# GPU Nuclear

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April 6, 1994
C311-94-2041

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

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating Licensing No. DPR-50
Docket No. 50-289
LER 94-001-00

This letter transmits Licensee Event Report (LER) No 94-001-00. During Cycle 10 Operation, the Pressurizer Spray Valve (RC-V1) was found to have experienced boric acid degradation of the body to bonnet bolted fasteners. Conclusions regarding the root cause or corrective actions are incomplete pending the results of laboratory analysis. This information will be provided in a supplementary report which is planned for submittal around June 15, 1994.

Public health and safety were not affected. The abstract provides a brief description of the event. For a complete understanding of the event, refer to the text of the report.

Sincerely,

T. G. Broughton

T. G. Broughton
Vice President and Director, TMI

MRK

Attachment

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





IE22

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

# LICENSEE EVENT REPORT (LER)

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

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
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| THREE MILE ISLAND, UNIT 1 | 05000289 | 1 OF 11 |

TITLE (4)
BORIC ACID DEGRADATION OF PRESSURIZER SPRAY VALVE (RC-V1) FASTENERS

| 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 05000 |
| 03 | 07 | 94 | 94 | -- 001 -- | 00 | 04 | 06 | 94 | FACILITY NAME | DOCKET NUMBER 05000 |

OPERATING MODE (9): N
POWER LEVEL (10): 75

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.402(b) | | 20.405(c) | | 50.73(a)(2)(iv) | | 73.71(b) |
| | 20.405(a)(1)(i) | | 50.36(c)(1) | | 50.73(a)(2)(v) | | 73.71(c) |
| | 20.405(a)(1)(ii) | | 50.36(c)(2) | | 50.73(a)(2)(vii) | | OTHER |
| | 20.405(a)(1)(iii) | | 50.73(a)(2)(i) | | 50.73(a)(2)(viii)(A) | | (Specify in Abstract below and in Text, NRC Form 366A) |
| | 20.405(a)(1)(iv) | X | 50.73(a)(2)(ii) | | 50.73(a)(2)(viii)(B) | | |
| | 20.405(a)(1)(v) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(x) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| M. R. Knight, TMI-1 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 |
|---|---|---|---|---|---|---|---|---|---|
| E | AB | V | V085 | YES | | | | | |
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|
| X | YES (If yes, complete EXPECTED SUBMISSION DATE). | NO | | 06 | 15 | 94 |


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

BORIC ACID DEGRADATION OF PRESSURIZER SPRAY VALVE (RC-V1) FASTENERS

On March 7, 1994, TMI-1 was operating at reduced power as a result of having located and isolated a body to bonnet leak from the pressurizer spray valve (RC-V1). RC-V1 was declared inoperable. Because of boric acid degradation exhibited by RC-V1 fasteners, this event was found to be reportable in accordance with 10CFR50.72(b)(1)(ii) and 10CFR50.73(a)(2)(ii). With RC-V1 isolated, the plant was evaluated to be in a safe condition and operation was permitted while preparations were being made to shut down and repair RC-V1. The plant was shut down on March 17, 1994 and the valve was repaired. Pending completion of a root cause analysis and laboratory examination of several of the removed studs, GPUN will provide conclusions and corrective actions in a supplemental LER which is planned for submittal around June 15, 1994.

BORIC ACID DEGRADATION OF PRESSURIZER SPRAY VALVE (RC-V1) FASTENERS

I. Plant Operating Conditions before Event:

On March 7, 1994, TMI-1 was operating at reduced power (75%) as a result of having located and isolated a body-to-bonnet leak from the pressurizer spray valve (RC-V1). The condition of the valve was under investigation and attempts to stop the leak were being discussed.

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

None.

