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ComEd

March 28, 1997

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Braidwood Nuclear Power Station, Units 1 and 2 Response to Request for Additional Information Regarding First 10-year ISI Facility Operating Licenses NPF-72 and NPF-77, NRC Docket Nos. 50-456 and 50-457

References:

1. Robert A. Capra (USNRC) letter to D. L. Farrar (ComEd), Safety Evaluation of the Inservice Inspection Program Relief Requests Nos. NR-20, NR-21, NR-22 for Braidwood Station Units 1 and 2 (TAC Nos. M 92120, M92121), dated September 1, 1995.

2. H. G. Stanley (ComEd) letter to Document Control Desk (USNRC) Revision 4 of the Inservice Inspection Plan for Braidwood Station Units 1 and 2, dated October 8, 1996.

3. George F. Dick (USRNRC) letter to Irene M. Johnson (ComEd), Request for Additional Information Regarding First 10-Year Inservice Inspection Program - Braidwood Station (TAC Nos. M97134 and M97135), dated March 19, 1997.

4 Teleconference between NRR and ComEd Regarding First 10-Year Inservice Inspection Program - Braidwood Station, on March 21, 1997.

Commonwealth Edison Company's (ComEd's) Braidwood Nuclear Power Station, Units 1 and 2 (Braidwood), performs inservice inspections (ISI) in accordance with Section XI of the 1983 Edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code through the Summer 1983 Addenda (ASME Code), as required by Title 10, Code of Federal Regulations, Part 50, Section 55a, Paragraph g, Subparagraph 4 [10 CFR 50.55a(g)(4)], except where alternatives have been authorized or relief has been requested and granted by the United States Nuclear Regulatory Commission (USNRC).

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As discussed in the teleconference in Reference 4, it is ComEd's understanding the Staff will review and approve Relief Requests Nos. NR-25, NR-26, and NR-30 before the end of the upcoming Unit 1 outage. The Staff is addressing NR-29 separately, as stated in Reference 3, ComEd has requested that this Relief Request be processed as soon as possible to facilitate the vessel inspection during the outage.

Relief Requests NR-27, NR-28, and NR-31 will be issued subsequent to the vessel inspection. Vessel inspection data will be provided to the Staff by April 30, 1997.

The information requested in Reference 3 is provided in Attachment A. The revised Relief Requests are also provided as attachments as well.

Please address any comments or questions regarding this matter to Ms. Patricia A. Boyle, at (815) 458-2801 extension 2519.

Very truly yours,

Gene Stanley

H. Gene Stanley
Site Vice President
Braidwood Nuclear Generating Station

Attachments
o:\confid\sitevp\corresp\1997\97026.doc

cc: A. B. Beach, Regional Administrator - RIII
G. F. Dick, Braidwood Project Manager - NRR
C. Phillips, Senior Resident Inspector - Braidwood
Office of Nuclear Facility Safety - IDNS
Michael T. Anderson, INEL Research Center

Attachment A

NRC RAI Item #3.1:

The staff has determine that licensees must state the specific paragraph of the regulations under which each proposed alternative or request for relief is submitted. The licensee should review the current submittal and provide the required references to ensure that each proposed alternative or request for relief is evaluated in accordance with the appropriate criteria, as discussed below.

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). When submitting a proposed alternative, the licensee must specify the appropriate regulatory basis. Under 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. Under 10 CFR 50.55a(a)(3)(ii), the licensee must show that compliance with the original requirements results in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Examples of hardship 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 increases in examination coverage.

In accordance with 10 CFR 50.55a(g)(5)(iii), a licensee may submit a request for relief from ASME Code requirements. If a licensee determines that conformance with certain ASME 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 this part of CFR to support the criteria for evaluation. The NRC may, giving due consideration to the burden placed on the licensee, impose an alternative examination requirement.

ComEd's Response:

Relief Requests NR-25, NR-26 and NR-30 have been revised and are attached to clearly identify under which provisions of 10 CFR 50.55a the licensee is requesting relief. NR-29 (revision 1) was reviewed and does reference the provisions of 10 CFR 50.55a under which the licensee is requesting relief. NR-29 is currently being addressed separately by the NRC Staff. As discussed in Item #3.2 response, NR-27, NR-28, NR-31 and a new relief request, NR-34 addressing augmented vessel examinations, will be revised/submitted as necessary after completion of the vessel inspections. These revisions will clearly identify the provisions of 10 CFR 50.55a under which the licensee is requesting relief.

NRC RAI Item #3.2:

Based on the review of Relief Requests NR-27, NR-28, and NR-31, that are associated with the reactor pressure vessel examinations, it appears that the Code volumetric coverages are based on those obtained on similar examination at the Byron Station. Requests for relief should be based on actual examination coverages on a plant specific basis. The Code requires that all examinations be performed to the extent practical. This may require a combination of manual and automated examinations.

It should be noted that because the licensee has not submitted plant specific information, these requests are considered unacceptable for review. As such, the licensee may want to consider the withdrawal of the current submittals and resubmit following the actual examinations. Describe the action the licensee proposes to take regarding this observation.

ComEd's Response :

Framatome Technologies (FTI) performed the underwater NDE Inspection Services for the Byron 1 reactor vessel examinations. Braidwood 1 and 2 are similar in design to Byron 1 and 2. Braidwood has contracted FTI to perform the reactor vessel examinations at Braidwood 1 & 2. During FTI's process for preparing for the reactor vessel inspections, as-built drawings were reviewed to develop examination scan plans. These scan plans are used as inputs to the computers which control/drive the underwater inspection equipment. After the scan plans were prepared for Braidwood 1, it was calculated that coverages achieved at Braidwood would be similar to those achieved at Byron. Relief Requests NR-27, NR-28, and NR-31 are based on this information. In addition to providing this information on examination coverage, the intent of submitting these relief requests prior to performing the reactor examination was to get prior approval of proposed examination alternatives.

As discussed during the Reference teleconference, the Braidwood Unit 1 reactor vessel examination is currently scheduled to start in early April and finish in mid April. NR-27, NR-28 and NR-31 are relief requests which address Section XI examination coverage requirements. Subsequent to that inspection, Braidwood will revise relief requests NR-27, NR-28 and NR-31 to specify actual examination coverages, as necessary. These revised Relief Requests will be resubmitted to the Staff by April 30, 1997. Additionally, ComEd will be submitting to the Staff for their review a new relief request, NR-34, which addresses limitations encountered during the 10 CFR augmented reactor vessel examination.

NRC RAI Item #3.3:

Provide the staff with the status of the augmented reactor pressure vessel examinations required by 10 CFR 50.55a(g)(6)(ii)(A), effective September 8, 1992 (Note: plants with greater than 40 months remaining in the interval on the effective date of the rule were required to perform the augmented examination in that interval) and provide a technical discussion describing how the regulation was/will be implemented at Braidwood Station, Units 1 and 2. Include in the discussion a description of the approach and any specialized techniques or equipment that was/will be used to complete the required augmented examination. It should be noted that requests for relief associated with the reactor pressure vessel examinations required by Section XI, Examination Category B-A, Item B1.10, cannot be evaluated until the augmented reactor pressure vessel requirements listed in 10 CFR 50.55a(g)(6)(ii)(A), are satisfied.

ComEd's Response :

At the time of issuance of the requirements for the augmented reactor pressure vessel examinations required by 10CFR50.55a(g)(6)(ii)(A), Braidwood Units 1 and 2 had more than 40 months remaining in their inspection interval; therefore, the site will be conducting the subject RPV examinations to satisfy the current interval Section XI requirements as well as the augmented reactor pressure vessel examinations. . Braidwood will conduct the subject RPV examinations during the A1R06 and A2R06 outages using Framatome Technologies underwater inspection equipment. The extent of coverage for Category B-A, Item B1.10 welds is in accordance with those of ASME Section XI 1989 Edition. The augmented RPV NDE procedures used for these examinations are consistent with the applicable rules of the ISI Code of record. For the Section XI and Augmented RPV examinations, Braidwood is proposing through NR-29 to substitute the qualified PDI examination techniques for those applicable UT techniques specified by the ISI Code of record. NR-31 was submitted to address limitations encountered during the Section XI scans on Item B1.10 welds. A new relief request, NR-34, will be submitted to address the same limitation encountered on Item B1.10 welds performed for the augmented vessel examination requirement.

