverse To The Weld Seam: Substitute: The angle beam examination for reflectors transverse to the weld shall be performed in two directions covering the minimum area from ½ inch from one side of the weld crown to ½ inch from the other side of the weld crown including the crown.

(3) Table III-3430-1 Calibration Notches – Substitute: depth 10% of *t*.

(4) Scanning from both sides of the weld is required where practical. Single side access limitations shall be noted in the examination data record. Cast items such as fittings, valve bodies, and pump casings may preclude meaningful examinations because of geometry and attenuation variables.

On data forms at KPS if examinations could not be performed to satisfy ASME Boiler and Pressure Vessel Code Section XI 1989 Edition Article III and Appendix III Supplement 4 of the required volume from Figures IWB-2500-8, IWC-2500-1, IWC-2500-2, or IWC-2500-7, the percentage of coverage not obtained was to be based on those criteria and listed as Code Percentage.

References

1. Letter from Michael Gaffney (NMC) to Document Control Desk (NRC), "Inservice Inspection Third 10-Year Interval June 16, 1994 - June 16, 2004 Limitation To Examination Relief Requests of the Kewaunee Nuclear Power Plant," dated June 23, 2005. (ADAMS Accession No. ML052440263)

Table 1

Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval

Component	Relief Request	Material	Potential Degradation Mechanism (1)	Most Likely Degradation Mechanisms (DM) For The Location	Basis For Whether The Achieved Examination Coverage Could Reasonability Be Expected to Detect This Degradation
Residual Heat Exchanger	RR-G-7-1 RR-G-7-8 RR-G-7-9	Shell-SA20 Type 304 Stainless Steel (SS) Attachment A285 Grade C Carbon Steel (CS)	Reference-PWR Piping Wrought SS	None (2) (3)	Basis for Code Case N-706 is no degradation has been observed in industry. Review of Risk-Based ISI Records Indicates no DM for Related Piping/Welds. Combination of the UT and PT examinations could reasonably be expected to detect a DM as the entire accessible examination region was examined and various examinations have been periodically performed over the lifetime of the plant. The examinations were performed using qualified procedures, equipment, and personnel. Should cracking occur due to fatigue, site engineers indicate that the expected crack morphology is on the order of a 6 to 1 aspect ratio, which corresponds to a long crack that could reasonably be expected to be detected with UT.
SI Pump	RR-G-7-2 RR-G-7-3	Shell-A266 Class 1 CS Support-A216 WCA CS Cladding 18-8	Reference-PWR Piping SS Weld & Clad and Carbon and Low Alloy Steel (C&LAS)	Mechanical fatigue (2) (3)	VT-2, VT-3, & MT confirm no evidence of degradation on exterior surface. Various examinations have been periodically performed over the lifetime of the plant. Review of Risk-Based ISI Records Indicates no DM for Related Piping/Welds. The examinations were performed using qualified procedures, equipment, and personnel.

Table 1

Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval

Component	Relief Request	Material	Potential Degradation Mechanism (1)	Most Likely Degradation Mechanisms (DM) For The Location	Basis For Whether The Achieved Examination Coverage Could Reasonability Be Expected to Detect This Degradation
Seal Water Injection Filter	RR-G-7-14 RR-G-7-15	SA-240 Type 304 SS	Reference-PWR Piping Wrought SS	None (2) (3)	Review of Risk-Based ISI Records Indicates no DM for Related Piping/Welds. The UT examination could reasonably be expected to detect a DM as the entire accessible examination region was examined and various examinations have been periodically performed over the lifetime of the plant. During the Second ISI Interval, KPS performed PT on the ID surface and RT. The examinations were performed using qualified procedures, equipment, and personnel. Should cracking occur due to fatigue, site engineers indicate that the expected crack morphology is on the order of a 6 to 1 aspect ratio, which corresponds to a long crack that could reasonably be expected to be detected with UT.
Letdown Heat Exchanger	RR-G-7-13	SA-240 Type 304 SS	Reference-PWR Piping Wrought SS	None (2) (3)	Review of Risk-Based ISI Records Indicates no DM for Related Piping/Welds. The UT examination could reasonably be expected to detect a DM as the entire accessible examination region was examined and various examinations have been periodically performed over the lifetime of the plant. The examinations were performed using qualified procedures, equipment, and personnel. Should cracking occur due to fatigue site engineers indicate that the expected crack morphology is on the order of a 6 to 1 aspect ratio, which corresponds to a long crack that could reasonably be expected to be detected with UT.

