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

Three Mile Island, Unit 1
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: TMI Unit 1 Response to NRC Request for Additional Information Regarding Relief Requests (RR-00-01 Through RR-00-13) Associated with the Third Ten Year Inservice Inspection (ISI) Interval Program Plan

Attached is the AmerGen Energy Company, LLC (AmerGen) response to the NRC request for additional information (RAI) dated February 25, 2002 regarding relief requests and alternatives associated with the third ten year ISI interval program plan for TMI Unit 1 which was submitted to the NRC on August 2, 2001. Unless the NRC RAI had a question on a particular request for relief or alternative to the ASME Boiler and Pressure Vessel Code Section XI requirements, the request is not addressed in this response.

Respectfully,

Michael P. Gallagher
Director – Licensing and Regulatory Affairs
Mid-Atlantic Regional Operating Group

Attachments: 1. AmerGen Response to NRC Request for Additional Information
2. Revised Relief Request Nos. RR-00-07, RR-00-08, RR-00-11, and RR-00-13

cc: H. J. Miller, USNRC, Regional Administrator, Region I
T. G. Colburn, USNRC, Senior Project Manager, TMI Unit 1
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File No. 01035

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ATTACHMENT 1

TMI Unit 1

AmerGen Response to NRC Request for Additional Information Regarding Relief Requests submitted with the Third Ten Year ISI Interval ISI Program Plan

AmerGen Response to NRC Request for Additional Information Regarding Relief Requests submitted with the TMI Unit 1 Third Ten Year ISI Interval ISI Program Plan

RR-00-01 - Examination of Pipe Welds Located Inside Reactor Vessel Primary Shield Wall

NRC Request:
The 1995 Edition including the 1996 Addenda, of the ASME Code, Section XI, Appendix VIII addresses performance demonstration requirements for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws. The licensee should address how it will meet the performance demonstration requirements of Appendix VIII for using ultrasonic examination techniques from the ID surface in detecting and sizing flaws on the OD surface.

Demonstrate that the techniques used for the B&W demonstration meet the objective of the Appendix VIII requirements providing assurance that surface flaws will be detected. What types/directions and size of flaws will the instrument, procedures and personnel be capable of detecting?

AmerGen Response:
The NRCs letter dated October 2, 2001 which approved this relief for second and third interval examinations conducted during the 1R14 Outage stated that the Performance Demonstration Initiative (PDI) project will be able to support performance demonstrations from the pipe inside surface by November 22, 2002. In addition, AmerGen is aware of the development of a Code Case that may affect this request and could obviate the need for it.

As suggested by the NRC in a conference call on January 31, 2002, AmerGen is withdrawing RR-00-01. We intend to resubmit this request when sufficient information is available and if NRC approval of an alternative is still needed subsequent to the resolution of Code Case activity. The next reactor vessel inspection is currently planned for fall of 2011.

RR-00-02 – Examination of Steam Generator Welded Attachment Welds

NRC Request:
Please describe your proposed VT-1 examination. Will the examination be a direct examination or a remote visual examination? What magnifications will be used? Describe the capabilities of your examination (such as resolution, detection and sizing capabilities). What percentage of the weld will the VT-1 examination cover (what percentage of the C-D surface area as shown in Figure IWB-2500-13 will be examined with this method)?

The NRC staff has also accepted UT examination from the accessible surface. (See safety evaluations for Brown's Ferry Nuclear Station, Units 2 and 3, 2nd 10-year ISI interval dated June 19, 2000, and Brown's Ferry Nuclear Station, Unit 2, 3rd 10-year interval, relief request 2-ISI-10, Revision 1, dated February 4, 2002.

AmerGen Response:
Visual examinations are conducted in accordance with Article 9 of Section V and IWA-2210 of Section XI. A procedure demonstration using a near distance vision test chart containing

text with lower case characters without an ascender or descender shall be performed. The weld to be examined shall be illuminated to attain a minimum of 50 foot candles at a maximum direct examination distance of 2 feet. In accordance with the Code visual examinations are "conducted to detect discontinuities and imperfections on the surfaces of components, including such conditions as cracks, wear, corrosion, or erosion." The proposed VT-1 examination will be a direct examination without the use of magnification. The examination will cover 100% of the area C-D as shown on Figure IWB-2500-13.

RR-00-03 – Examination of Letdown Cooler Manifold Welds

NRC Request:
For the letdown coolers longitudinal welds, the Code requires the examination of 1 foot of longitudinal weld intersecting the circumferential weld?

