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September 9, 2011
L-11-267

10 CFR 50.55a

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
Response to Request for Additional Information Related to the 10 CFR 50.55a Requests In Support of the Third 10-Year Inservice Inspection Interval (TAC Nos. ME5376, ME5377, ME5379, ME5380, and ME5381)

By letter dated January 24, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110320065), the FirstEnergy Nuclear Operating Company (FENOC) submitted multiple 10 CFR 50.55a requests associated with the Perry Nuclear Power Plant Inservice Inspection Program. These requests apply to the third 10-year inservice inspection interval. By letter dated July 26, 2011 (ADAMS Accession No. ML112020459), the Nuclear Regulatory Commission (NRC) staff requested additional information to complete its review. Responses to the NRC staff's questions are provided in the attachment.

As identified in the response to request 5, FENOC agreed to withdraw the proposed alternative applicable to the top guide grid. An updated copy of IR-056, Revision 1, with the top guide grid information removed, is enclosed to assist in the review process. Revision bars were used to designate the areas of change.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

Sincerely,

Mark B. Bezilla

Attachment:

Response to Request for Additional Information Related to Third 10-Year Inservice Inspection Interval for Perry Nuclear Power Plant

Enclosure:

Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-056, Revision 1

cc: NRC Region III Administrator
NRC Resident Inspector
Nuclear Reactor Regulation Project Manager

By letter dated July 26, 2011, the Nuclear Regulatory Commission (NRC) staff requested additional information to complete its review of alternatives proposed in FirstEnergy Nuclear Operating Company (FENOC) correspondence dated January 24, 2011. FENOC's responses for the Perry Nuclear Power Plant (PNPP) are provided below. The NRC staff's questions are presented in bold type, followed by FENOC's responses.

REQUEST FOR ADDITIONAL INFORMATION

1. **Proposed Alternative IR-013, Revision 2, ASME Code, Section XI, Examination Category C-G, Item C6.10, Pressure Retaining Welds in Pumps and Valves**

For ASME Code, Class 2, pump casing welds, the ASME Code requires 100 percent surface examination be performed on either the inner or outer surface of the weld on at least one pump from each group that has a similar design, size, function and service in a system. In accordance with 10 CFR 50.55a(a)(3)(i) or 10 CFR 50.55a(a)(3)(ii), the licensee may propose an alternative provided that: (1) The licensee demonstrates that the proposed alternatives would provide an acceptable level of quality and safety; or (2) the licensee demonstrates that the ASME Code requirements are a hardship, or unusual difficulty, in complying with ASME Code examination requirements that are adequately described and the licensee can demonstrate that complying with the ASME Code requirements would result in no compensating increase in quality and safety.

The licensee has provided descriptions of the pumps, which are located in the concrete flooring, and would require disassembly for examination of the subject welds; however, the stated reason the pump would have to be disassembled is too general and insufficient to demonstrate it would provide a hardship or unusual difficulty.

Since the licensee requested the alternative under hardship, they should discuss the specific causes of, or unusual difficulty, in support of an evaluation under 10 CFR 50.55a(a)(3)(ii). Examples of hardship or unusual difficulty include, but are not limited to: having to enter multiple technical specifications' limiting conditions for operations, as low as reasonably achievable (ALARA) concerns (including personnel exposure estimates), or creating significant hazards to plant personnel.

The second part of the licensee's proposal must show that, even if the pumps were to be disassembled and the subject welds were to be examined, these activities would not result in a compensating increase in the level of quality and safety. Discuss why the ASME Code-required examinations on the inside surface of the welds, requiring disassembly of the pumps, would not provide an increase in quality and safety.

The licensee has provided a typical description and sketch of a pump casing. The description implies that the floor is preventing examinations of all of the casing welds.

(a) Provide a sketch showing the restrictions. If the restrictions are associated with a pit, provide the approximate depth and annulus dimensions. Show the locations where the pump is in intimate contact with the supporting structure. If the casing below the floor is completely encased in concrete, provide the approximate depth of the encased casing.

(b) Provide a discussion on the accessibility for remote visual examinations performed from the outside or inside casing surface (including welds below the floor).

(c) Provide a pecent estimate of a combined visual and surface examination of the pump casing.

(d) Identify the number of pumps in the group being represented by the residual heat removal pump A.

(e) Provide a discussion on the effects a crack pump casing weld would have on the functionality of the pump and the effects on the region outside the casing.

The subject pumps are Byron Jackson vertical pumps installed for PNPP residual heat removal (RHR) (that is, the low pressure coolant injection mode), low pressure core spray (LPCS), and high pressure core spray (HPCS) emergency core cooling systems (ECCS). Other than the number of stages, the basic design of the pumps are similar. The inservice examination program sketch for the RHR A pump that was provided in the January 24, 2011 submittal request as a typical sketch shows that the pump casings or barrels are below the floor elevation. Additional details can be seen in the provided drawings. The pump barrels are encased in a steel-lined concrete pit. The weights of the major pump components, the number of stages, and the depths of the pump barrels are provided in the following table.

Description	RHR Pumps	LPCS Pump	HPCS Pump
Barrel Weight (lbs)	7,000	4,800	10,000
Motor Weight (lbs)	7,800	11,500	20,900
Discharge Head-Column-Bowl Assembly Weight (lbs)	16,000	13,100	28,000
Number of Stages	3	5	13
Approximate Depth of Pump Barrel	21 ft	23 ft	23.5 ft

In addition to disassembly of the piping and electrical connections, disassembly of the pumps would involve significant rigging and heavy load lifts. Furthermore, due to the impeller assembly lengths, the pumps cannot be pulled without partial disassembly of the pump internals, thus making disassembly even more difficult. Accurate man-hours involved with disassembly of these pumps are not available because there is little experience with disassembly. It is estimated to take several hundred man-hours to disassemble each pump. This estimate does not include the man-hours required for reassembly. General area dose rates in the areas of the pumps vary, with the lowest being less than 2 millirem/hour in the area of the LPCS pump to the highest being over 35 millirem/hour in the area of the RHR A pump. It is estimated that dose for disassembly and reassembly would result in at least 1 rem (LPCS pump) to more than 10 rem (RHR A pump). Thus, substantial personnel exposure would be necessary to disassemble and reassemble the pumps. With the sizes and weights of the components, significant rigging, lifts, and dose, disassembly and reassembly of these pumps would provide a hardship or unusual difficulty.