NRC RAI Item #3.4:

For Request for Relief NR-30, the licensee has proposed to satisfy the Code pressure test for Class 2 piping at containment penetration in conjunction with Appendix J leakage tests. When implementing the alternatives to Code requirements contained in Code Case N-522, Pressure Testing of Containment Penetration Piping, the NRC staff finds this alternative to Code requirements acceptable only if the licensee commits to performing the pressure test at peak design pressure and implements a procedure for the detection and location of through-wall flaws. Describe the action the licensee proposes to take regarding these conditions for acceptance.

ComEd's Response:

The Proposed Alternate Provision section of NR-30 has been revised to include the statement that when this relief request is invoked, testing will be performed at peak design pressure and that procedures for the detection and location of through-wall flaws will be used. Revision 1 of Relief Request NR-30 is attached.

RELIEF REQUEST NR-25

COMPONENT IDENTIFICATION

Code Classes:	1
Reference:	IWB-2500-1
Examination Categories:	B-H
Item Numbers	B8.20
Description:	Alternate rules for the Inservice Inspection of the Pressurizer Seismic Lug Welds.
Component Numbers:	1PZR-01-PSL-01, 1PZR-01-PSL-02 1PZR-01-PSL-03, 1PZR-01-PSL-04 2PZR-01-PSL-01, 2PZR-01-PSL-02 2PZR-01-PSL-03, 2PZR-01-PSL-04

CODE REQUIREMENT

Subsection IWB, Table IWB-2500-1, Examination Category B-H, Item B8.20 requires surface or volumetric examination of Integrally Welded Attachments to the Pressurizer (Reference Figure IWB-2500-15).

BASIS FOR RELIEF

ComEd's Braidwood Nuclear Power Station, Units 1 and 2, conducts ISI activities in accordance with the 1983 Section XI Edition, 1983 Summer Addenda as required by Title 10, Code of Federal Regulations, Part 50, Section 55a, Paragraph g, Subparagraph 4 [10 CFR 50.55a(g)(4)]. Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested on the basis that compliance with the specified Code requirement has been determined to be impractical.

Braidwood Units 1 and 2 Pressurizer seismic lugs are welded to the Pressurizer shell (reference Attachment 1). There are 4 seismic lugs per unit, located 90 degrees apart (reference Attachment 5). In order to perform examinations on the seismic lug welds, the outside surface of the lower vessel shell to lug area must be accessible. The exam surface is not accessible since it is covered by the seismic lug restraint and lower Pressurizer shell insulation(reference Attachment 3 and 4). Also, the configuration of the Pressurizer coffin limits access to the seismic lugs. The impact of removing the seismic lug restraint, altering the Pressurizer coffin and removing the lower shell insulation is presented below.

The seismic restraint (Reference Attachment 1 and 2), which surrounds the lug, prohibits access needed to perform a meaningful surface exam. There are 4 restraints located about the 428' elevation, one for each lug, which were not designed for removal. The top of the concrete floor at this location is at 428' 3" elevation. This floor, which is 2'6" thick, interferes with access to 2 of the 4 lugs (Reference Attachment 2, 3 and 5). Also, the Pressurizer coffin itself severely limits access to remaining 2 seismic restraints (Reference Attachment 5). All of the restraints, which are embedded in the concrete, would require major modification to the existing Pressurizer coffin to allow for removal and access. This modification would require the redesign of the seismic restraint and Pressurizer coffin to allow for periodic removal and access to the seismic restraints. Implementation of this redesign would require significant engineering resources, construction resources and significant dose to plant personnel.

RELIEF REQUEST NR-25 (cont.)

Only the upper panels were designed with clips to provide for removal. Insulation on the lower shell of the Pressurizer prohibits access needed to perform a meaningful surface examination of the seismic lug weld areas. The removal of the insulation covering the lower Pressurizer shell to seismic lug area will result in high radiation exposure to plant personnel. The insulation on the pressurizer consists of panels which are fastened together. The lower panels are fastened together with screws. To provide access from below would require scaffolding from the 401' elevation grating to the 428' elevation of the seismic restraint. Also, to remove the Pressurizer shell insulation would require removal of the screw fasteners. Access to these screws is limited by the floor and Pressurizer coffin(reference Attachment [illegible] and 5). As stated above, the insulation could be removed from the upper portions of the lugs. This can only be accomplished for 2 of the [illegible] seismic lugs, because access is prohibited by the Pressurizer coffin configuration (Reference Attachment 5). The current configuration of the seismic restraint also only allows limited access for visual examination. To provide suitable access for all 4 seismic lug restraints would require major modifications and significant resources.

Even if the non removable insulation is removed (Reference Attachment 3, 4 & 5), full surface examination of the seismic lugs would not be achieved. The Pressurizer coffin, concrete floor and seismic restraint geometry would greatly limit access to all sides. The resulting coverage would only be a small percentage of the weld volume. The limited data obtained from these examinations do not provide a compensatory increase in quality and safety to justify the hazards of personnel radiation exposure to obtain the data. When the removable insulation panels are removed, it is estimated that 5.74" of surface per accessible lug will be achievable. This accessible portion of surface can be visually inspected. It is expected that only a best effort Liquid Penetrant (PT) exam can be performed on the accessible exposed surfaces. Access and clearance interferences will limit how well the surface of the examination volume can be prepped for the PT examination. Because the examination is being performed on slightly rusted carbon steel components, which will receive a best effort surface prep, that a white to pinkish back ground will be expected after developing. Even with a pinkish background, detection of relevant indications will still be possible. Also, bleed out from the lower edge of the non removable insulation will interfere with some of the accessible exam volume. This volume of interference will depend upon the amount of bleed out and will mask any relevant indication.

PROPOSED ALTERNATE PROVISIONS

A VT-1 of the upper surfaces of the 2 accessible lugs when the removable insulation panels are removed. It is estimated that 5.74" per accessible lug (4" of the top and .87" of each side) will be achievable when the removable insulation panels are removed. This is approximately [illegible] of the total exam volume for one lug. Also, a best effort surface inspection (Liquid Penetrant) will be performed on those portion of the lug that are inspectable when the removable insulation panels are removed. In conjunction with the above proposed alternative technique, the periodic VT-2 examinations in accordance with the requirements of ASME Section XI, Table IWB-2500-1, Examination Category B-P and applicable Reactor Coolant system monitoring requirements specified in the Technical Specifications will provide reasonable assurance of continued structural integrity of the Pressurizer shell.

