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SOUTHERN COMPANY
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November 21, 2002

Docket Nos. 50-321
50-366

HL-6329

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

Edwin I. Hatch Nuclear Plant
Third 10-Year Interval Inservice Inspection (ISI) Program,
Response to Request for Additional Information for
Requests for Relief RR-34 and RR-37

Ladies and Gentlemen:

On September 28, 2001, Southern Nuclear Operating Company (SNC) submitted Inservice Inspection (ISI) Program Request for Relief RR-34 (ref. HL-6129). Subsequently, industry and regulatory communications resulted in SNC revising the request for relief (Rev. 1) and the development of another request for relief (RR-37) which were both submitted on May 3, 2002 (HL-6221). In late September, 2002, SNC was verbally requested to review a similar request for relief from the Tennessee Valley Authority (TVA) for Brown's Ferry Nuclear Plant and provide responses for six questions posed to TVA. On November 7, 2002, a conference call was conducted between SNC and NRC personnel to discuss various NRC questions related to the two relief requests and make sure that SNC understood the questions requiring response. Subsequently, the staff provided a Request for Additional Information (RAI) via e-mail dated November 8, 2002, which included ten (10) NRC questions.

SNC reviewed the RAI and has prepared a response for each question. SNC also made some minor changes to each relief request resultant to the RAI. Therefore, the below listed enclosures are included for your review.

Request for Relief RR-34, Revision 2
Request for Relief RR-37, Revision 1
SNC Response to NRC RAI

Since the original submittal was dated more than a year ago, was revised and resubmitted six-months ago based on the information applicable at that time, and SNC is now responding to additional NRC requests based on even later information, SNC would appreciate a timely review of this submittal. Examination of at least three nozzle inner radius regions is scheduled during the Hatch Unit 2 Outage in March, 2003. Therefore, SNC is requesting NRC review by December 31, 2002.

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Should you have any questions in this regard, please contact this office.

Respectfully submitted,

H. L. Sumner, Jr.

IFL/eb

Enclosures:
1. ISI Request for Relief RR-34, Revision 2
2. ISI Request for Relief RR-37, Revision 1
3. SNC Response to NRC RAI

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. Joseph Colaccino, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector – Hatch

ENCLOSURE 1

Edwin I. Hatch Nuclear Plant
Third 10-Year Interval Inservice Inspection (ISI) Program,
Response to Request for Additional Information for
Requests for Relief No. RR-34 Revision 2

SOUTHERN NUCLEAR OPERATING COMPANY
EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2
INSERVICE INSPECTION PROGRAM
REQUEST FOR RELIEF NO. RR-34 REVISION 2

I. System/Component(s) for Which Relief is Requested: Examination of Reactor Pressure Vessel (RPV) nozzle inner radii, ASME Code Category B-D, Item B3.100, other than Feedwater (FW) nozzles.

II. Code Requirement: ASME Code, Section XI, 1989 edition, Table IWB-2500-1, Examination Category B-D, Item B3.100, requires a volumetric examination of the inner radius region of all Reactor Pressure Vessel (RPV) nozzles welded with full penetration welds as shown in Figures IWB-2500-7(a) through (d).

III. Code Requirement for Which Relief is Requested: Relief is requested from the requirement of performing volumetric examination.

IV. Basis for Relief: Pursuant to 10CFR50.55a(a)(3)(i), SNC is requesting relief from the ASME Section XI requirement to perform volumetric examination as described above, except for the FW nozzles (note that the Control Rod Drive (CRD) return line at Plant Hatch has been cut and capped on both units.).

V. Justification for Granting Relief: Early in the development of ASME Section XI, examination requirements were applied to all nozzles welded with full penetration welds. RPV nozzle inner radius examinations are the only non-welded areas requiring ultrasonic examination, and no service related cracking or degradation has ever been found in the nozzle inner radius region in any of the BWR fleet plant nozzles other than on Feedwater or operational CRD return line nozzles. Examination of Feedwater nozzles will continue to be performed in accordance with augmented examination program commitments (i.e. NUREG-0619). For all nozzles other than Feedwater, there is no significant thermal cycling during operation, therefore, from a risk perspective, there is no need to perform volumetric examination on any other nozzles. Southern Nuclear Operating Company (SNC) believes that application of a visual examination alternative for the subject nozzle inner radius regions ensures an acceptable level of quality and safety.