As indicated in the request, similar pump casing welds accessible above the floor elevation are examined on a continuing basis. Since the construction and operating conditions of these pump casing welds are identical to those of the inaccessible welds, it is reasonable to apply satisfactory results from examined welds to the unexamined welds. With the acceptable initial condition and the capability to examine the similar accessible welds on a continuing basis, it is concluded that disassembling the pumps to perform the applicable Code examinations would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

The barrel flange (the sole plate for the pump) is grouted and bolted to the floor and is not designed for routine removal. Thus, the annular space between the pump barrels and the pit walls, which is approximately 4.5 inches for the RHR and LPCS pumps and 2.5 inches for the HPCS pump, is not accessible for surface or remote visual examination. As can be seen in the provided drawings of the pumps, the interior surfaces of the pump barrel welds are only accessible through the suction side of the pump. The only access for remote visual examination from inside the pump barrels is through a 3/4-inch vent connection in the head shell. From there, lowering the fiberscope and locating the barrel welds would be difficult and only those portions of the

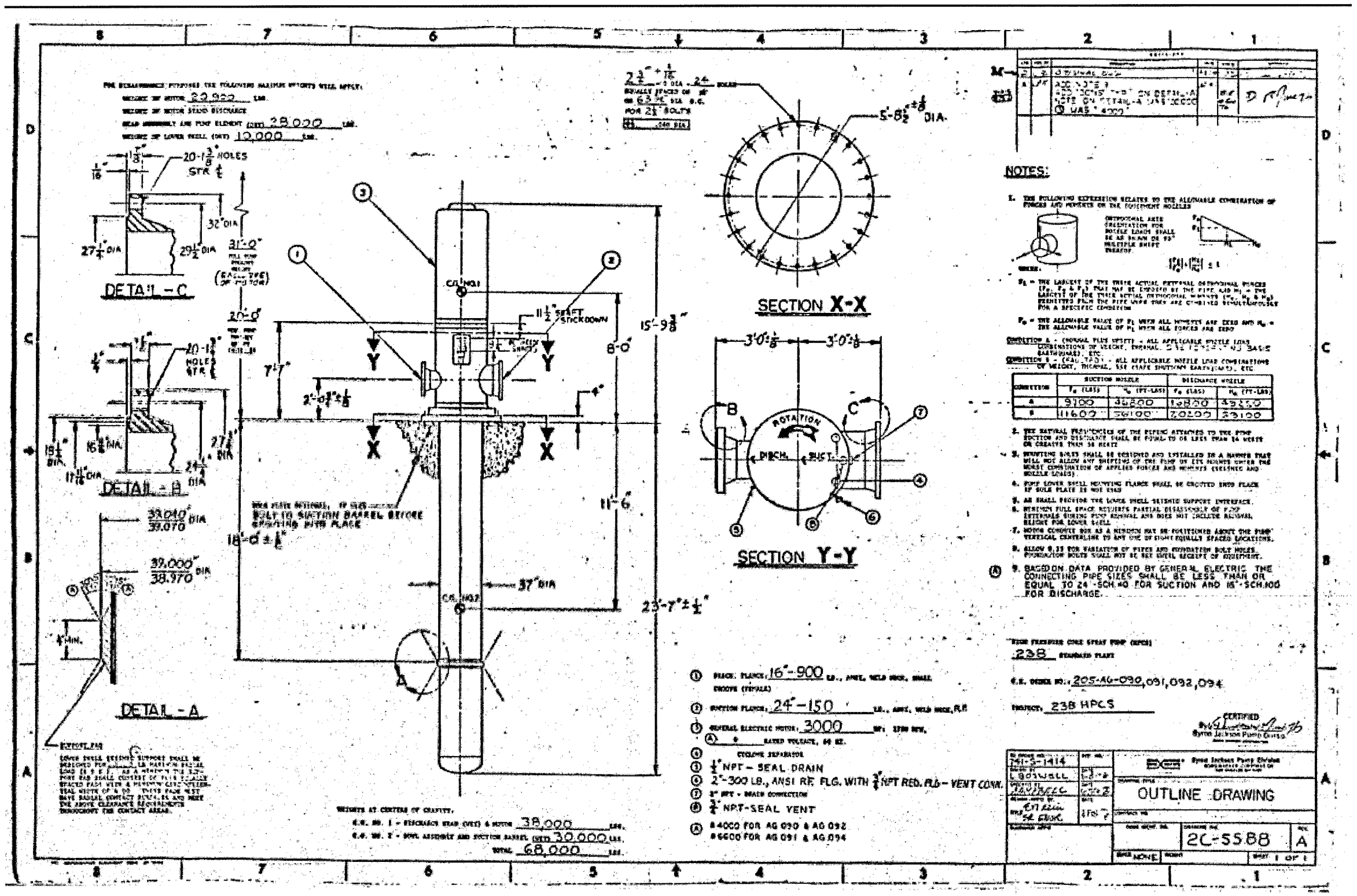
circumferential barrel welds directly below the vent connection are able to be inspected. Lowering the fiberscope from the vent connection does not allow for control of the focal distances, making it unlikely that images of sufficient quality could be obtained, even for the very limited areas that could be seen. Based on the above, it is unlikely that any percentage of visual examinations could be credited for distance, lighting, and acuity requirements.

Each of the ECCS pumps has 15 Category C-G, Item C6.10, casing welds of which eight (approximately 53 percent) are accessible. Those on the selected pumps receive the required surface examinations each inspection interval. For the seven casing welds in the barrel below the floor, no examinations are performed.

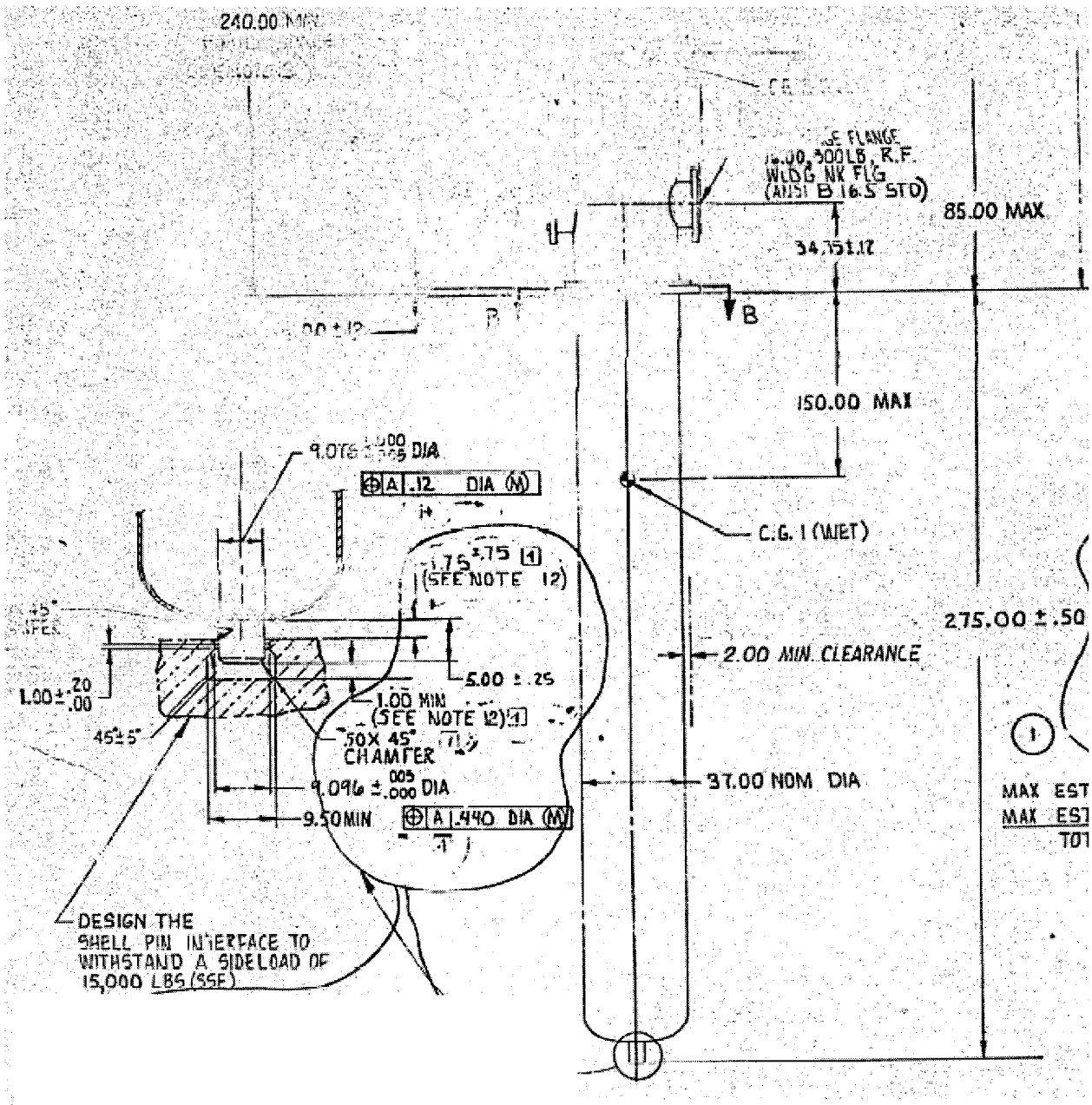
As indicated in the request, the design and weld configuration for RHR pump A is representative of the five ECCS pumps. With regard to selection for examination, Category C-G Note 1 of the table states that in the case of multiple pumps of similar design, size, function and service in a system, required weld examinations may be limited to all the welds in one pump in the same group. In this context, RHR pump A is only representative of RHR pumps A and B. The design, size and function of RHR pump C is similar, but RHR pump C performs the service of low pressure coolant injection (LPCI) while the RHR pumps A and B perform additional functions; therefore, PNPP groups it separately from the A and B pumps. The functions and service of the LPCS and HPCS pumps are different and also grouped separately. Thus, under Category C-G, four of the five ECCS pumps are selected for examination.