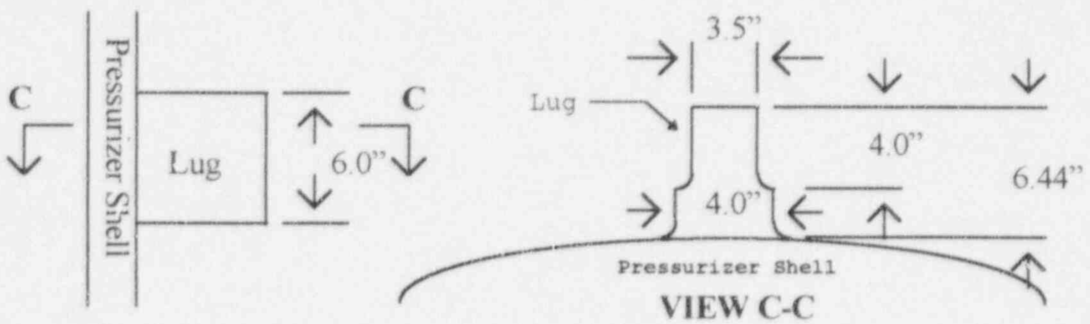
APPLICABLE TIME PERIOD

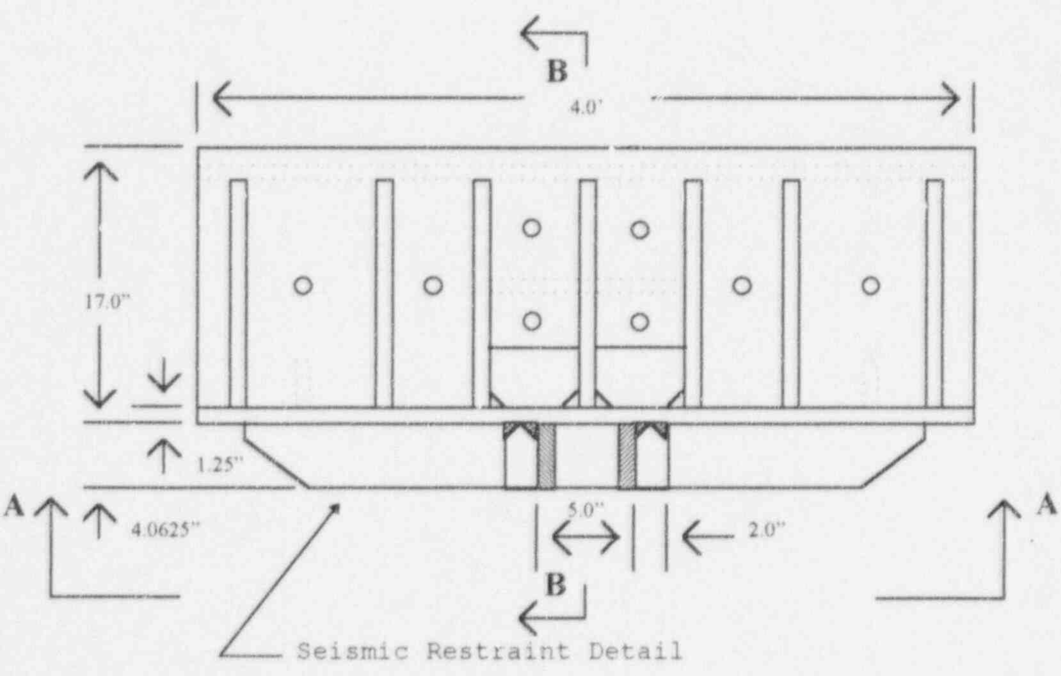
This relief request will be required for the first ten-year Inspection Interval.

APPROVAL STATUS

Pending NRC review

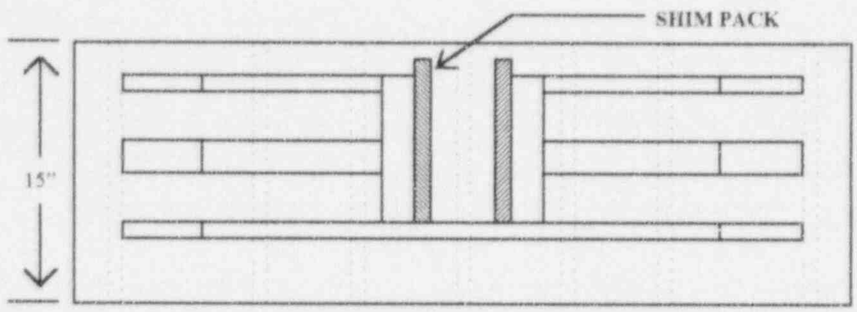
(Drawing not to scale)



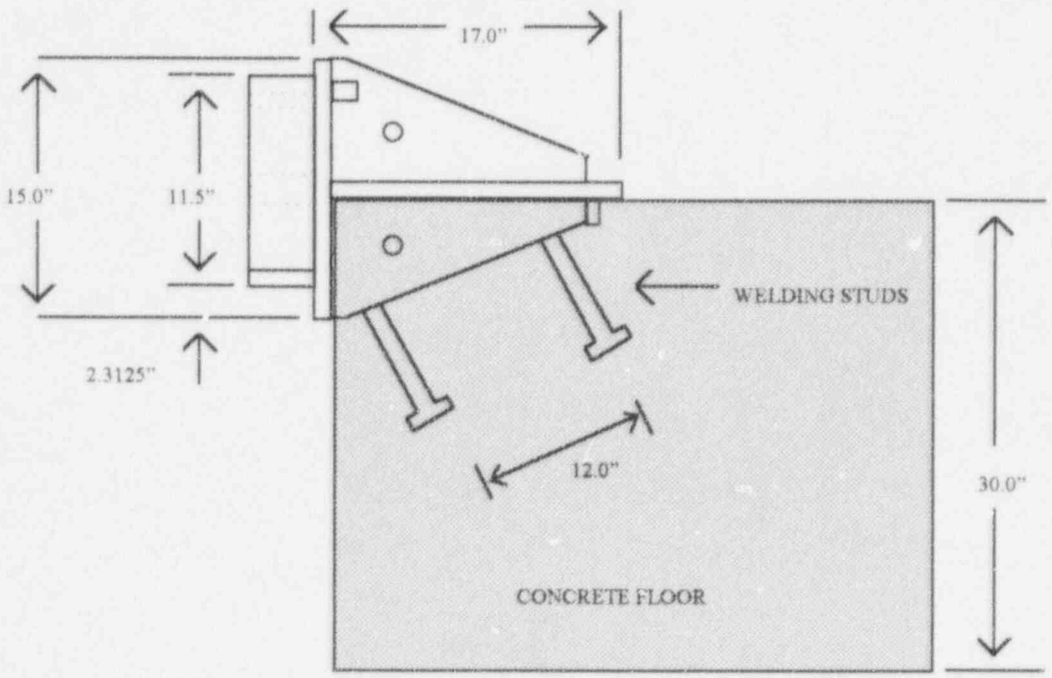


Pressurizer Seismic Lug and Restraint Detail

(Drawing not to scale)



