AmerGen
AmerGen Energy Company, LLC
200 Exelon Way
Suite 345
Kennett Square, PA 19348
www.exeloncorp.com
An Exelon/British Energy Company

July 2, 2001
5928-01-20188

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

SUBJECT: THREE MILE ISLAND, UNIT 1 (TMI UNIT 1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
ASME SECTION XI RELIEF REQUESTS ASSOCIATED WITH REACTOR VESSEL HEAD REPAIR

Dear Sir or Madam:

During the upcoming TMI Unit 1 Cycle 14 Refueling Outage (1R14), which is currently scheduled for October 5, 2001, visual inspections for leakage/boric acid deposits of all Reactor Vessel Head Thermocouple (TC) nozzles and Control Rod Drive Mechanism (CRDM) nozzles will be conducted. Nozzles showing evidence of leakage will be repaired.

In order to conduct the repairs efficiently and insure personnel exposure is kept to a minimum, relief requests from portions of the ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda must be approved. Pursuant to 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(a)(3)(ii), attached for your review are proposed relief requests (RR-01-14 through RR-01-17).

The repair plans seek to significantly reduce exposures by instituting machine remote processes for CRDM repair similar to that used at the Oconee Nuclear Station – Unit 2. The repair process for the CRDM is provided for information in Enclosure 1 of the attachments. Based on the Oconee Nuclear Station – Unit 2 experience of repairing manually verses repairing with machine remote processes, it is estimated that at TMI Unit 1, a significant radiological dose savings of 11 REM will be realized for each CRDM nozzle repaired. There are 69 CRDM nozzles and 8 TC nozzles on the TMI Unit 1 reactor vessel head. If repairs are necessary, approval of these requests are required in order to realize the above radiological dose savings.

A047

Approval of these requests will allow repairs to TC and CRDM nozzles utilizing alternatives to the use of material, welding processes and examinations requirements of several ASME Code sections for the repair of Class A Reactor Vessel head components. The relief requests have been evaluated and determined that the alternatives described in each relief request provide an acceptable level of quality and safety. Thermocouple nozzle repairs, if required, will be performed manually, with the majority of the work being performed from the top side of the Reactor Vessel head, such that worker dose is minimized. The repair will consist of either nozzle plugging or nozzle sleeving, depending on replacement of individual nozzle functional requirements.

Attachment A and its enclosures contain information proprietary to Framatome ANP (FRA-ANP). The proprietary information is enclosed by brackets "[ ]". An affidavit from FRA-ANP is included which sets forth the basis on which the information may be withheld from public disclosure by the NRC pursuant to 10 CFR 2.790. Attachment B provides a non-proprietary version of this request. A table of contents has been provided which identifies all documents.

We request that this relief be approved by August 31, 2001, for use, if necessary, in the upcoming TMI Unit 1 Cycle 14 Refueling Outage (1R14) which is currently scheduled for October 5, 2001.

Very truly yours,

James A. Hutton
Director - Licensing
Mid-Atlantic Regional Operating Group

Attachment: A) Proprietary Version
B) Non-Proprietary Version

cc: H. J. Miller, USNRC, Regional Administrator, Region I
T. G. Colburn, USNRC, Senior Project Manager, TMI Unit 1
J. D. Orr, USNRC, Senior Resident Inspector, TMI Unit 1
File No. 01055

Table of Contents

TMI UNIT 1
ASME Section XI Relief Requests Associated With Reactor Vessel Head Repair

Attachment A—(Proprietary Version)

Framatome ANP Affidavit	
Enclosure 1	Machine Repair Process
Figure 1	CRDM Repair Configuration
Relief Request RR-01-14	Use of Alloy 690 (Alloy 52/152) Based Weld Material
Relief Request RR-01-15	Eliminate 5 Inch Band Examination
Relief Request RR-01-16	Eliminate 48 Hour Hold Time
Attachment 1	PQ7109-00
Attachment 2	PQ7153-00
Relief Request RR-01-17	Eliminate Monitoring of InterpassTemperature
Attachment 1	Weld Interpass Temperature Evaluation

Attachment B—(Non-Proprietary Version)