What is the appropriate length of a longitudinal weld that the licensee is able to examine? Why can't the licensee examine additional longitudinal welds and/or both coolers?

Provide a drawing(s) showing welds and identify accessible areas. What percentage of the required 1 foot are you able to obtain on one cooler?

What was examined in the first and second ISI intervals? Were there any unacceptable indications? What percentage of the required examination was attained?

Are there other examinations performed that help justify that the proposed inspection provides reasonable assurance of structural integrity?

AmerGen Response:
Except for the straight portion of accessible manifold (approximately 1¼ inch length on the inlet side and approximately 2¾ inch on the outlet side) that will be examined, the balance of the longitudinal welds intersecting the circumferential weld is covered by the shell side of the heat exchanger. Additional longitudinal weld length is not accessible since it is totally enclosed within the Class 3 (heat sink) side of the vessel. Access to additional longitudinal weld length would require removal of fabrication welds and destruction of the unit. Table IWB-2500-1, Category B-B states that the examination may be limited to one vessel among the group of vessels performing a similar function.

As shown in Figure 1, the letdown coolers are counter flow, spiral tube heat exchangers that reduce the temperature of letdown flow to protect the demineralizer resins and the Reactor Coolant Pump seals. The secondary shell side is cooled by the Intermediate Closed Cooling Water (ICCW) System. Longitudinal welds run the full length of both sides of the two 3¾ inch outer diameter (OD) "cylinders" which comprise the primary inlet and outlet manifolds that connect to the tube ends of the (30) ¾ inch OD helical coils within the external casing shell. The secondary shell side has been fabricated around these components and only the inlet and outlet manifold stubs protruding from the cooler are accessible. Each manifold has two longitudinal seam welds and each letdown cooler has four such welds.

Section "B-B" from drawing NU-D-1044-1 is a cross section of one of the manifolds within the cooler unit showing these longitudinal welds. Section "A-A" from drawing NU-D-1044-1 shows the short section of an inlet manifold protruding from the tubesheet that is accessible for volumetric examination.

Figure 1 is a photograph of the Letdown Cooler internals (cooling coils and manifolds) with the casing removed for destructive examination. In this photograph the component is laying on its face with the accessible portion of the longitudinal welds on the underneath side and therefore not visible. Cooler leaks in 1978 and 1987 were found to have originated where the coils are tack welded to each other. Procedure changes to reduce the flow and the heat load by operating both coolers in parallel and to ramp the flow more slowly have resolved the problem and subsequent replacement has not been necessary.

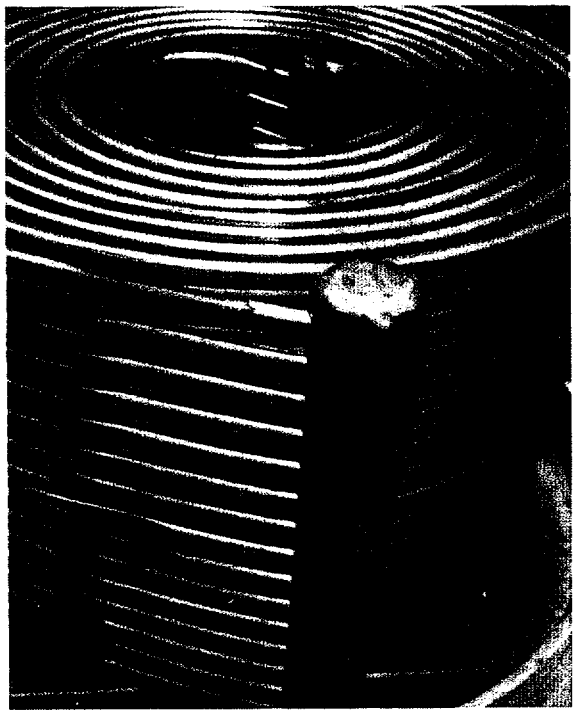

Figure 1 - Letdown Cooler Internals

The accessible approximately 1¼ inch length of the inlet manifold seam welds and approximately 2¾ inch of the outlet side manifold seam welds represent approximately 10.4% and 22.9% of the required 1-foot length of each weld. Examinations conducted for the first and second ISI intervals were essentially the same in accordance with relief granted by the NRC on March 20, 1987 (first interval) and October 8, 1992 (second interval). No unacceptable indications were identified. Although the first and second examination coverage was the same, the Code requirement applicable to the first interval only required 10% of each weld (3.1 inches vs. 12 inches). Therefore, for the first ISI interval the percentage of examination coverage was greater (40.3% and 88.7% of the required lengths of inlet and outlet manifold seam welds respectively) for the first interval examination compared to that of second interval examination and that planned for the third interval. The examinations planned