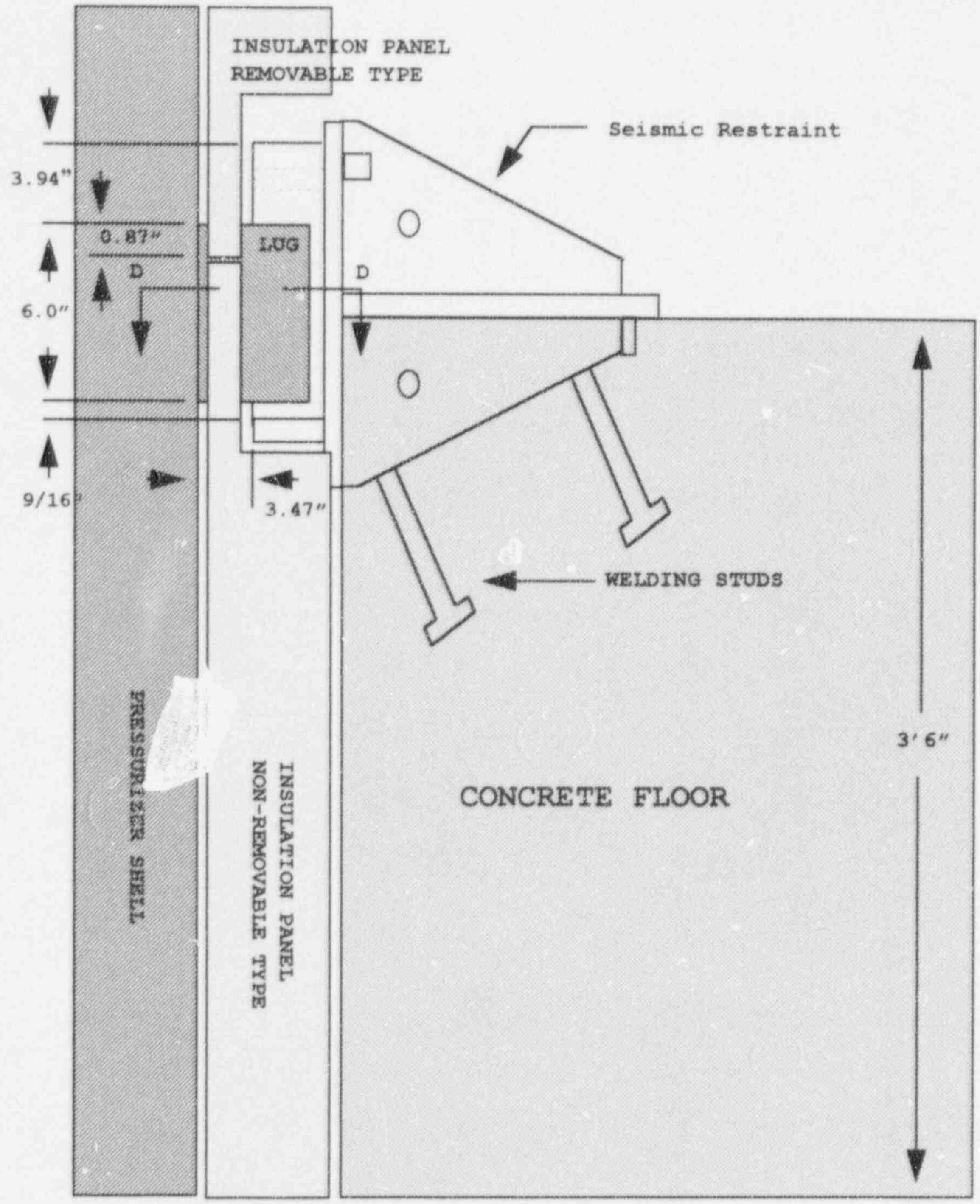
VIEW A-A



VIEW B-B

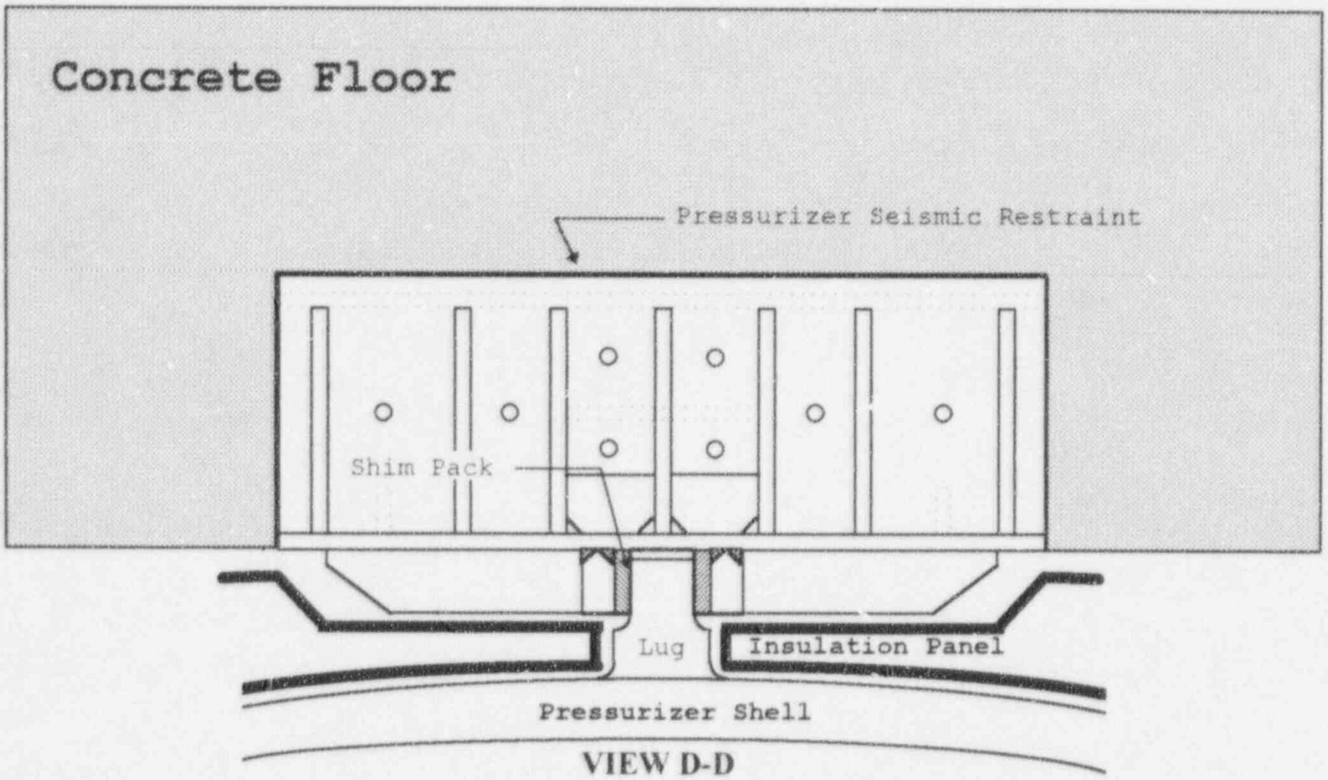
Pressurizer Seismic Lug and Restraint Detail (continued)

(Drawing not to scale)



Pressurizer Seismic Lug and Restraint Insulation Detail

(Drawing not to scale)



Pressurizer Seismic Lug and Restraint Insulation Detail
(continued)

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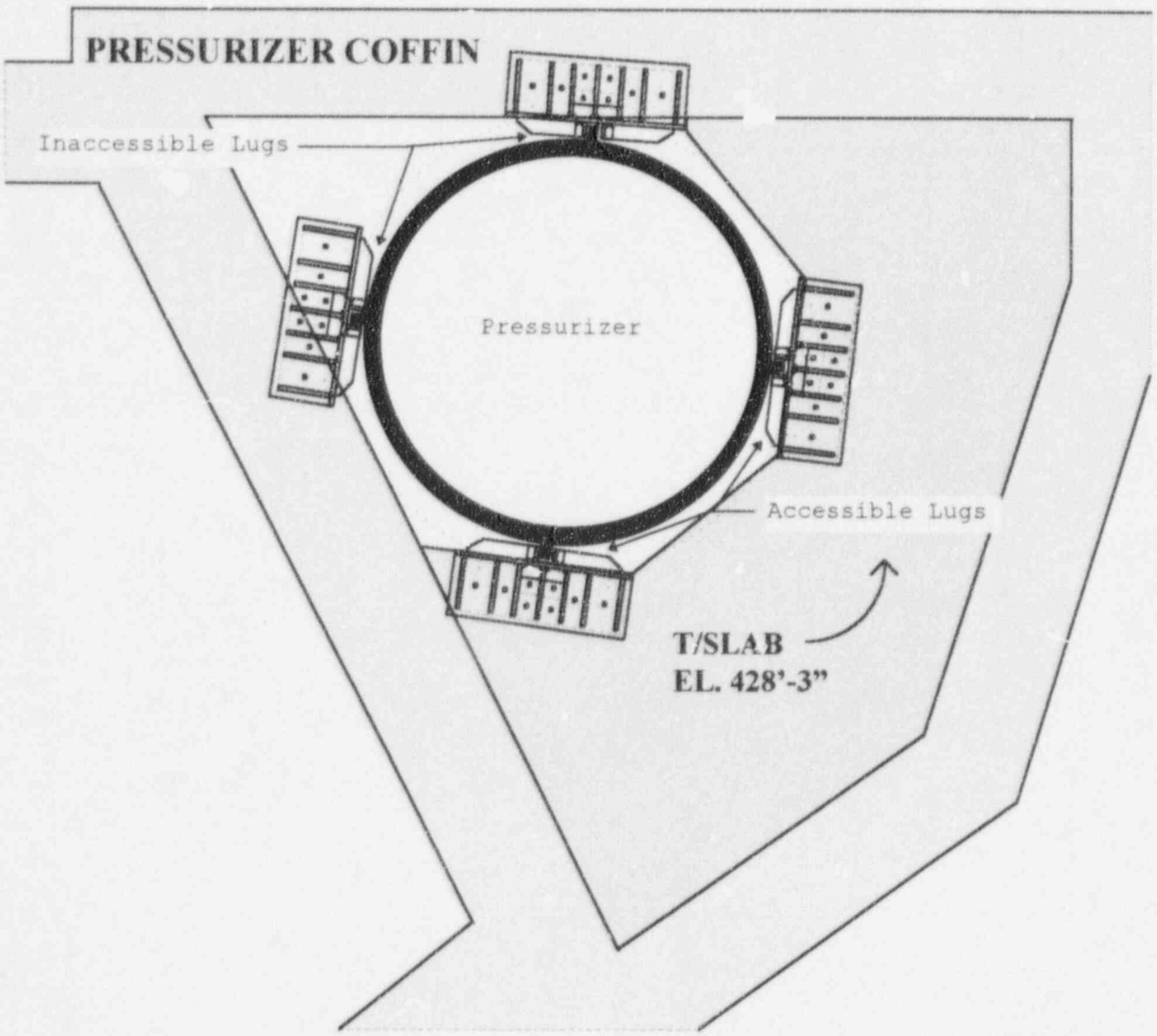