Enclosure 1	Machine Repair Process
Figure 1	CRDM Repair Configuration
Relief Request RR-01-14	Use of Alloy 690 (Alloy 52/152) Based Weld Material
Relief Request RR-01-15	Eliminate 5 Inch Band Examination
Relief Request RR-01-16	Eliminate 48 Hour Hold Time
Attachment 1	PQ7109-00
Attachment 2	PQ7153-00
Relief Request RR-01-17	Eliminate Monitoring of Interpass Temperature
Attachment 1	Weld Interpass Temperature Evaluation

ATTACHMENT B
(Non-Proprietary Version)

TMI UNIT 1
ASME Section XI Relief Requests Associated
With Reactor Vessel Head Repair

Enclosure 1

Machine Repair Process

Visual inspections for leakage/boric acid deposits of all Reactor Vessel Head Thermocouple (TC) nozzles and Control Rod Drive Mechanism (CRDM) nozzles will be conducted. Nozzles showing evidence of leakage will be repaired. The repair plans seek to significantly reduce exposures for CRDM repair by instituting machine remote processes similar to those used at the Oconee Nuclear Station – Unit 2.

Thermocouple nozzle repairs, if required, will be performed manually, with the majority of the work being performed from the top side of the Reactor Vessel head, such that worker dose is minimized. The repair will consist of either nozzle plugging or nozzle sleeving, depending on replacement of individual nozzle functional requirements.

A description of the CRDM machine repair process is as follows:

<u>Control Rod Drive Mechanism Nozzle Repair:</u>

[

]

The CRDM nozzle repair configuration is illustrated in Figure 1.

Figure 1

TMI Unit 1 CRDM Machining

Three Mile Island Unit 1
Request for Relief RR-01-14
"Use of Alloy 690 Based Weld Filler Material"

COMPONENT IDENTIFICATION

Code Class:	Class 1
Reference:	ASME, Section XI; 1995 Edition through 1996 Addenda
Examination Categories:	B-O (Section XI)
Item Number:	B14.10 (Section XI)
Description:	Use of Alloy 690 (Alloy 52/152) welding filler material
Component Numbers	All as necessary

CODE REQUIREMENTS FROM WHICH AN ALTERNATIVE IS REQUESTED

The Code to be utilized for the repairs to Thermocouple (TC) nozzles and Control Rod Drive Mechanism (CRDM) nozzles is ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda.

ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda allows, by reference, the use of Alloy 600 based weld filler material (Alloy 82/182) but does not include the use of Alloy 690 (Alloy 52/152) based weld filler material (Alloy 52/152).

BASIS FOR ALTERNATIVE

Alloy 690 (Alloy 52/152) weld materials possess a high resistance to primary water stress corrosion.

Alloy 690 (Alloy 52/152) material has been shown to be superior to Alloy 600 (Alloy 82/182) material in resisting Primary Water Stress Corrosion Cracking (PWSCC). The NRC approved the use of Alloy 690 (Alloy 52/152) material in the construction of the replacement steam generators to be installed at Oconee, McGuire Nuclear Station Units 1 & 2, and Catawba Nuclear Station Unit 1. The NRC also approved requests for use of Alloy 690 (Alloy 52/152) material in the repairs of the Oconee Unit 1 thermocouple and CRDM nozzles, and the Unit 3 CRDM nozzles. At TMI-1, Alloy 690 (Alloy 52/152) has been approved for use for steam generator plugs and sleeves.

Three Mile Island Unit 1
Request for Relief RR-01-14
"Use of Alloy 690 Based Weld Filler Material"

ASME Code Cases 2142-1 and 2143-1 establish the uniform chemical and material properties and the classification of the weld material with respect to its welding characteristics. Code Case 2142-1 establishes the F-No. for the American Welding Society (AWS) specification AWS A5.14 and Unified Numbering System (UNS) designation UNS N06052 (Alloy 52) as F-No. 43 for both procedure and performance qualification purposes. Code Case 2143-1 establishes the F-No. for AWS A5.11 and UNS designation W86152 (Alloy 152) for a coated electrode as F-No. 43 for procedure and performance qualification purposes. These sets of specifications and F-No. assignments completely describe this material for welding purposes as similar in their welding characteristics to other Code approved nickel based weld metals.