VI. Alternate Examination:

SNC proposes the substitution of a visual, VT-1 type examination in lieu of the volumetric examination requirements. Direct visual examination of the RPV head spray (N6A(B)) and RPV head vent (N7) nozzles will be performed and the remaining nozzles inner radii regions will be examined using remote visual examination techniques. For both direct and remote visual examinations, the resolution sensitivity will be established using a 1-mil (.001 inch) wire standard, or equivalent. The visual examination coverage will include virtually 100% of the surface M-N as shown in ASME XI Figures IWB-2500-7(a) through (d). No examination coverage limitations exist for the below listed RPV nozzle inner radius regions.

Nozzle Inner Radius Visual Examination Summary *With No* Limitations		
RPV Nozzle	Quantity per Unit	Nozzle Description
N1	2	Recirculation Outlet Nozzle
N3	4	Main Steam Line Nozzle
N6	2	RPV Head Spray Nozzle
N7	1	RPV Vent Nozzle
N9	1	CRD Return Line Nozzle (Cut & Capped - both Units)

If crack-like surface flaws are detected by visual examination, the flaws will be characterized in accordance with Table IWB-3512-1. When applying Table IWB-3512-1 criteria, the crack depth will be assumed to be equal to one-half the measured crack length. Once the flaw characteristics are established, the flaws will be evaluated in accordance with ASME Section XI Code section IWB-3140.

VII. Implementation Schedule: This request for relief is applicable to examinations using the 1989 Edition of ASME Section XI for the remainder of the Third Ten-Year ISI Inspection Interval (1/1/96 – 12/31/05).

VIII. Relief Request Status: This request for relief is awaiting NRC approval.

ENCLOSURE 2

Edwin I. Hatch Nuclear Plant
Third 10-Year Interval Inservice Inspection (ISI) Program,
Response to Request for Additional Information for
Requests for Relief No. RR-37 Revision 1

SOUTHERN NUCLEAR OPERATING COMPANY
EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2
INSERVICE INSPECTION PROGRAM
REQUEST FOR RELIEF NO. RR-37, REVISION 1

I. System/Component(s) for Which Relief is Requested: Examination of Reactor Pressure Vessel (RPV) nozzle inner radii, ASME Code Category B-D, Item B3.100, other than Feedwater (FW) nozzles.

II. Code Requirement: ASME Code, Section XI, 1989 edition, Table IWB-2500-1, Examination Category B-D, Item B3.100, requires a volumetric examination of the inner radius section of all Reactor Pressure Vessel (RPV) nozzles welded with full penetration welds as shown in Figures IWB-2500-7(a) through (d).

III. Code Requirement for Which Relief is Requested: Relief is requested from the requirement of performing volumetric examination.

IV. Basis for Relief: Pursuant to 10CFR50.55a(a)(3)(ii), SNC is requesting relief from the ASME Section XI requirement to perform volumetric examination as described above, except for the FW nozzles (note that the Control Rod Drive (CRD) return line at Plant Hatch has been cut and capped on both units.).

VII. Justification for Granting Relief: Early in the development of ASME Section XI, examination requirements were applied to all nozzles welded with full penetration welds. RPV nozzle inner radius examinations are the only non-welded areas requiring ultrasonic examination, and no service related cracking or degradation has ever been found in the nozzle inner radius region in any of the BWR fleet plant nozzles other than on Feedwater or operational CRD return line nozzles. Examination of Feedwater nozzles will continue to be performed in accordance with augmented examination program commitments (i.e. NUREG-0619). For all nozzles other than Feedwater, there is no significant thermal cycling during operation, therefore, from a risk perspective, there is no need to perform volumetric examination on any other nozzles. Southern Nuclear Operating Company (SNC) believes that continued volumetric examination of the RPV nozzle inner radius regions is an unwarranted burden with little or no added safety benefit. SNC believes that application of a visual examination alternative for the subject nozzle inner radius regions provides an acceptable level of quality and safety.

