GPU Nuclear

GPU Nuclear Corporation
Route 441 South
P.O. Box 480
Middletown, Pennsylvania 17057-0480
(717) 944-7621
Writer's Direct Dial Number:

717-948-8005

February 16, 1993
C311-93-2004

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

Gentlemen:

Subject: Three Mile Island Nuclear Station Unit 1, (TMI-1)
Docket No. 50-289
Operating License No. DPR-50
Relief from ASME Section XI Code Inservice Inspection (ISI)
Requirements

The NRC Safety Evaluation Report (SER) dated October 8, 1992 responded to GPU Nuclear's April 19, 1991 submittal of the revised Inservice Inspection (ISI) Program for the Second 10-Year Interval. The revised program submittal contained requests for specific relief where Code requirements were found to be impractical. The ISI program was prepared to comply with the 1986 Edition of the ASME Section XI Code.

In accordance with 10 CFR 50.55a(g)(5)(iii), if the licensee determines that conformance with certain Code requirements is impractical for its facility, the licensee shall notify the Commission and submit information to support the determination. 10 CFR 50.55a(a)(3) states that proposed alternatives to the Section XI requirements may be used, when authorized by the NRC, if the applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety or that compliance with the requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The purpose of this letter is to request additional relief from the requirements of ASME Section XI Code beyond that which was granted by the SER. Attached are three specific requests for relief from Code requirements which are consistent with the criteria of 10 CFR 50.55a(a)(3). These requests are summarized as follows:

A. Item No. 1 is a request for relief from the schedular requirements of the ASME Section XI Code to perform a hydrostatic test of the Class 3 Main Steam System piping segments extending from MS-V2A/B to the turbine driver for the Turbine-driven Emergency Feedwater (EFW) Pump, EF-P1, during each 10-year interval. GPU Nuclear requests relief to defer the

test to TMI-1's Cycle 11 Refueling Outage (11R), currently scheduled for September 1995. This will result in a postponement of the test approximately five months beyond the one year extension allowed by the Code.

GPU Nuclear's request for relief, provided in Attachment 1, demonstrates the impracticality of the current schedule for performing this test based on 1) alternative testing, 2) hardship or unusual difficulty, 3) no benefit to be gained from the test, and 4) the likelihood that the requirement for this test will be deleted by the ASME and NRC prior to Outage 11R.

B. Item No. 2 is an extension of relief granted by the SER for Relief Request No. 15 of GPU Nuclear's April 19, 1991 submittal. Additional relief is sought from the requirement to remove insulation from the Pressurizer heater connections during each refueling outage. Removal of the insulation from around the Pressurizer heaters would subject the heater cables to potential damage without resulting in a commensurate safety benefit. A significant number of heaters, approximately one-third, have already been lost. Although loss of heaters to date cannot be attributed directly to maintenance activities, wear and tear of the heater cable jackets has been noted to result from activities such as removal and replacement of insulation.

GPU Nuclear proposes to perform a VT-2 Leakage Test each refueling interval during Hot Shutdown conditions without removing the insulation. As described in the attachment, removal of the Pressurizer insulation around the heater connections during each refueling outage is not practical due to the potential for damage to heater cables and connectors that could result from these additional maintenance activities.

C. Item No. 3 is a request for relief from the requirement to perform a VT-2 Leakage Test of two 6" EFW piping segments running from the EF-V30 Control Valves (EF-V30A, EF-V30B, EF-V30C, and EF-V30D) to the EF-V12 Check Valves (EF-V12A and EF-V12B) at nominal operating pressure. Approval is requested to allow a reduced pressure VT-2 Leakage Test of affected piping at nominal Once Through Steam Generator (OTSG) pressure or slightly below (~900 psig). The proposed test pressure is approximately 200 psig below nominal EFW System operating pressure.

There are no alternatives which would significantly reduce the impact of extending the EFW Pump run time sufficient to accomplish a VT-2 Leakage Test of the affected piping without the risks involved with introducing cold oxygenated water into the OTSGs. Modification of the tilting disk EF-V12 check valves or fabricating temporary plugs is also considered to be impractical.

Testing these components to the Code criteria would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Compared to the risk involved, the benefit to be gained by conducting the test at a slightly higher test pressure clearly does not justify the actions that would be needed to meet the Code requirement.

The three specific issues described above were not discussed in the previous GPU Nuclear request because the need for relief had not been identified. Copies of the ISI drawings listed in the attached relief requests are being provided to the NRC Project Manager. In order to support our planning effort for the TMI-1 Cycle 10 Refueling (10R) Outage, NRC approval is needed prior to May 1993.

Sincerely,

T. G. Broughton
Vice President and Director, TMI-1

MRK

Attachment

cc: Administrator, Region I
TMI-1 Senior Resident Inspector
TMI-1 Senior Project Manager

RELIEF REQUEST 1

COMPONENT DESCRIPTION:
Main Steam System piping from MS-V2A/B to the turbine driver, EF-U1, for the turbine-driven Emergency Feedwater (EFW) Pump (EF-P1)

ISI CLASS: 3

ISI DRAWING NO.: 1D-ISI-FD-001 - Main Steam System and Drainage

DESIGN PRESSURE:
Piping header from MS-V2A/B to MS-V6 - 1050 psig
Piping header from MS-V6 to EF-U1 - 750 psig

NORMAL OPERATING PRESSURE:
During normal operation MS-V13A/B and MS-V10A/B are closed and piping from MS-V2A/B to MS-V13A/B and MS-V10A/B experiences Once Through Steam Generator (OTSG) pressure of approximately 900 psig. During surveillance testing of the turbine-driven Emergency Feedwater (EFW) Pump (EF-P1), MS-V13A/B is open and pressure from MS-V13A/B and MS-V10A/B to MS-V6 is approximately equal to OTSG pressure. MS-V6 controls the pressure from MS-V6 to EF-U1 at approximately 165 psig.

CODE REQUIREMENT:
The 1986 Edition of ASME Section XI, Table IWD-2500-1 Examination Category D-B, Item No. D2.10 requires a hydrostatic test of Class 3 piping each 10-year interval. In accordance with IWD-5223(a), the Hydrostatic test pressure is 1.25 times the lowest pressure setting among the safety or relief valves provided for overpressure protection within the boundary of the system to be tested. The piping from MS-V2A/B to MS-V6 is protected by the Main Steam Safety Valves (MSSVs). The lowest MSSV setpoint is 1040 psig. The piping from MS-V6 to EF-U1 is protected by safety valves MS-V22A/B. MS-V22A is set at 260 psig; MS-V22B set at 280 psig.

RELIEF BEING REQUESTED:
GPU Nuclear requests schedular relief for the above requirement until the TMI-1 Cycle 11 Refueling Outage (11R) pending action by the ASME Section XI Code Committee to eliminate this requirement for a hydrostatic test. Outage 11R is currently scheduled to begin in September, 1995. This would result in an extension of approximately five months in addition to the one year extension allowed by the Code.

It is noteworthy that TMI-1 lost four months from the current period of the ISI program as a result of the update to the 1986 Edition of the Code on April 20, 1991. This shift of the schedule occurs because the newer Edition of the Code divides the 10-year ISI interval into three unequal periods (1st period = 36-months, 2nd period = 48-months, and 3rd period = 36-months) rather than three equal 40-month periods.

ALTERNATIVE EXAMINATION:

As an alternative to the hydrostatic test of the Main Steam System piping from MS-V2A/B to EF-U1, an Operational VT-2 Leak Test will be performed in conjunction with quarterly pump and valve IST of the EFW System when main steam is available.

BASIS FOR REQUESTING RELIEF:

A hydrostatic test of the piping between MS-V2 and EF-U1 would require blanking or gagging of relief valves MS-V22A/B, blanking of the EF-U1 turbine discharge, and pinning of the affected piping supports. The piping from MS-V2A/B to MS-V6 would require a hydrostatic test pressure of 1300 psig (1.25 x 1040 psig). The piping from MS-V6 to EF-U1 would require a hydrostatic test pressure of 325 psig (1.25 x 260 psig). If MS-V6 seat leakage exceeds the capacity of the hydrostatic test pump, the required hydrostatic test pressure for the higher pressure upstream piping may not be achievable. GPU Nuclear believes that it would be impractical to remove or blank the relief valves, blank the EFW steam turbine, and pin the affected piping supports only for the purpose of performing the Code-required hydrostatic test of these relatively short sections of piping. MS-V2A/B to MS-V6 is approximately 50 feet; MS-V6 to EF-U1 is approximately 40 ft of piping.