for the third interval will be the same as the second interval consisting of the accessible approximately 1¼ inch length of the welds on both sides of the inlet manifold and the accessible approximately 2¾ inch length of the welds on both sides of the outlet manifold of one cooler. Since the one foot requirement has not changed from that required during the second interval, the percentage of coverage will be the same.

Both letdown coolers, (MU-C-1A & MU-C-1B) receive a Visual, VT-2, examination prior to plant startup following each reactor refueling outage in accordance with the requirements of ASME Section XI Examination Category B-P, Item No. B15.40.

These heat exchangers are a weldseal design with no removable parts; internal inspection is not possible. If trouble is experienced, the manufacturer recommends that the complete cooler unit be returned to them for service. Therefore, no meaningful examination of these longitudinal welds within the casing shell can be performed without destroying the cooler unit.

The letdown coolers are located in the letdown cooler room in the Reactor Building basement where previous plant history indicates radiation fields of 100 mr/hr to 250 mr/hr general radiation, with occasional hot spots in excess of 1R/hr. When the letdown coolers were replaced in 1987, with shielding installed to reduce personnel exposure, approximately 15 person-rem were expended (14.543 person-rem was recorded for the letdown cooler exposure tracking number). Therefore AmerGen concludes that it is impractical to replace the letdown coolers only to satisfy the Section XI Code examination requirement.

RR-00-05 – Examination of Decay Heat Removal Cooler Shell to Retaining Ring Welds

NRC Request:
How frequently are the decay heat removal coolers repaired/maintained so as to allow access to the weld location? Does the repair/maintenance program assure that one of the coolers will be disassembled each interval? Were either of the welds (DH-0399 and DH-0404) examined in the previous interval?

AmerGen Response:
Decay Heat Removal (DHR) coolers DH-C-1A and DH-C-1B are located in a High Radiation Area and have not been dismantled for repairs or maintenance. Because of the need to maintain personnel radiation exposure as low as reasonably achievable (ALARA), the DHR coolers are not included in a scheduled maintenance program for periodic disassembly. The subject welds receive a Section XI visual VT-2 examination each refueling outage in accordance with procedure 1303-11.16, "Refueling DHR Testing." Detection of leakage from these welds would require repairs. Neither weld DH-0399 nor DH-0404 was UT examined during the second ISI interval.

RR-00-07 – Alternative Requirements for Qualification of VT-2 Examination Personnel

NRC Request:
Code Case N-546 is listed in the Draft Regulatory Guide DG-1091 which is out for public comment. The ADAMS accession number for DG-1091 is ML013120019. DG-1091 lists three conditions for the use of Code Case N-546. Please incorporate the three conditions. Should resolution of the public comments change the conditions the licensee will be expected to follow the requirements in the Code Case and any additional requirements listed in the future revision to Regulatory Guide 1.147.

AmerGen Response:
Relief Request No. RR-00-07 has been revised in Attachment 2 to incorporate Code Case N-546 to include the conditions of DG-1091 for personnel performing VT-2 Visual inspections.

RR-00-08 – Removal of Insulation During System Pressure Tests (Ref Borated Systems)

NRC Request:
Initially the licensee references Class 1 and Class 2 under the heading of COMPONENT IDENTIFICATION. In the proposed alternative the licensee references only Class 1 systems. Will the alternative be applied to only Class 1 or to Class 1 and Class 2.
(Code Case N-533-1 with a condition is also referenced in the Draft Regulatory Guide DG-1091)

AmerGen Response:
The alternative will be applied to Class 1 only. RR-00-08 has been revised in Attachment 2 to delete the reference to Class 2.

Please note that the condition applied to Code Case N-533-1 in DG-1091 that "A 4-hour hold time must be maintained prior to the VT-2 visual examination" has already been incorporated into the relief request.

RR-00-11 – Proposed Actions for Bolt Removal at Leaking Connections

NRC Request:
The alternative should be drafted in terms of Code Case-566-1 (Vogtle SER dated 5/4/2001)

AmerGen Response:
RR-00-11 has been revised in Attachment 2 to reference Code Case N-566-1, similar to the request approved by the NRC for South Texas Project in an SER dated February 9, 2000.

