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

October 9, 1995
C311-95-2413

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

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
LER 95-003-00

The purpose of this letter is to transmit TMI-1 Licensee Event Report (LER) No. 95-003-00 regarding a reactor coolant leak caused by a cracked weld in the drain line at the bottom of the "B" Reactor Coolant Pump suction. The abstract provides a brief description of the event.

The event did not adversely affect the health and safety of the public.

Sincerely,

J. Knubel
Vice President and Director, TMI

MRK
Attachment

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

160100

IE22
/11

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
THREE MILE ISLAND, UNIT 1	05000289	1 OF 15

TITLE (4)

REACTOR COOLANT LEAK DUE TO A CRACKED WELD IN A COLD LEG DRAIN LINE

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
09	09	95	95 --	003 --	00	10	09	95		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): N

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
M. R. KNIGHT, TMI LICENSING ENGINEER	(717) 948-8554

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

REACTOR COOLANT LEAK DUE TO A CRACKED WELD IN A COLD LEG DRAIN LINE

On September 9, 1995 TMI-1 was in Hot Shutdown and in the process of cooling down for the Cycle 11 Refueling Outage. The Reactor Coolant System (RCS) was at 535°F and 2000 psig. At approximately 7:00 am, a Reactor Building radiation monitor indicated an increase in iodine activity and at approximately 10:15 am a leak was found in a weld on a nonisolable 2 inch diameter cold leg drain line. The leak was estimated at approximately 20 drops per seconds. This event is reportable in accordance with 10 CFR 50.73.a.2.ii. There were no adverse safety consequences or safety implications that resulted from this event, and this event did not affect the health and safety of the public.

Metallurgical evaluations have determined that the failure was fatigue induced and it appears that the fatigue occurred over a long period of time. The most probable root cause for growth of the crack from an initial flaw is reactor coolant turbulent penetration into the drain line coupled with thermal stratification causing fatigue cycles in the weld. The failed line was replaced. Welds in similar lines have been inspected and evaluated to be acceptable for operation. Planned actions include insulation of the Reactor Coolant Pump suction drain lines, additional inspections, testing, and evaluation to confirm the adequacy of GPU Nuclear response to this event.

REACTOR COOLANT LEAK DUE TO A CRACKED WELD IN A COLD LEG DRAIN LINE

I. Plant Operating Conditions before Event:

The plant was at Hot Shutdown conditions with temperature 535 degrees F and pressure 2000 psig while cooling down for the Cycle 11 Refueling (11R) Outage.

II. Status of Structures, Components, or Systems that were Inoperable at the Start of the Event and that Contributed to the Event:

No systems, structures or components were out-of-service that contributed to this event.

III. Event Description:

TMI-1 began shutting down for the 11R Outage on September 8, 1995 at 5:00 pm.

On September 9, 1995 at approximately 7:00 am, RB radiation monitor (RM-A2) indicated an increase in iodine activity. TMI-1 was at 0% power. Tave was 535°F and Reactor Coolant System (RCS) pressure was 2000 psig. An investigation for an RCS leak was initiated in response to the increase in iodine activity. At approximately 10:15 am a leak was found in a weld on the 2 inch diameter "B" Reactor Coolant Pump (RCP) suction drain line upstream of valve RC-V6B [AB/V]*. The leak was estimated at approximately 20 drops per second.

Portable ventilation with charcoal and particulate filtration was placed at the leak location to contain and reduce the radiological effects of the leak. At approximately 11:00 am, RCS pressure was at 1080 psig and the leakrate had decreased significantly.

The leak occurred in a nonisolable section of the 2 inch drain line. TMI-1 Technical Specification (TS) 3.1.6.4 states that, "If any reactor coolant leakage exists through a nonisolable fault in an RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the Cold Shutdown condition shall be initiated within 24 hours of detection." The reactor was subcritical when the leak was discovered, and preparations for cooldown were proceeding; therefore the TS requirement was satisfied.

This event was found to be reportable in accordance with 10 CFR 50.73.a.2.ii as a condition found during shutdown that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principle safety barriers, being seriously degraded. The NRC Operations Center was notified via the Emergency Notification System (ENS) line at 11:38 am on September 9, 1995.

