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W3F1-97-0014

A4.05

PR

February 13, 1997

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

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Inservice Inspection (ISI) Relief Request ISI-017, Revision 1,
Request Authorization to Substitute Visual (VT-3) Examination for
the Code-required Surface or Volumetric Examination of the four
Reactor Pressure Vessel Support Integral Attachment Welds
(ISI Exam Number 01-074)

Gentlemen:

By letter dated June 3, 1996, Entergy Operations, Inc. requested approval to delete the Code required surface examination on the Reactor Pressure Vessel (RPV) support integral attachment welds as defined by the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, 1980 Edition, Winter 1981 Addenda, Table IWB-2500-1, Examination Category B-H, Item No. B8.10. This Code requires surface or volumetric examination of Waterford's integral attachment welds. After further review, Entergy now requests authorization to perform a visual (VT-3) examination in lieu of the Code-required examinations based on personnel exposure concerns and limited examination area access. Furthermore, Entergy considers the VT-3 examination to be more effective in detecting the type of degradation that is most likely to occur.

By letter dated December 16, 1996, the NRC's requested additional information regarding the subject relief request. The requested information is supplied in Appendix 1, and is included in Revision 1 to Relief Request ISI-017, supplied as Appendix 2. As requested, a copy of this letter is also being sent to INEL, the NRC's contractor.

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Request to Substitute Visual (VT-3) Examination for Surface or Volumetric Examination of the four Reactor Pressure Vessel Support Integral Attachment Welds (Relief Request ISI-017)

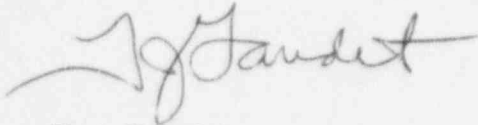
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Based on the discussion provided in Appendix 2, Relief Request ISI-017, Revision 1, Waterford 3 requests authorization to substitute visual (VT-3) examination for the Code-required surface or volumetric examination on the RPV support integral attachment welds. This request has been discussed with the Waterford 3, NRR, Project Manager and is similar to relief granted to other facilities, such as Grand Gulf Nuclear Station. Waterford 3 respectfully requests a timely review of this Relief Request to allow for appropriate planning of Refuel 8 ISI inspection activities, currently scheduled for April, 1997.

Should you have any questions regarding this matter, please contact me at (504) 739-6666.

Very truly yours,



T. J. Gaudet
Acting Director
Nuclear Safety & Regulatory Affairs

TJG/DMU/tjs

Attachment

cc:

L. J. Callan, NRC Region IV
C.P. Patel, NRC-NRR
R.B. McGehee
N.S. Reynolds
NRC Resident Inspectors Office
M. T. Anderson, INEL Research Center

Appendix 1
Response to the NRC's
RAI for Relief Request ISI-017

Question:

A) State the specific paragraph of the Regulations (10 CFR 50.55a) under which the request is submitted and provide supporting justification as discussed below.

The regulations provide that a licensee may propose an alternative to CFR or Code requirements in accordance with 10 CFR 50.55a(a)(3)(i) or 10 CFR 50.55a(a)(3)(ii). Pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative must be shown to provide an acceptable level of quality and safety, i.e., essentially, be equivalent to the original requirement in terms of quality and safety. Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee must show that compliance with the original requirement results in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Examples of hardships and/or unusual difficulty include, but are not limited to, excessive radiation exposure, disassembly of components solely to provide access for examinations and development of sophisticated tooling that would result in only minimal increase in examination coverage. A licensee may also submit a request for relief from ASME requirements. In accordance with 10 CFR 50.55a(g)(5)(iii), if a licensee determines that conformance with certain Code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in §50.4, information to support that determination. When a licensee determines that an inservice inspection requirement is impractical, e.g., the system would have to be redesigned, or a component would have to be replaced to enable inspection, the licensee should cite 10 CFR 50.55a(g)(5)(iii). The NRC may, giving due consideration to the burden placed on the licensee, impose an alternative examination requirement.