ASME Code Case N-498, which allows substitution of operational leakage tests for hydrostatic tests of Class 1 and 2 piping, has been approved by the NRC. A revision of Code Case N-498 has been drafted which will eliminate the hydrostatic test requirements for Class 3 piping. The revised Code Case now awaits approval of the ASME Section XI Code Committee.

GPU Nuclear considers a Section XI Code hydrostatic test of the Main Stea. System piping from MS-V2A/B to the EFW turbine would be a hardship with essentially no benefit. Relief is needed for Cycle 10 operation in order to use the alternative test for this relatively short run of Class 3 piping. It is likely that Code Case N-498 will be revised to allow the alternative testing for Class 3 piping before the end of TMI-1 Cycle 10 which is currently scheduled to end in September, 1995. Therefore, relief is being requested until the TMI-1 Cycle 11 Refueling Outage (11R) pending action by the ASME Section XI Code Committee to eliminate the requirement for a hydrostatic test.

It is recognized in Code Case N-498 that the increase in pressure to perform a hydrostatic test does not significantly stress or challenge the pressure boundary. The use of an alternative to the hydrostatic test would provide an equivalent assurance of safety and avoid potential damage to system components from removal/blanking of relief valves, blanking off of the Turbine-driven EFW Pump, and pinning the piping supports that would be affected.

GPU Nuclear has completed approximately thirteen (13) system hydrostatic tests at TMI-1 and never found weld or piping through wall leakage. We believe that there would be insufficient benefit in performing a hydrostatic test of this piping compared to the VT-2 leakage examination at nominal operating pressure. Leakage would not be appreciably more apparent using the hydrostatic pressure test as compared with a VT-2 visual examination at nominal operating pressure because the applicable system hydrostatic test pressure would not apply significant additional stress to the piping.

To assure compliance with Code requirements without action by the Code Committee followed by NRC approval, a hydrostatic test of the Main Steam System piping from MS-V2A/B to the EFW turbine must be performed prior to April 19, 1994. Without Code relief, the last scheduled outage for performing this test is 10R which is scheduled for September 1993. GPU Nuclear requests that the NRC approve relief from Code requirements and grant an extension of the ISI schedule for performing this hydrostatic test until Outage 11R, which is scheduled for September 1995. GPU Nuclear believes that action will be taken by the ASME and NRC to eliminate the Code requirement for hydrostatic test of Class 3 Systems. Extension of the ISI schedule for this test until Outage 11R would allow additional time[1] for the ASME Section XI Code committee to take action and approve the revision of Code Case N-498.

GPU Nuclear concludes that hydrostatic testing of Main Steam System piping from MS-V2A/B to the EFW turbine would not significantly challenge the pipe integrity and would result in hardship or unusual difficulty without yielding a compensating increase in the level of quality and safety.

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[1] This amounts to a five month extension in addition to the 12 month extension allowed by the Code to permit examinations to coincide with plant outages.

RELIEF REQUEST 2

COMPONENT DESCRIPTION:
Pressure Retaining Bolted Connections (Pressurizer Heaters)

ISI CLASS: 1

ISI DRAWING NO.: ID-ISI-FD-019

DESIGN PRESSURE: 2500 psig

NORMAL OPERATING PRESSURE: 2155 psig

CODE REQUIREMENTS:
The 1986 Edition of ASME Section XI, Table IWB-2500-1, Examination Category B-P, Item B15.20 and B15.21 requires a system leakage test each refueling outage and a hydrostatic test once each ten year interval, respectively. IWA-5242 states, "For systems borated for the purpose of controlling reactivity, insulation shall be removed from pressure retaining bolted connections for visual examination VT-2."

The NRC Safety Evaluation Report (SER) dated October 8 1992 granted relief "from the VT-2 Visual Test for the Class 1 bolted connections provided the Licensee's alternative examination require: (1) a minimum of 4 hours at nominal operating pressure before the VT-2 visual examination of the insulated connections, and (2) that the insulation be removed from the affected Class 1 bolted connections during each refueling outage for a VT-3 visual examination."

RELIEF BEING REQUESTED:
GPU Nuclear requests additional relief from the requirement imposed by the SER to remove insulation from the Pressurizer in the area near the Pressurizer heater connections for VT-3 Visual Test at Cold Shutdown during refueling outages. This request is an extension of Relief Request No. 15 from GPU Nuclear's April 19, 1991 submittal where relief has been granted by the SER.

ALTERNATIVE EXAMINATION:
GPU Nuclear proposes to perform a VT-2 Leakage Test of the Pressurizer in the area around the Pressurizer heater connections each refueling interval during Hot Shutdown conditions without removing the insulation.

BASIS FOR REQUESTING RELIEF:
Removal of the Pressurizer insulation from around the Pressurizer heater connections due to the potential for damage to heater cables and connectors that could result from the maintenance activities that would be required.

The Pressurizer insulation in the affected area is made up of fiberglass cloth covered by stainless steel sheet metal. The Pressurizer heater

electrical connections are potted in ceramic material. To remove and reinstall the insulation requires the use of scaffolding. Removal of the insulation in this area poses the risk that these fragile components could become damaged by the sheet metal or the scaffold installation and removal process.

There is a limited number of heaters that can be lost due to failure of the element or connector pins without costly and radiologically significant repairs to replace an entire heater bundle. A significant number of heaters, approximately one-third, have already been lost. Although loss of heaters to date cannot be attributed directly to maintenance activities, wear and tear of the heater cable jackets has been noted to result from activities such as removal and replacement of insulation. Figure 1 shows the configuration of a heater bundle (one of three total).

Currently only 79 of the 117 elements remain inservice. (63 of these heater elements are required to maintain an RCS heatup rate of 100°F/hr, the maximum Pressurizer heatup rate allowed by TMI-1 Technical Specification 3.1.2.3). Exposure of the heater cables to potential damage from removal and replacement of insulation poses a risk that could result in costly repairs and possibly additional critical path outage time to install a new heater bundle assuming the parts were available.

Removal of the insulation from around the Pressurizer heaters would subject the heater cables to potential damage without contributing a commensurate safety benefit. The possibility of leakage is very low for this closure for the following reasons (see Figure 1):

1) As depicted in Detail F, the seal weld between the diaphragm (Part 13) and the Pressurizer heater belt shell (Part 4) makes this closure different from the typical bolted connection which normally uses a gasket. The cover plate (Part 12) studs (Part 75) provide clamping force on the cover and diaphragm.

2) As depicted in Detail G, each immersion heater (Part 28) is seal welded to the diaphragm (Part 13). If this seal weld were to leak, this could be seen without insulation removal because the ends of the immersion heaters, where they penetrate the cover (Part 12), are not insulated.

Granting of additional relief would result in personnel exposure savings in addition to the amount of time craft personnel would need to spend in respirators in an area where they are subjected to heat stress. Although the personnel exposure dose associated with removal and reinstallation of Pressurizer insulation would not be expected to be high, there would be a benefit toward keeping doses as low as reasonably achievable (ALARA) if insulation removal were not required. GPU Nuclear considers that Pressurizer heater insulation removal presents a hardship where these savings are significant because there is essentially no benefit to removing the insulation for inspection purposes.

The damage to the Pressurizer heaters which has occurred at TMI-1 affects the electrical capability of the heaters, but would not be expected to cause leakage. On October 24, 1991 at Cold Shutdown during the Cycle 9 Refueling Outage (9R), the insulation was removed from the three heater bundles and a VT-2 Leakage Test was performed. No leakage or unusual conditions were found. TMI-1 has never experienced leakage from Pressurizer heater connections.

GPU Nuclear concludes that the benefit to be derived from removal of the insulation from the Pressurizer in the area of the heater connections does not justify actions which could potentially cause significant damage to Pressurizer heater electrical connections.

RELIEF REQUEST 3

COMPONENT DESCRIPTION:
Emergency Feedwater System (EFW) piping from the EF-V30 Control Valves (EF-V30A, EF-V30B, EF-V30C, and EF-V30D) to the EF-V12 Check Valves (EF-V12A and EF-V12B). The affected 6" piping consists of sections ~100 ft in length for Train A (EF-V30A/D to EF-V12A) and approximately 75 ft in length for Train B (EF-V30B/C to EF-V12B).