System/Component Description:

The TMI-1 RCS has two Once Through Steam Generators (OTSGs) [AB/HX]* with four RCPs (two in OTSG Loop "A" and two in OTSG Loop "B"), and includes piping and instrumentation. Each RCP [AB/P]* suction line (28 inch diameter) has a 2 inch drain. Each drain line contains two manual valves in series. The drain lines are routed to a header connected to the suction of the Reactor Drain Pump [WD/P]*.

The 2 inch drain lines for the "A," "B," and "D" suction lines are connected to 1.5 inch nozzles (with inconel safe ends) by 1-1/2 inch by 2 inch reducing 90 degree elbows. The "C" suction leg is drained through the 2.5 inch diameter RCS letdown line which taps off of the bottom of the suction leg.

Location/Identification of Leaking Weld (ISI Weld No. RC-187):

The leak is located at ISI Weld No. RC-187, which is the weld between the 1.5 inch x 2 inch, 90 degree reducing elbow and the 2 inch diameter horizontal pipe. The reducing elbow is the first elbow in the drain line off the RCS Loop "A" suction line to the "B" RCP. The elbow and horizontal pipe is type 316 stainless steel. The elbow is welded to the vertical 1.5 inch diameter Inconel safe end of the RCP suction drain nozzle. (See Figures 1 and 2).

Description of Weld Indication at Leak:

Visual and ultrasonic (UT) inspections were performed on the leaking weld (RC-187). Visually the indication is circumferential in the weld, two tenths (0.2) inches long starting 0.3 inches counter-clockwise (CCW) and stopping 0.5 inches CCW from top dead center looking in the direction of flow, from the nozzle and elbow to the first isolation valve, RC-V6B.

The UT examinations revealed that the indication is approximately 0.55 to 1.20 inches long on the outside diameter (OD) and approximately 2 to 3 inches long on the inside diameter (ID). Axially from the center of the weld, the indication favors the horizontal pipe side of the weld. The indication appeared to be located within the weld area including the weld root. Figures 3 and 4 illustrate the indication found in the affected pipe joint.

IV. Component Failure Data:

The 28 inch RCP suction piping is A-106, Grade C carbon steel clad with 18-8 austenitic stainless steel. The drain line consists of a 1.5 inch nozzle of the same materials as the suction piping, a 1.5 inch Schedule 160 Inconel safe-end, a 1.5 inch x 2 inch, A-403, WP 316 stainless steel, schedule 160 reducing elbow and 2 inch, schedule 160, A312/376 Type 316 stainless steel pipe. All connections are butt-welded.

The design, fabrication, inspection, and testing of the reactor coolant piping is in accordance with USAS B31.7, Code for Pressure Piping, Nuclear Power Piping, February 1968 Draft including June 1968 Errata. Nozzles on the reactor coolant piping comply with USAS B31.7 February 1968 Draft including June 1968 Errata. The drain lines were designed in accordance with B31.1 - 1967; and fabricated, erected and tested in accordance with B31.7 February 1968 Draft including June 1968 Errata. The suction drain line to the second isolation valves (RC-V7B and RC-V1031) is classified ASME Section XI Class 1 (IWB).

V. Automatic or Manually Initiated Safety System Responses:

No safety system responses occurred or were required to occur.

VI. Assessment of the Safety Consequences and Implications of the Event:

The leak occurred while the reactor was shut down. Plant cooldown was proceeding and the effects of the leak were minimal.

TMI-1 TS 3.1.6.6 states: "Action to evaluate the safety implication of reactor coolant leakage shall be initiated within four hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the limits of TS 3.22.2.1." TS 3.22.2.1 identifies the dose rate limits due to radioactive materials released in gaseous effluent for the site.

Actions were initiated within four hours of detecting the leak to evaluate the safety implications. The safety evaluation determined that the nature of the leak was such that the leakrate was not expected to significantly increase. Leakage diminished as RCS pressure was decreased. The leak was very small and did not present a concern for RCS makeup capability or waste handling capacity. The leak was monitored, including the radiological effects within the RB. Local and RB radiological effects were significantly reduced when the portable ventilation unit was placed by the leak. Additionally, the RB Purge filters were in operation and the RB purge line would have been automatically isolated in the event of a significant release of activity inside the RB. The Plant Review Group (PRG) concluded that the exposure of offsite personnel to radiation would remain a small fraction of the limits of TS 3.22.2.1.