In conclusion, the use of Alloy 690 (Alloy 52/152) welding filler material (Alloy 52/152) and the associated ASME Code Cases 2142-1 and 2143-1 for the repairs to TMI-1 TC and CRDM nozzles will provide superior corrosion protection over that provided by Alloy 600 (Alloy 82/182) material. The use of Alloy 690 (Alloy 52/152) has been previously authorized for new construction and other repair activities. Therefore, the proposed alternative provides an acceptable level of quality and safety.

PROPOSED ALTERNATIVE PROVISIONS

In lieu of the Code requirement, the use of Alloy 690 (Alloy 52/152) weld filler material is proposed for the repair of all TC nozzles and CRDM nozzles located on the TMI-1 Reactor Vessel (RV) head.

Pursuant to 10 CFR 50.55a(a)(3)(i), an alternative is requested on the basis that the proposed alternative provides an acceptable level of quality and safety.

ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda allows, by reference, the use of Alloy 600 based weld filler material (Alloy 82/182) but does not include the use of Alloy 690 (Alloy 52/152) based weld filler material (Alloy 52/152).

Code cases 2142-1 and 2143-1 introduce and classify new nickel based weld metals that closely match Alloy 690 (Alloy 52/152). Code Case 2142-1 establishes welding classifications and other requirements for bare wire filler metal (UNS N06052 Ni-Cr-Fe). Code Case 2143-1 establishes welding classifications and other requirements for a coated electrode (UNS W86152 Ni-Cr-Fe). These two Code cases have not been incorporated by reference into the regulations; therefore, their use requires NRC approval.

Three Mile Island Unit 1
Request for Relief RR-01-15
"Eliminate 5 Inch Band Examination"

COMPONENT IDENTIFICATION

Code Class:	Class 1
Reference:	ASME, Section XI; 1995 Edition through 1996 Addenda
Examination Categories:	B-O (Section XI)
Item Number:	B14.10 (Section XI)
Description:	Examination Requirements
Component Numbers	All as necessary

CODE REQUIREMENTS FROM WHICH AN ALTERNATIVE IS REQUESTED

The Code to be utilized for the repairs to the Control Rod Drive Mechanism (CRDM) nozzles is ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda.

IWA-4634 of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda states, "The weld as well as the preheated band shall be examined by the liquid penetrant method after the completed weld has been at ambient temperature for at least 48 hours. The weld shall be volumetrically examined."

The first sentence of IWA-4610(a) of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda states, "The area to be welded plus a band around the area of at least 1-1/2 times the component thickness or 5 in., whichever is less, shall be preheated and maintained at a minimum temperature of 350 degrees F for the SMAW process and 300 degrees F for the GTAW process during welding."

BASIS FOR ELIMINATION

The configuration of the new pressure boundary welds limits the ability to examine the band area as defined by IWA-4610(a). For the repairs, the GTAW process will be utilized. Due to the thickness of the RV head, the 5-inch minimum is utilized for definition of the band area.

[

Three Mile Island Unit 1
Request for Relief RR-01-15
"Eliminate 5 Inch Band Examination"

] the surfaces of the new pressure boundary weld receive both the PT and UT inspections. [

]

Remote enhanced video will be used during the welding operation to insure welding quality. The combination of the PT and UT examinations on the weld surfaces and weld procedure qualification will provide an acceptable level of quality and safety.

Compliance with the post-repair examination areas to the requirements of IWA-4634 of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda presents a [

]

PROPOSED ALTERNATIVE PROVISIONS

Pursuant to 10 CFR 50.55a(a)(3)(ii), Compliance with the requirements of IWA-4634 and IWA-4610(a) of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. An alternative is requested on the basis that the proposed alternative provides an acceptable level of quality and safety.

Due to the unique geometry of the CRDM [ ] Reactor Vessel (RV) head repair, it is not practical to inspect the band area defined by IWA-4610(a). [

] Post-repair inspections of the repaired areas will be done by a combination of remote and manual methods. In lieu of inspecting the Code required band area, it is proposed that the repair area be inspected by PT and UT.