The pump casings are only subject to the suction pressure. The design suction pressure is 115 pounds per square inch (psi) for the two core spray pumps and 215 psi for the three RHR pumps. During plant operation with the ECCS pumps in standby, the suction pressure for the pumps are approximately 9 psi for the LPCS and RHR pumps and 23 psi for the HPCS pump. In the unlikely event of cracking (the welds are carbon steel welds that are only subjected to low pressure conditions and operating experience review found no reported degradation of these welds), there could be some leakage through the casing. The affect of cracking on pump functionality would be indeterminate; however, based on engineering judgment, it would be expected to have little impact on the capacity of the pump. Likewise, the leakage would be expected to have little affect on the annular space between the barrel and the pit, the region outside the casing, because the concrete walls of the pit are greater than 1 foot thick and have a 1/4-inch steel leak-tight barrier. If there was sufficient leakage to fill the annular space, it would then leak out from the pump casing sole-plate-to-floor interface and would be detectable during VT-2 walkdowns. The leakage would also run into the ECCS pump room floor drains and to the sumps where there are high level sump alarms that would alert Control Room operators to have the pump rooms inspected for leakage.

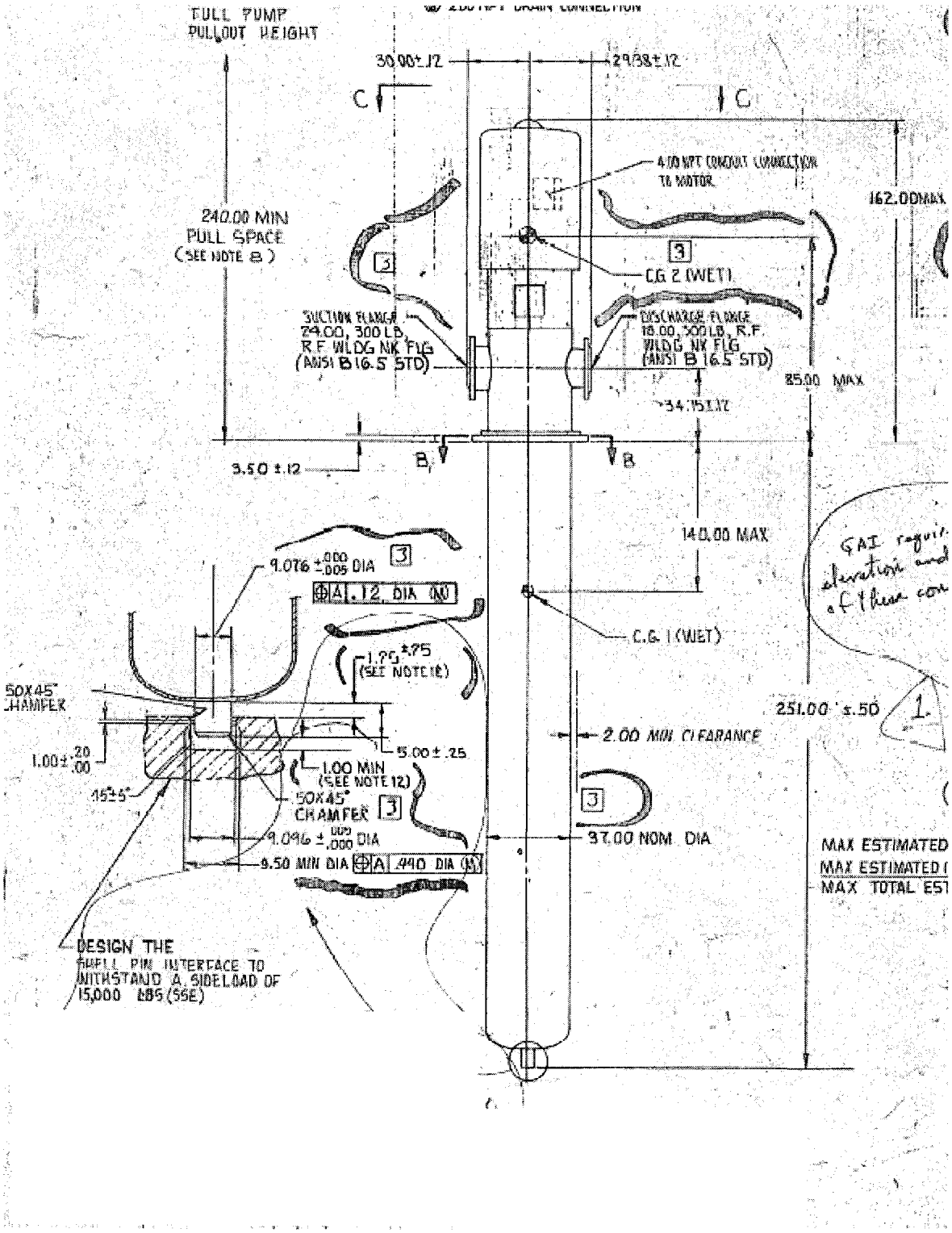
HPCS Pump Outline Drawing
Typical of Byron Jackson ECCS Pumps



LPCS Pump


240.00 MIN
85.00 MAX
34.75±.12
B
150.00 MAX
C.G. 1 (WET)
2.00 MIN. CLEARANCE
275.00 ±.50
37.00 NOM DIA
.12 DIA (M)
1.75 ±.75
(SEE NOTE 12)
5.00 ±.25
1.00 MIN
(SEE NOTE 12)
.50X 45°
CHAMFER
9.096 ±.005/.000 DIA
9.50 MIN
.440 DIA (M)
1.00±.20/.00
45°±5°
DESIGN THE
SHELL PIN INTERFACE TO
WITHSTAND A SIDELOAD OF
15,000 LBS (SSE)

RHR Pump
(Applicable to RHR A, B, and C)