The visual examination coverage will include all accessible areas of the surface M-N as shown in Figures IWB-2500-7(a) through (d). RPV internal component configurations (e.g., thermal sleeves, spargers, vessel internal attachments, instrumentation lines, etc.) prevent placement of the remote visual examination camera in positions necessary to examine surface M-N over the full circumference of the nozzle inner radius. However, examinations will be performed on the accessible nozzle inner radius region to the maximum extent practicable.

All nozzle forgings were examined during the fabrication process (volumetric and surface techniques) and have subsequently been examined in accordance with inservice inspection program requirements. No indication of fabrication defects or service induced cracking has been detected by these examinations to date.

Obtaining additional visual examination coverage would result in significant hardship due to the limitations of existing remote visual examination equipment and inability to remove or alter RPV internals components to allow additional coverage. Removal, or alteration, of the internal interference could result in damage to the components, requires specialized removal equipment, could require replacement with new components which are not readily available and are extremely expensive, and removal/re-installation requires significant expenditure of man-power. The tables below provide estimates of the examination coverage in the circumferential direction for each affected nozzle inner radius region.

Nozzle Inner Radius Visual Examination Summary *With* Limitations				
RPV Nozzle	Quantity per Unit	Description	Coverage Estimate [1]	Limitation
N2	10	Recirculation Inlet Nozzle	50%	Thermal Sleeve
N5	2	Core Spray Nozzle	40%	Thermal Sleeve & Sparger T
N8	2	Jet Pump Instrument Nozzle	Unit 1-40%	Instrument lines & Shroud Gusset
			Unit 2-50%	Instrument lines

[1] Circumferential coverage of surface M-N as shown in Figures IWB-2500-7(a) through (d).

The limited visual examination coverage does not significantly reduce the level of plant quality and safety for the following reasons:

1) There are no mechanisms of damage other than fatigue for the nozzle inner radius section, and for other than Feedwater nozzles, there is no significant thermal cycling. Therefore, the primary flaw of concern would be a flaw that was not detected during the manufacturing process[2]. All of the nozzles were examined during and after fabrication by surface and volumetric examination techniques. Additionally, preservice and inservice ultrasonic examinations have detected no flaws. It is very unlikely that any flaws would be initiated by the fatigue mechanism.
2) After approximately 27 years of reactor operation for Unit 1, and 23 years for Unit 2, no cracking of any kind has been detected in the subject nozzle inner radius regions.
3) Approximately 42% of the total nozzle population will receive a complete (100%) examination of the inner radius region (see Relief Request RR-34).
4) Visual examination of the accessible nozzle inner radius surface (zone M-N) provides reasonable assurance that deep flaws are not present. Additionally, when flaws are initiated by fatigue mechanisms, they typically are encountered over a significant portion of the nozzle circumference as was the case for cracking of Feedwater nozzles address in NUREG-0619.

[2 & 3] Conclusions made in ASME NDE Subcommittee Report ISI-99-26, "Technical Basis for Elimination of Reactor Vessel Nozzle Inner Radius Inspections."

In summary, fatigue cracking is the only applicable degradation mechanism for the nozzle inner radius region, and for all nozzles other than Feedwater there is no significant thermal cycling during normal operation. Therefore, from a risk perspective, there is no reason to perform volumetric examination of any nozzles other than Feedwater at Plant Hatch. The four (4) Feedwater nozzle inner radiuses on each Hatch Unit will continue to be examined with ultrasonic examination techniques that were developed and have been qualified in accordance with previous SNC commitments to the NRC. The Feedwater nozzles alone represent approximately 17% of all nozzles currently requiring volumetric examination, which exceeds industry accepted risk informed sampling scopes. Additionally, Relief Request RR-34 provides for 100% examination of the inner radius region for ten (10) nozzles (42%). SNC believes that the partial visual examination alternative for the nozzle inner radius regions above provides an acceptable level of quality and safety.