RR-00-13 – Alternative Requirements to Required Percentages of Examinations

NRC Request:
Rewrite in terms of CC N-598. (See Vogtle SER dated 5/4/2001).
Licensee's proposed alternative did not address Table IWD.2412-1.

AmerGen Response:
RR-00-13 has been revised in Attachment 2 in terms of Code Case N-598. The revised alternative addresses Table IWD-2412-1.

ATTACHMENT 2

TMI Unit 1

Revised Relief Requests Associated with the Third 10-yr ISI Interval ISI Program Plan

AmerGen Energy Company
Three Mile Island Unit 1
Third 10-Year Interval
Request for Relief RR-00-07

COMPONENT IDENTIFICATION

Code Class:	Class 1, Class 2, and Class 3
Reference:	ASME, Section XI; 1995 Edition, 1996 Addenda; IWA-2300
Examination Categories:	Not Applicable
Item Numbers:	Not Applicable
Description:	Alternative Requirements for Qualification of VT-2 Examination Personnel
Component Numbers:	Class 1, Class 2, and Class 3 Pressure Retaining Components

CODE REQUIREMENTS

ASME Code, Section XI, Rules for Inservice Inspection of nuclear Power Plant Components, 1995 Edition with 1996 Addenda, Subarticle IWA-2300 and Paragraph IWA-2312, require personnel performing nondestructive examinations not listed in SNT-TC-1A to be qualified and certified to a comparable level of qualification as defined in SNT-TC-1A and the Employer's written practice.

CODE REQUIREMENTS FROM WHICH AN ALTERNATIVE IS REQUESTED

Relief is requested from the requirements of IWA-2300 which requires personnel performing VT-2 examinations to be qualified and certified to comparable levels of qualifications as defined in SNT-TC-1A and the Employer's written practice.

BASIS FOR ALTERNATIVE

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

Section XI currently requires personnel conducting VT-2 inspections to be qualified and certified to comparable levels of qualifications as defined in SNT-TC-1A and the Employer's written practice. However, unlike the nondestructive testing methods addressed within SNT-TC-1A, or VT-1 and VT-3 examination methods, VT-2 examinations do not require any special knowledge of underlying technical principles to perform the examination. It is only a straightforward examination to look for evidence of leakage or structural distress. No special skills or technical training are required in order to observe water dripping from a component or bubbles forming on a wetted joint. As such, VT-2 personnel should not be subject to the same qualification and certification requirements that were established for nondestructive testing personnel. Code Case N-546 **and DG-1091 provide** more appropriate requirements for the qualification and certification of VT-2 examination personnel.

Code Case N-546 **and the additional requirements of DG-1091** require personnel performing VT-2 visual inspections meet the following requirements:

(1) Personnel shall have at least forty (40) hours of plant walkdown experience.
(2) Personnel shall receive a minimum of four (4) hours of training on Section XI requirements and plant specific procedures.
(3) Examination personnel shall be qualified by test to demonstrate knowledge of Section XI and plant specific procedures for VT-2 visual examination.
(4) Examination personnel shall be re-qualified by examination every three years.
(5) Personnel shall pass the vision test requirements of IWA-2321, 1995 Edition.
(6) This Code Case is applicable only to the performance of VT-2 examinations.

This alternative to the existing Code requirements reduces the administrative burden of maintaining a Section XI qualification and certification program for VT-2 examiners, and allows for the use of personnel most familiar with the walkdown of plant systems, such as licensed and non-licensed operators, local leak rate test personnel, system engineers and examination personnel. The quality of VT-2 visual examinations will be maintained by using the alternative qualification criteria of the Code Case **along with the conditions stated in DG-1091.**

Code Case N-546 was approved by the ASME Boiler and Pressure Vessel Code Committee on August 24, 1995, but is not yet included in the most recent listing of NRC approved code cases provided in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability – ASME, Section XI, Division 1." **However, Draft Regulatory Guide (DG) 1091 has conditionally accepted Code Case N-546, and those conditions have been incorporated into this relief request.**

PROPOSED ALTERNATIVE PROVISIONS

TMI Unit 1 will use the provision of Code Case N-546 **with the conditions of DG-1091** as an alternative to the requirements of Section XI, IWA-2300 for qualifying VT-2 visual examiners.

