Entergy

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Director
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December 19, 2000

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Mail Stop OP1-17
Washington, DC 20555-0001

Subject: Entergy Operations, Inc.
Request for Relief from 10CFR50.55a Examination Requirements

Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29

CNRO-2000-00037

Ladies and Gentlemen:

Pursuant to 10CFR50.55a(a)(3)(ii), Entergy Operations, Inc. (Entergy) requests relief from performing surface and volumetric examinations on circumferential piping welds as defined in ASME Section XI, Figure IWB-2500-8. As documented in Request for Relief GG-ISI-001 (see attachment), Entergy believes there is adequate evidence to indicate these examinations may be deleted due to hardship without a compensating increase in the level of quality and safety.

This request applies to Grand Gulf Nuclear Station (GGNS). Entergy requests the NRC approve GG-ISI-001 prior to the beginning of the upcoming spring refueling outage at GGNS, which begins on April 13, 2001.

This letter contains no commitments.

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Should you have any questions regarding this submittal, please contact Guy Davant at (601) 368-5756.

Very truly yours,

Ron Byrd for MAK

MAK/GHD/baa
attachment
cc: Mr. W. A. Eaton (GGNS)
Mr. G. R. Taylor (ECH)

Mr. T. L. Hoeg, NRC Senior Resident Inspector (GGNS)
Mr. E. W. Merschoff, NRC Region IV Regional Administrator
Mr. S. P. Sekerak, NRC Project Manager (GGNS)

REQUEST FOR RELIEF
GG-ISI-001, Rev. 0

Components/Numbers:	See Table 1 below
Code Classes:	1
References:	ASME Section XI, 1992 Edition, Table IWB-2500-1
Examination Category:	B-J
Item Number:	B9.11 and B9.21
Description:	Examination of piping welds inside containment penetrations
Unit / Inspection Interval Applicability:	Grand Gulf Nuclear Station – second (2^{nd}) 10-year interval

Table 1

System	Penetration Weld	Line #	Line Size	Possible % Coverage Volume/Surface
Feedwater A	1B21G026-W2	DBA-013	24"	24/13
Feedwater B	1B21G026-W18	DBA-013	24"	24/29
Main Steam A	1B21G12-A1-A	MSA-003	28"	27/27
Main Steam B	1B21G12-B1-A	MSA-003	28"	23/20
Main Steam C	1B21G12-C1-A	MSA-003	28"	32/18
Main Steam D	1B21G12-D1-A	MSA-003	28"	23/20
RWCU	1G33G002-W18	DBA-009	6"	58/56
RCIC Steam Inlet	1E51G004-W7	DBA-024	10"	100/35
RHR/RCIC Head Spray	1E51G001-W12	DBA-030	6"	*
RHR Pump Suction	1E12G012-W47	DBA-064	20"	19/19
Main Steam Drain	1B21G021-W9	DBA-023	3"	100% Surface

* This weld was not required for examination during the first interval in accordance with the 1977 Edition with the Summer of 1979 Addenda of Section XI. However, it has been selected for examination in the second interval in accordance with the 1992 Edition of Section XI.

I. Code Requirement(s)

ASME Section XI, Table IWB-2500-1, Examination Category B-J, Item B9.11 requires a surface examination and a volumetric examination on all piping welds as defined by Figure IWB-2500-8. Item B9.21 requires a surface examination of the weld as defined by Figure IWB-2500-8.

II. Code Requirement from Which Relief is Requested

Pursuant to 10CFR50.55a(a)(3)(ii), Entergy Operations, Inc. (Entergy) requests relief from performing the Code-required surface and volumetric examinations on the circumferential welds listed in Table 1, above.

III. Basis for Relief

The high-energy piping that penetrates the containment was designed as a flued head-type penetration that includes a guard pipe similar in design to Figure 1, below. Additionally, these penetrations were designed such that the penetrations are anchored to the containment building. Fins are provided for cooling.

All these lines are designed to 575°F and a pressure ranging from 1060 to 1180 psig depending on application. The process pipe is either ASME SA 155 KCF 70, ASME 106 Grade B, or ASME SA 106 Grade C. Guard pipes are ASME SA 155 KCF 70, ASME SA 106 Grade B, or ASME SA 105.