Analysis has shown that the cracked pipe weld in the "B" RCP suction drain line satisfied "leak before break" criteria for the end of Cycle 10. Analyses have concluded that the line would not have ruptured under faulted conditions (seismic).

However, if a leak were to develop in a suction leg drain line the reactor coolant leakage would be identified by one or more of the following methods: 1) Containment Radiation Monitor (RM-A2), 2) Reactor Building Sump level readings, 3) Humidity Monitor alarm in the Control Room, and 4) RCS leakrate trending. In the event of an unidentified RCS leak exceeding 1 GPM, TS requires the reactor to be placed in Hot Shutdown within 24 hours of detection. If the leak is through a nonisolable fault in an RCS strength boundary, the reactor is to be shutdown, and cool-down to cold shutdown condition initiated within 24 hours of detection. The 1 GPM limit on unidentified leakage can be accurately measured while leakage is sufficiently low to ensure early detection. Leakage of this magnitude can be reasonably detected within a matter of hours, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

Plant operators are trained and procedures are in place to ensure the response to a reactor coolant leak is timely and sufficient to protect the health and safety of plant personnel and the public. If the RCS leakage limits are exceeded or containment radiation monitoring indicates high airborne activity, appropriate actions will be initiated by the control room operators as specified in plant procedures. Such actions include placing the plant in the appropriate condition as required by plant technical specifications, identifying the source of the leak, and evaluating the safety implication of the leak.

The limiting break size for a rupture of the "A," "B," and "D" drain lines is 1.5 inches. This would result in a Small Break Loss Of Coolant Accident (SBLOCA), which is within the design and licensing basis of TMI-1. (See TMI-1 Updated FSAR Section 14.2.2.4.3). Procedures, drills, as well as simulator training are conducted to provide confidence that such an event would be handled without endangering the health and safety of the public.

There were no adverse safety consequences or safety implications that resulted from this event, and this event did not affect the health and safety of the public.

VII. Failure Analysis of the Weld

Metallographic examination of the fracture surface showed it to have initiated at the ID surface of the pipe. The crack appears to have initiated at the toe of the weld in an area of very shallow (approximately 3 mils) intergranular attack (i.e., less than 1/2 grain deep). In addition the weld toe, where the crack originated, formed a sharp angle with the ID of the pipe creating a potential stress riser. The geometry on the elbow side of the joint was much smoother in comparison.

The weld prep for this weld consisted of a 37.5 degree bevel, with a small (1/16 inch) land associated with the 3/32 inch root gap at the ID. The crack propagated through the pipe wall in a transgranular mode with a circumferential orientation. The crack penetrated the first 85 mils (0.085 inch) in the coarse grained region of the pipe heat affected zone (HAZ). With the crack propagating transverse to the pipe wall, it intersected the weld fusion line and then continued to propagate through weld metal until it reached the OD surface.

Visual examination of the fracture surface shows a fatigue crack propagation mode. "Beach markings" found are definitely characteristic of fatigue failure and indicate periods of crack arrest. Approximately forty one (41) "beach markings" were counted across the fracture surface, possibly correlating to heatup/cooldown cycles of the unit. Since startup, forty two (42) heatup/cooldown cycles have been recorded. Scanning Electron Microscope (SEM) examination of the fracture surface showed fatigue striations over a number of areas, all corresponding to regions where the crack propagated through the HAZ material. From the "beach marks," the crack has an aspect ratio that varies between 4:1 and 6:1. As the crack begins to propagate in the weld metal, the aspect ratio is around 4:1 and appears to remain constant as the crack reaches the OD surface. Closer to the ID surface, the aspect ratio of the crack is higher, at around 6:1. In the weld material, the striations were extremely difficult to identify, and those areas in the weld metal where striations were detected, a reliable striation spacing measurement was not possible. For those areas where striation spacing could be measured, which was near the ID (where the crack propagated in the HAZ material), the spacing was approximately $0.2\mu m$. Assuming a constant striation spacing over the entire crack length from ID to OD, this would correspond to just under 44,000 cycles to propagate a crack through the 0.344 inch wall thickness.