Response:

A) Authorization to implement an alternative to the Code requirement is being requested pursuant to 10CFR50.55a(a)(3)(ii). The purpose of the authorization is to seek relief from an ASME Code examination requirement that imposes an unusual difficulty on Waterford-3 due to plant configuration, accessibility, and radiation exposure. Authorization is being requested for ISI Exam Number 01-074, Reactor Vessel Support Attachment Welds, Code Category B-H, Item No. B8.10. Relief is not being requested for ISI Exam Number 01-073, Reactor Vessel Support, Code Category F-C, since the four supports were completely and satisfactorily examined per ASME Code Section XI requirements during the most recent refueling outage (Refuel 7).

Question:

B) This request describes the limitations associated with examining the reactor vessel support integral attachment welds. However, the configuration is not adequately described. Provide a detail sketch of the examination area for this request, including all limitations.

Response:

B) The attached detailed sketches (see Appendix 3) show the Reactor Pressure Vessel Nozzles, integrally-welded attachments, supports, and interferences. As can be seen from these sketches, accessibility to the attachment welds is severely restricted even after access to the Reactor Cavity area has been obtained.

Question:

C) The Code requires surface examination of all four integral attachment welds. The proposed alternative examination is to VT-3 examine only two of the subject welds. Provide a basis for examination of only two welds when the Code requires examination of all four.

Response:

C) After further review, we have determined that it is possible to visually (VT-3) examine the remaining two attachment welds with small additional radiation exposures to plant personnel. Consequently, Relief Request ISI-017, Revision 1 (Appendix 2), includes VT-3 examination of all four welds in lieu of the Code-required surface or volumetric examination.

Question:

D) Provide the cumulative radiological exposure associated with the VT-3 visual examinations of the two integral welded attachments proposed and the exposure associated with performing the VT-3 visual examination on all four reactor vessel integrally welded attachments.

Response:

D) The radiation dose associated with only the VT-3 examination of the two attachment welds (excluding the VT-3 of the four supports) is estimated to be approximately 300 mr. It is estimated that the radiation dose associated with a VT-3 examination of the remaining two attachment welds during the next refueling outage will also be 300 mr. The total dose for the four attachment welds is expected to be approximately 600 mr.

APPENDIX 2
RELIEF REQUEST
ISI-017, Revision 1

1. Component:

ISI Examination Number 01-074, which consists of four reactor pressure vessel (RPV) cold leg nozzle-to-vessel support pad integral attachment welds.

2. Code:

The RPV support welds are designed, fabricated and certified to the 1971 Edition, Summer 1971 Addenda of ASME Section III, Subsection NB (Class 1).

3. Code requirements:

ASME Section XI, 1980 Edition, Winter 1981 Addenda, Table IWB-2500-1, Examination Category B-H, Item No. B8.10 requires surface or volumetric examination of integral attachment welds.

4. Relief Requested:

Authorization is requested pursuant to 10CFR50.55a(a)(3)(ii) to substitute a visual (VT-3) examination for the Code-required surface or volumetric examination of the four RPV support integral attachment welds.

5. Basis for relief:

The Reactor Pressure Vessel is supported by pads welded to the underside of the four inlet nozzles. These pads rest on spherical bearings. The spherical bearings rest on the containment support plates. The spherical bearings and the containment support plates are held down by four six inch diameter studs/nuts. These spherical bearings contain Stellite, which became a radioactive source due to its cobalt content. Surveys taken during Refuel 7, revealed contact dose rates at the spherical bearings near the integral attachments are 10 to 20 R/hr. Installation of lead shielding reduced these contact dose rates to 1 R/hr. The dose rates at 30 cm from the spherical bearing (with shielding installed) are 600 mr/hr and the general area dose rates are 300 mr/hr.

The as-found surface condition of the integral attachment welds were not adequate for the performance of either magnetic particle (MT) or dye

penetrant (PT) examination. The surfaces had small patches of tightly adhering scale, a general coating of tightly adhering rust and minor surface irregularities that would create non-relevant indications, interfering with the interpretation of the MT or PT exam results. At minimum, power tool preparation of the weld surfaces is required. Additionally, the examination area is limited in two ways. First, the integral attachment weld surface facing the reactor vessel is completely inaccessible for base metal preparation due to the close proximity to the reactor vessel wall (approximately 3") and the underlying vessel support structure. Secondly, approximately 50% of each of the two sides of the integral attachment weld surface is blocked due to the close proximity of the six inch studs which hold down the spherical bearings and containment support plates. This obstruction prevents access for base metal preparation required for either MT or PT. The only area with unobstructed access for surface preparation/examination is the attachment weld surface facing outward from the vessel. Therefore, the best possible surface examination coverage is estimated to be 50%.