ASME Section III (1974 with Summer 1975 Addenda and 1980 Edition for GE piping and 1974 Edition and Summer Addenda through Summer 1975 Addenda for Bechtel-supplied) was used for the design of the flued head and guard pipe. The process pipe was also designed to ASME Section III, Subsection NB 1974 Edition with Summer 1975 Addenda.

The circumferential welds for which relief is requested are composed of carbon steel. As such, in a typical BWR environment they are not susceptible to stress corrosion cracking. Design fatigue cumulative usage factors (CUF) for the subject welds are less than 0.1. Therefore, the potential to develop fatigue cracks is extremely low (see the stress analysis review discussion below). Other potential failure mechanisms [e.g., general corrosion, pitting, flow-accelerated corrosion (FAC), etc.] are also considered low probability events, based upon both the operating parameters of the systems and the fact that inservice inspection (ISI) of other welds in these systems has shown no evidence of service-related degradation to date. In addition, any leakage would return to the drywell, leading to an increase in the unidentified leak rate and an increase in the drywell temperature.

Pre-service inspection (PSI) of these welds has detected no relevant surface indications and no recordable volumetric indications. The ISI performed on these welds to date has also detected no relevant and no recordable indications. Should the conditions in the systems change, examination of the remaining welds in the systems will likely detect the onset of service-related degradation.

Each of the lines identified in Table 1 has a pressure-retaining circumferential weld that was previously accessible for partial examination via an inspection port included in the penetration. The original design of these access ports included bolted gasketed covers that required the performance of periodic local leak rate tests (LLRTs). These had a history of LLRTs failures (approximately 25% failure rate). Therefore, the access ports were welded closed to provide assurance of minimal leakage. Thus the Code-required examinations would require removing the access port welds to gain access to the process pipe welds and re-welding the covers following the examinations. A personnel exposure of approximately 24 Rem would be expected to complete the limited Code-required examinations of these welds over the interval. Even after this level of effort of opening the access ports, the extent of weld examination is limited because of space restrictions between the guard pipe and the process pipe, as listed in Table 1.

Furthermore, nine of these welds are in the MEB 3-1 High-Energy Line population. Therefore, 100% of the welds in these lines that fall within the no-break zone are volumetrically examined in accordance with ASME Section XI. This is a much larger population than the 25% requirement of Section XI.

In addition, leakage was postulated to occur from cracks initiated in these lines originating from a size equal to the process pipe cross section (non-mechanistic). The guard pipe design provides for leakage return to the drywell, which is designed for such an event. Additionally, the guard pipes are designed for the process pipe design conditions, as stated above.

The process pipes were tested to the required ASME Code hydrostatic test pressure and the guard pipes were tested to the process pipe operating pressure conditions.

Entergy has performed a stress analysis and a risk analysis of this relief request. These topics are discussed below.

- Stress Analysis Review

  Entergy has performed a review of the various stress analyses to determine both the state of stress and the cumulative fatigue usage (design) for the welds in question. The design requirements for these lines inside the penetrations required that either:

  1) The maximum stress range as calculated using equation (10) of ASME Section III NB-3653 not exceed 2.4 S_m and the cumulative usage factor must be less than 0.1, or

  2) The stress range calculated using equation (12) or (13) of the Code not exceeds 2.4 S_m

  Review of calculations and revisions thereto performed by General Electric and Bechtel show that both of these criteria were met and that the usage factors never approached 0.1. Therefore the likelihood of a pipe break in non-IGSCC sensitive

materials makes the probability of failure extremely remote. In addition, any leakage would return to the drywell leading to an increase in the unidentified leak rate and an increase in the drywell temperature.

- Risk Discussion

  The upper bound core damage frequency (CDF) increase associated with the relief from inspections of specific welds is equal to 4.31E-07 per year, which is considered "non-risk significant". This conclusion is based on the *EPRI PSA Applications Guide* for permanent facility changes. This is a bounding value, which is conservative.

A more realistic estimate of the CDF increase due to the weld inspection interval change is 3.23E-08 per year. This increase corresponds to approximately less than a 1% increase in the total CDF. The large early release frequency (LERF) increase due to the proposed change is also insignificant since the change impacts welds inside the primary containment.

In conclusion, the proposed relief from inspection of specific welds does not significantly increase the total CDF or LERF.