Plan at Elevation 428'-7 1/2"

UNIT 1 AS SHOWN
UNIT 2 OPP. HAND

RELIEF REQUEST NR-26

COMPONENT IDENTIFICATION

Code Class(es):	2
Reference:	IWC-2500-1
Examination Categories:	C-C
Item Numbers:	C3.20
Description:	Alternative rules for the Inservice Inspection of Inaccessible Welds on Welded Attachments
Component Number(s):	Unit 1 Welds: *1FW-06-33, *1FW-10-29, *1FW-11-29, *1FW-12-29 *1RH-03-37B, *1RH-04-73A, *1RH-05-21B, 1RH-07-25A *1RH-08-02A, 1RH-09-45A, *1SI-04-02A, *1SI-12-23A 1SI-26-09A Unit 2 Welds: *2FW-06-01, *2FW-10-24, *2FW-11-25, *2FW-12-25 *2RH-04-03, *2RH-05-35, *2RH-08-03, *2RH-09-28 *2SI-04-03. *2SI-12-19A, *2SI-26-03 (*) denotes welds selected for inspection during the interval.

CODE REQUIREMENT

Subsection IWC, Table IWC-2500-1, Examination Category C-C, Item C3.20 requires surface examination of the Integrally Welded Attachments to Piping (Reference Figure IWC-2500-5). |

BASIS FOR RELIEF

ComEd's Braidwood Nuclear Power Station, Units 1 and 2, conducts ISI activities in accordance with the 1983 Section XI Edition, 1983 Summer Addenda as required by Title 10, Code of Federal Regulations, Part 50, Section 55a, Paragraph g, Subparagraph 4 [10 CFR 50.55a(g)(4)]. Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is being requested on the basis that compliance with the specified Code requirement has been determined to be impractical.

Some penetrations at Braidwood were originally designed where one of the integral attachment welds is inside the flued head penetration assembly, thus making the welds inaccessible for inservice inspection. Access from outside of the closed end of the flued head penetration assembly for examiners is prohibited by the integral attachment. Access from the open end of the penetration is severely restrained due to geometry and clearance. See Attachments 1, 2, 3, 4 and 5 for penetration details. The integral attachment weld is set back some distance inside the flued head penetration assembly and the clearance between the pipe and penetration sleeve is small. See Table 1 on Attachment 6.

To satisfy the Code requirement to perform a surface examination of this weld, modification to the flued head penetration assembly and/or piping to allow access would be required. Braidwood would incur significant engineering and installation costs to perform such a modification without a compensating increase in the level of quality and safety to justify such modifications.

RELIEF REQUEST NR-26 (cont.)

PROPOSED ALTERNATE PROVISIONS

When a weld is scheduled for inspection, a surface examination of the accessible weld on the exposed outside surface of the penetration will be performed. In conjunction with the above proposed alternative technique, the periodic VT-2 examinations in accordance with the requirements of ASME Section XI, Table IWC-2500-1, Examination Category C-H will provide reasonable assurance of continued structural integrity of the piping systems.

APPLICABLE TIME PERIOD

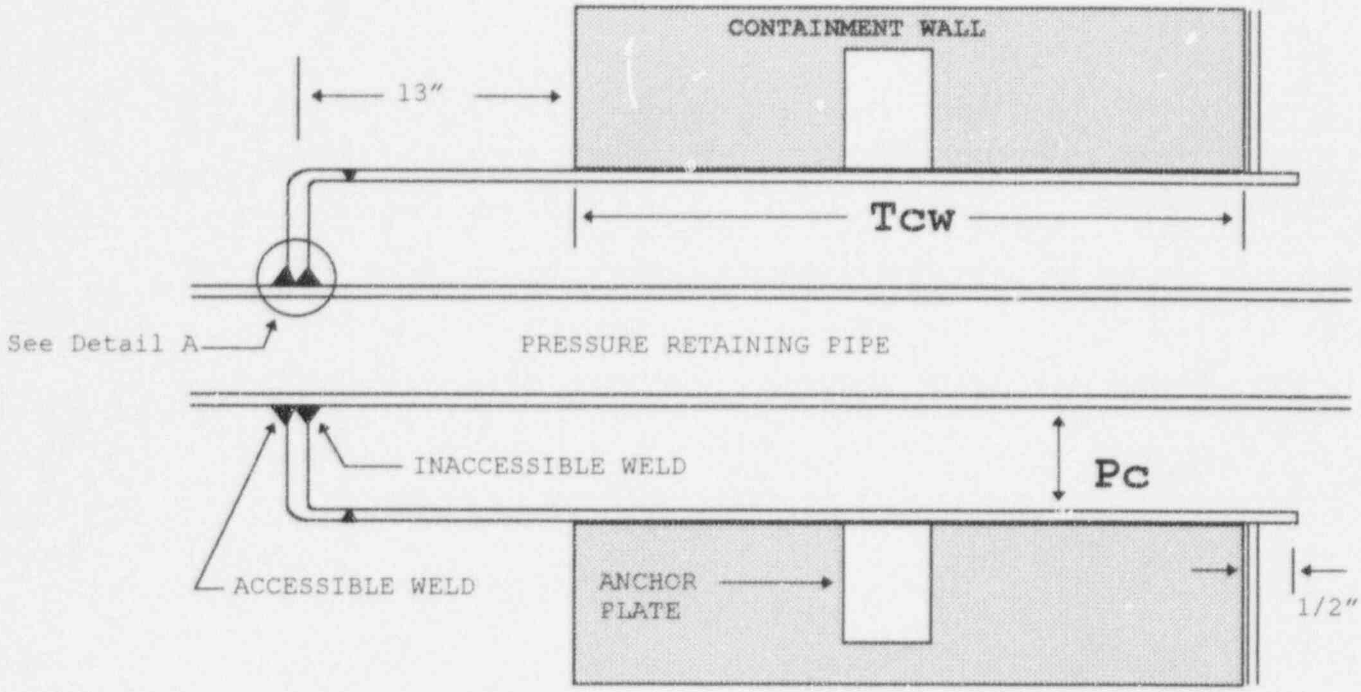
This relief request will be required for the First Ten Year Inspection Interval.