Three Mile Island Unit 1
Request for Relief RR-01-16
"Eliminate 48 Hour Hold Time"

COMPONENT IDENTIFICATION

Code Class:	Class 1
Reference:	ASME, Section XI; 1995 Edition through 1996 Addenda
Examination Categories:	B-O (Section XI)
Item Number:	B14.10 (Section XI)
Description:	Examination Requirements
Component Numbers	All as necessary

CODE REQUIREMENTS FROM WHICH AN ALTERNATIVE IS REQUESTED

The Code to be utilized for the repairs to the Control Rod Drive Mechanism (CRDM) nozzles is ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda.

IWA-4634 of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda states, "The weld as well as the preheated band shall be examined by the liquid penetrant method after the completed weld has been at ambient temperature for at least 48 hours. The weld shall be volumetrically examined."

BASIS FOR ELIMINATION

IWA-4634 of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda specifies that the weld region shall undergo volumetric examination after the weld repair area has been at ambient temperature for a minimum of 48 hours. The 48-hour hold is specified to assure that no delayed cold cracking in the ferritic steel HAZ has occurred. The weld consumables to be used in the new pressure boundary weld consist of bare wire with no flux. The welding will be performed at 300 degrees F minimum preheat temperature using the GTAW process, as required by IWA-4610(a).

The post-weld heat soak requirement of IWA-4633.2(d) is to assure that no delayed cold cracking in the ferritic steel HAZ occurs. The weld consumables to be used will consist of bare wire with no flux. The preheat temperature of 300 degrees F will be maintained during the post-weld soak for four hours. The combination of the low moisture absorbing GTAW weld process and maintaining the post-weld soak temperature at 300 degrees F for four hours will eliminate the possibility of hydrogen induced cracking.

Industry experience has found that delayed hydrogen cracking requires a hydrogen concentration above 5ml/100g of deposited weld metal, and a weld and Heat Affected Zone (HAZ) with low ductility/toughness. Delayed hydrogen cracking tends to occur in carbon and alloy steel welds produced by processes which use a flux, e.g. shielded metal arc welding (SMAW), submerged arc welding (SAW), and flux cored arc welding (FCAW). The flux in these processes can pick up moisture that breaks down during welding to produce atomic hydrogen. The atomic hydrogen is partially absorbed by the weld metal and HAZ. Absorption of hydrogen, in sufficient quantity in low ductility material, may cause delayed hydrogen cracking. The GTAW process uses Argon gas as the shielding medium.

Moisture contaminated shielding gas or high humidity environments may introduce hydrogen into GTAW welds. The Electric Power Research Institute (EPRI) performed tests (Reference 1) where argon shielding gas was bubbled through a cylinder of water and then mixed with welding grade argon having a dew point of –70 degrees F to produce gas mixtures with dew points from –60 degrees F to +60 degrees F. At +60 degrees F dew point (an unrealistically high dew point), the measured hydrogen concentration in test welds was 4.6 ml/100g of as deposited weld metal. This value falls in the extra low hydrogen range specified by American Welding Society (AWS). The EPRI study also measured the hydrogen content of bare filler material and found it to be less than 1 ml/100g of as deposited weld metal.

The EPRI work further showed that a 450 degrees F post-weld heat soak would reduce the already low hydrogen content to infinitesimally small values. Work by Coe and Moreton, as documented in table 2-6 of Reference 1 determined that it takes only 0.3 hours at 450 degrees F to remove 95% of any hydrogen present. At 300 degrees F, the diffusivity rate measurements showed that only 0.7 hours is required to remove 95% of any hydrogen that is present. The proposed alternative will hold the post-weld heat soak at 300 degrees F for four hours.

In addition to the compelling data promulgated in the EPRI report, FRA-ANP has qualified the GTAW temper-bead process in support of ASME approval of Code Case N-606-1, "Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique for BWR CRD Housing/Stub Tube Repairs" (Reference 2). The supporting welding PQR's for this work, PQ7109-00 (Reference 3) and PQ7153-00 (Reference 4), are given in Attachment 1 and 2 respectively. These qualifications were performed at room temperature with cooling water to limit the maximum interpass temperature to a maximum of 100 degrees F. These noted qualifications were performed on the same P-3 Group-3 base material as proposed for the CRDM repairs, using the same filler material, i.e. Alloy 52 AWS Class ERNiCrFe-7, with similar low heat input controls as will be used in the repairs. The qualifications did not include a post-weld heat soak. As noted above, the repairs described herein will be made to the

Three Mile Island Unit 1
Request for Relief RR-01-16
"Eliminate 48 Hour Hold Time"

ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda. However, Code Case N-432, Revision 1[1], deletes the 48-hour hold period requirement. In summary, the proposed elimination of the 48-hour period prior to performing NDE is based on the: 1) use of bare wire with no flux with the 300 degree F preheat such that delayed hydrogen induced cracking is eliminated; and, 2) the recently approved Code case that allows elimination of the 48-hour hold period. These items, as well as the weld procedure qualification provisions described above, assure an acceptable level of quality and safety.