The same inaccessibility that hampers the surface examination also applies to any attempt to perform volumetric examination of the welds. Radiography (RT) is impractical due to the non-symmetrical design of the attachments as well as the extreme thickness of the components. Very thick steel components require very large RT sources which can not be positioned due to the limited access available. Such RT examinations would also require very long exposure times. Performing RT examinations would also stop all work in the Containment Building during a refueling outage and would therefore add considerable duration and cost to the outage without a compensating increase in the level of quality and safety.

Ultrasonic examination (UT) from the surface of the welded attachments is also impractical due to the irregular design of the lugs. UT from the inside diameter (ID) would also necessitate the design and fabrication of special tooling. The UT from the nozzle ID also requires that the Reactor Vessel be defueled and core barrel removed. Access to the ID of these inlet nozzles is only possible during the 10-year Reactor Vessel ISI examination which has already occurred for this Interval. It is also worthy to note that these attachment welds have never received any form of volumetric examination. Performing either RT or UT would be of limited value since there is no baseline examination with which to compare results.

Because of the difficult access conditions, the necessity for insulation removal and the amount of surface preparation, it is estimated that a minimum of 12 man-hours is required to prepare and examine the welds using the magnetic particle method. Based on the 600 mr/hr dose rates (with shielding installed and working at least 30 cm from the bearing), accumulated exposure to perform surface examination is estimated to be 7.2 man-rem.

All support pad base and weld materials were procured in compliance with ASME III NB-2300 to assure adequate protection against non-ductile failure. Integral attachment weld material was required to exhibit a reference nil-ductility transition temperature of not greater than 10° F as established by Charpy V-notch and/or dropweight tests. Rapid crack propagation from service conditions is unlikely.

During normal operating conditions (including heat-up and cooldown) and during design basis earthquakes, the support pads are primarily loaded in compression. The only occurrence that produces tensile stresses in the attachment weld is a reactor coolant system pipe rupture, which has not occurred at Waterford 3. The integral attachment weld is E8018-C3 shielded metal arc and Manganese Molybdenum Nickel (MnMoNi) submerged arc weld metal. Weld metal material test reports indicate these materials have at least 25% elongation. In general, ductile materials do not fail (by fracture) in compression but tend to deform in response to imposed loads. Therefore, the most likely failure mode of the support is distortion failure (deformation) caused by compressive stress. Any normal service induced degradation would be evidenced by structural deformation as a result of compressive stresses. Although surface (PT or MT) and volumetric (RT or UT) examinations are capable of detecting small defects, they are most effective in finding cracking, which would not be the first indicator of damage in this instance. As a first indicator of a structural problem, the VT-3 examination provides a better examination of the condition of the welds than would a surface or volumetric examination, especially since these examinations would be severely limited due to inadequate accessibility.

During Refuel 7, two of the four attachment welds were VT-3 visual examined expending approximately 300 mr. It is estimated that the radiation dose associated with a VT-3 examination of the remaining two attachment welds during the next refueling outage is also 300 mr. The total dose for the four attachment welds is expected to be approximately 600 mr.

Substitution of the visual (VT-3) examination for the Code-required surface or volumetric examination does not significantly affect the assurance of continued structural integrity of the vessel supports. The VT-3 examination would be more effective in detecting the type of degradation that is most likely to occur.

Based on the high radiation exposure anticipated to perform a partial surface examination and the degradation most likely to occur, Entergy Operations believes that requiring the Code required examination be performed on the subject welded attachments would be a hardship without a compensating increase in the level of quality and safety.

6. Alternate Testing:

Perform a VT-3 visual examination on the subject inlet nozzle support pad welds.