VI. Alternate Examination:

SNC proposes the substitution of a visual, VT-1, type examination in lieu of the volumetric examination requirements. All nozzle inner radii regions will be examined using remote visual examination techniques. These remote visual examinations will be performed in accordance with the ASME Section XI Code, 1989 edition, paragraph IWA-2211(c) except the resolution sensitivity will be established using a 1-mil (.001 inch) wire standard, or equivalent, which is superior to that used for other RPV internals examinations.

If crack-like surface flaws are detected by visual examination, the flaws will characterized in accordance with Table IWB-3512-1. When applying Table IWB-3512-1 criteria, the crack depth will be assumed to be equal to one-half the measured crack length. Once the flaw characteristics are established, the flaws will be evaluated in accordance with ASME Section XI Code section IWB-3140.

VII. Implementation Schedule: This request for relief is applicable to examinations using the 1989 Edition of Section XI for the remainder of the Third Ten-Year ISI Inspection Interval (1/1/96 – 12/31/05).

VIII. Relief Request Status: This request for relief is awaiting NRC approval.

ENCLOSURE 3

Edwin I. Hatch Nuclear Plant
Third 10-Year Interval Inservice Inspection (ISI) Program,
Response to Request for Additional Information:
SNC's Response to NRC RAI

ENCLOSURE 3

SOUTHERN NUCLEAR OPERATING COMPANY
EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2
INSERVICE INSPECTION PROGRAM
SNC RESPONSE TO NRC RAI

NRC Question No. 1

Provide a list/table of the nozzles, nozzle sizes, percent of coverage from prior examinations, and when (month/year) the examinations occurred.

SNC Response

See tables for Hatch Unit 1 and Unit 2 below.

Hatch Unit 1 RPV Nozzle Inner Radius Inspections						
Nozzle	Nozzle Description	Size	Last UT Inspection	UT Coverage	Estimated Visual Coverage	Visual Limitation
N1A	Recirculation Outlet	28"	Fall/97	100%	100%	None
N1B	Recirculation Outlet	28"	Fall/00	100%	100%	None
N2A	Recirculation Inlet	12"	Fall/97	100%	50%	Thermal Sleeve
N2B	Recirculation Inlet	12"	Fall/00	100%	50%	Thermal Sleeve
N2C	Recirculation Inlet	12"	Fall/94	100%	50%	Thermal Sleeve
N2D	Recirculation Inlet	12"	Fall/00	100%	50%	Thermal Sleeve
N2E	Recirculation Inlet	12"	Fall/00	100%	50%	Thermal Sleeve
N2F	Recirculation Inlet	12"	Fall/94	100%	50%	Thermal Sleeve
N2G	Recirculation Inlet	12"	Fall/00	100%	50%	Thermal Sleeve
N2H	Recirculation Inlet	12"	Fall/00	100%	50%	Thermal Sleeve
N2J	Recirculation Inlet	12"	Fall/94	100%	50%	Thermal Sleeve
N2K	Recirculation Inlet	12"	Fall/00	100%	50%	Thermal Sleeve
N3A	Main Steam	24"	Fall/97	100%	100%	None
N3B	Main Steam	24"	Fall/97	100%	100%	None
N3C	Main Steam	24"	Fall/00	100%	100%	None
N3D	Main Steam	24"	Fall/00	100%	100%	None
N5A	Core Spray	10"	Fall/94	96%	40%	Thermal Sleeve & Sparger
N5B	Core Spray	10"	Fall/97	100%	40%	Thermal Sleeve & Sparger
N6A	RPV Head Spray	6"	Fall/00	93%	100%	None
N6B	RPV Head Spray	6"	Fall/00	93%	100%	None
N7	RPV Vent	4"	Fall/00	95%	100%	None
N8A	Jet Pump Instrument	4"	Spring/99	100%	40%	Thermal Sleeve & Shroud Gussets
N8B	Jet Pump Instrument	4"	Fall/00	100%	40%	Thermal Sleeve & Shroud Gussets
N9	CRD	4"	Fall/97	100%	100%	None