Elimination of the 48-hour hold period meets the NRC's criteria for a hardship case per 10 CFR 50.55a(a)(3)(ii). Section IWA-4634 of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda requires a post-weld 48-hour hold period prior to performing the NDE required by Section XI. Compliance with the requirement for a post-weld 48-hour hold period prior to performing the NDE required by IWA-4634 would result in the addition of 2 days to the refueling outage schedule. The additional time and delay of plant startup will constitute unusual hardships and burdens that are not necessary considering that the NDE that could be performed in a shorter time period following the repair would provide an acceptable level of assurance of the quality and safety of the weld repairs. Any weld defects or cracking would be identified by the NDE performed before the 48-hour hold time. The proposed approach will provide assurance of the structural integrity of the CRDM nozzles as demonstrated by a Section III analysis of the new weld configuration, in addition to the above described low hydrogen producing welding process, weld procedure qualification, and NDE procedures and processes.

As previously described: (1) the purpose of the 48-hour hold period is to assure that no undetected delayed hydrogen induced cold cracking in the ferritic steel HAZ has occurred; and (2) the welding process used (GTAW) avoids delayed cold cracking. In recognition that the 48-hour hold period is an unnecessary hardship for temper bead weld repairs using the GTAW welding process with 300°F minimum preheat, Code Case N-432, Revision 1, has deleted the 48-hour hold period requirement of IWA-4634.

The purposes of the 48-hour hold period is not necessary when using the GTAW welding process with 300°F preheat. Accordingly, compliance with this requirement would not provide a compensating increase in the level of quality and safety. The quality and safety of the repair is not increased by the 48-hour hold period and, therefore the additional 2 day outage extension would be an unnecessary hardship without a compensating increase in the level of quality and safety.

---

[1] *Approved by the ASME Main Committee (Action No. ISI-99-34) on February 16, 2001.*

Three Mile Island Unit 1
Request for Relief RR-01-16
"Eliminate 48 Hour Hold Time"

PROPOSED ALTERNATIVE PROVISIONS

Pursuant to 10 CFR 50.55a(a)(3)(ii), Compliance with the specified requirements of IWA-4634 of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda would result in a hardship without a compensating increase in the level of quality and safety. An alternative is requested on the basis that the proposed alternative provides an acceptable level of quality and safety.

It is proposed that the 48-hour hold time be eliminated and that the volumetric and surface inspections be performed after the welds are completed and conditions have reached near ambient temperatures.

REFERENCES

1. Electric Power Research Institute (EPRI), Document TR103354, "Temperbead Welding Repair of Low Alloy Pressure Vessel Steels: Guidelines," December 1993, Section 2, "Diffusible Hydrogen in Low Alloy Steel Gas Tungsten-arc Welds" D. Gandy & S. Findlan.

2. ASME Code Case N-606-1, "Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper-Bead Technique for BWR CRD Housing/Stub Tube Repairs."

3. Framatome-ANP Welding Procedure Qualification Record PQ7109-00, dated February 23, 2000 (See Attachment 1)

4. Framatome-ANP Welding Procedure Qualification Record PQ7153-00, dated May 8, 2001 (See Attachment 2).

Attachment 1

PQ7109-00

[ ]

Attachment 2

PQ7153-00

[ ]

Three Mile Island Unit 1
Request for Relief RR-01-17
"Eliminate Monitoring of Interpass Temperature"

COMPONENT IDENTIFICATION

Code Class:	Class 1
Reference:	ASME, Section XI; 1995 Edition through 1996 Addenda
Examination Categories:	B-O (Section XI)
Item Number:	B14.10 (Section XI)
Description:	Examination Requirements
Component Numbers	All as necessary

CODE REQUIREMENTS FROM WHICH AN ALTERNATIVE IS REQUESTED

The Code to be utilized for the repairs to Control Rod Drive Mechanism (CRDM) nozzles is ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda.