2. **Proposed Alternative IR-027, Revision 2, ASME Code, Section XI, Examination Category D-A, Item D1.10, Welded Attachments for Vessels, Piping, Pumps, and Valves**

For welded attachments to ASME Code Class 3 pressure vessels, the ASME Code requires 100 percent visual VT-1 examination on one vessel from each group that has a similar design, function, and service in a system. In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee has proposed an alternative. When proposing an alternative under 10 CFR 50.55a(a)(3)(ii), the licensee must demonstrate that compliance with the specified requirements of the ASME Code would result in (1) hardship, or unusual difficulty, and (2) no compensating increase in the level of quality and safety would result. The licensee stated that the subject welds are covered in Pyrocrete, which is a hard and rigid material used for fire protection. In order to remove this material from the welds, cutting and chipping of the Pyrocrete would be required; however, this brief description of removal activities does not adequately demonstrate a specific basis for hardship or unusual difficulty. Provide a discussion on the specific causes of hardship, or unusual difficulty, in support of an evaluation under 10 CFR 50.55a(a)(3)(ii). Examples of hardship or unusual difficulty include, but are not limited to having to enter multiple technical specifications' limiting conditions for operations, ALARA concerns (including personnel exposure estimates), or creating significant hazards to plant personnel.

The Pyrocrete on the emergency diesel generator fuel oil day tanks is a fire retardant coating that was applied as a permanent coating, is brittle in nature, and difficult to remove. Pyrocrete is designed to provide a three-hour fire barrier between the fuel oil and the diesel engine. Pyrocrete is a cementitious inorganic formulation that when cured forms a hard inorganic coating that resists abrasion, vibration, impact, and other forms of physical abuse. For the diesel generator fuel oil day tanks, the applied thickness is 1 7/16 inches on the supports and 3 inches over expanded metal lath for the tank. Removal would require chipping and/or grinding the Pyrocrete off of the welds and cutting away the underlying wire mesh. Removal would be further complicated by welds that may not be ground flush. Damage to the subject weld or to surrounding areas could also occur during the removal process. Based on the removal process required and the resulting potential damage to the subject welds, it is concluded that removing the Pyrocrete to perform the applicable examinations would result in hardship or unusual difficulty.

3. **Proposed Alternative IR-043, Revision 2, ASME Code, Section XI, Examination Catetory B-M-1, Items B12.30 and B12.40, Pressure Retaining Welds in Valve Bodies**

When proposing an alternative under 10 CFR 50.55a(a)(3)(ii), the licensee must demonstrate that compliance with the specified requirements of the ASME Code would result in (1) hardship, or unusual difficulty, and (2) no compensating increase in the level of quality and safety. The licensee stated that no failures of the valve body welds have been experienced to date, no degradation mechanism has been identified for these welds, and further, the 2008 Edition of the ASME Code eliminated the surface and volumetric examinations for these welds. However, the licensee's statements do not satisfy the requirements for use of 10 CFR 50.55a(a)(3)(ii). Specifically, discuss why there is not a compensating increase in the level of quality and safety if the licensee were to inspect the subject valve body welds.

State the materials of construction for the gate valves listed in Table 2.3.1 below.

Table 2.3.1 – Examination Category B-M-1		
ASME Code Item	Weld ID	Weld Type
B12.30	1G33-F0101-SEAM	Reactor Water Clean-Up 3" Gate Valve
B12.40	1G33-F0100-SEAM	Reactor Water Clean-Up 4" Gate Valve
B12.40	1G33-F0106-SEAM	Reactor Water Clean-Up 4" Gate Valve
B12.40	1G33-F0001-SEAM	Reactor Water Clean-Up 6" Gate Valve
B12.40	1G33-F0004-SEAM	Reactor Water Clean-Up 6" Gate Valve
B12.40	1E12-F0019-SEAM	Residual Heat Removal, 6" Check Valve
B12.40	1E51-F0013-SEAM	Reactor Core Isolation Cooling, 6" Gate Valve
B12.40	1E51-F0063-SEAM	Reactor Core Isolation Cooling, 10" Gate Valve
B12.40	1E51-F0064-SEAM	Reactor Core Isolation Cooling, 10" Gate Valve
B12.40	1E12-F0039A-SEAM	Residual Heat Removal, 12" Gate Valve
B12.40	1E12-F0039B-SEAM	Residual Heat Removal, 12" Gate Valve
B12.40	1E12-F0039C-SEAM	Residual Heat Removal, 12" Gate Valve
B12.40	1E12-F0042A-SEAM	Residual Heat Removal, 12" Gate Valve
B12.40	1E12-F0042B-SEAM	Residual Heat Removal, 12" Gate Valve
B12.40	1E12-F0042C-SEAM	Residual Heat Removal, 12" Gate Valve
B12.40	1E21-F0005-SEAM	Low Pressure Core Spray, 12" Gate Valve
B12.40	1E21-F0007-SEAM	Low Pressure Core Spray, 12" Gate Valve
B12.40	1E22-F0036-SEAM	High Pressure Core Spray, 12" Gate Valve

The latest Edition of 10 CFR 50.55a only lists the 2004 Edition of the ASME Code as the latest edition approved for use. State why it is appropriate to cite the unapproved 2008 Edition of ASME Code Section XI, or provide the date of a safety evaluation (SE) that allows the use of the 2008 Edition of ASME Code Section XI.

The hardship for examining these valves with surface or volumetric techniques is the amount of dose received from insulation removal and reinstallation, and the actual examination. Since the ASME Code only requires inspections of 8 of the 18 welds listed in Table 2.3.1, state if any of these valves are located in low dose areas and if there are other similar gate valves in these systems that can be examined that would not result in high personnel exposure as when examining the valves listed above.

The gate valves listed in Table 2.3.1 are carbon steel. As described in the request, the operating experience for valve body welds has been excellent with no reported weld failures or unacceptable indications. Risk-informed insights have not identified any degradation mechanism for these welds. With no active inservice degradation mechanisms, performance of examinations on the valve body welds would provide no compensating increase in the level of quality or safety. As a result, the American Society of Mechanical Engineers (ASME) deleted the Category B-M-1 weld examinations from ASME Section XI in the 2008 Addenda, which was approved and incorporated into 10 CFR 50.55a on July 21, 2011.

The 2008 Edition of ASME Code Section XI was cited by FENOC in the context that with the valve weld examinations eliminated in that Edition, it was further evidence that excellent performance of valve body welds has been recognized.

The valves selected as representative valves for the eight valve groupings were originally selected based on accessibility and dose considerations and are generally the lower dose valves among the groupings. As the groupings involve valves within the same systems and same service, dose rates among the valves within a grouping are similar. From one operating cycle to the next, dose rates may vary and the low dose valve within a particular valve grouping may change, but typically the change is not enough to justify varying from the successive examination requirements of IWB-2420(a), which requires that inspections be repeated on the same components from one inspection interval to the next to the extent practical.