ISI CLASS: 2

ISI DRAWING NO.: ID-ISI-FD-009

DESIGN PRESSURE: 1100 psig

NORMAL OPERATING PRESSURE: ~1050 psig

CODE REQUIREMENTS:
ASME Section XI Table IWC-2500-1 Category C-H requires a VT-2 Leakage Test each inspection period (3, 4, and 3 years respectively). IWC-5221 states that the nominal operating pressure shall be acceptable as the system test pressure.

RELIEF BEING REQUESTED:
GPU Nuclear requests relief from the requirement to test the piping which extends from the EF-V30 control valves to the EF-V12 check valves at nominal EFW System operating pressure.

ALTERNATIVE EXAMINATION:
Each ISI inspection period, a reduced pressure VT-2 Leakage Test of the piping between the EF-V30 Control Valves and the EF-V12 Check Valves will be performed at nominal Once Through Steam Generator (OTSG) operating pressure or slightly below OTSG pressure, ~900 psig, instead of EFW nominal operating pressure as required by the Code. This is approximately 200 psig below nominal EFW System operating pressure.

BASIS FOR REQUESTING RELIEF:
The TMI-1 EFW System is a standby system operated only for emergency feedwater addition and required surveillance tests.

The EF-V12 valves (depicted in Figure 2) are tilting disc check valves and cannot be closed to prevent forward flow. The primary safety grade suction source for the EFW Pumps is the Condensate Storage Tanks (CSTs); the backup safety grade source is river water. The addition of cold oxygenated water (~50-70°F) from the CSTs to the hot OTSG tubes causes a concern regarding the increased likelihood for initiation or propagation of OTSG tube cracks and is not practical for testing purposes due to the risk involved.

Actuation of EFW with RCS temperature greater than 300°F adds a thermal cycle to the EFW nozzles. TMI-1 has used 5 thermal cycles since the

nozzles where replaced in 1985 with only 40 thermal cycles total allowable. In accordance with Technical Specification (TS) 4.9.1.5, Surveillance Procedure (SP) 1303-11.42 tests the EFW System once each refueling outage and following each Cold Shutdown that exceeds 30-days in length. In accordance with the NRC's October 8, 1992 Safety Evaluation Report (SER) for the second 10-year IST interval, testing is limited to this frequency to limit exposure of the OTSG internals to oxygenated water.[1]

A reduced pressure VT-2 Leakage Test of the piping between the EF-V30 Control Valves and the EF-V12 Check Valves will be performed at nominal OTSG operating pressure or slightly below OTSG pressure, ~900 psig, instead of EFW nominal operating pressure as required by the Code. The test pressure will be at OTSG operating pressure if EF-V12A/B has seat leakage and the leakage is greater than seat leakage through the normally closed boundary valves (the EF-V30 valves, EF-V11A/B, and EF-V13). If the EF-V12 valves do not have seat leakage or if the leakage is less than that of the boundary valves, a hydrostatic test pump will be used to test the affected piping to a pressure slightly below OTSG pressure. In this case, the test pressure must be less than OTSG pressure to avoid opening the EF-V12 check valves. A higher test pressure might open or unseat EF-V12A/B which could result in steam binding of the EFW Pumps[2] when the piping is returned to service after the test.

In addition to the reduced pressure VT-2 Leakage Test, volumetric and surface examination will be performed in accordance with TMI-1 Second 10-Year ISI Plans and Schedules.[3] The Code requires volumetric and surface examination of eleven welds from the EF-V30s to the OTSGs. Although only three of the eleven welds are located in the piping sections between the EF-V30 valves and the EF-V12 valves, the piping that contains the other eight welds experiences similar service conditions. Therefore, the results from examination of the welds between the EF-V12 valves and the OTSGs will serve as an indication of the condition of the piping between the EF-V30 valves and the EF-V12 valves. If examination of these eleven welds identifies defects exceeding the Code acceptance criteria, additional welds will be examined.

The piping from the EF-V30 valves to EF-V12A/B is 6" Seismic Class I, ASTM A 106 Grade B Schedule 80, carbon steel pipe with butt welded ends. There are no bolted connections between the EF-V30 valves and EF-V12A/B. The affected piping experiences standby service only. Except for EFW actuations and during testing in accordance with SP 1303-11.42, the affected piping is idle. The affected piping contains CST water of good

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[1] Letter, NRC to Hukill, dated October 3, 1986 (5211-86-3262).

[2] Steam binding of EFW Pumps was the subject of NRC Bulletin 85-01 and NRC Generic Letter 88-03.

[3] Letter, Broughton to NRC, dated April 19, 1991 (C311-91-2010).

quality and experiences only cold flow (design temperature is 110°F). Consequently, the thermal expansion stresses are minimal.

The piping from the EFW Pump discharge to EF-V30 valves has a design pressure of 1300 psig and design temperature of 110°F. Pump and valve testing (IST)[4] and quarterly Heat Sink Protection System (HSPS) testing cause the piping from the pump to the EF-V30 valves to be pressurized to the maximum EFW Pump discharge pressure head (~1420 psig). If a failure (i.e., pipe crack) were to occur, it would be discovered by the operator during these tests. TMI-1 has not experienced any cracks or leakage in the piping from the EF-V30 valves to EF-V12A/B. Using this experience as an indicator, the existence of cracks or leakage in the affected piping is considered to be very unlikely.

SP 1303-11.42 is performed at Cold Shutdown conditions to avoid having to spray cold water into the hot OTSGs. Even though a trailer of nitrogen is brought in and the CSTs are sparged with nitrogen immediately prior to the performance of SP 1303-11.42, the EFW Pumps are only run long enough to verify accident design flow rate. Longer run times are not permitted in order to limit the quantity of oxygenated water delivered to the OTSGs. There is insufficient time during the performance of SP 1303-11.42 to complete a VT-2 Leakage Test which, as required by the Code, includes a hold time of at least 10 minutes prior to the examination. Alternatives do not exist that would significantly reduce the impact of extending the EFW Pump run time sufficient to accomplish a VT-2 Leakage Test of the affected piping. Two alternative which were evaluated are as follows:

1. Throttling the EF-V30 control valves to a small flow rate would limit the quantity of oxygenated water addition, but the pressure of the affected piping would be reduced significantly below the normal EFW operating pressure due to the pressure drop across the EF-V30 valves with the OTSGs at atmospheric pressure.

2. Conducting the test at reduced plant power drawing low oxygenated water from the Condenser Hotwell would also introduce cold water into the OTSG tubes resulting in a thermal cycle on the EFW nozzles and risking damage to the OTSG tubes.

Modification of the tilting disk EF-V12 check valves or installing temporary plugs has also been considered as follows:

1. Replacement of the EF-V12 valves with a stop check valve or modification of the existing valves is considered to be impractical considering the costs involved where the only benefit would be a relatively small increase in the test pressure attainable.