IWA-4610(a) of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda states, "The area to be welded plus a band around the area of at least 1-1/2 times the component thickness or 5 in., whichever is less, shall be preheated and maintained at a minimum temperature of 350 degrees F for the SMAW process and 300 degrees F for the GTAW process during welding. The maximum interpass temperature shall be 450 degrees F. Thermocouples and recording instruments shall be used to monitor the process temperatures. Their attachment and removal shall be in accordance with Section III."

BASIS FOR ALTERNATIVE

Due to the difficulty in placing thermocouples adjacent to the new pressure boundary welds [ ] direct monitoring of the interpass temperature is a hardship and is physically impossible. [

] In lieu of monitoring the interpass temperature via adjacent

Three Mile Island Unit 1
Request for Relief RR-01-17
"Eliminate Monitoring of Interpass Temperature"

thermocouples, a calculation has been performed justifying the actual interpass temperature at the weld location based on a maximum allowable welding heat input; weld bead placement travel speed, and conservative preheat temperature assumptions. This calculation is provided as Attachment 1 to this relief request. The calculation supports the conclusion that using the maximum heat input through the third layer of the weld, the interpass temperature returns to the preheat temperature. Heat input beyond the third layer will not have a metallurgical affect on the low alloy steel HAZ.

The calculation is based on a typical inter-bead time interval of five minutes. The five minute inter-bead interval is based on the time:

3) required to explore the previous weld deposit with the two remote cameras housed in the weld head,
4) to shift the starting location of the next weld bead circumferentially away from the end of the previous weld-bead, and
3) [ ]

A welding mockup on the full size Midland RV closure head was used to demonstrate the welding technique described herein. During the mockup, thermocouples were placed to monitor the resistance heating of the head. These locations will be retained for the actual repairs. During the mockup, thermocouples were placed on the outside diameter of the RV head within a 5-inch band surrounding the CRDM nozzle. Three other thermocouples were placed on the RV head inner diameter. One of the three thermocouples was placed 1-1/2 inches from the CRDM penetration, on the lower hillside. The other inner diameter thermocouples were placed at the edge of the 5-inch band surrounding the CRDM nozzle, one on the lower hillside, the second on the upper hillside. During the mockup, all thermocouples fluctuated less than 15 degrees F throughout the 18-hour welding cycle. Based on past experience, it is believed that the temperature fluctuation was due more to the resistance heating variations than the low heat input from the welding process.

Controlling the parameters determined by the referenced calculation will assure that the maximum interpass temperature is not exceeded and thus provides an acceptable level of quality and safety.

The Code requirement for monitoring the weld interpass temperature is to prevent excessive local heat input into the low alloy steel. The control of the heat input; weld head travel speed, and the calculation of the dissipation of the heat into the large thermal mass of the RV head will insure that the heat input is prevented from exceeding Code limitations. The conclusions of the referenced calculation have been validated in

Three Mile Island Unit 1
Request for Relief RR-01-17
"Eliminate Monitoring of Interpass Temperature"

demonstrations on the Midland RV head. These measures will provide an acceptable level of quality and safety compared to Code requirements.

Compliance with IWA-4610(a) of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda constitutes a hardship per 10 CFR 50.55a(a)(3)(ii). It is physically impossible to locate thermocouples adjacent to the new pressure boundary weld region for the purposes of monitoring interpass temperature. The alternatives described will provide an acceptable level of quality and safety when compared to the Code requirements to directly monitor weld interpass temperature.

PROPOSED ALTERNATIVE PROVISIONS

Pursuant to 10 CFR 50.55a(a)(3)(ii), compliance with the requirement of IWA-4610(a) of ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition through 1996 Addenda would result in hardship without a compensating increase in the level of quality and safety. [ ] it is physically impossible to directly monitor interpass temperature. It is proposed that the requirement for monitoring of the welding interpass temperature be eliminated.

Attachment 1

Weld Interpass Temperature Evaluation

[ ]