4. **Proposed Alternative IR-054, Revision 1, ASME Code, Section XI, Examination Category B-D, Items B3.90 and B3.100, Full Penetration Welded Nozzles in Vessels**

The licensee proposed, in lieu of performing examinations on 100 percent of the reactor vessel nozzle-to-vessel welds and nozzle inside radius sections, to incorporate Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds Section XI, Division 1," which requires a minimum of 25 percent of nozzle inner radii and nozzle-to-shell welds, including at least one nozzle from each system and nominal pipe size. NRC Regulatory Guide 1.193, Revision 3, "ASME Code Cases Not Approved for Use," states that:

> **The applicability of Code Case N-702 must be shown by demonstrating that the criteria in Section 5.0 of NRC Safety Evaluation dated December 19, 2007 (ADAMS Accession No. ML073600374) regarding BWR Vessel and Internals Project (BWRVIP)-108: "BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii, Electric Power Research Institute (EPRI) Technical Report 1003557, October 2002" (ADAMS Accession No. ML023330203) are met. The evaluation demonstrating the applicability of the ASME Code Case shall be reviewed and approved by the NRC prior to the application of the Code Case.**

The five criteria are related to the driving force of the probabilistic fracture mechanics (PFM) analyses for the recirculation inlet and outlet nozzles. It was stated in the December 19, 2007 SE that the nozzle material fracture toughness-related nil-ductility transition reference temperature (RT_{NDT}) values used in the PFM analyses were based on data from the entire fleet of BWR reactor pressure vessels (RPVs). Therefore, the BWRVIP-108 report PFM analyses are bounding with respect to fracture resistance, and only the driving force of the underlying PFM analyses needs to be evaluated. It was also stated in the December 19, 2007 SE that, except for the RPV heat-up/cool-down rate, the plant-specific criteria are for the recirculation inlet and outlet nozzles only because the probabilities of failure for other nozzles are an order of magnitude lower.

FENOC provided their calculations and results, which meet the criteria set forth in Section 5 of the NRC SE mentioned above. However, there is a discrepancy in the specific values provided for the RPV inner radius (r) and wall thickness (t) at PNPP. The values provided in proposed alternative

IR-054 for the RPV inner radius and wall thickness are 119 inches and 7.19 inches, respectively. The values provided in BWRVIP-108, Table 3-1 for the PNPP RPV inner radius and wall thickness are 120.2 inches (RPV inner diameter was provided in the table as 240.4 inches) and 6 inches, respectively. If the PNPP values stated in BWRVIP-108 are used, the licensee would not be in compliance with the SE for the recirculation outlet nozzle-to-vessel welds.

Verify and state the specific values for the RPV inner radius and wall thickness provided in proposed alternative IR-054 and explain why there is an inconsistency in the values provided in the BWRVIP-108 report and those provided in your submittal dated January 24, 2011.

Additionally, verify and state the specific values for the nozzle inner and outer radius for the recirculation inlet and outlet nozzles provided in proposed alternative IR-054.

PNPP is a BWR/6 plant and the value used in IR-054 reflects the nominal radius of 119 inches. In the shell course drawings, the radius is 120 inches, not including 3/16 inches of stainless steel cladding. Thus, the radius to the low alloy steel RPV vessel wall is 120.1875 inches, which can be rounded up to 120.2 inches and matches the value provided in BWRVIP-108. The RPV shell thickness is 7 3/16 inches, or 7.19 inches, including the 3/16-inch stainless steel cladding, or 7 inches without the cladding. PNPP does have a shell course that is only 6 9/32 inches thick, but no shell courses that are exactly 6 inches thick. The shell course with N1 and N2 nozzles is 7 inches. The 6-inch shell thickness listed for PNPP in BWRVIP-108, Table 3-1, is assumed to be a rounded down value.

Using the more conservative values for RPV inner radius (120.2 inches) and RPV wall thickness (7 inches), the affected calculations for IR-054, Revision 1, are as follows:

The applicability of the BWRVIP-108 report to the PNPP is demonstrated by showing the criteria within Section 5 of the SE are met. The generic terms to be used in the SE Section 5 applicability evaluations are:

C_{RPV} = recirculation inlet or outlet nozzles (from BWRVIP-108 model)
$\underline{C_{RPV} = 19332 \text{ psi}}$ (recirculation inlet nozzles)
$\underline{C_{RPV} = 16171 \text{ psi}}$ (recirculation outlet nozzles)

The PNPP-specific terms to be used in the SE Section 5 applicability evaluations are:

p = reactor pressure vessel (RPV) normal operating pressure, <u>p = 1045 psig</u>
(maximum reactor steam dome pressure per Technical Specification 3.4.12)
r = RPV inner radius, <u>r = 120.2"</u>
t = RPV wall thickness, <u>t = 7"</u>

The updated calculations are:

(2) Recirculation Inlet (N2) Nozzles

Equation to meet criterion: $(pr/t)/C_{RPV} < 1.15$

$(1045 \times 120.2 \div 7) \div 19332 < 1.15$

The PNPP result is 0.93, which is less than 1.15.

(4) Recirculation Outlet (N1) Nozzles

Equation to meet criterion: $(pr/t)/C_{RPV} < 1.15$

$(1045 \times 120.2 \div 7) \div 16171 < 1.15$

The PNPP result is 1.11, which is less than 1.15.

Conformance with the criteria within Section 5 of the NRC SE for BWRVIP-108 is met.

The nozzle inner and outer radius for the recirculation inlet and outlet nozzles provided in IR-054 were obtained from the nozzle forging drawings and are correct values. The following values remain unchanged:

- inner radius for recirculation outlet (N1) nozzles is 10 inches
- inner radius for recirculation inlet (N2) nozzles is 5.813 inches
- outer radius for recirculation outlet (N1) nozzles is 17.594 inches
- outer radius for recirculation inlet (N2) nozzles is 11.125 inches

5. **Proposed Alternative IR-056, Revision 1, ASME Code, Section XI, Examination Category B-N-1, Item B13.10, Interior of Reactor Vessel, and Examination Category B-N-2, Item B13.40, Welded Core Support Structures and Interior Attachments to Reactor Vessels**

   The licensee is requested to withdraw any proposed alternative in IR-056 that applies the requirements of BWRVIP-183, "Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines." BWRVIP-183 is currently under review by the NRC staff, and it would not be appropriate to consider this portion of the alternative at this time.

FENOC agrees to withdraw the proposed alternative applicable to the top guide grid, which applies the requirements of BWRVIP-183. An updated copy of IR-056, Revision 1, with the top guide grid information removed, is enclosed to assist in the review process. Revision bars were used to designate the areas of change.

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. American Society of Mechanical Engineers (ASME) Code Components Affected

Core Support Structure Components	Code Class
Reactor Vessel Interior	1
Shroud Support Plate	1
Shroud Support Legs	1
Shroud Horizontal Welds	1
Shroud Vertical Welds	1
Shroud Repairs	1
Top Guide	1
Core Support Plate	1
Control Rod Guide Tubes	1

2. Applicable Code Edition and Addenda

ASME Section XI, 2001 Edition through the 2003 Addenda

3. Applicable Code Requirement

Table IWB-2500-1, Examination Category B-N-1, Item No. B13.10 requires accessible areas of the reactor vessel interior to be examined each inspection period by the visual, VT-3 method. Examination Category B-N-2, Item No. B13.40 requires accessible surfaces of the core support structure to the reactor vessel to be examined by the visual, VT-3 method each interval.

4. Reason for Request

FENOC requests to use the Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines, endorsed by the Nuclear Regulatory Commission (NRC) and implemented by the industry, to perform examinations in accordance with industry initiatives because Code inspection requirements have not evolved with boiling water reactor (BWR) inspection experience.

5. Proposed Alternative and Basis for Use

In lieu of the ASME Section XI examination requirements, FENOC proposes to perform examinations pursuant to the requirements within the identified BWRVIP guidelines.