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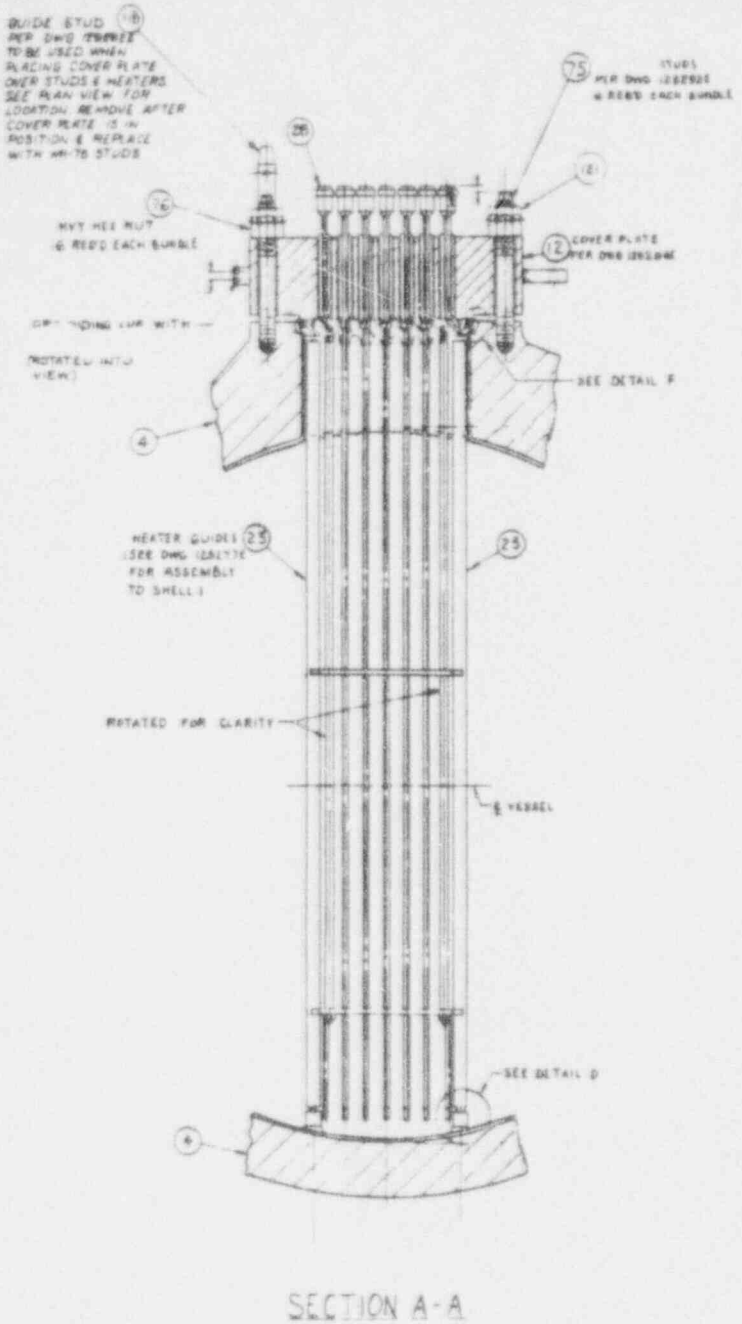
[4] Letter, Broughton to NRC, dated November 12, 1992 (C311-92-2089), Technical Specification Change Request (TSCR) 219 requests a change to quarterly EFW Pump testing in accordance with the ASME Section XI 1986 Edition.

2. Installation of temporary plugs in the EF-V12 valves would require draining down of the OTSGs. Access to the valve bonnet assemblies is limited by the proximity to the overhead. A plug can not be fabricated ahead of time because the valve internal dimensions are not available. Therefore, taking measurements for fabrication of a custom plug and fitup would require an extended amount of time with the valve opened. This poses the risk of disturbing the nitrogen blanket which protects the OTSG internals from exposure to oxygen and also creates a personnel hazard from release of nitrogen through the disassembled valve. Installation of a plug would risk possible damage to the valve internal components an create the possibility that the valve's metallic pressure seal ring will leak after reassembly. Flow through the valve could become blocked if materials were inadvertently left there. Procedures and training should prevent this from occurring, however events of this nature are possible.

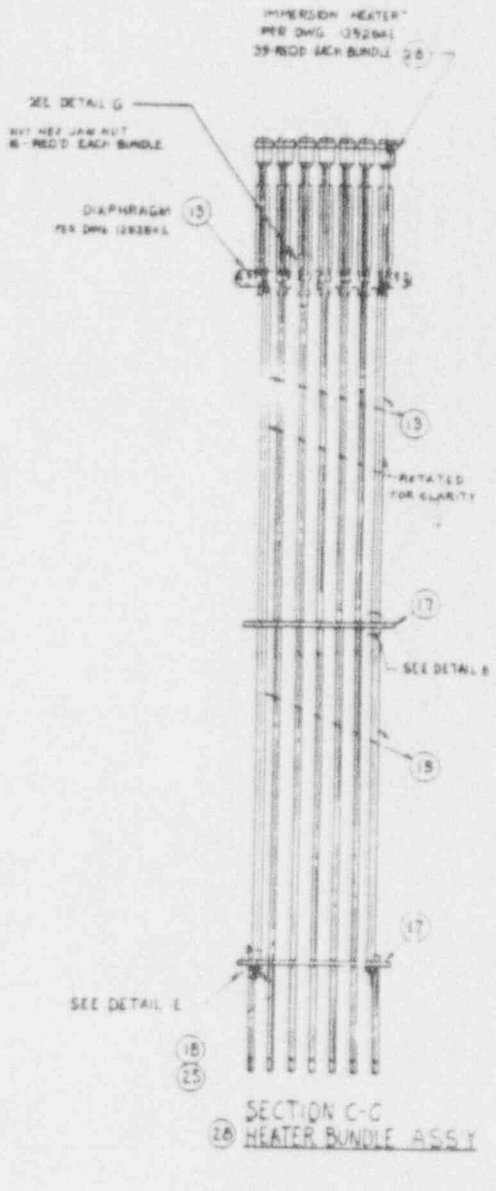
Compared to the risk involved, the benefit to be gained by conducting the test at only a slightly higher test pressure clearly does not justify the actions that would be needed to meet the Code requirement.