Hatch Unit 2 RPV Nozzle Inner Radius Inspections						
Nozzle	Nozzle Description	Size	Last UT Inspection	UT Coverage	Estimated Visual Coverage	Visual Limitation
2N1A	Recirculation Outlet	28"	Spring/00	100%	100%	None
2N1B	Recirculation Outlet	28"	Spring/94	100%	100%	None
2N2A	Recirculation Inlet	12"	Spring/97	100%	50%	Thermal Sleeve
2N2B	Recirculation Inlet	12"	Spring/94	100%	50%	Thermal Sleeve
2N2C	Recirculation Inlet	12"	Spring/00	100%	50%	Thermal Sleeve
2N2D	Recirculation Inlet	12"	Spring/97	100%	50%	Thermal Sleeve
2N2E	Recirculation Inlet	12"	Spring/00	100%	50%	Thermal Sleeve
2N2F	Recirculation Inlet	12"	Spring/94	100%	50%	Thermal Sleeve
2N2G	Recirculation Inlet	12"	Spring/97	100%	50%	Thermal Sleeve
2N2H	Recirculation Inlet	12"	Spring/00	100%	50%	Thermal Sleeve
2N2J	Recirculation Inlet	12"	Spring/94	100%	50%	Thermal Sleeve
2N2K	Recirculation Inlet	12"	Spring/97	100%	50%	Thermal Sleeve
2N3A	Main Steam	24"	Spring/97	100%	100%	None
2N3B	Main Steam	24"	Spring/97	100%	100%	None
2N3C	Main Steam	24"	Spring/00	100%	100%	None
2N3D	Main Steam	24"	Spring/00	100%	100%	None
2N5A	Core Spray	10"	Spring/94	100%	40%	Thermal Sleeve & Sparger
2N5B	Core Spray	10"	Spring/00	100%	40%	Thermal Sleeve & Sparger
2N6A	RPV Head Spray	6"	Spring/00	100%	100%	None – Direct VT
2N6B	RPV Head Spray	6"	Spring/00	100%	100%	None – Direct VT
2N7	RPV Vent	4"	Spring/00	100%	100%	None – Direct VT
2N8A	Jet Pump Instrument	4"	Spring/97	100%	50%	Thermal Sleeve
2N8B	Jet Pump Instrument	4"	Spring/97	100%	50%	Thermal Sleeve
2N9	CRD	4"	Spring/97	100%	100%	None

NRC Question No. 2

Discuss the changes to examination coverage as a result of using VT (% surface) in lieu of the UT (% volume).

SNC Response

For the nozzles included in relief request RR-34, the coverage is basically 100% of volume for UT and 100% of surface for VT. There are no internal interferences for the Recirculation Outlet (N1 nozzles), Main Steam (N3 nozzles), and Control Rod Drive (CRD) Return Line (N9 – cut & capped) nozzles, therefore the coverage is 100% for UT and VT. Access is suitable for direct visual examination from under the reactor pressure vessel (RPV) head for the Head Spray Nozzles (N6 nozzles), and Vent Nozzle (N7). Therefore, 100% access to the surface is available for remote visual examination of these three nozzles, whereas UT coverage is slightly less than 100% of the volume.

For the nozzles included in relief request RR-37, the remote visual examination surface coverage will be less than the UT volume coverage due to internal interferences (e.g., thermal sleeves, internal piping, shroud gussets for Unit 1, etc.). The visual examination coverage indicated in the above tables are estimates based on experience with remote visual examination equipment, limitations for positioning the remote camera, and physical limitation posed by internal interference. Each nozzle, which is remotely examined, will be examined on a best effort basis.