The BWRVIP Inspection and Evaluation (I&E) guidelines have recommended aggressive specific inspection by Boiling Water Reactor (BWR) operators to identify material condition issues with BWR components. A wealth of inspection data has been gathered during these inspections across the BWR industry. The I&E guidelines focus on specific and susceptible components, specify appropriate inspection methods capable of identifying real anticipated degradation mechanisms, and require re-examination at conservative intervals. In contrast, the Code inspection requirements were prepared before the BWRVIP initiative and have not evolved with BWR inspection experience.

Not all the components addressed by these guidelines are Code components. The guidelines applicable to the subject Code components are:

BWRVIP-03, "Reactor Pressure Vessel and Internals Examination Guidelines"
BWRVIP-18-A, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines"
BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines"
BWRVIP-26-A, "BWR Top Guide Inspection and Flaw Evaluation Guidelines"
BWRVIP-27-A, "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines"
BWRVIP-38, "BWR Shroud Support Inspection and Flaw Evaluation Guidelines"
BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"
BWRVIP-42-A, "LPCI Coupling Inspection and Flaw Evaluation Guidelines"
BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
BWRVIP-48-A, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"
BWRVIP-76, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines" (see Note 1)
BWRVIP-100-A, "Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds"

Note 1: If flaw evaluations are required for BWRVIP-76 examinations, the fracture toughness values of BWRVIP-100-A will be utilized.

Table 1 compares current ASME Examination Category B-N-1 and B-N-2 requirements with the current BWRVIP guideline requirements, as applicable to the Perry Nuclear Power Plant (PNPP). Table 2 provides the inspection history for the PNPP reactor core support structures.

Any deviations from the referenced BWRVIP guidelines for the duration of the proposed alternative will be appropriately documented and communicated to the NRC, per the BWRVIP Deviation Disposition Process. Currently, the PNPP does not have any deviations from the BWRVIP guidelines.

As part of Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Material Issues," BWRs are required to examine reactor internals in accordance with BWRVIP guidelines. These guidelines have been written to address the safety significant vessel internal components and to examine and evaluate the examination results for these components using appropriate methods and re-examination frequencies. The BWRVIP has established a reporting protocol for examination results and deviations. The NRC has agreed with the BWRVIP approach in principle and has issued Safety Evaluations for these guidelines (References 1 – 12). Therefore, use of these guidelines as an alternative to the subject Code requirements provide an acceptable level of quality and safety and will not adversely impact the health and safety of the public.

The Attachment, "Comparison of Code Examination Requirements to BWRVIP Examination Requirements," identifies specific examples that compare the inspection requirements of Table IWB-2500-1, Item Nos. B13.10 and B13.40, to the inspection requirements in the BWRVIP documents. Specific BWRVIP documents are cited as examples. This comparison also includes a discussion of the inspection methods. These comparisons demonstrate that use of these guidelines, as an alternative to the subject Code requirements, provides an acceptable level of quality and safety and will not adversely impact the health and safety of the public.

6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019.

7. Precedent

NRC letter to FirstEnergy Nuclear Operating Company, December 16, 2008, Subject: Perry Nuclear Power Plant, Unit No. 1 – Request for Relief Related to Inservice Inspection Relief Requests Nos. IR-056 and IR-057 (TAC Nos. MD8198 and MD8199).

NRC letter to Entergy Nuclear Operations, September 19, 2005, Subject: Safety Evaluation of Relief Request RI-01, Vermont Yankee Nuclear Power Station (TAC No. MC0690).

8. References

1. NRC letter to BWRVIP, June 30, 2008, Subject: Safety Evaluation for Electric Power Research Institute (EPRI) Boiling Water Reactor Vessel and Internals Project (BWRVIP) Report TR-105696-R6 (BWRVIP-03), Revision 6, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines" (TAC No. MC2293).

2. NRC letter to BWRVIP, September 6, 2005, Subject: NRC Approval Letter of BWRVIP-18-A, "BWR Vessel and Internals Project Boiling Water Reactor Core Spray Internals Inspection and Flaw Evaluation Guidelines" (Accession No. ML052490002).

3. NRC letter to BWRVIP, December 19, 1999, Subject: Final Safety Evaluation of BWRVIP Vessel and Internals Project, "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25)," EPRI Report TR-107284, December 1996 (TAC No. M97802).

4. NRC letter to BWRVIP, September 9, 2005, Subject: NRC Approval Letter of BWRVIP-26-A, "BWR Vessel and Internals Project Boiling Water Reactor Top Guide Inspection and Flaw Evaluation Guidelines" (Accession No. ML052490550)

5. NRC letter to BWRVIP, June 10, 2004, Subject: Proprietary Version of NRC Staff Review of BWRVIP-27-A, "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines."

6. NRC letter to BWRVIP, July 24, 2000, Subject: Final Safety Evaluation of the "BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)," EPRI Report TR-108823 (TAC No. M99638).

7. NRC letter to BWRVIP, February 4, 2001, Subject: Final Safety Evaluation of the "BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)," (TAC No. M99870).

8. NRC letter to BWRVIP, September 9, 2005, Subject: NRC Approval Letter of BWRVIP-42-A, "BWR Vessel and Internals Project Boiling Water Reactor Low Pressure Coolant Injection and Flaw Evaluation Guidelines" (Accession No. ML052490557).

9. NRC letter to BWRVIP, September 9, 2005, Subject: NRC Approval Letter of BWRVIP-47-A, "BWR Vessel and Internals Project Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines" (Accession No. ML052490537).

10. NRC letter to BWRVIP, July 25, 2005, Subject: NRC Approval Letter of BWRVIP-48-A, "BWR Vessel and Internals Project Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines" (Accession No. ML052130284).

11. NRC letter to BWRVIP, July 27, 2006, Subject: Safety Evaluation of Proprietary EPRI Report , "BWR Vessel and Internals Project, BWR Core Shroud and Inspection and Flaw Evaluation Guidelines (BWRVIP-76)."

12. NRC letter to BWRVIP, November 1, 2007, Subject: NRC Approval Letter with Comment for BWRVIP-100-A, "BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds" (Accession No. ML073050135).

TABLE 1 – Page 1 of 2

Comparison of ASME Examination Category B-N-1 and B-N-2 Requirements With BWRVIP Guidance Requirements for BWR/6 (1)

ASME Item No. Table IWB-2500-1	Core Support Structure Components	ASME Exam Scope	ASME Exam	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
B13.10	Reactor Vessel Interior	Accessible Areas (Non-specific)	VT-3	Each period	BWRVIP-18-A, 26-A, 38, 41, 42-A, 47-A, 48-A, 76	Overview examinations of components during BWRVIP examinations are performed to satisfy Code VT-3 inspection requirements.		
B13.40	Shroud Support Plate	Accessible Surfaces	VT-3	Each 10-year Interval	BWRVIP-38, 3.2.2, Figures 3-4, 3-5	Welds H8 and H9(2)	EVT-1 or UT	Based on as-found conditions, to a maximum 6 years for one side EVT-1, 10 years for UT
	Shroud Support Legs	Accessible Surfaces (beneath core plate; rarely accessible)			BWRVIP-38, 3.2.3	Welds H10, H11 and H12	Per BWRVIP-38 NRC SER (7/24/00), inspect with appropriate method(4)	When accessible