Table 1

Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval

Component	Relief Request	Material	Potential Degradation Mechanism (1)	Most Likely Degradation Mechanisms (DM) For The Location	Basis For Whether The Achieved Examination Coverage Could Reasonability Be Expected to Detect This Degradation
Reactor Coolant Pump Main Flange Bolting	RR-G-7-10	SA540 Grade B24 Class 4 Carbon Steel	Reference-PWR Piping C&LAS	Wastage & Pitting	The likely DM is related to RCS leakage at the flange joint. The UT examination could reasonably be expected to detect a DM. VT-2, VT-3 and UT examinations are adequate for identification of the likely DM. Various examinations have been periodically performed over the lifetime of the plant. The examinations were performed using qualified procedures, equipment, and personnel.
Reactor Vessel	RR-G-7-20 RR-G-7-21 RR-G-7-71 RR-G-7-72 RR-G-7-73	A508-64 Class 2 Carbon Steel with Stainless Steel Cladding	Reference-PWR Reactor Pressure Vessel C&LAS	None (3)	The UT examination could reasonably be expected to detect a DM. Examinations were performed using multiple probes with different frequencies and angles. The entire accessible examination region was examined, and various examinations have been periodically performed over the lifetime of the plant. The examinations were performed using qualified procedures, equipment, and personnel.

Table 1

Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval

Component	Relief Request	Material	Potential Degradation Mechanism (1)	Most Likely Degradation Mechanisms (DM) For The Location	Basis For Whether The Achieved Examination Coverage Could Reasonability Be Expected to Detect This Degradation
Regenerative Heat Exchanger	RR-G-7-64	Shell-A351 CF8 Tubesheet A182 F 304	Reference-PWR Piping Wrought SS and CASS	None (3)	Basis for Code Case N-706 is no degradation has been observed in industry. Review of Risk-Based ISI Records Indicates no DM for Related Piping/Welds. The UT examination could reasonably be expected to detect a DM as the entire accessible examination region was examined, and various examinations have been periodically performed over the lifetime of the plant. The examinations were performed using qualified procedures, equipment, and personnel. Should cracking occur due to fatigue, site engineers indicate that the expected crack morphology is on the order of a 6 to 1 aspect ratio, which corresponds to a long crack that that could reasonably be expected to be detected with UT.
Steam Generator Primary Side	RR-G-7-12	SA508 Class 3A Carbon Steel with Stainless Steel Cladding	Reference-PWR SG Shell C&LAS	None (3)	The UT examination could reasonably be expected to detect a DM. Examinations were performed using multiple probes with different frequencies and angles. The entire accessible examination region was examined and various examinations have been periodically performed over the lifetime of the plant. The examinations were performed using qualified procedures, equipment, and personnel.

Table 1

Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval

Component	Relief Request	Material	Potential Degradation Mechanism (1)	Most Likely Degradation Mechanisms (DM) For The Location	Basis For Whether The Achieved Examination Coverage Could Reasonability Be Expected to Detect This Degradation
Steam Generator Primary Nozzle to Safe-End Butt Welds	RR-G-7-22 RR-G-7-23	Alloy 600 Weld with Alloy 690 ID Cladding to Safe End of SA 336 Class F 316LN Stainless Steel and Alloy 690 Weld to SA508 Class 3A Carbon Steel Steam Generator Nozzle	Reference - PWR SG Shell Ni-base Welds & Clad	None (3)	Review of Risk-Based ISI Records Indicates no DM for the Safe-Ends. The ID surface of the Safe-Ends is clad with Alloy 52, which is resistant to PWSCC. Combination of the UT and PT examinations could reasonably be expected to detect a DM as the entire accessible examination region was examined. The Safe-End Welds were fully examined by RT during steam generator replacement (SGR). The examinations were performed using qualified procedures, equipment, and personnel.
Steam Generator Secondary Side	RR-G-7-16	SA-533 Grade A Class 1 Carbon Steel	Reference-PWR SG Shell C&LAS	Pitting & Fatigue	The UT examination in combination with a surface examination (MT) could reasonably be expected to detect the likely DM. The entire accessible examination region was examined by UT. A surface examination (MT) was completed on the weld ID, during SGR. Various examinations have been periodically performed over the lifetime of the plant. The examinations were performed using qualified procedures, equipment, and personnel. Should cracking occur due to fatigue, site engineers indicate that the morphology is on the order of a 6 to 1 aspect ratio, which corresponds to a long crack that could reasonably be expected to be detected with UT.