APPROVAL STATUS

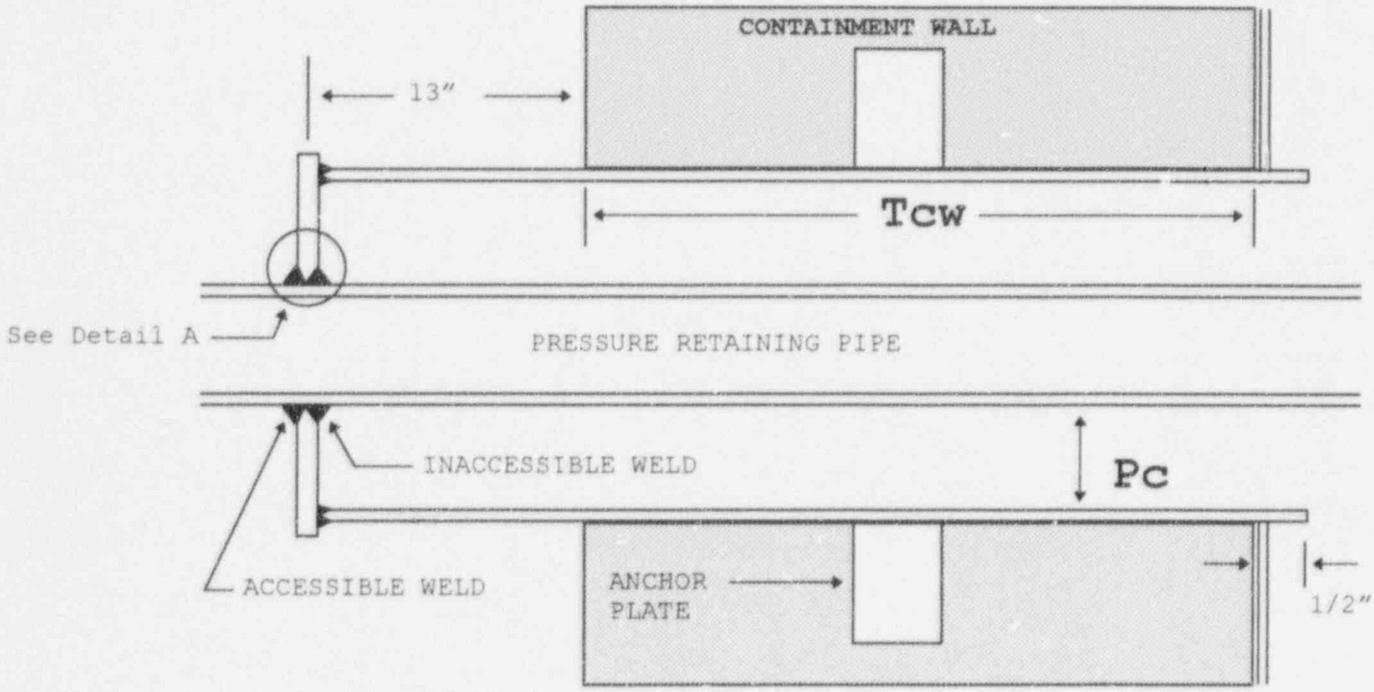
Pending NRC review.

ATTACHMENT 1

(Drawing not to scale)