Visual examination of the deposit found on the fracture surface could be used to qualitatively evaluate the age of the crack. The coloration of the deposit was such that the portion of the crack closer to the ID had a darker deposit on the surface than the portion of the crack closer to the OD. This suggested that the crack had been propagating for some time and, in the estimation of the metallurgical laboratory, for more than one outage cycle.

VIII. Previous Events of a Similar Nature:

There have been no previous LERs at TMI-1 related to failure of a Reactor Coolant System pressure boundary weld.

IX. Assessment of "A" and "D" Drain Lines

The configuration of the "A" and "D" RCP suction drain lines is generally similar to the "B" drain line which experienced a cracked weld. Although the "A" and "D" lines presently have no indications, an assessment of the integrity of these lines was performed. The configuration of the "C" RCP suction drain line is significantly different from that of the "A" and "D" lines. The "C" RCP drain line was determined to not be susceptible to the same failure mechanism.

Metallographic evidence obtained from a specimen containing the cracked weld shows thousands of fatigue striations starting from an ID initiation site and extending completely through-wall. Following the same general pattern, there is evidence of approximately forty (40) separated "beach marks," indicating a change of loading conditions from that producing the striations. These "beach marks" are indicative of stress cycle changes. The overall flaw aspect ratio of the through-wall crack comprised overwhelmingly of striations is conservatively estimated at 5:1, indicating a focused initiation site and high through-wall stress.

Given only these observations, without introducing any physical deterioration mechanism, linking cause and effect, it is possible to conservatively estimate the extent of potential damage to like welds in the "A" and "D" cold leg drain lines that could accumulate over the course of the next operating cycle (Cycle 11). Having this estimate, it is possible to show by calculation that such damage as described above will not cause fracture of the pressure boundary even when a load combination including the faulted condition (seismic) is applied at the end of Cycle 11. Standardized calculations from the 1992 Edition of ASME, Section XI, Appendix C are used in order to satisfy the code required safety factors for prevention of the failure mode known as "net-section collapse." This is appropriate when considering ductile materials such as the grade 316L stainless steel.

Crack growth rate per year through Cycle #10 can be estimated using only the striations. The "beach marks" are assumed to be a result of heatup/cooldown cycles. These closely match in number the actual number of such cycles that have occurred at TMI since plant start-up. Initiation of the crack in the "B" drain line therefore occurred early in plant life. The contribution to total crack growth due to heatup/cooldown cycles is much smaller than

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

that due to the striations so that only these should be considered in determining crack growth rate.

The actual pipe wall that is lost to progressive fatigue fracture is not the design wall thickness but slightly less. This is due to the fact that an initial crack due to loss of fatigue resistance must form before progressive fatigue fracture can begin. This initial crack size is assumed to be the minimum detectable or 0.025 in. The average crack growth rate per year through Cycle 10 is found from the difference between the design wall thickness less the initial crack size divided by the operating time (in effective full power years) through Cycle 10. The average crack growth rate was determined to be 0.027 inches per year.

Assuming: 1) that the deterioration mechanisms that contributed to the "B" RCP drain line crack were present in the "A" and "D" drain lines, and 2) that an initial flaw of the largest size that could have been missed by the recent UT inspection (10% wall or 0.034 inches) exists in the drain lines, then the final flaw size at the end of Cycle 11 would be no greater than 0.083 inches or 24.13% through-wall.

X. Identification of Root Cause

The most probable root cause for growth of a crack from an initial flaw is reactor coolant turbulent penetration into the drain line coupled with thermal stratification causing fatigue cycles in the weld. Testing/monitoring of the drain line(s) during startup from the current (11R) outage as well as additional reviews and evaluations that are planned will provide confirmation of the root cause.

Based upon the our evaluations, the leak in the "B" RCS drain line is a unique condition. Although additional work is needed to confirm and validate the root cause, there is no compelling argument to believe what caused the problem in the "B" line will cause a problem in the other drain lines. Inspections of the other drain lines show no indications or any type of weld discontinuity to raise a concern. Interaction with other plants having configurations and conditions as close to TMI's as possible indicate that a similar failure has not occurred. Although no plant has exactly the same configuration, the fact that similar cracked welds have not been experienced at other plants provides additional confidence that this event is not part of an industry problem.