FIGURE 1

TMI-1 PRESSURIZER HEATER DETAILS

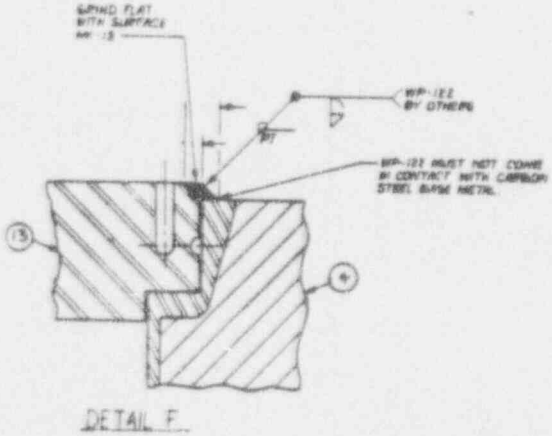


SECTION A-A

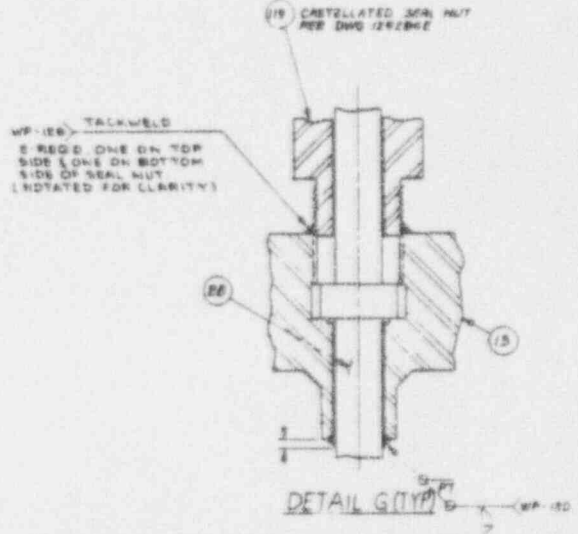


SECTION C-C

(28) HEATER BUNDLE ASSY



DETAIL F



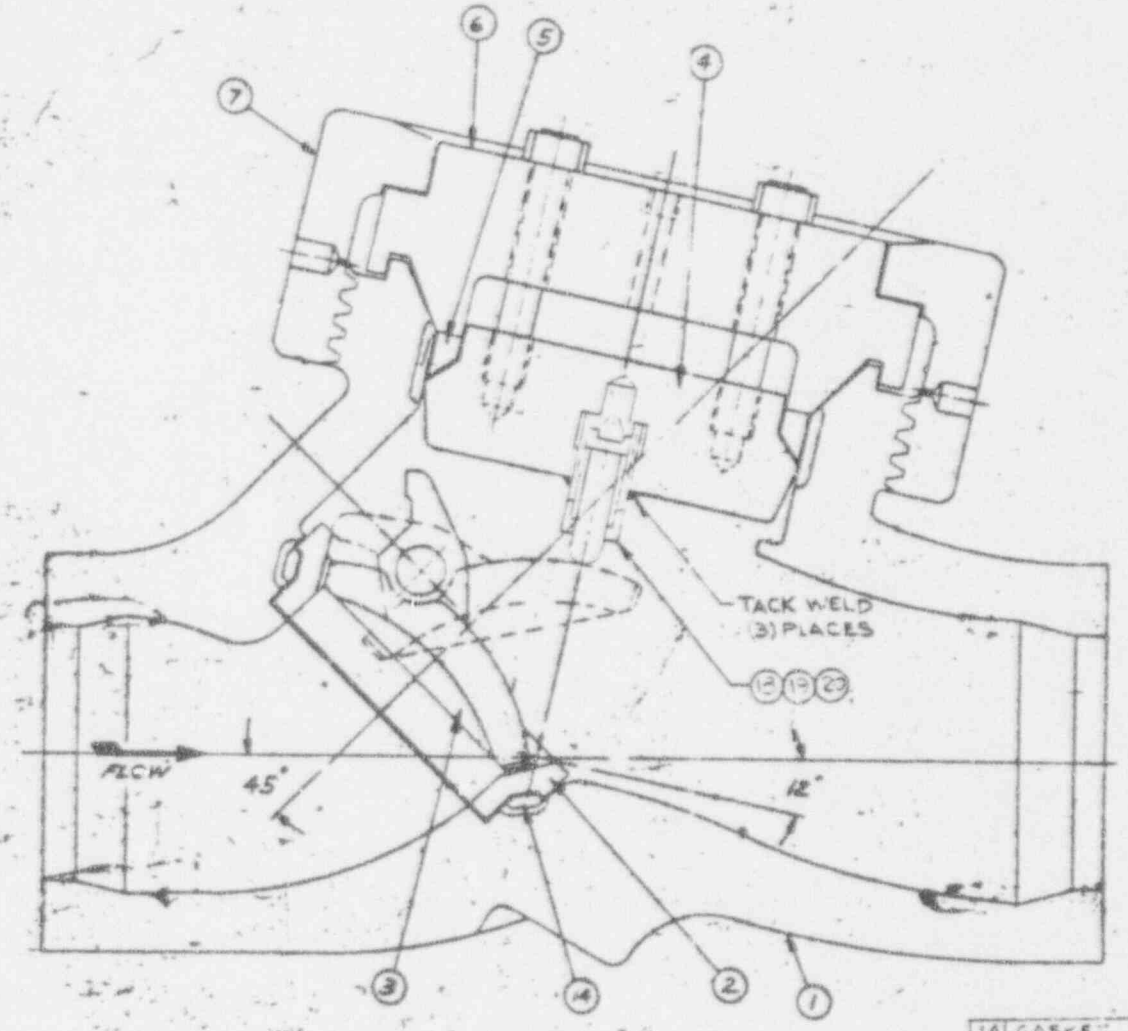
DETAIL G (TYP)

FIGURE 2

TMI-1 CRANE TILTING DISC

CHECK VALVE ASSEMBLY

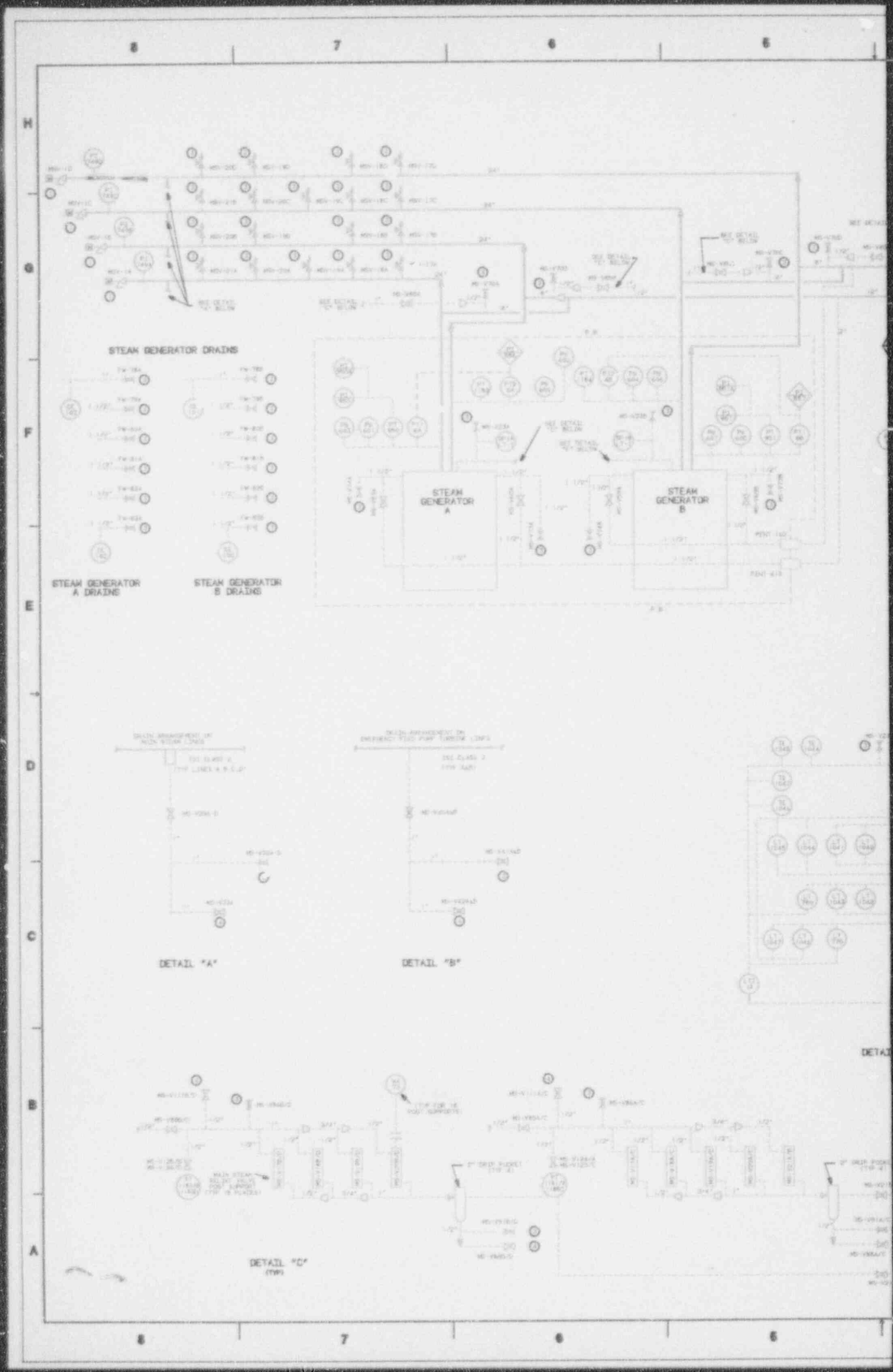
(EF-V12 A/B)



MODIFIED DESIGN

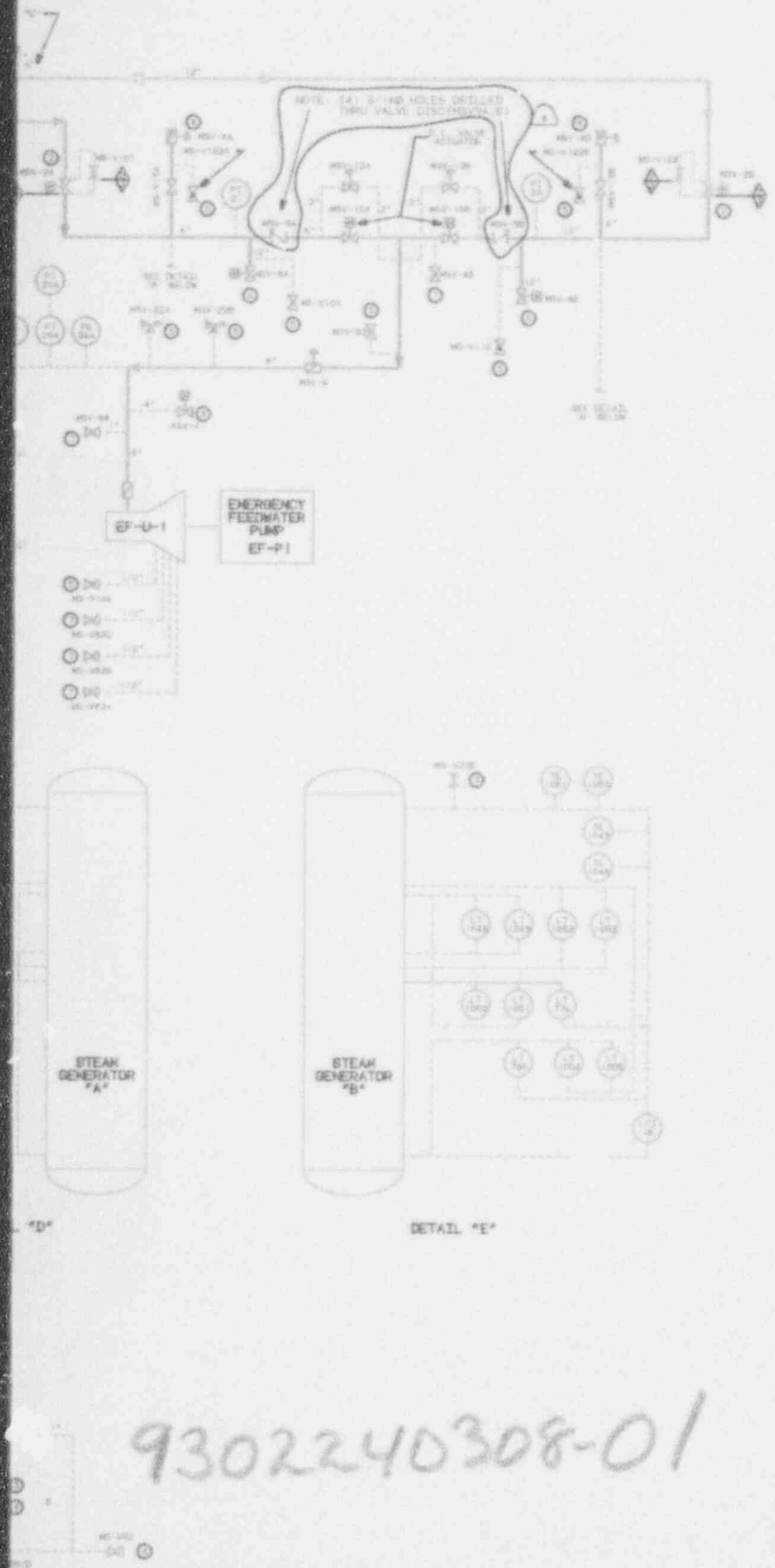
*	21	PIVOT PIN	1	A276 316 MALC
	20	SLEEVE	1	A276-316
	19	PLUNGER	1	A276-316
	18	SPRING	1	A313 S.S.
		METALLIC KEEPER	2	A.I.S. [illegible]
*	17	ROLL PIN	2	A.I.S. 302 S.S.
	16	LOCK PLATE	2	A240-316