TABLE 1 (continued) – Page 2 of 2

Comparison of ASME Examination Category B-N-1 and B-N-2 Requirements With BWRVIP Guidance Requirements for BWR/6 (1)

ASME Item No. Table IWB-2500-1	Core Support Structure Components	ASME Exam Scope	ASME Exam	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
B13.40	Shroud Horizontal welds	Accessible Surfaces	VT-3	Each 10-year Interval	BWRVIP-76, 2.2 Figure 2-2(3)	Welds H1-H7 as applicable	EVT-1 or UT	Based on as-found conditions, to a maximum 6 years for one side EVT-1, 10 years for UT
	Shroud Vertical welds				BWRVIP-76, 2.3, 3-3, Figures 2-4, 3-2, 3-3	Vertical and Ring Segment Welds	EVT-1 or UT	Maximum 6 years for one-sided EVT-1, 10 years for UT; only required when horizontal welds are found to contain flaws exceeding certain limits or the shroud is a repaired shroud
	Shroud Repairs (3)				BWRVIP-76, 3.5, 3.6	Tie-Rod Repair	VT-3	Per repair designer recommendations per BWRVIP-76
	Top Guide				BWRVIP-26-A 3.2 Table 3-2	Top Guide Studs	VT-3	Each 10-year Interval
	Core Support Plate				BWRVIP-25 3.2 Table 3.2	None for BWR/6	N/A	N/A
	Control Rod Guide Tubes (CRGTs)				BWRVIP-47-A 3.2 Table 3.3	CRGT Body Welds and Fuel Support Pins and Lugs	EVT-1 of body welds and VT-3 of pins and lugs	10% of the CRGT Assemblies within 12 years

NOTES:

1) This Table provides an overview of the requirements. For more details, refer to ASME Section XI, Table IWB-2500-1, and the appropriate BWRVIP document.
2) For Perry, this results in a requirement of 10 percent of the weld length. However, for H9 essentially 100 percent of the weld length was ultrasonically examined.
3) Perry's shroud is a Category B un-repaired shroud.
4) When inspection tooling and methodologies are available, they will be utilized to establish a baseline inspection of these welds. Until such time, and as committed to in BWRVIP-47-A, Section 3.2.5, visual inspections of the lower plenum area (which includes the shroud support legs) will be performed to the extent practical when access is made available through non-routine refueling outage activities (for example, jet pump disassembly).

TABLE 2 - Page 1 of 3
Perry Nuclear Power Plant
Reactor Core Support Structures Inspection History

Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summary of Inspection Results, Repairs, Replacements, Re-inspections
Reactor Vessel Interior (BWRVIP-48-A)	1989 (RF1)	VT-3	In Refueling Outage (RF) 1, VT-3 of 80% of total reactor pressure vessel (RPV) area from the shroud support plate to the flange. The remaining 20% was inaccessible due to the physical lay-out of the Jet Pump (JP) area. There were no relevant indications.
	1992 (RF3)	VT-3 & VT-1	In RF3, VT-3 of the accessible areas was performed along with a VT-1 exam of the vessel wall area near the Feedwater Sparger Spray nozzle ruptures (found during RF3, Nonconformance Report 92-S-045).
	1996 (RF5)	VT-3	In RF5, VT-3 of the accessible areas was performed. There were no revelant indications.
	1999 (RF7)	VT-3	In RF7, VT-3 of the accessible areas was performed. There were no revelant indications.
	2003 (RF9)	VT-3	In RF9, VT-3 of the top head interior was performed. Exam found unusual crud deposits on the upper (i.e., steam region) vessel inside diameter (ID) cladding. Under Condition Report (CR) 03-01995 the hard deposits were evaluated as acceptable for continued operation.
	2007 (RF11)	VT-3	In RF11, VT-3 of 100% of accessible areas above the top guide flange was performed. No indications beyond previously addressed RPV crud.
Core Shroud (BWRVIP-76)	1994 (RF4)	VT-3 & EVT-1	In RF4, VT-3 of entire shroud interior and EVT-1 of the H-3 and H-4 weld inside surfaces at 4 approx. 1-foot long sample locations. No indications.
	1997 (RF6)	VT-3	In RF6, a Code VT-3 exam was performed on all accessible shroud exterior areas. No indications.
	1999 (RF7)	UT	In RF7, UT examination of the H-3, H-4, H-6A and H-7 welds was performed in accordance with the Category B Plant guidelines of BWRVIP-01. No indications.
	2005 (RF10)	UT	In RF10, UT exams of the H-3 and H-4 welds with the Tecnatom ID tool and H-6A and H-7 with the GE OD Tracker. H-4 and H-6A were two sided exams and H-3 and H-7 were one-sided exams. Shallow cracking was found in H-7. It was less than 10% of the inspected length of 67% of the weld and evaluated as acceptable per BWRVIP-76.

TABLE 2 (continued) - Page 2 of 3
Perry Nuclear Power Plant
Reactor Core Support Structures Inspection History

Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summary of Inspection Results, Repairs, Replacements, Re-inspections
Shroud Support (BWRVIP-38)	1990 (RF2)	VT-3 & VT-1	In RF2, VT-3 of shroud support plate and VT-1 of the shroud support plate access hole cover. No indications.
	1996 (RF5)	VT-3 & VT-1	In RF5, VT-3 of shroud support plate and VT-1 of the shroud support plate access hole cover. No indications.
	1999 (RF7)	EVT-1	In RF7, baseline EVT-1 exams of the H-8 and H-9 were performed in accordance with BWRVIP-38. No Indications.
	2001 (RF8)	VT-1	In RF8, re-seating of JP # 5 provided access to the H-10, H-11 and H-12 welds of the shroud support leg at 90° and approx. 10° of the underside of H-8 and H-9 so they were visually examined with at least VT-1 resolution. No indications.
	2007 (RF11)	EVT-1 & VT-1	In RF11, JP #6 was removed and re-seated due to excess leakage at the transition piece. While disassembled approx. 10° of the underside of H-8 and H-9 were examined with at least VT-1 resolution. Also, the H-10, H-11 and H-12 welds of the shroud support legs at 90° and 120° were examined with EVT-1 resolution. Coverage was approx. 35-50% for the welds of the 90° leg and 25% for the welds of the 120° leg. No indications.
Top Guide (Rim, and so forth) (BWRVIP-26-A)	1989 (RF1)	VT-3	Top Guide periphery, including 90 studs and tack welds, examined in RF1. No indications.
	1994 (RF4)	VT-3	Top Guide grid examined in RF4. No indications.
	1999 (RF7)	VT-1 & VT-3	In RF7, performed VT-3 of the Top Guide assembly in accordance with ASME Category B-N-2 and VT-1 of the studs and tack welds in accordance with BWRVIP-26. No indications.
	2005 (RF10)	VT-3	Code B-N-2 exam of accessible portions of Top Guide grid. Due to ID Core Shroud exams, a significant number of the grid cells were vacated and accessible for inspection. No indications.
Core Plate (Rim, and so forth) (BWRVIP-25; not applicable to BWR/6s)	1989 (RF1)	VT-3	Accessible core plate areas and fuel support castings examined in RF1. No indications.
	1994 (RF4)	VT-3	All of the hold down bolts examined from shroud interior in RF4. No indications.
	1999 (RF7)	VT-3	In RF7, performed VT-3 exam of the core plate areas made accessible by replacement of 5 Control Rod blades in accordance with ASME Category B-N-2. No indications.