III. Background:

During normal plant operation, Reactor Coolant System (RCS) [AB/--][1] pressure is controlled by the pressurizer [AB/PZR] steam cushion in conjunction with pressurizer spray and pressurizer heaters. The pressurizer spray line originates at the discharge of the "A" reactor coolant pump. Pressurizer spray flow is controlled by a motor operated globe valve (RC-V1) in response to pressure set points. Motor operated valve (RC-V3), in series with RC-V1, provides a backup means of securing spray flow in the event the spray valve [AB/V] should stick open.

A small bypass line with a throttle valve (RC-V24) is provided around the spray valve (RC-V1). Bypass spray flow is provided to prevent thermal shock to the spray nozzle at the pressurizer shell-to-spray line interface. Bypass spray flow also allows a small amount of continuous flow through the pressurizer surge line which minimizes differences in boron concentration between the pressurizer and the RCS.

[1] 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).

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

During a down power transient, steam demand decreases causing the primary coolant temperature leaving the Once Through Steam Generators (OTSGs) to increase. The increase in the RCS temperature causes water to insurge into the pressurizer. As the pressurizer steam space is compressed some of the steam will condense to limit the pressure increase. If the transient is large enough, RC-V1 will automatically open to spray cooler water from the RCS cold leg into the pressurizer steam space to condense the steam and reduce RCS pressure.

Pressurizer spray flow is provided by the driving force created through the pressure differential between the reactor coolant pump (RC-P1A) discharge and the pressurizer. RC-V1 is capable of automatic or manual operation. When operated in manual control, a jog circuit is provided to control flow through the valve to control the rate of RCS depressurization. In the automatic control mode, the valve will open at a high pressure setpoint and remain open until the pressure decreases to the lower predetermined setting.

None of the conclusions regarding design basis transients or accidents identified in TMI-1 FSAR Chapter 14 rely upon operation of pressurizer spray[2]. RC-V1 is not required to shut down the plant, to maintain the plant in a safe shutdown condition, or to mitigate the consequences of any design basis accident. The only safety related function of RC-V1 is to serve as part of the reactor coolant pressure boundary unless it is isolated by closure of a manual valve (RC-V31) and a motor operated valve, RC-V3.

RC-V1 is a 2-1/2", 1500 Class Velan Motor Operated globe valve (Figure B9-374B-13MS) with a Limitorque motor operator. The valve has a bolted bonnet design with eight (8), 5/8" diameter studs and nuts. The studs are ASTM A193 Grade B7 which thread into the valve body. The nuts are ASTM A194 Grade 2H nuts. The bonnet-to-body joint is sealed with a spiral wound gasket. The gasket rests in a machined counterbore in the body and is compressed by the flat flange surface of the bonnet.

[2] Pressurizer spray was assumed to function during loss of electric load events; however, the spray had no significant impact on the event.

IV. Event Description:

On March 5, 1994 while at 100% power during the 1500-2300 shift, an increase of approximately .05 gpm RCS leakage was identified after receiving a Reactor Building (RB) particulate radiation monitor (RMA-2) alarm. A second RCS leakrate calculation (mass balance) was subsequently performed which validated the increase in leakage noted earlier. At 0730 on March 6, 1994 a control room operator observed steam in the vicinity of the pressurizer during a shift tour using the Reactor Building closed circuit video camera system. Plans were immediately initiated for a RB entry to investigate the leakage and assess the significance of the leak in accordance with Technical Specification (TS) 3.1.6.6. Total RCS leakage rate was approximately 0.21 gpm at that time. The leak rate was steady.

At 1246 on March 6, 1994 GPUN personnel entered the RB to assess the leak. The leak was identified as an RC-V1 body-to-bonnet leak. The insulation was removed from RC-V1 and a steady wisp of steam was observed to emanate from the body-to-bonnet joint in the vicinity of the (capped) packing leak-off line. The steam plume produced insignificant impingement force and was not directed toward any equipment that could be damaged by the plume. The accessible studs appeared to be in acceptable condition. However, due to the limited access around the valve, the direction of the leak and the configuration of the insulation on the valve, only half of the bolting was clearly visible.