	14	GASKET	1	A240 [illegible]
*	13	LOCK WIRES	2	18-8
	12	HELICOILS	2	INSERT TYPE SERIES
*	11	LOCK PLATE BOLTS	2	18-8
* *	10	SPRING WASHERS	2	17-7 PH
*	9	BUSHINGS	2	A276 316 MALC
* *	3	PIVOT PINS	2	A276 316 MALC
	7	LOCK RING	1	A351 CF8M
	6	RETAINING CAP	1	A351 CF8M
	5	SPRING RING	1	A240 [illegible]
	4	BONNET	1	A240 [illegible]
*	3	DISC STELLITED	1	A351 CF8M
* *	2	SEAT RING STELL.	1	A351 CF8M
	1	BODY	1	A351 CF8M
				WATER AL


STEAM GENERATOR DRAINS
STEAM GENERATOR A DRAINS
STEAM GENERATOR B DRAINS
STEAM GENERATOR A
STEAM GENERATOR B
DETAIL "A"
DETAIL "B"
DETAIL "C" (TYP)

SI APERTURE CARD

Also Available On Aperture Card



COLOR LEGEND

CLASS 2 NON-EXEMPT COMPONENT

CLASS 2 EXEMPT COMPONENT

CLASS 3 CATEGORY D-A NON-EXEMPT COMPONENTS

CLASS 3 CATEGORY D-B EXEMPT COMPONENTS

INFORMATION ONLY

GENERAL NOTES

1. THIS SKETCH REFLECTS ISI CLASSIFICATIONS 2&3 FOR SYSTEMS SHOWN ON GAI FLOW DIAGRAMS [illegible] AND [illegible]

2. THESE DRAWINGS ARE TO BE USED FOR ISI ONLY. THE CATEGORIES SHOWN ARE USED FOR CLASSIFICATION OF PIPING FOR IN SERVICE INSPECTION (ISI) IN ACCORDANCE WITH ASME B & PV CODE SECT. XI. THESE CATEGORIES ARE NOT INTENDED TO CORRESPOND TO PIPING DESIGN SAFETY CLASSIFICATIONS [illegible] THE CLASSIFICATIONS HEREIN ARE NOT TO BE USED TO DETERMINE SAFETY OR SERVICE CLASSIFICATIONS.

SPECIFIC NOTES

CLASS 2 SYSTEM BOUNDARY PER REG. GUIDE 1.26 - OUTERMOST CONTAINMENT ISOLATION VALVE

CLASS 2 TO CLASS 3 SYSTEM BREAK PER REG. GUIDE 1.26 - OUTERMOST CONTAINMENT ISOLATION VALVE

CLASS 2 SYSTEM BOUNDARY PER REG. GUIDE 1.26 - FIRST VALVE - NORMALLY CLOSED

CLASS 2 BOUNDARY NORMALLY OPEN VALVE [illegible] VALVE CRITERIA. FAILURE OF LOWER CLASS COMPONENT WILL NOT CAUSE LOSS OF SAFETY FUNCTION. STEAM LEAK OF THIS SIZE IS ENVELOPED BY MAIN STEAM LINE BREAK [illegible] INSIDE THE INTERMEDIATE BUILDING.

CLASS 2 SYSTEM BOUNDARY PER REG. GUIDE 1.26 - FIRST VALVE NORMALLY CLOSED

CLASS 2 SYSTEM BOUNDARY PER REG. GUIDE 1.26 - VALVES CAPABLE OF AUTOMATIC CLOSURE

CLASS 2 BOUNDARY NORMALLY OPEN VALVE [illegible] VALVE CRITERIA. FAILURE OF LOWER CLASS COMPONENT WILL NOT CAUSE LOSS OF SAFETY FUNCTION. STEAM LEAK OF THIS SIZE IS ENVELOPED BY MAIN STEAM LINE BREAK [illegible] INSIDE THE INTERMEDIATE BUILDING.