Table 1

Information Regarding Potential and Likely Degradation Mechanisms Regarding KPS Relief Requests for Third ISI Interval

Component	Relief Request	Material	Potential Degradation Mechanism (1)	Most Likely Degradation Mechanisms (DM) For The Location	Basis For Whether The Achieved Examination Coverage Could Reasonability Be Expected to Detect This Degradation

(1) Degradation Matrix transmitted under NEI letter dated November 2, 2004 from Marvin S. Fertel to Chief Nuclear Officers.

(2) The stainless steel items with limited examination coverage are fabricated from 18 Cr – 8 Ni stainless steels and: a) have a high corrosion resistance with low contribution of corrosion products to the coolant, b) have good mechanical properties, and c) are highly weldable. Chemistry controls on chlorides, fluorides, sulfides, and dissolved oxygen are mandated by plant Technical Specifications and other administrative procedures at KPS to ensure that favorable conditions for stress corrosion cracking are precluded. Consideration of the potential degradation mechanisms has been eliminated, unless otherwise indicated, based upon system application, particular weld or component location, water chemistry, or operating temperature and flow rate.

(3) No likely degradation postulated for these components/items.

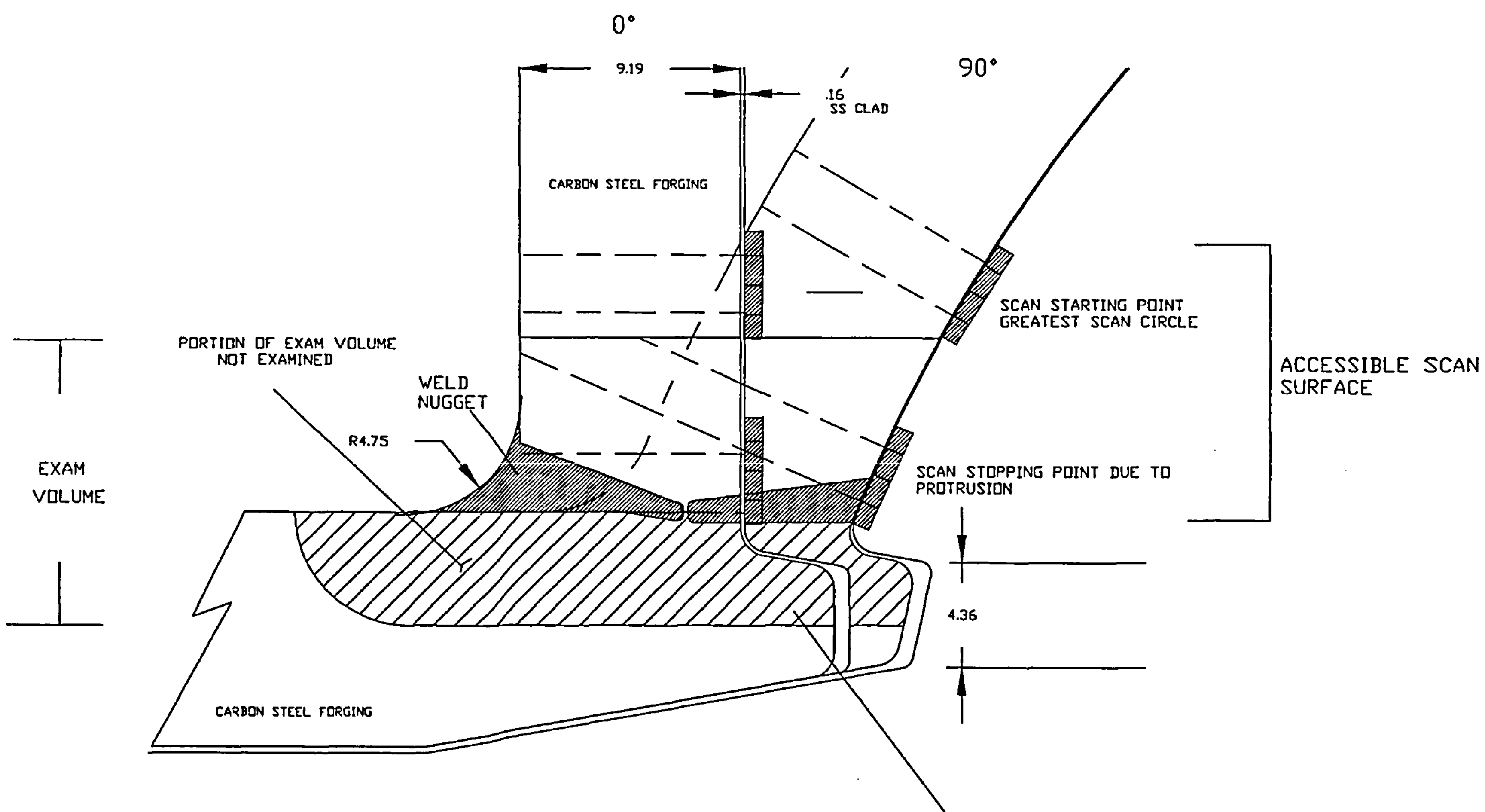

FIGURE 1
RR-G-7-20 SUPPLEMENTAL

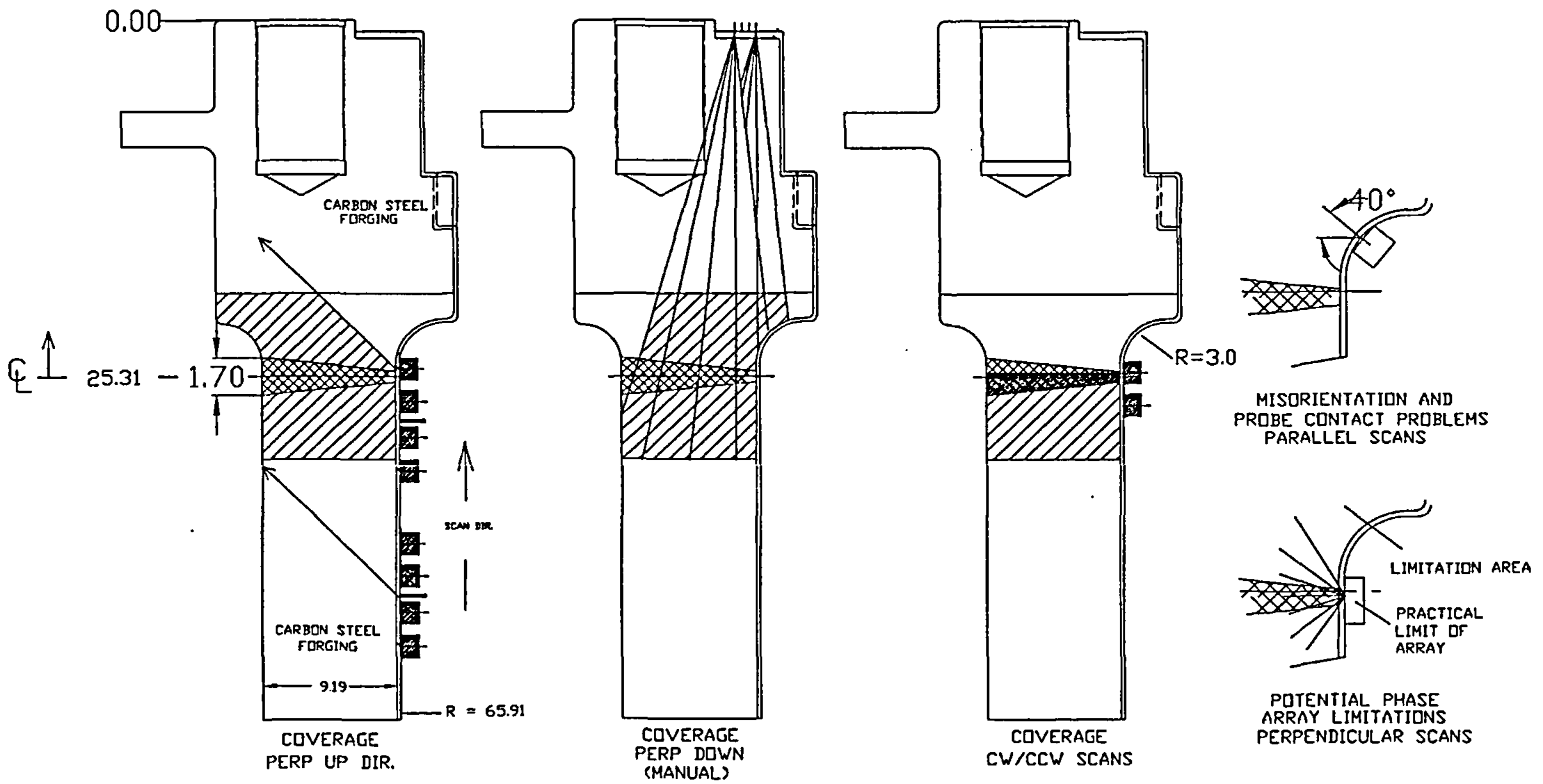

FIGURE 2
RR-G-7-71 SUPPLEMENTAL