The Plant Review Group (PRG) met at 1340 hours on March 6, 1994 to assess the safety significance of the leak. It was noted that the increase in total RCS leakage rate contributed by RC-V1 was approximately 0.06 to 0.10 gpm, well within the TS 3.1.6.1 limit of 10 gpm for identified leakage. Also, leakage was from a gasket in a bolted connection and not from a failure through a nonisolable fault in an RCS strength boundary (such as the reactor vessel, piping, valve body, etc.) The leak was isolable by closing RC-V3 and RC-V31. The PRG concluded that TS 3.1.6.6 was met, and that the leak did not require a plant shutdown at that time. It was decided to continue shiftly monitoring of RCS leakrate and monitoring the leak twice a shift using the RB video camera system.

Radiological Controls determined that the leak did not pose a problem for RB entry. There was no RB purge in progress, therefore there was no effect on offsite dose. It was also noted that initiation of RB purge would have had no significant effect on offsite dose.

On March 7, 1994, personnel entered the RB to attempt tightening the valve body-to-bonnet joint. The stud nuts were to be torqued to 120 ft-lbs as recommended by Plant Engineering. The vendor-recommended minimum torque was 95 ft-lbs. Torquing the nut nearest the leak resulted in no movement of the nut. A second nut (approximately 180° opposite the first nut torqued) was turned about one flat and the valve leakage suddenly increased. The Maintenance personnel later stated that movement of the fastener "felt soft." Although the torque wrench was placed on a third nut, Maintenance decided to stop work and exit the area due to increased leakage and the nut was not turned. Total RCS Leakrate was calculated at approximately 3 gpm after this attempt to torque the bonnet stud nuts.

In subsequent planning for a second attempt to retorque the bonnet stud nuts, it was determined that the valve had been in the closed position when the first torquing attempt was made. With the valve closed it was felt that the valve bonnet flange joint could have moved and caused the leakage to increase. A revised torquing plan was developed for the second attempt.

After further planning it was decided to reduce reactor power to about 75% and to isolate RC-V1 by closing the downstream valve, RC-V3, (motor operated from the Control Room) and the upstream valve, RC-V31 (local manual valve). This power level was chosen to reduce the rate of pressurizer insurge or outsurge associated with small load changes and hence offer greater stability to the plant with RC-V1 out of service. The valve was confirmed isolated by Operations personnel. RC-V1 was then cracked off its seat in preparation for retorquing the bonnet stud nuts.

At approximately 1835 on March 7, 1994 Maintenance and Engineering personnel attempted to tighten the valve joint by retorquing the studs. No movement was achieved when torquing of the first nut was attempted, which was about 90° counterclockwise from the leak area. Upon attempting to position the torque wrench on a second stud, located approximately 90° clockwise from the area of the leak, the stud/nut came out in the individual's hand. Closer inspection showed that three (3) studs immediately adjacent to the failed stud were significantly degraded. As noted above, this area had previously been difficult to examine closely because the leak was in an area where visibility was not good. The work area was secured and personnel exited the RB.

Because of the degraded condition exhibited by the RC-V1 studs, this event was found to be reportable on March 8, 1994 at 1405 hours in accordance with 10 CFR 50.72(b)(1)(ii) and 10 CFR 50.73(a)(2)(ii) and the NRC was notified at 1427 hours. With RC-V1 isolated, the plant was evaluated to be in a safe condition and operation could continue as preparations were made to shut down the plant and make the necessary repair to RC-V1. It was decided that the plant would be maintained at reduced power to lessen the pressure effects of transients. Also, loss of a feedwater pump from this reduced power would be less likely to result in a reactor trip.

On March 17, 1994 the plant was taken to Hot Shutdown conditions to repair the RC-V1. This outage is referred to as the second unscheduled outage during operating cycle 10 (10U2). The bonnet studs, nuts, and gasket were removed and replaced. Four of the eight studs were found severely degraded; of the four, two had completely failed. The remaining four studs showed only minor thread degradation. All studs had boron deposits in the threads. No drilling or cutting was required to remove the studs from the valve body or nuts from the studs.