NRC Question No. 3

Identify the ASME Code Inservice Inspection Program: either A or B. Reference the appropriate IWB-2500-1 item number: either B3.20 or B3.100.

SNC Response

SNC uses ASME Section XI Code ISI Program B at Plant Hatch. Therefore, Table IWB-2500-1, Item B3.100 is applicable.

NRC Question No. 4

Discuss any changes in radiation exposure to personnel as a result of the change in examination method.

SNC Response

SNC did not discuss radiation exposure reductions in the original relief requests because we did not feel confident in quantifying the reductions. SNC has been very proactive in reducing the nuclear source term at Plant Hatch which has resulted in significant reductions in radiation dose rates inside the primary containment (drywell). UT examination of all of these nozzle inner radius (IR) regions requires personnel to be in very close proximity to the RPV shell or head. Some of the nozzle configurations (e.g., thermal sleeves) also allow for entrapment of radioactive material that could result in high personnel dose rates due to "hot-spots". SNC has even made efforts in the past to perform nozzle flushing prior to examination to reduce these "hot-spots" requiring 12-16 hours of outage time. Since no nozzle IR examinations have been performed since these dose rate reduction efforts have been implemented, SNC did not feel confident in trying to estimate the savings. The IR UT examinations would be performed at the same time as the nozzle-to-shell weld examinations utilizing automated examination equipment. Performing the IR examinations does require reconfiguration of the automated scanner which required personnel to be in close proximity of the nozzle and RPV shell. Therefore, even though SNC feels uncomfortable providing accurate estimates, the personnel exposure time would be reduced approximately 1-3 hours per nozzle. The dose rates under the RPV head (nozzles N6A, N6B, and N7) are typically low (< 10 mr) therefore, radiation exposure for these examinations is expected to be approximately the same for VT or UT. However, since all other IR areas would be remotely examined, the radiation dose will be less than that for UT since the dose for remote visual is negligible.

NRC Question No. 5

Provide a technical discussion that supports structural integrity, such as the Oak Ridge report mentioned in RR-37. Provide sufficient data for third party retrieval of the Oak Ridge report.

SNC Response

The reason for the differences in justification provided in relief requests RR-34 and RR-37 was due to SNC's understanding that additional justification was needed for nozzles with VT examination coverage < 100%. However, the basic justifications included in both relief requests are applicable to all nozzles. Note that SNC has deleted reference to the Oak Ridge report in RR-37 since it was identified in another technical report (ISI-99-26, see below) and we have been unable to retrieve a copy or find information allowing easy retrieval. However, additional technical justification is provided below.

The ASME Section XI Task Group ISI Optimization developed a technical justification for the elimination of UT of the nozzle IR region during development of Code Case N-648. This report[1] was developed to justify elimination of examinations entirely; however, subsequent discussions with NRC personnel resulted in a compromise of VT in lieu of UT. NRC staff member, Mr. T. K. McLellan, is a member of this Task Group and should be able to provide a copy of the subject report for reference. Note that this report utilized prior work performed by Oak Ridge National Laboratories which is specifically referenced in the ASME report.

Development of the above reference Code Case (N-648) was coordinated with the Westinghouse Owners Group (WOG), ASME, and the NRC. On May 9, 2000, the WOG met with NRC personnel to discuss issues related to the proposed elimination of examination requirements for RPV nozzle IR regions. Although justification was presented to eliminate any examination of RPV nozzle IR regions (excluding BWR Feedwater and CRD nozzles), a consensus was reached between the WOG and the NRC, to replace the volumetric examination of the RPV nozzle (non Feedwater and CRD) with a visual (VT-1) examination. Subsequent to that agreement, additional provisions have been added related to remote visual examination resolution criteria.

[1] ASME Section XI Subcommittee White Paper ISI-99-26, "Technical Basis for Elimination of Reactor Vessel Nozzle Inner Radius Inspections."