CLASS 2 BOUNDARY PER REG. GUIDE 1.26 - NORMALLY CLOSED VALVE

THIS IS A COMPUTER GENERATED DWG. DO NOT REVISE IT MANUALLY

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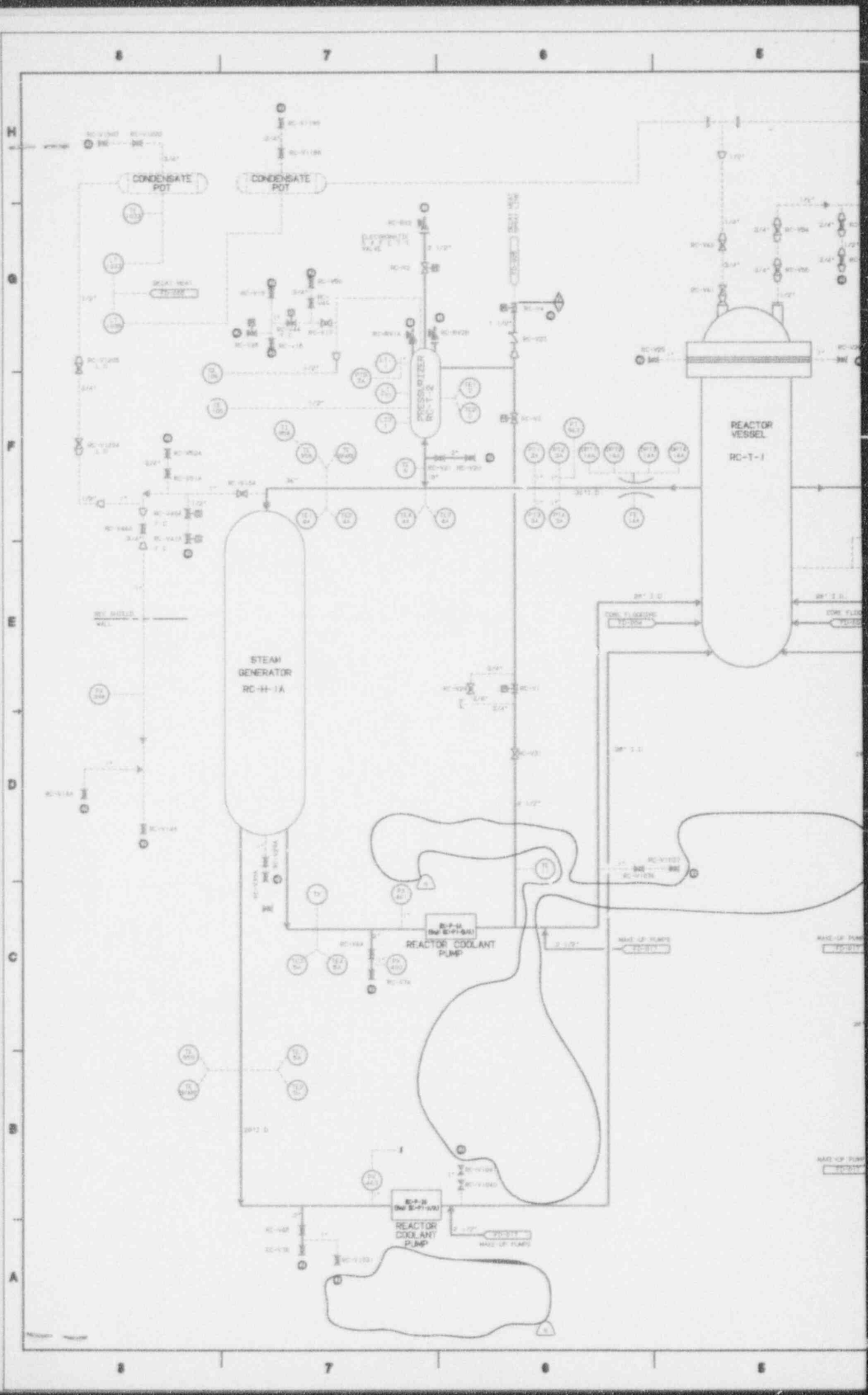
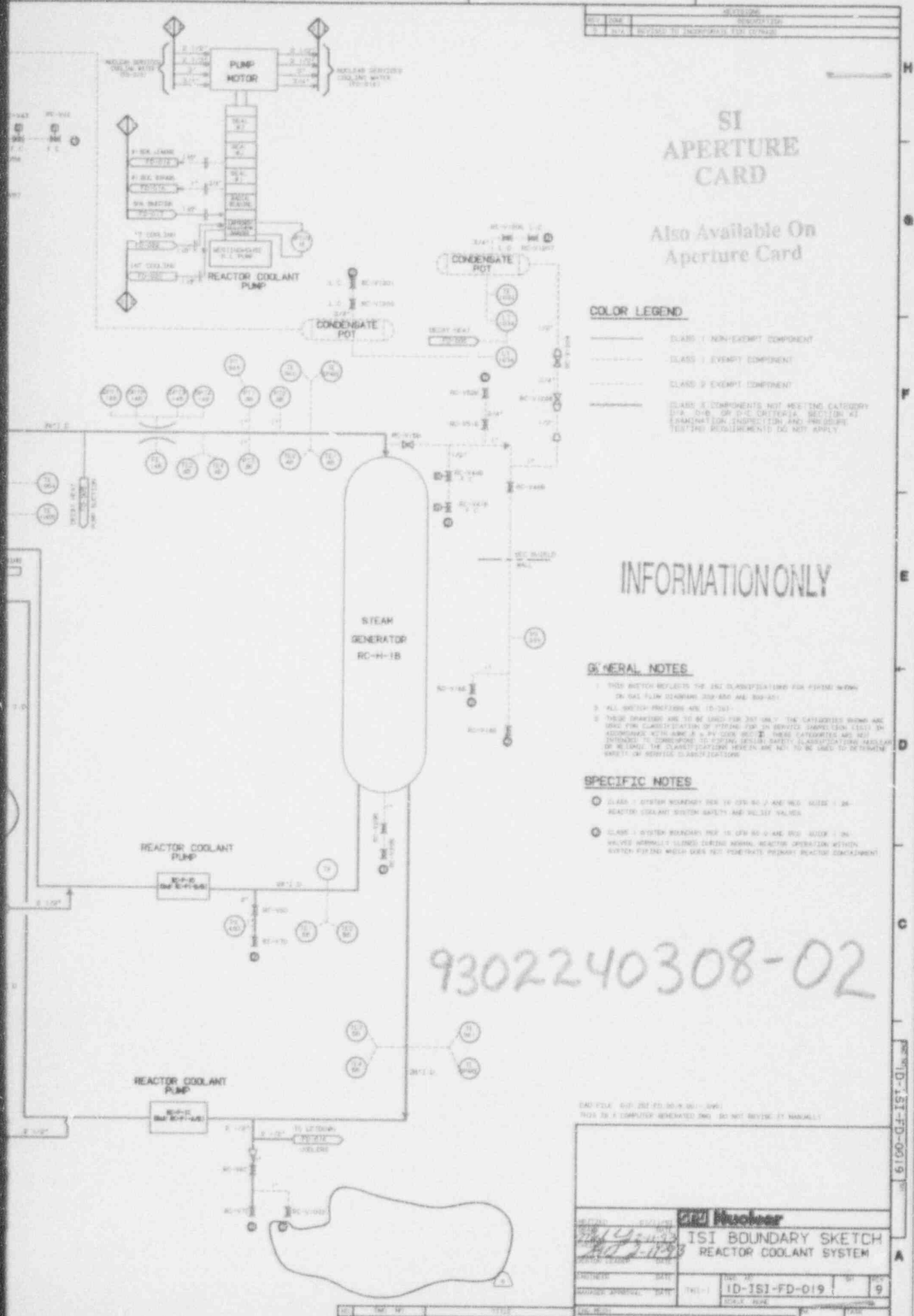
ISI BOUNDARY SKETCH