The RC-V1 gasket seating surfaces (body and bonnet) were found in good condition with no steam cuts or pitting. An inspection of the spiral wound (SST/asbestos) bonnet gasket revealed that the asbestos filler material was washed out for over 1/4 of the gasket circumference. Although gasket orientation was not recorded before the gasket was removed, it is judged that there would be a direct correlation between the area where the gasket was degraded and with the area of the steam leak. The failed SST/asbestos spiral wound gasket was replaced with a SST/graphite filled spiral wound gasket provided by the valve manufacturer, Velan.

V. Analysis of the Event:

During the Cycle 10 Refueling Outage (10R)[3], mirror reflective metal and temporary fiber blanket insulation on the spray line (including RC-V1) was replaced with fiber blanket type insulation. The personnel who replaced the insulation noted no significant buildup of boric acid crystals around the valve. It should also be noted that during 10R, RC-V1 was repacked. This was a preventive maintenance action initiated because the valve had last been repacked in 1989. Upon repacking the valve it was identified that only four rings of packing were able to be installed. This deviated from previous repacking where five packing rings had been installed.

On October 14, 1993, during the heatup following the 10R Outage, a visual exam (VT-2)[4] of RC-V1 for leakage was performed at Hot Shutdown conditions in accordance with Surveillance Procedure (SP) 1303-8.1, "Reactor Coolant System." No indication of leakage at RC-V1 was noted.

On November 14, 1993, while shutting down to replace a leaking Pressurizer Code Safety Valve[5], Operations personnel noted a body-to-bonnet leak at RC-V1 at Hot Shutdown conditions and initiated a Maintenance Work Request for repair (J.O. 80474). Plant Maintenance performed an inspection and determined from their observations that RC-V1 leakage was from the valve packing and not body-to-bonnet. Therefore, the work request for repair of body-to-bonnet leakage was canceled and a work request was initiated to repair packing leakage.

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[3] The 10R Outage began on September 10, 1993 and ended on October 16, 1993.

[4] Technical Specification 4.2.1 and 10 CFR 50.55a(g) requires Inservice Inspection (ISI) compliance with the ASME Code Section XI, 1986 Edition. ASME Section XI, IWB-5210 requires a VT-2 Visual Examination prior to startup following each refueling outage. Surveillance Procedure 1300-6, "VT-2 Leakage Exam" provides additional guidance for performing the examinations.

[5] The first unscheduled outage during operating cycle 10 (10U1) began on November 14, 1993 and ended on November 19, 1993.

Repairs to the packing during 10U1 (J.O. 80478) included replacement of a cracked carbon bushing, installation of five rings of packing, and correcting an error found in the installation of Belleville spring washers on the packing bolts.

It is possible that the cracked bushing may have existed or occurred during the 10R repacking of RC-V1, thereby preventing the installation of all five packing rings and contributing to packing leakage. At Hot Shutdown conditions during the startup following the 10U1 packing repair, the packing was inspected and no leakage was identified. These conditions, along with evidence of boron on the underside of the motor operator, lent further credibility to the belief that the packing was leaking.

In addition to the packing repairs, Plant Maintenance had obtained torque values and approval from Plant Engineering to retorque the flange bolting. As a precautionary measure during the valve packing work, Maintenance personnel verified that the bonnet studs were torqued to the maximum recommended torque of 120 ft-lbs. No movement at 120 ft-lbs was noted by the Maintenance worker when the torque was verified.

When Maintenance performed their inspections during 10U1, the valve insulation around the bolted bonnet joint was removed from only one side (south side) of RC-V1; the leakage was directed northwest. During the visual inspection which resulted in canceling the work request for repair of a body-to-bonnet leak, Plant Maintenance recalls noting no degraded studs or significant boric acid deposits.