XI. Corrective Actions:

A. Immediate Corrective Action:

Upon location of the leak, portable ventilation with particulate and charcoal filtration was positioned locally to minimize the spread of contamination and reduce the radiological effects in the Reactor Building. The plant was cooled down and depressurized to the Cold Shutdown condition.

B. Additional Actions Taken:

1. A UT examination was performed on the cracked "B" suction drain line weld. Two (2) independent examinations were performed and each confirmed the results of the other examination. The description of the indication provided above was obtained by the UT examination. The UT signal from the indication was sharp and clean with no distinct branching. The indication appeared to propagate from the inside diameter root area, on the pipe side of the joint, straight through the weld joint. The signal was not characteristic of intergranular stress corrosion cracking (IGSCC).

2. The original radiographs taken of the cracked weld during construction were reviewed. No reportable discontinuities were found in the original radiographs.

3. The affected "B" RCP suction drain line was replaced from the nozzle safe end to approximately four feet along the horizontal section of pipe after the reducing elbow.

4. The "B" RCP suction drain line safe-end to elbow weld (the weld upstream of the cracked weld) was radiographed prior to replacement and found to meet the construction code (USAS B31.7, 1969) acceptance criteria. This weld had been examined by liquid penetrant (PT) and radiography (RT) in 1986 as part of the ISI program. No reportable discontinuities were found in 1986 or prior to replacement during the current outage.

5. UT examinations of the "A" and "D" RCP suction drain line welds during the current outage showed no reportable indications. No indications were reported from a UT examination of one of the welds in 1988 and a visual examination of the other in 1979.

6. Review of the Certified Material Test Reports for the 90 degree reducing elbow, pipe, weld wire, and consumable insert for the weld which cracked showed that the material and test requirements were properly documented and the material was of the correct material composition.

7. Because an oil residue from the RCP motors was found to have collected on the drain line which leaked, samples of RCP oil (new and used) were analyzed and our evaluation concluded that the likelihood that the oil contributed to failure of the weld is extremely remote.

8. The piping supports on all four RCP drain lines have been reconfigured to reduce thermal stresses. Analyses were performed to support these configuration changes.

9. A review of other piping connected to the RCS was made using the EPRI screening criteria for turbulent penetration and thermal stratification cycling and concluded that these lines would not be subject to the postulated phenomenon for the 'B' drain line failure.

C. Action Planned to Prevent Recurrence:

1. Insulation will be installed on the RCP "A," "B," and "D" suction drain lines. This will minimize any potential thermal effects. Insulation will be installed on the "A" and "D" lines prior to startup from the current (11R) outage. The RCP "B" drain line will be insulated at the next outage of opportunity but no later than the next refueling outage (12R). The RCP "B" drain line is not being insulated at this time so that it can be instrumented with thermocouples for determining if thermal stratification or thermal cycling is occurring. Operating for one cycle (approximately 24 months) without insulation will not pose a risk with regard to thermal fatigue because this line was replaced during the current outage and the number of fatigue cycles required to fail this line would not occur over the course of one cycle.

2. Prior to startup for Cycle 11, the "B" drain line will be instrumented with thermocouples and measurements of displacement and acceleration will be taken during heatup to collect information which will aid in understanding conditions which the line is subjected to. Thermocouples will be placed on the RCP "B" line and the RCP "D" line to determine if thermal stratification or thermal cycling is occurring and if these thermal phenomena are minimized by the addition of insulation. This will allow comparison of the potential thermal effects on an insulated drain line compared with the uninsulated line.

3. The "B" drain line weld that experienced the crack, along with the corresponding welds in the "A" and "D" drain lines, will be included in the Augmented ISI Program and inspected during the next refueling outage (12R).

4. A metallurgical evaluation has been performed on the RCP "B" suction drain line containing the weld crack. The evaluation incorporated fractographic, metallographic and chemical analysis. Preliminary results of these analyses have been used to determine the failure mechanism and assist in identifying the root cause. GPU Nuclear expects the final metallurgical report to be issued within a few weeks.