MAIN STEAM SYSTEM AND DRAINAGE

TMI-1	DWG NO. ID-ISI-FD-001
SCALE	NONE
REV	8

ID-ISI-FD-001

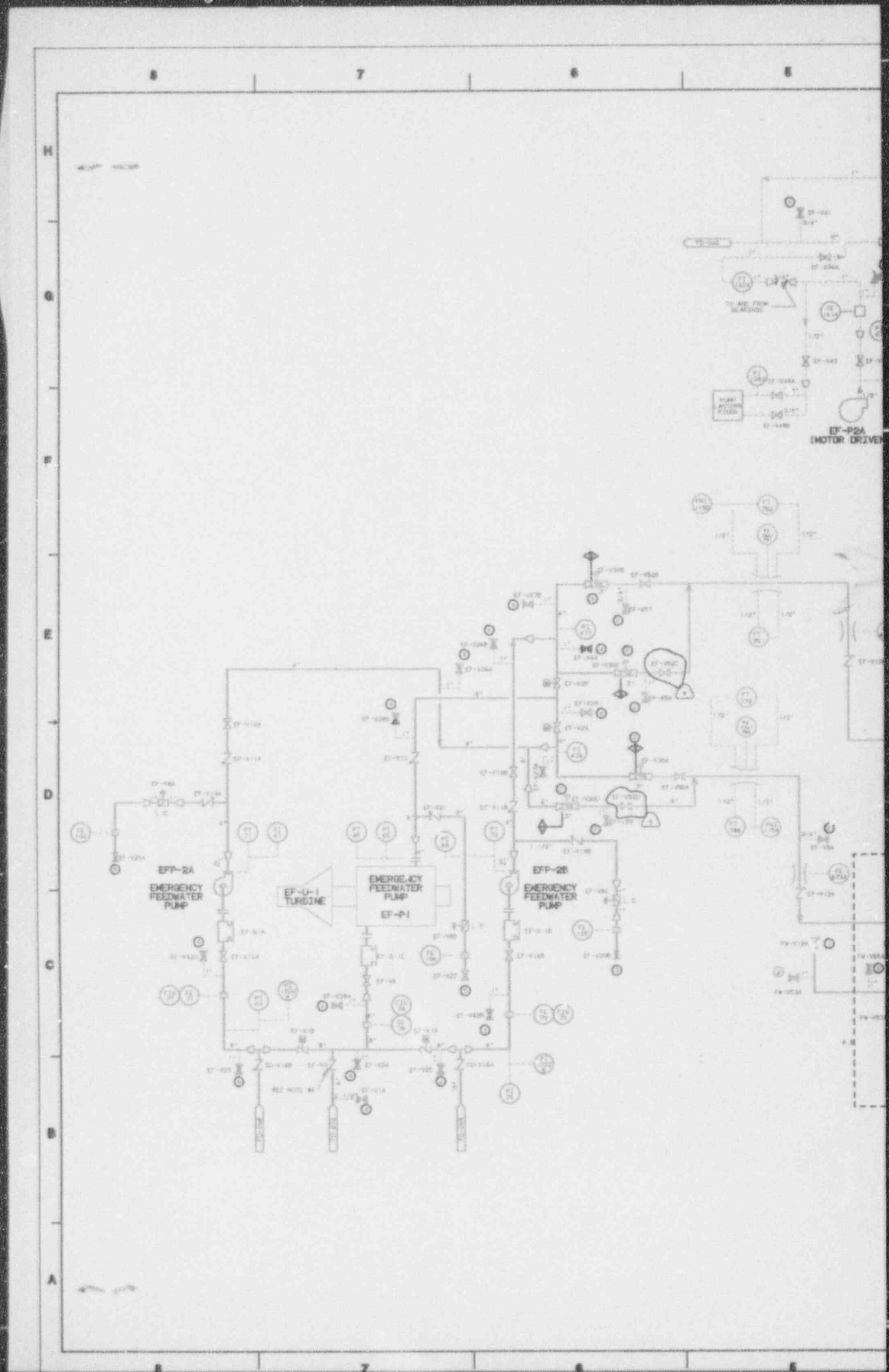

CONDENSATE POT
CONDENSATE POT
PRESSURIZER RC-T-2
REACTOR VESSEL RC-T-1
STEAM GENERATOR RC-H-1A
REACTOR COOLANT PUMP
REACTOR COOLANT PUMP

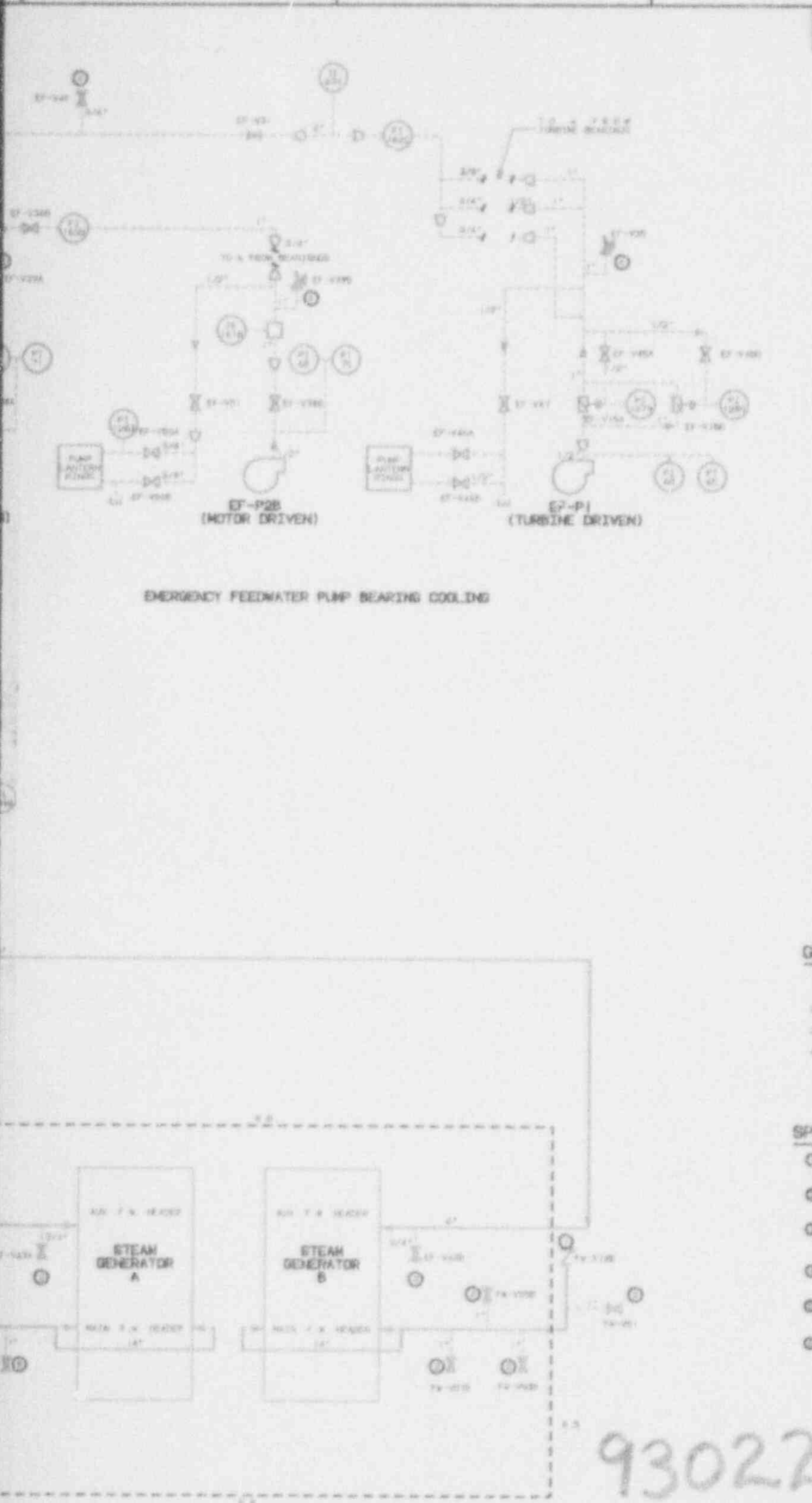


9302240308-02

ISI BOUNDARY SKETCH
REACTOR COOLANT SYSTEM

ID-ISI-FD-019 | 9


EF-P2A
(MOTOR DRIVEN
EFP-2A
EMERGENCY
FEEDWATER
PUMP
EF-U-1
TURBINE
EMERGENCY
FEEDWATER
PUMP
EF-P1
EFP-2B
EMERGENCY
FEEDWATER
PUMP



COLOR LEGEND

CLASS 2 NON-EXEMPT COMPONENT

CLASS 2 EXEMPT COMPONENT

CLASS 3 CATEGORY D-G NON-EXEMPT COMPONENTS

CLASS 3 CATEGORY D-B EXEMPT COMPONENTS

GENERAL NOTES

1. THIS SKETCH REFLECTS ISI CLASSIFICATIONS FOR SYSTEM SHOWN ON GAI FLOW DIAGRAM 302-082 & 302-081
2. PIPING ON THIS SKETCH IS ISI CLASS 2 & 3
3. ALL SKETCH PREFIXES ARE 1D-ISI
4. THESE DRAWINGS ARE TO BE USED FOR ISI ONLY. THE CATEGORIES SHOWN ARE USED FOR CLASSIFICATION OF PIPING FOR IN SERVICE INSPECTION (ISI) IN ACCORDANCE WITH ASME B & PV CODE SECT XI. THESE CATEGORIES ARE NOT INTENDED TO CORRESPOND TO PIPING DESIGN SAFETY CLASSIFICATIONS NUCLEAR OR SEISMIC. THE CLASSIFICATIONS HEREIN ARE NOT TO BE USED TO DETERMINE SAFETY OR SERVICE CLASSIFICATIONS.

SPECIFIC NOTES

(1) CLASS 2 SYSTEM BOUNDARY PER REG. GUIDE 1.26 - CHECK VALVES PREVENTING FLOW FROM THE STEAM GENERATORS (OUTBOARD) CONTAINMENT ISOLATION VALVE
(2) CLASS 2 SYSTEM BOUNDARY PER REG. GUIDE 1.26 - VENTS, DRAINS AND RELIEF VALVES - NORMALLY CLOSED
(3) CLASS 2 TO CLASS 3 SYSTEM BREAK PER REG. GUIDE 1.26 AND ANSI 18.2A VALVE CAPABLE OF AUTOMATIC CLOSURE. FLOW CONTROL VALVES PREVENTING FLOW TO AND FROM STEAM GENERATORS
(4) CLASS 3 SYSTEM BOUNDARY PER REG. GUIDE 1.26 - VENTS, DRAINS AND RELIEF VALVES - NORMALLY CLOSED
(5) CLASS 3 SYSTEM BOUNDARY WHEN NO VALVE EXISTS PER ANSI N18.2A FAILURE OF LOWER CLASS COMPONENT WILL NOT CAUSE LOSS OF NUCLEAR SAFETY FUNCTION
(6) VALVE INTERNALS REMOVED

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ISI BOUNDARY SKETCH
EMERGENCY FEEDWATER

1D-ISI-FD-009 | REV 9