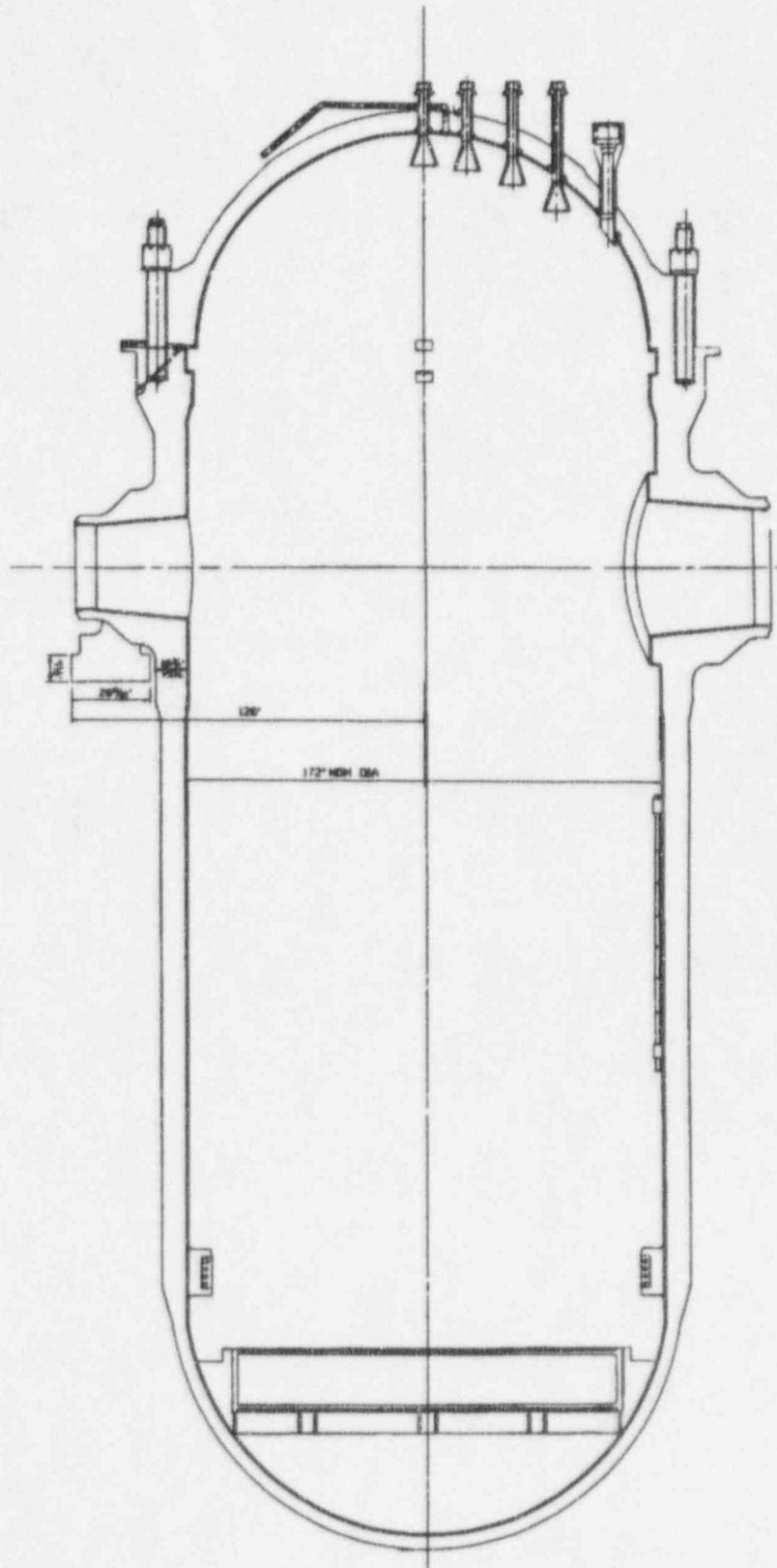
7. Schedule for Implementation:

First Ten Year Interval

8. NRC discussion statement (Revision 1):

Upon receipt of the SER from the NRC, NRC discussion statements contained in the SER are added later, as applicable.

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Appendix 3 (Page 2 of 2)