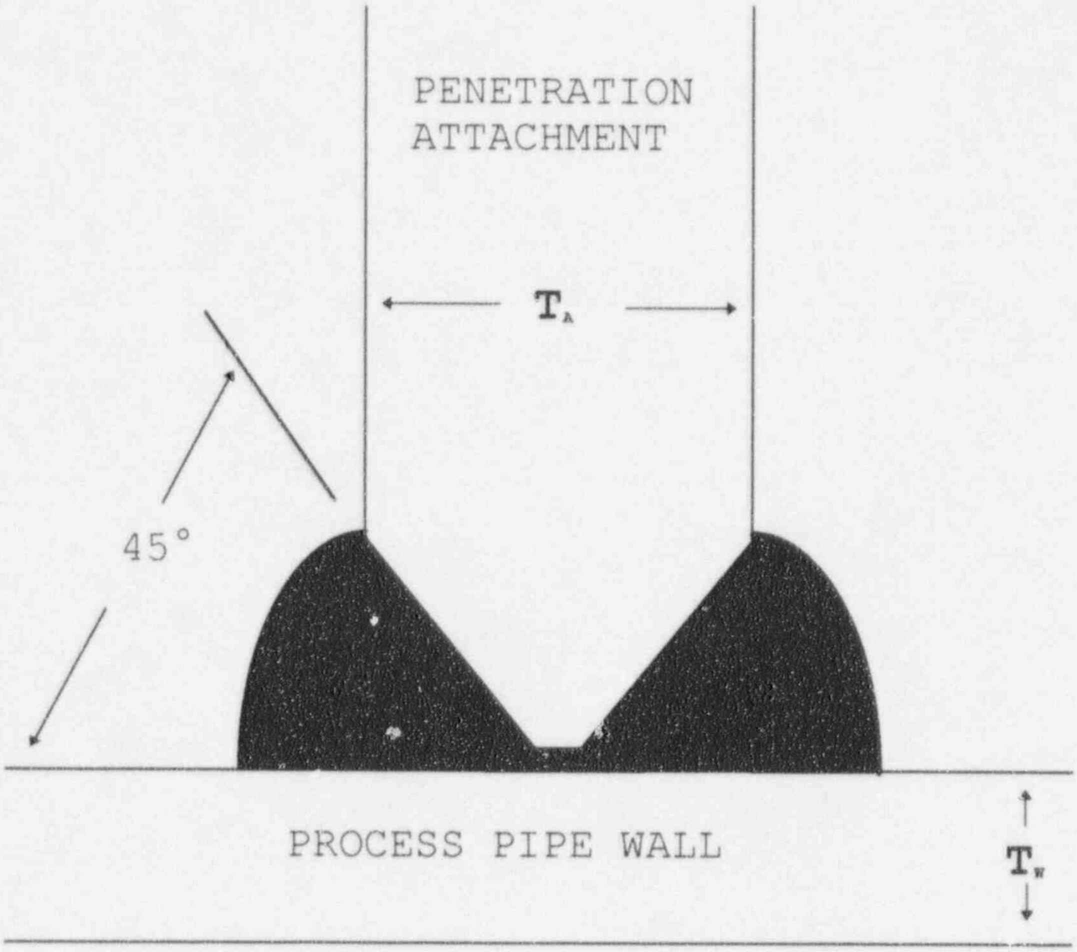


TYPICAL TYPE 2 PENETRATION



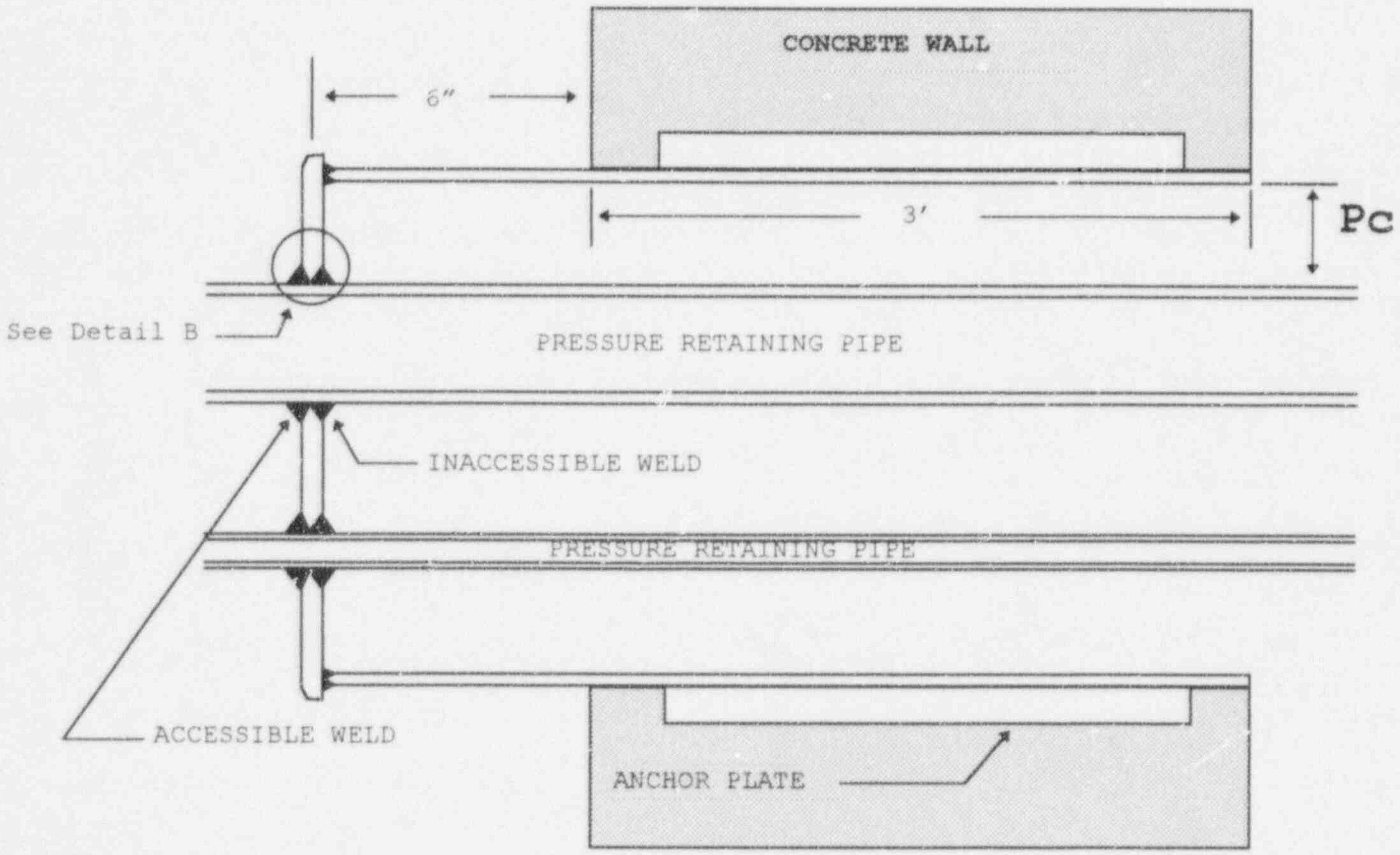
TYPICAL TYPE 3 PENETRATION

(Drawing not to scale)



DETAIL A

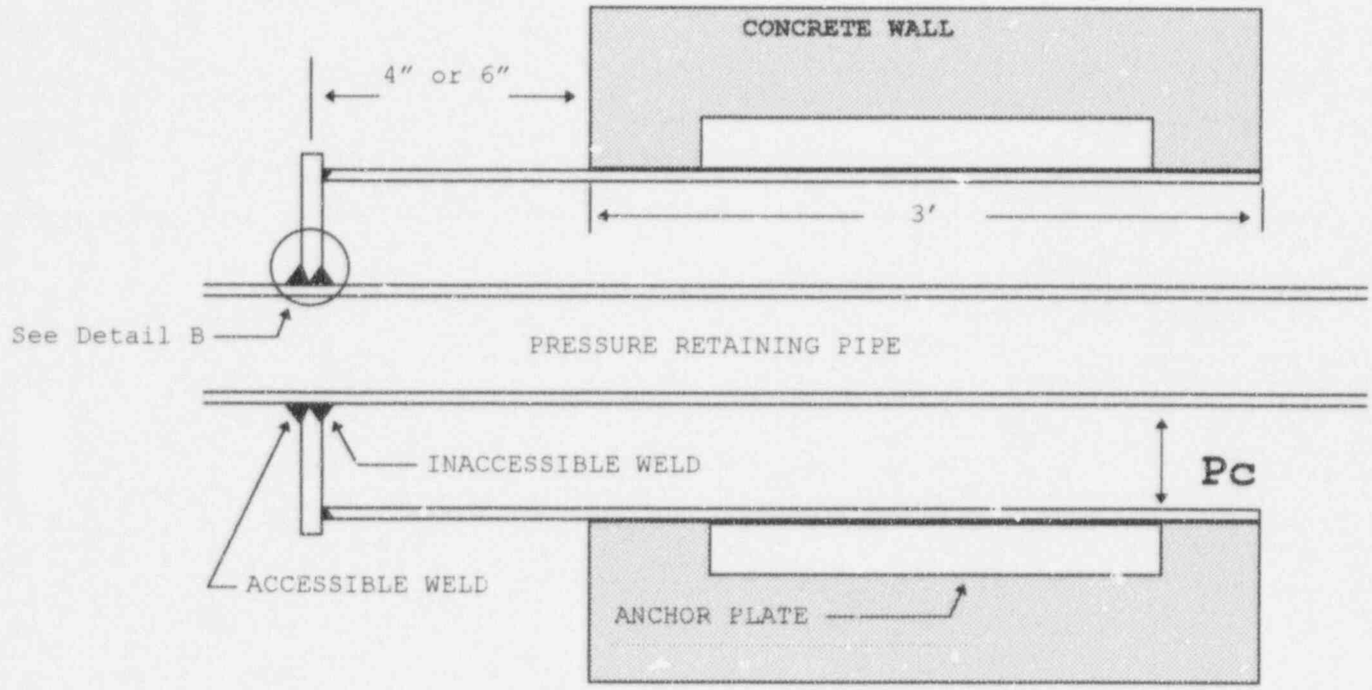
(Drawing not to scale)



TYPICAL PENETRATION (DETAIL 27)

ATTACHMENT 4

(Drawing not to scale)