5. An independent Advisory Panel is being assembled to review this event. This review will provide added confidence in the effectiveness of the actions taken and those planned. A copy of the final report from the panel will be available at the TMI-1 site for NRC review.

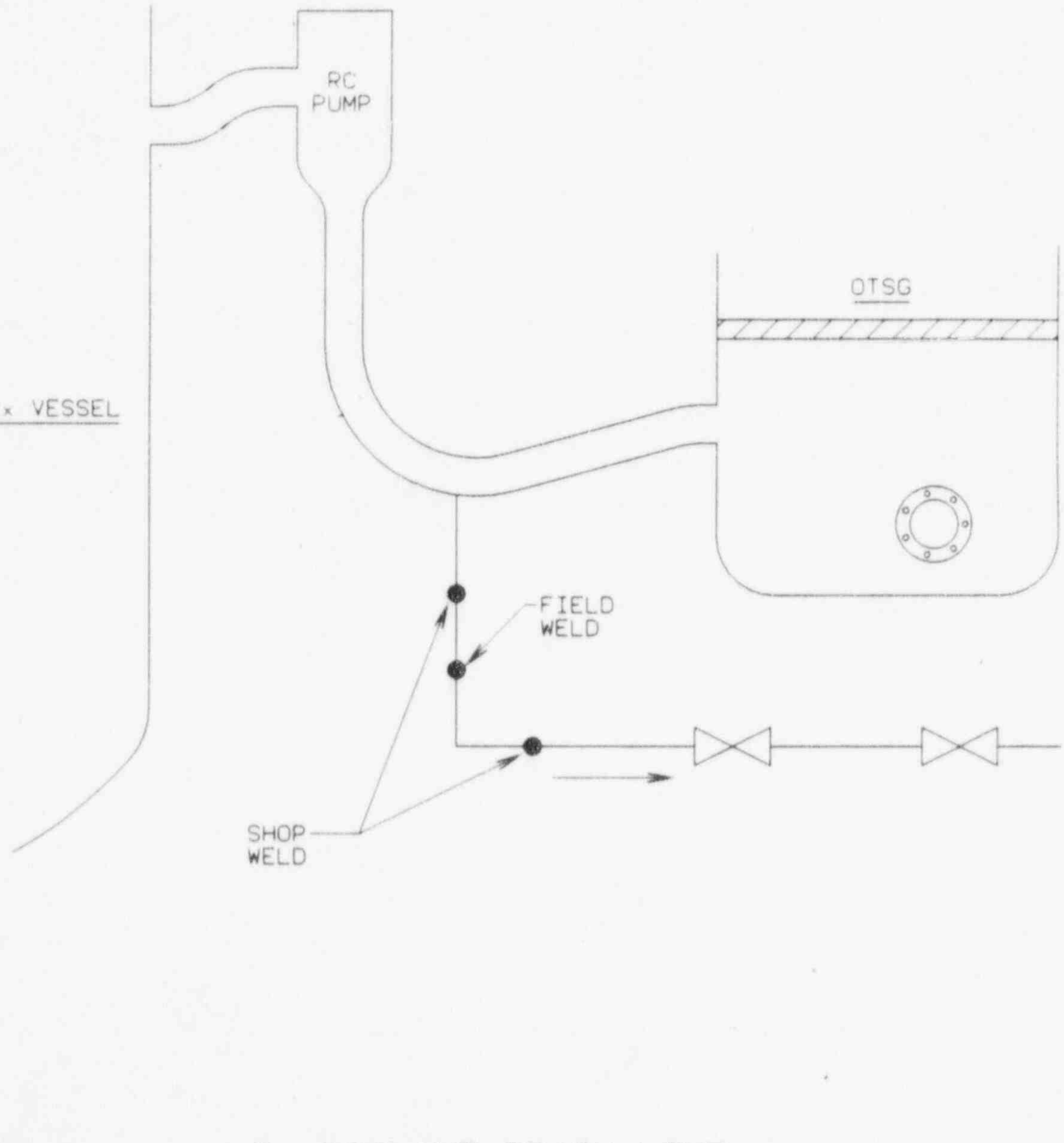
* The Energy Industry Identification System (EIIS), System Identification (SI) and Component Function Identification (CFI) Codes are included in brackets, "[SI/CFI]", where applicable, as required by 10 CFR 50.73(b)(2)(ii)(F).