PERIOD FOR WHICH AN ALTERNATIVE IS REQUESTED

Relief is requested for the third ten-year inspection interval of the Inservice Inspection Program for TMI Unit 1.

AmerGen Energy Company
Three Mile Island Unit 1
Third 10-Year Interval
Request for Relief RR-00-08

COMPONENT IDENTIFICATION

Code Class:	Class 1
Reference:	ASME, Section XI; 1995 Edition, 1996 Addenda; IWA-5242(a)
Examination Categories:	Not Applicable
Item Numbers:	Not Applicable
Description:	Alternative requirements to insulation removal specified in IWA-5242(a).
Component Numbers:	Bolted connections in systems borated for the purposes of controlling reactivity.

CODE REQUIREMENTS

ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition with 1996 Addenda, Paragraph IWA-5242(a) states, "For systems borated for the purpose of controlling reactivity, insulation shall be removed from the pressure retaining bolted connections for VT-2 visual examination." The VT-2 visual examination is required to be performed coincident with the system leakage test at nominal operating pressure and temperature.

CODE REQUIREMENTS FROM WHICH AN ALTERNATIVE IS REQUESTED

An alternative is requested from the requirements of IWA-5242(a) which requires the removal of insulation from bolted connections to perform VT-2 visual examinations at system pressure for systems borated for the purpose of controlling reactivity.

BASIS FOR ALTERNATIVE

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

An alternative is requested to the requirement to remove insulation from the subject bolted connections during the system leakage test at nominal pressure and temperature for the following reasons:

1. Code Class 1 systems borated for the purpose of controlling reactivity are large extensive systems covering many areas and elevations. Scaffolding is required to access many of the bolted connections. In addition, many of the bolted connections are located in difficult to access areas and in medium to high radiation areas. Insulation removal combined with scaffolding requirements will increase refuel outage durations,

personnel exposure, financial costs and generation of radwaste associated with the performance of VT-2 visual examinations.

2. The VT-2 visual examination of Class 1 systems, primarily the Reactor Coolant System (RCS) piping and components which are located inside containment are performed at hot shutdown conditions. As required by IWB-5221, the RCS is at normal operating conditions. Removal/reinstallation of insulation for Class 1 systems pose significant radiological considerations. In addition, performance of the VT-2 visual examination, removal/reinstallation of insulation, and assembly/disassembly of scaffolding at bolted connections under these operating conditions also presents significant personnel safety considerations.

The following TMI Unit 1 bolting examination commitments in conjunction with the Proposed Alternative Provisions provide an acceptable level of safety and quality for bolted connections in systems borated for the purpose of controlling reactivity.

In response to NRC Generic Letter 88-05, TMI Unit 1 has established a program for engineering to examine all boric acid leaks discovered in the containment building and to evaluate the impact of those leaks on carbon steel or low alloy steel components. Any evidence of leakage, including dry boric acid crystals or residue, is examined and evaluated regardless of whether the leak was discovered at power or during an outage. Issues such as the following are considered in the examination and evaluation.

1. Evidence of corrosion or metal degradation.
2. Effect the leakage may have on the pressure boundary.
3. Possibility of boric acid traveling along the inside of insulation on piping.
4. Possibility of dripping or spraying on other components.

Based on this evaluation, TMI Engineering initiates appropriate corrective actions to preclude reoccurrence of the leakage and to repair or replace, if necessary, any degraded materials or components.

This relief was previously granted for the second inservice inspection interval at TMI Unit 1 in the NRC's Safety valuation Report (SER), dated October 8, 1992, which references the GPUN submittal dated April 19, 1991.

PROPOSED ALTERNATIVE PROVISIONS

As an alternative to the Code required examination, AmerGen proposes the following:

For Class 1 systems borated for the purpose of controlling reactivity, a system leakage test shall be performed in accordance with the frequency required by Table IWB-2500, Category B-P without the removal of insulation at bolted connections. A minimum 4-hour hold time at normal system operating pressure prior to the VT-2 visual examination to allow for leakage propagation from the insulation shall be required. Additionally, the insulation shall be removed from Class 1 bolted connections and a VT-2 visual examination shall be conducted with the system

depressurized. The frequency for these depressurized VT-2 visual examinations shall be in accordance with the system examination frequencies specified in Table IWB-2500, Category B-P, each refueling outage. The propose alternative is consistent with the requirements of Code Case N-533. These examinations shall be implemented through application of the TMI Unit 1 surveillance program to assure they are performed within the prescribed time periods.