NRC Question No. 6

Describe the direct/enhanced visual examination systems and the resolution sensitivity that will be used during the examinations. Discuss how the 1-mil (0.0001 inch) resolution sensitivity will be demonstrated for each system. Provide the same for the remote enhanced visual examination systems.

SNC Response

The direct visual examination of the RPV head nozzles (N6A, N6B, and N7) IR regions, will be performed in accordance with 1989 Edition of ASME Section XI, paragraph IWA-2211, VT-1 requirements. However, the resolution sensitivity will be enhanced to require that of a 1-mil, or smaller, wire. A resolution standard containing a 1-mil, or smaller, wire will be placed on the surface adjacent to the nozzle IR and the examiner will verify resolution capability using natural or artificial light as appropriate.

Examination of all other nozzles will be performed remotely using RPV & internals visual inspection techniques consistent with the BWR Vessel and Internals Project (BWRVIP), Reactor Pressure Vessel and Internals Examination Guidelines, BWRVIP-03. Camera resolution will be demonstrated to be capable of resolving a 1-mil (.001") wire (or better) resolution standard at the distance and lighting conditions present when the examinations are performed.

NRC Question No. 7

Provide a description for the performance of the proposed enhanced direct visual examination of the RPV nozzles that will assure a 1-mil resolution.

SNC Response

The direct visual examination of the RPV head nozzle IR regions will be performed in accordance with the 1989 Edition of ASME Section XI, paragraph IWA-2211 (i.e., VT-1 requirements). However, the resolution sensitivity will be demonstrated utilizing a standard containing a 1-mil, or smaller, wire at a maximum distance of 2 feet in lieu of the 1/32 inch black line on an 18% neutral gray card. Also, see the response for Question #6 above.

NRC Question 8

For each unit, provide the appropriate Code edition of the ASME Section XI, current inservice inspection (ISI) interval, and when the current ISI interval is over.

SNC Response

The ISI Program for Hatch Unit 1 and Unit 2 was developed using the ASME Section XI Code, 1989 Edition, with no Addenda. The current ISI interval for both units began on January 1, 1996 and ends on December 31, 2005.

NRC Question No. 9

For RR-37, discuss the obstructions for each nozzle or group of nozzles that limit full coverage.

SNC Response

For the ten (10), 12 inch diameter, Recirculation Inlet Nozzles (N2 nozzles), the thermal sleeve and inlet riser elbow to thermal sleeve connection limit access to the bottom side of the nozzle and the area adjacent to the inside bend radius of the elbow. Therefore, review of fabrication drawings results in an estimated coverage of 50% for both units. This 50% total coverage results from examinations on each side of the nozzle IR region.

For the 10 inch diameter Core Spray Nozzles (N5 nozzles), access is limited by the thermal sleeve, Core Spray T-Box, and internal piping on each side of the T-Box. Therefore, review of fabrication drawings results in an estimated coverage of 40% for both units. This 40% coverage results from examinations near the top and both sides of each nozzle IR region.

For the 4 inch, Jet Pump Instrument Nozzles (N8 nozzles), several 1 inch instrument lines converge to exit the RPV through these nozzles. These 1 inch lines limit access to the bottom side of the nozzles. Additional limitations exist for Hatch 1 due to the shroud support ledge and a shroud support gusset resulting in approximately 40% total coverage. For Hatch 2, the shroud support ledge is also directly below the nozzle, and coverage is limited to approximately 50% total. The coverage for these nozzles results from examinations near the top and each side of the IR region.

NRC Question No. 10

For RR-37, if leak-before-break is a part of the bases for structural integrity of the vessel, provide the supporting technical justification.

SNC Response

SNC understands that the NRC does not consider leak-before-break to be an acceptable option for the RPV. However, the report referenced in Question 5 (i.e., ISI-99-26) does include discussion of research work which indicated that such would occur. Therefore, SNC does not pose leak-before-break as acceptable justification for these relief requests, but does pose the information contained in the reference report as being worthy of NRC review.