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
THREE MILE ISLAND, UNIT 1	05000289	95 --	003 --	00	12 OF 15

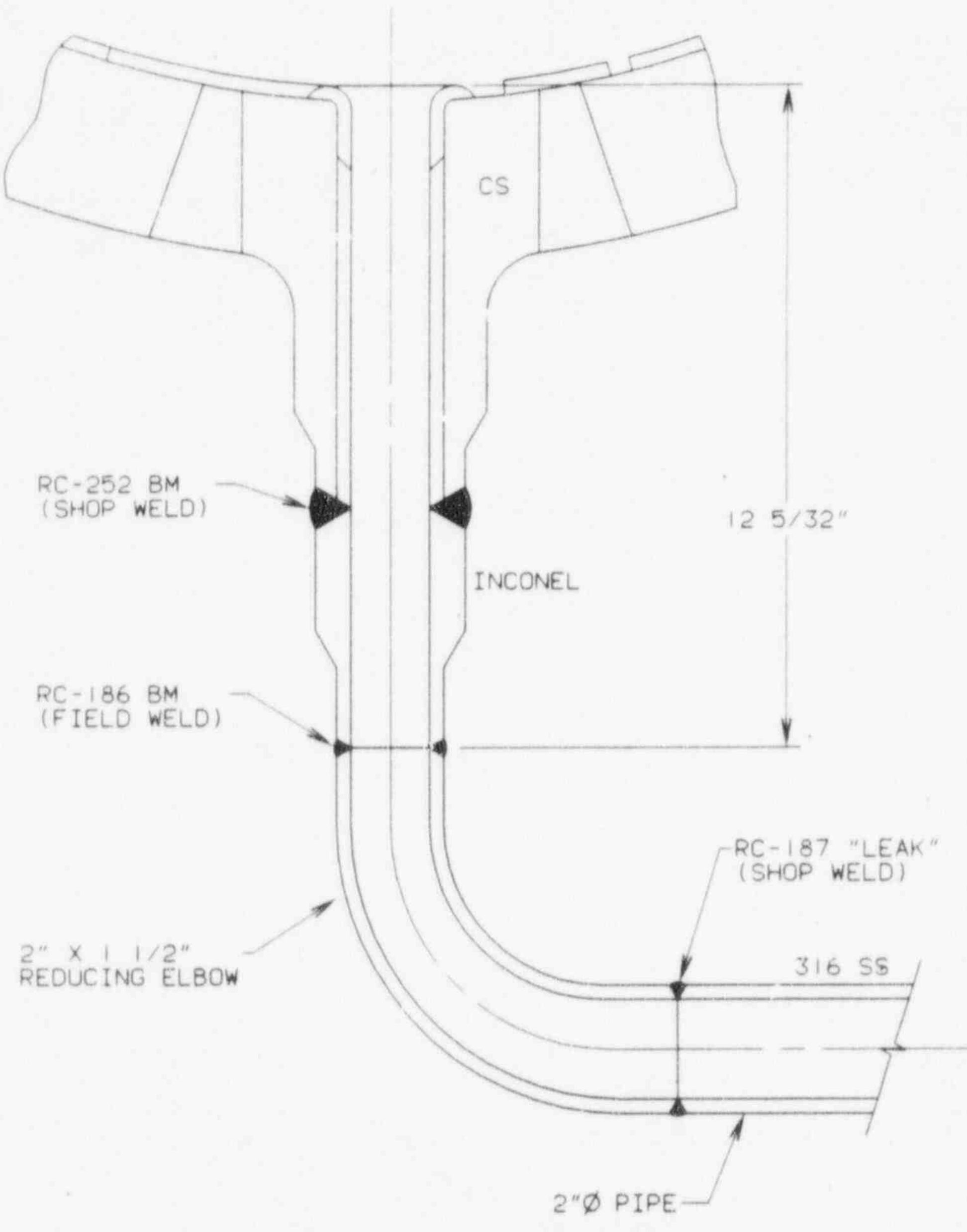
TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