TABLE 2 (continued) - Page 3 of 3
Perry Nuclear Power Plant
Reactor Core Support Structures Inspection History

Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summary of Inspection Results, Repairs, Replacements, Re-inspections
CRD Guide Tube (BWRVIP-47-A)	1999 (RF7)	VT-1 & EVT-1	In RF7, performed VT-1 of alignment pins and EVT-1 of the welds of 5 Control Rod Guide Tubes in accordance with BWRVIP-47. No indications.
	2001 (RF8)	VT-1 & EVT-1	In RF8, performed VT-1 of alignment pins and EVT-1 of the welds of an additional 4 Control Rod Guide Tubes in accordance with BWRVIP-47 to meet the 5% completion requirements of BWRVIP-47. No indications.
	2005 (RF10)	VT-1 & EVT-1	In RF10, performed VT-1 of alignment pins and EVT-1 of the welds of an additional 5 Control Rod Guide Tubes in accordance with BWRVIP-47. No indications.
	2007 (RF11)	VT-1 & EVT-1	In RF11, performed VT-1 of alignment pins and EVT-1 of the welds of an additional 4 Control Rod Guide Tubes in accordance with BWRVIP-47 to meet the 10% completion (i.e., 18 out of 177) requirements of BWRVIP-47. No indications.
Access Hole Cover (AHC) (BWRVIP-180)	1996 (RF5)	VT-1	VT-1 examination of the access hole cover welds in accordance with SIL-409. No indications.
	2007 (RF11)	EVT-1	EVT-1 examination of the access hole cover welds in accordance with the draft BWRVIP AHC Inspection and Evaluation Guidelines. No indications.

COMPARISON OF CODE EXAMINATION REQUIREMENTS TO BWRVIP EXAMINATION REQUIREMENTS

The following discussion provides a comparison of the examination requirements provided in ASME Section XI, Examination Table IWB-2500-1, Item Nos. B13.10, and B13.40, to the examination requirements in the BWRVIP guidelines. Specific BWRVIP guidelines are cited as examples for comparisons. This comparison also includes a discussion of the examination methods.

Code Requirement - B13.10 - Reactor Vessel Interior Accessible Areas (B-N-1)

The ASME Section XI Code requires a VT-3 examination of reactor vessel accessible areas, which are defined as the spaces above and below the core made accessible during normal refueling outages. The frequency of these examinations is specified as the first refueling outage, and at intervals of approximately three years, during the first inspection interval, and each period during each successive 10-year inspection interval. Typically, these examinations are performed every other refueling outage of the inspection interval. This examination requirement is a non-specific requirement that is a departure from the traditional Section XI examinations of welds and surfaces. As such, this requirement has been interpreted and satisfied differently across the industry. The purpose of the examination is to identify relevant conditions such as: distortion or displacement of parts; loose, missing, or fractured fasteners; foreign material, corrosion, erosion, or accumulation of corrosion products; wear; and structural degradation.

Portions of the various examinations required by the applicable BWRVIP guidelines require access to accessible areas of the reactor vessel during each refueling outage. Examination of core spray piping and spargers (BWRVIP-18-A), top guide (BWRVIP-26-A), jet pump welds and components (BWRVIP-41), interior attachments (BWRVIP-48-A), core shroud welds (BWRVIP-76), shroud support (BWRVIP-38), low pressure coolant injection couplings (BWRVIP-42-A), and lower plenum components (BWRVIP-47-A) provides such access. Locating and examining specific welds and components within the reactor vessel areas above, below (if accessible), and surrounding the core (annulus area) entails access by remote camera systems that essentially perform equivalent VT-3 examination of these areas or spaces as the specific weld or component examinations are performed. This provides an equivalent method of visual examination on a more frequent basis than that required by the ASME Section XI Code. Evidence of wear, structural degradation, loose, missing, or displaced parts, foreign materials, and corrosion product buildup can be, and has been observed during the course of implementing these BWRVIP examination requirements. Therefore, the specified BWRVIP guideline requirements meet or exceed the subject Code requirements for examination method and frequency of the interior of the reactor vessel. Accordingly, these BWRVIP examination requirements provide an acceptable level of quality and safety as compared to the subject Code requirements.

Code Requirement - B13.40 - Core Support Structure (B-N-2)

The ASME Code requires a VT-3 examination of accessible surfaces of the integrally welded core support structure each 10-year interval. In a BWR/6 boiling water reactor, the welded core support structure has primarily been considered the shroud itself and the shroud support structure, including the shroud support plate (annulus floor) the shroud support ring, the shroud support welds, and the shroud support legs (if accessible). Historically, this requirement has been interpreted and satisfied differently across the industry. Category B-N-2 is titled, "Integrally Welded Core Support Structures and Interior Attachments to Reactor Vessels." However, since the title for Item No. B13.40 simply states, "Core Support Structure," some plants, including Perry, have also applied the examination requirements to other core support structures such as the control rod guide tubes, core plate and top guide assembly. The proposed alternate examinations replace this ASME requirement with specific BWRVIP guidelines that examine susceptible locations for known relevant degradation mechanisms.

- The Code requires a VT-3 of accessible surfaces each 10-year interval.

- The BWRVIP requires, as a minimum, the same examination method (VT-3) as the Code for integrally welded core support structures, and for specific areas, it requires either an enhanced visual examination technique (EVT-1) or ultrasonic examination (UT).

BWRVIP recommended examinations of core support structures are focused on the known susceptible areas of this structure, including the welds and associated weld heat affected zones. As a minimum, the same or superior visual examination technique is required for examination at the same frequency as the Code examination requirements. In many locations, the BWRVIP guidelines require a volumetric examination of the susceptible welds at a frequency identical to the Code requirement.

The BWRVIP guidelines require an EVT-1 or UT of core support structures. The core shroud and shroud support plate are used as examples for comparison between the Code and BWRVIP examination requirements as shown below.

Comparison to BWRVIP Requirements - BWR Core Shroud Examination and Flaw Evaluation Guidelines (BWRVIP-76)

- The Code requires a VT-3 examination of accessible surfaces every 10 years.

- BWRVIP-76 requires an EVT-1 examination from the inside and outside surface, where accessible, or UT examination of select circumferential welds that have not been structurally replaced with a shroud repair, at a calculated "end of interval" that will vary depending upon the amount of flaws present, but not to exceed 10 years.

Comparison to BWRVIP Requirements - BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)

- The Code requires a VT-3 examination of accessible surfaces every 10 years.

- The BWRVIP requires examinations of the support plate to shroud weld (H8) and support plate to reactor vessel weld (H9). Examination coverage is required to be (100 percent - Flaw Tolerance) or 10 percent of the weld length, whichever is greater. Examinations are to be performed by EVT-1 or UT from the annulus or UT from the RPV outside surface. Reinspection depends upon the amount of flaws present, but not to exceed six years for EVT-1 or 10 years for UT.

In summary, the BWRVIP recommended examinations specify locations that are known to be vulnerable to BWR relevant degradation mechanisms rather than "all surfaces." The BWRVIP examination methods (EVT-1 or UT) are superior to the Code required VT-3 for flaw detection and characterization. The BWRVIP examination frequency is equivalent to or more frequent than the examination frequency required by the Code. The superior flaw detection and characterization capability, with an equivalent or more frequent examination frequency and the comparable flaw evaluation criteria, results in the BWRVIP criteria providing a level of quality and safety equivalent to or superior to that provided by the Code requirements.