IV. Proposed Alternative Examinations

None

V. Conclusion

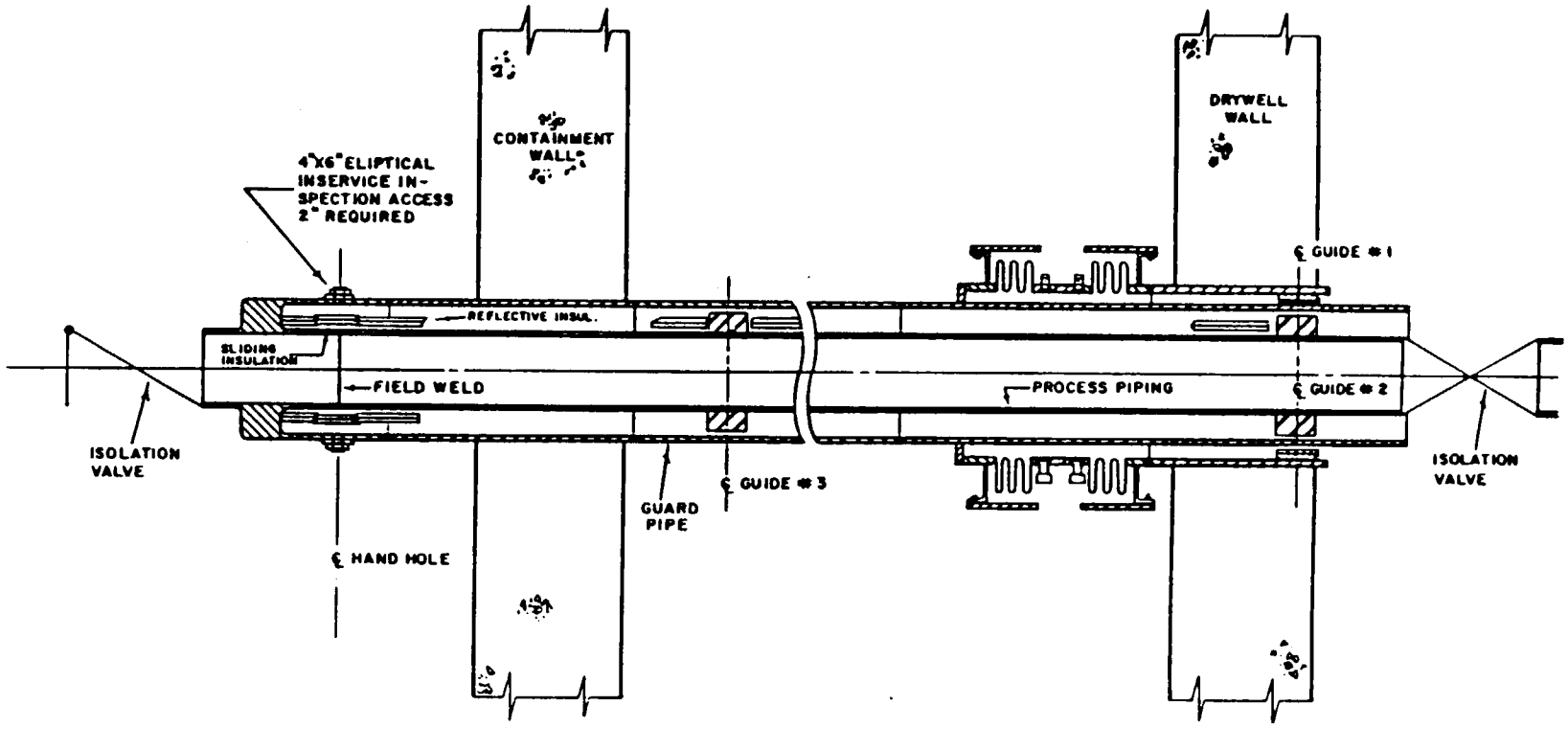
10CFR50.55a(a)(3) states:

"Proposed alternatives to the requirements of (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

Entergy believes that requiring the access ports to be cut opened and then re-welded to perform partial examinations of the subject welds results in a hardship without a compensating increase in the level of quality and safety. Therefore, we request the proposed alternative be authorized pursuant to 10CFR50.55a(a)(3)(ii).



VERTICAL SECTION THRU GUARD PIPE ASSEMBLY

GRAND GULF NUCLEAR STATION
UNIT I

FIGURE 1