FIGURE 1



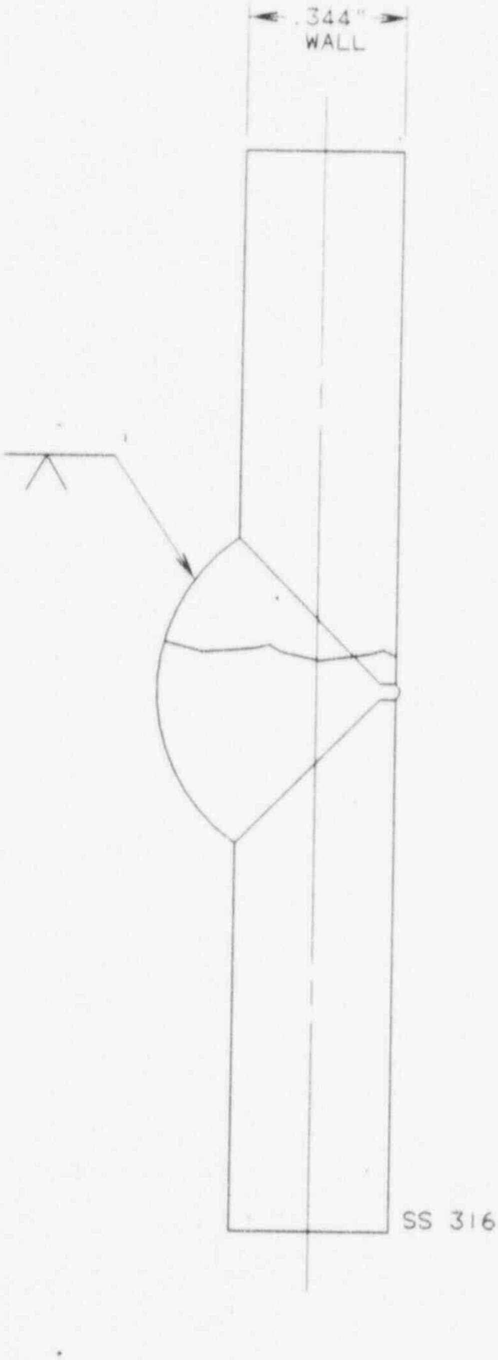
Rx COOLANT DRAIN LINE
(TYPICAL)

FIGURE 2



DRAIN LINE CONFIGURATION
(TYPICAL) "B"SHOWN

FIGURE 3

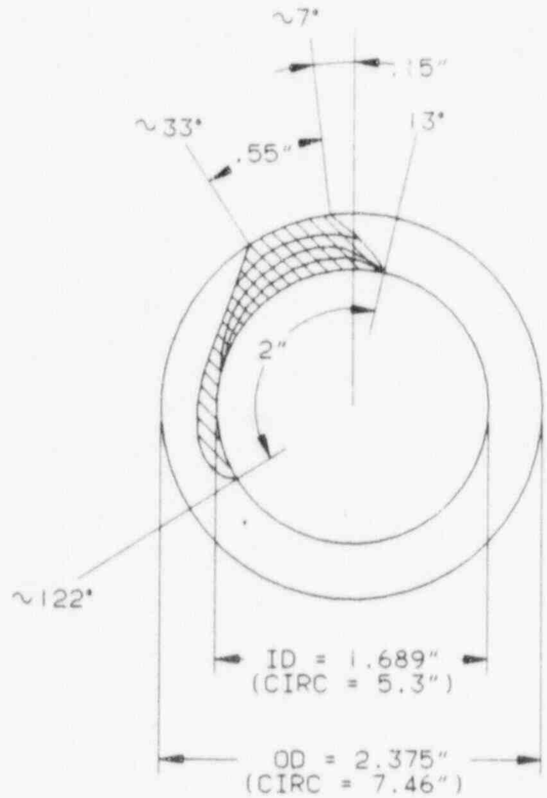


LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
THREE MILE ISLAND, UNIT 1	05000289	95 --	003 --	00	15 OF 15

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

FIGURE 4



CRACKING CONFIGURATION FROM UT REPORT