PERIOD FOR WHICH AN ALTERNATIVE IS REQUESTED

An alternative is requested for the third ten-year inspection interval of the Inservice Inspection program for TMI Unit 1.

AmerGen Energy Company
Three Mile Island Unit 1
Third 10-Year Interval
Request for Relief RR-00-11

COMPONENT IDENTIFICATION

Code Class:	Class 1, Class 2, and Class 3
Reference:	ASME, Section XI; 1995 Edition, 1996 Addenda; IWA-5250(a)(2)
Examination Categories:	Not Applicable
Item Numbers:	Not Applicable
Description:	Alternative requirements to inspections specified in IWA-5250(a)(2).
Component Numbers:	Class 1, Class 2, and Class 3 bolted connections.

CODE REQUIREMENTS

ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition with 1996 Addenda, Subsection IWA-5250(a)(2) requires that if leakage occurs at a bolted connection on other than a gaseous system, one of the bolts shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100. The bolt selected shall be the one closest to the source of leakage. When the removed bolt has evidence of degradation, all remaining bolting in the connection shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100.

CODE REQUIREMENTS FROM WHICH AN ALTERNATIVE IS REQUESTED

An alternative is requested from the Code requirements of Subsection IWA-5250(a)(2) which requires that if leakage occurs at a bolted connection on other than a gaseous system, one of the bolts shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100.

BASIS FOR RELIEF

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

Removal of pressure retaining bolting at mechanical connections for VT-3 visual examination and subsequent evaluation, in locations where leakage has been identified, is not always the most discerning course of action to determine acceptability of the bolting. The Code requirement to remove, examine, and evaluate bolting in this situation does not allow the owner to consider other factors which may indicate the acceptability of the mechanical joint bolting.

Other factors which should be considered when evaluating bolting acceptability when leakage has been identified at a mechanical joint include, but are not limited to: joint bolting material,

service age of joint bolting material, location of leakage, history of leakage at the joint, evidence of corrosion with the joint assembled, and corrosiveness of process fluid.

Performance of the pressure test while the system is in service may identify leakage at a bolted connection that, upon evaluation may conclude that the integrity and pressure retaining ability of the joint is not challenged. It would not be prudent to negatively impact the availability of a safety system to perform its safety function.

A situation frequently encountered at AmerGen nuclear facilities is the complete replacement of bolting materials (studs, bolts, nuts, washers, etc.) at mechanical joints during plant outages. When the associated system piping is pressurized during plant startup, leakage may be identified at those joints. The root cause of this leakage is most often due to thermal expansion of the piping and bolting materials at the joint and subsequent fluid seepage at the joint gasket. Proper re-torquing of the joint bolting, in most cases, stops the leakage. Removal of the joint bolting to evaluate for corrosion would be unwarranted in this situation due to the new condition of the bolting materials.

PROPOSED ALTERNATIVE PROVISIONS

As an alternative to the bolt removal requirements of Subsection IWA-5250(a)(2), AmerGen proposes the following **requirements from ASME Section XI Code Case N-566-1 be implemented in response to detection of leakage at a pressure-retaining bolted connection during a VT-2 visual examination:**

(a) Stop the leak and evaluate the bolting and the component material for joint integrity as described in (c) below.
(b) If the leakage is not stopped, evaluate the joint for joint integrity in accordance with IWB-3142.4. This evaluation shall include the considerations listed in (c) below.
(c) The evaluation in (a) and (b) is to determine the susceptibility of the bolting to corrosion and failure. The evaluation shall include the following:
- **The number and service age of bolts;**
- **Bolt and component material;**
- **Corrosiveness of process fluid;**
- **Leakage location and system function;**
- **Leakage history at the connection or other system components; and**
- **Visual evidence of corrosion at the assembled connection.**

When the pressure test is performed on a system that is in service or that Technical Specifications require to be operable, and the bolting is susceptible to corrosion, the evaluation shall address the connection's structural integrity until the next component/system outage of sufficient duration. If the evaluations conclude the system can perform its safety related function, removal of the bolt closest to the source of the leakage and a VT-3[1] visual examination of the bolt will be performed when the system or component is taken out of service for sufficient duration (to accomplish other system maintenance activities.)