Pending completion of a root cause analysis and laboratory examination of several of the removed studs, GPUN will provide conclusions and corrective actions in a supplementary LER which is planned for submittal around June 15, 1994.

VI. Component Data:

Spray valve RC-V1 is a 2-1/2" 1500 Class motor operated globe valve (Figure B9-374B-13MS) manufactured by Velan Engineering Company and provided with a Limitorque motor operator. The valve has a bolted bonnet design with eight (8) 5/8" A-193, Grade B7 studs and A-194, Grade 2H nuts. The valve was designed in accordance with ASME Section III (Summer 1969

Addenda and March 1970 Draft Addenda) and ANSI B16.5. The studs thread into the valve body and the body-to-bonnet joint is sealed with a spiral wound gasket. The body and bonnet materials are ASME SA-182, Grade F316 stainless steel. The valve design pressure and temperature are 2500 psig and 670°F respectively. Normal operating pressure and temperature of the spray line are approximately 2205 psig and 555°F respectively.

VII. Automatic or Manually Initiated Safety System Responses:

No safety system responses occurred or were required to occur.

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

There were no safety consequences experienced from the degraded condition of the RC-V1 bonnet bolting and resultant leakage, however GPUN recognizes there are several significant safety implications. The RCS leakage never exceeded the Technical Specification 3.1.6.1 limit of 10 gpm for identified leakage. No RB purge was in progress or initiated during the valve joint leak. No other safety related equipment was damaged or rendered inoperable by the leak.

Increased leakage rate would have forced a shutdown upon reaching the Technical Specification leakage limits for identified leakage (10 gpm). However, if RC-V1 had been allowed to leak excessively over a long period of time with no action taken to isolate the leak, with accelerated corrosion due to a steady flow of boric acid across the studs, the joint could have failed. If the bolting had failed completely and the bonnet separated, a Small Break Loss of Coolant Accident (SBLOCA) could have resulted. SBLOCAs are within the design basis, and covered by plant procedure and training.

IX. Previous Events of a Similar Nature:

There have been no previous LERs related to degraded RCS component bolting.

X. Corrective Actions:

1. Initial Actions:

   a. RC-V1 was repaired during the 10U2 Outage and boric acid leakage in the area was cleaned up. The failed SST/asbestos spiral wound gasket was replaced with a SST/graphite filled spiral wound gasket provided by the valve manufacturer, Velan. It was noted that insulation had partially covered the body-to-bonnet joint. The studs do not extend through the lower flange, so insulation removal from the lower flange is not necessary. RC-V1 insulation was modified to ensure that no portion around the circumference of the body-to-bonnet joint is blocked from view.

   b. Prior to the 10U2 Outage, Engineering developed a list of other bolted bonnet valves[6] in the RB and Auxiliary Building which incorporate fasteners that might be susceptible to boric acid degradation in a fashion similar to RC-V1. Only one of the valves (MU-V113), although not leaking, showed evidence of prior leakage. MU-V113 exhibited some boric acid degradation, although not as severe as that of RC-V1. Three out of the eight studs on MU-V113 required replacement; the other five were acceptable for use.

   c. Three of the studs/nuts from RC-V1 have been sent to B&W Nuclear Technology (BWNT) for fractographic evaluation, metallographic evaluation, and chemical analysis. Additionally, at the request of the NRC, one stud/nut was sent to Brookhaven National Laboratory.

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[6] It is noteworthy that these valves were previously inspected in accordance with the ASME Code during startup in mid October 1993. However, because of the RC-V1 degradation, another thorough examination was performed prior to and during the 10U2 Outage.

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

2. Followup Action:

Followup corrective actions to prevent recurrence, which are being developed as part of the GPUN evaluation of the root cause(s), will be addressed in a supplement to this LER. Submittal of the supplement is planned for June 15, 1994.