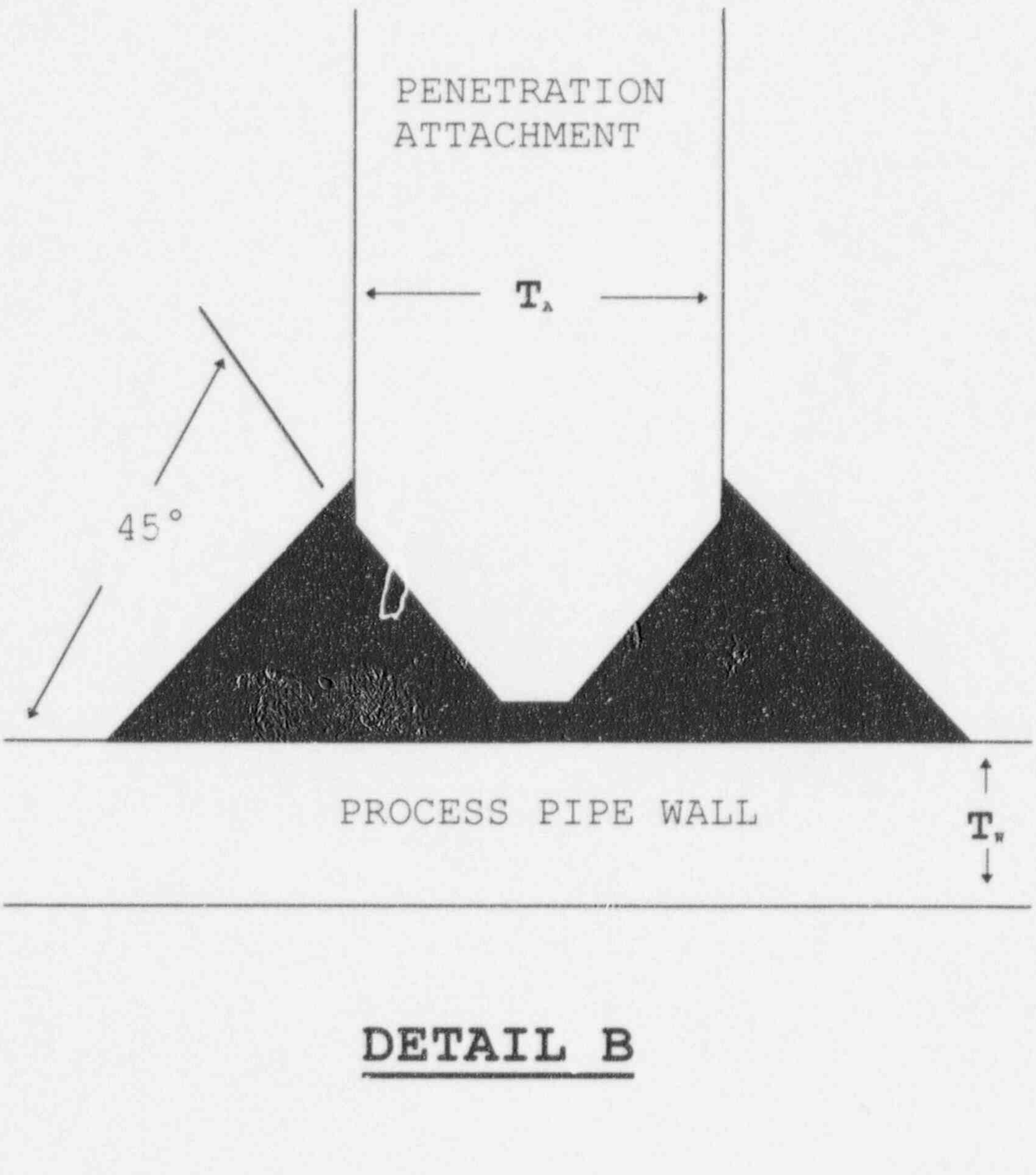


TYPICAL PENETRATION (DETAIL 4)

TABLE 1

Weld ID	Pipe Size	Pipe Thickness T_W	Attachment Thickness T_A	Thickness Concrete T_{CW}	Penet. Size	Penet. Clearance P_C
TYPE 2 PENETRATIONS (ATTACHMENT 1)						
1RH-04-73A	12″	0.375″	2.0″	3′6″	24″	4.9″
1RH-08-02A	12″	0.375″	2.0″	3′6″	24″	4.9″
1SI-04-02A	8″	0.906″	2.0″	3′6″	24″	7.0″
1SI-12-09A	12″	1.125″	2.0″	3′6″	24″	4.9″
2RH-04-03	12″	0.375″	2.0″	3′6″	24″	4.9″
2RH-08-03	12″	0.375″	2.0″	3′6″	24″	4.9″
2SI-04-03	8″	0.906″	2.0″	3′6″	24″	7.0″
2SI-12-19A	12″	1.125″	2.0″	3′6″	24″	4.9″
TYPE 3 PENETRATIONS (ATTACHMENT 1)						
1FW-06-33	6″	0.432″	2″	4′6″	16″	3.8″
1FW-10-29	6″	0.432″	2″	4′6″	16″	3.8″
1FW-11-29	6″	0.432″	2″	4′6″	16″	3.8″
1FW-12-29	6″	0.432″	2″	4′6″	16″	3.8″
1SI-26-09A	8″	0.906″	2″	3′6″	24″	7.0″
2FW-06-01	6″	0.432″	2″	4′6″	16″	3.8″
2FW-10-29	6″	0.432″	2″	4′6″	16″	3.8″
2FW-11-29	6″	0.432″	2″	4′6″	16″	3.8″
2FW-12-29	6″	0.432″	2″	4′6″	16″	3.8″
2SI-26-03	8″	0.906″	2″	3′6″	24″	7.0″
DETAIL 4 PENETRATIONS (ATTACHMENT 4)						
1RH-03-37B	8″	0.375″	1″	3′0″	18″	4.3″
1RH-07-25A	8″	0.375″	1″	3′0″	18″	4.3″
DETAIL 27 PENETRATIONS (ATTACHMENT 3)						
1RH-05-21B	8″	0.375″	1″	3′0″	20″	5.2″
1RH-09-45A	8″	0.375″	1″	3′0″	20″	5.2″
2RH-05-35	8″	0.375″	1″	3′0″	20″	5.2″
2RH-09-28	8″	0.375″	1″	3′0″	20″	5.2″

(Drawing not to scale)



DETAIL B

RELIEF REQUEST: NR-30

COMPONENT IDENTIFICATION

Code Class:	2
Reference:	Table IWC-2500-1
Examination Category:	C-H
Item Numbers:	C7.30, C7.40, C7.70, and C7.80
Description:	Alternate Rules to Table IWC-2500-1, Category C-H: Pressure Testing of Containment Penetration Piping with Attached Nonclassed Piping

CODE REQUIREMENT

ASME Section XI, Table IWC-2500-1, Examination Category C-H, requires the performance of a visual VT-2 examination during a system pressure test on Code Class 2 pressure retaining components. Note 7 of this table states, "The pressure boundary includes only those portions of the system required to operate or support the safety system function up to and including the first normally closed valve (including a safety or relief valve) or valve capable of automatic closure when the safety function is required."

BASIS FOR RELIEF

ComEd's Braidwood Nuclear Power Station, Units 1 and 2, conducts ISI activities in accordance with the 1983 Section XI Edition, 1983 Summer Addenda as required by Title 10, Code of Federal Regulations, Part 50, Section 55a, Paragraph g, Subparagraph 4 [10 CFR 50.55a(g)(4)]. Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative would provide an acceptable level of quality and safety.

Specifically, Braidwood Station requests relief to perform 10 CFR 50 Appendix J leakage testing in lieu of the pressure test required by ASME Section XI, Table IWC-2500-1, Examination Category C-H on the Code Class 2 Containment Penetration piping with attached nonclassed piping.

The applicable components are piping lines and valves which are portions of non-safety related systems that penetrate the primary reactor containment. At each containment penetration, the process pipe is classified Code Class 2 and provided with isolation valves that are either locked shut during normal operation, capable of automatic closure, or capable of remote closure to support the containment safety function. The balance of piping outside the isolation valves is non-code and therefore outside the scope of the ASME Boiler & Pressure Vessel Code, Section XI. These components perform no other safety function. The only reason that the penetration piping is classified as Class 2 is because of its function as part of the containment pressure boundary. The remaining portion of the system is non nuclear related and the integrity of the system in relation to its primary function is not within the scope of Section XI. Since containment integrity is the only safety related function, it is logical to test the Class 2 penetration portion of the system to the Appendix J criteria.

The primary reactor containment integrity, including all containment penetrations, is periodically verified by performing leakage tests in accordance with 10 CFR 50, Appendix J. The Appendix J test frequency provides assurances that the containment pressure boundary is being maintained at an acceptable level while monitoring for deterioration of seals, valves and piping. If a pipe existed with a through-wall

RELIEF REQUEST NR-30 (cont.)

flaw, the isolation valves located on both sides of the containment wall would prevent any release outside containment. Multiple through-wall flaws or leakage paths occurring simultaneously inside and outside of containment between the isolation valves in a pipe segment is unlikely. Each of the Code Class 2 lines and their associated isolation valves are tested during an Appendix J leakage test at a pressure not less than 44.4 psig (Peak calculated containment pressure). The Appendix J leakage tests are performed at intervals in accordance with the requirements of the Braidwood Technical Specifications.

Performance of these Appendix J leak tests will verify the integrity of the subject Code Class 2 lines and valves at the Containment penetrations. The performance of ASME Section XI, Examination Category C-H pressure tests on these same lines will provide little, if any, additional verification of primary reactor containment integrity and impose a burden of duplicate testing. Duplicate testing results in a significant increase in total amount of work force and radiological exposure without a compensating increase in the level of quality or safety.

Per the preceding information, Braidwood Station requests relief to use the Appendix J test as an optional alternative to ASME Section XI requirements for pressure testing the Code Class 2 containment penetration components on the basis that the Proposed Alternate Provisions provide an acceptable level of quality and safety. The proposed alternative is consistent with the requirements of Code Case N-522.

PROPOSED ALTERNATE PROVISIONS

Braidwood Station will perform 10 CFR 50, Appendix J leakage tests as an optional alternative to the Section XI required pressure test on the subject primary reactor containment penetration piping and associated valves. When implementing the Appendix J leakage test and invoking this relief request, peak design pressure and procedures for the detection and location of through-wall flaws will be used.

PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for the first inspection interval.

APPROVAL STATUS

Pending NRC review.