For bolting that is susceptible to corrosion, and when the initial evaluation indicates that the connection cannot conclusively perform its safety function until the next component/system outage of sufficient duration, the bolt closest to the source of the leakage will be removed, and a VT-3 visual examination will be performed and results evaluated in accordance with IWA-3100(a).

1 The acceptance criteria for the Visual. VT-1 will be used to access acceptability of the bolting.

This relief was previously granted for the second inservice inspection interval at TMI Unit 1 in the NRC's Safety Evaluation Report (SER), dated December 2, 1998, which references the GPUN submittal dated June 3, 1998.

PERIOD FOR WHICH AN ALTERNATIVE IS REQUESTED

Relief is requested for the third ten-year inspection interval of the Inservice Inspection Program for TMI Unit 1.

AmerGen Energy Company
Three Mile Island Unit 1
Third 10-Year Interval
Request for Relief RR-00-13

COMPONENT IDENTIFICATION

Code Class:	Class 1, Class 2, and Class 3
Reference:	ASME, Section XI; 1995 Edition, 1996 Addenda; Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, and IWF-2410-2
Examination Categories:	Not Applicable
Item Numbers:	Not Applicable
Description:	Alternative requirements to examination percentage completion.
Component Numbers:	Class 1, Class 2, and Class 3 components and supports.

CODE REQUIREMENTS

ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1995 Edition with 1996 Addenda, Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, and IWF-2410-2 list the required percentages that must be performed per inspection period in accordance with Inspection Program B. Per these tables, the number of examinations to be completed during the first period shall be between 16% and 34%. For the second period, the total number of examinations to be completed shall be between 50% and 67%, and by the end of the third period, 100% of the examinations for the interval shall be completed.

CODE REQUIREMENTS FROM WHICH AN ALTERNATIVE IS REQUESTED

An alternative is requested from the Code requirements for examination percentage completion identified in Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, and IWF-2410-2.

BASIS FOR ALTERNATIVE

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

The inspection program percentage tables of ASME Code Section XI were originally established such that approximately one-third of the non-deferred component examinations would be performed each period. The emergence of longer plant operating fuel cycles coincident with efforts to reduce the length of refueling outages have limited the amount of time available to perform examinations. These factors make it difficult to plan and complete the Code required percentages of examinations in allotted critical path time.

The alternative provision was developed to address these issues. Expansion of the range for examination completion percentages shown in Table 1 allows component examinations to be more evenly distributed between outages. In addition, this expansion minimizes the need to

schedule excessive numbers of examinations during a specific outage, and allows for a more uniform distribution between outages that is more conducive to performing quality examinations. Repetitive costs associated with inspections, such as the erecting and disassembly of scaffolding, labor costs associated with acquiring inspectors each outage, can be minimized through balancing the inspection percentages.

Two additional factors were considered when evaluation the impact of the percentage requirements of Table 1 on plant safety. The first was that the existing examination percentage tables of Section XI allow up to 50% of the examinations to be performed in the second and third periods, but only 34% can be performed in the first period. Therefore, the Section XI Inspection Plan B schedules are biased towards delaying examinations until the end of the interval. The more flexible percentages required by Table 1 allows for more examinations to be performed earlier in the interval. This should improve safety because any degradation, should it exist, would be detected earlier in the interval.

The second factor that was considered was that some minimum amount of examinations should be required in each period. To address this consideration, Note 1, is included in the table, so the examinations will be required during all three inspection periods.

Based on the factors identified above, AmerGen considers that the alternative provisions provide an acceptable level of quality and safety.

PROPOSED ALTERNATIVE PROVISIONS

As an alternative to the Code requirements for the determination of examination percentage completion identified in Section XI Tables IWB-2412-1, IWC-2412-1, **IWD-2412-1,** and IWF-2410-2, AmerGen proposes the use of the examination percentages **derived from Code Case N-598 and** identified in Table 1 below.

Table 1

Inspection Interval	Inspection Period, Calendar Years of Plant Service Within the Interval	Minimum Examinations Completed, %	Maximum Examinations Credited, %
3rd	3	16	50
	7	50[1]	75
	10	100	100

NOTE:
(1) If the first period completion percentages for any examination category exceeds 34%, at least 16% of the required examinations shall be performed in the second period.

PERIOD FOR WHICH ALTERNATIVE IS REQUESTED

An alternative is requested for the third ten-year inspection interval of the Inservice Inspection Program for TMI Unit 1.