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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

NOV 22 1977

Docket No. 50-346

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Toledo Edison Company
ATTN: Mr. James S. Grant
Vice President - Energy Supply
Edison Plaza
300 Madison Avenue
Toledo, OH 43652

Gentlemen:

This refers to the inspection conducted by Messrs. T. N. Tambling and T. L. Harpster of this office on September 26-30; October 5-7, 18-21, and 27, 1977, of activities at Davis-Besse Nuclear Power Station, Unit 1, authorized by NRC Operating License No. NPF-3 and to the discussion of our findings with Mr. T. Murray and members of your staff at the conclusion of the inspection.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

During this inspection, certain of your activities appeared to be in noncompliance with NRC requirements, as described in the enclosed Appendix A.

This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office within twenty days of your receipt of this notice a written statement or explanation in reply, including for each item of noncompliance: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further noncompliance; and (3) the date when full compliance will be achieved.

As discussed during the exit interview, it is requested that you submit a final followup report on the September 24, 1977 event. This report should include the chronology of events, pertinent transient data, evaluation of the transients and any long term effects, results of any testing and short and long term corrective action. This report will serve as a basis for a generic review of unusual transients.

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In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter, the enclosures, and your response to this letter will be placed in the NRC's Public Document Room, except as follows. If the enclosures contain information that you or your contractors believe to be proprietary, you must apply in writing to this office, within twenty days of your receipt of this letter, to withhold such information from public disclosure. The application must include a full statement of the reasons for which the information is considered proprietary, and should be prepared so that proprietary information identified in the application is contained in an enclosure to the application.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

Gaston Fiorelli, Chief
Reactor Operations and
Nuclear Support Branch

Enclosures:
1. Appendix A, Notice of Violation
2. IE Inspection Report No. 50-346/77-32

cc w/encl:
Central Files
Reproduction Unit NRC 20b
PDR
Local PDR
NSIC
TIC
C. Young Park, Power Siting Commission

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OFFICE ▶	RIII	RIII	RIII	RIII	RIII	
SURNAME ▶	Tambling/ls	Webster WW	Knop RCK	Little	Fiorelli RCK for	
DATE ▶	11/14/77					

Appendix A

NOTICE OF VIOLATION

Toledo Edison Company Docket No. 50-346

Based on the inspection conducted September 26-30, October 5-7, 18-21, and 27, 1977, it appears that certain of your activities were in noncompliance with NRC requirements below. The item is a deficiency.

Contrary to the approved Quality Assurance Manual and Criterion V of 10 CFR 50, Appendix B, Administrative Procedure 1823.00 was not completely adhered to in the logging and review of jumper - lift wires.

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U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report No. 50-346/77-32

Docket No. 50-346 License No. NPF-3

Licensee: Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, OH 43652

Facility name: Davis-Besse Nuclear Power Station, Unit 1

Inspection at: Davis-Besse Site, Oak Harbor, OH

Inspection conducted: September 26-30, October 5-7, 18-21, and 27, 1977

Inspectors: T. N. Tambling 11/15/77
T. L. Harpster 11/18/77

Other Accompanying Personnel:
W. Little, September 30, 1977
L. Engle, September 30, 1977
V. Leung, September 30, 1977
A. Szuklewicz, September 30, 1977
J. Mazetis, September 30, 1977
J. Rajan, September 30, 1977
J. Pittman, October 19-20, 1977
A. Plumber, October 19-20, 1977
R. Denning, October 19, 1977

Approved by: R. C. Knop, Chief 11/21/77
Reactor Projects Section 1

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Inspection Summary

Inspection on September 26-30, October 5-7, 18-21, and 27, 1977 (Report No. 50-346/77-32)

Areas Inspected: Investigated the causes, evaluation and corrective action associated with the sudden depressurization of the reactor coolant system on September 24, 1977, routine, unannounced inspection of plant operation, tour of plant areas, followup of modification to electrical grid system to meet licensee commitment, and nonroutine event reports. The inspection involved 110 inspector-hours onsite by two NRC inspectors.

Results: Of the four areas inspected, no items of noncompliance or deviations were found in three areas; one apparent item of noncompliance and two unresolved items were found in one area (deficiency - failure to properly implement procedure for jumper-lifted wire - Paragraph 6.b, and unresolved - open water tight door, - Paragraph 7 - apparent defect in a cable penetration seal, Paragraph 7).

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DETAILS

1. Persons Contacted

J. Evans, Station Superintendent
*T. Murray, Assistant Station Superintendent
*L. Stalter, Technical Engineer
*W. Green, Administrative Coordinator
*J. Buck, Operations QA Manager
*W. Derivan, Acting Operations Engineer
L. Grime, Reliability Engineer
D. Briden, Chemistry and Health Physicist
F. Faist, B&W Site Operations Manager

*Denotes those attending the exit interview.

The inspector also talked with and interviewed other licensee employees including members of the technical and engineering staffs, reactor shift crews, and startup test leaders. The inspector also participated in a meeting on September 30, 1977 at Davis-Besse that included representatives of NRR, TECo Engineering, TECo Corporate Management, Babcox-Wilcox Company and Bechtel Corporation.

2. Licensee Action on Previous Inspection Findings

(Closed) Noncompliance (50-346/77-16): Failure to properly document a review of a reportable occurrence and proper adherence to the administrative procedures for processing deviation reports. The inspector found that Administrative Procedure AD 1807.00 was revised to clarify the apparent awkwardness in the procedure and that the procedure is being implemented to insure required review of reportable and nonreportable occurrences tracked by the deviation report system. (Paragraph 6.a)

3. Loss of Steam Generator Feedwater Supply and Depressurization of Reactor Coolant System[1/]

On September 25, 1977, the licensee reported to Region III that a spurious trip signal in the Steam Feedwater Rupture Control System (SFRCS) on September 24, 1977 initiated a series of events that resulted in the loss of feedwater supply to the No. 2 Steam Generator, depressurization of the Reactor Coolant System (RCS) and

1/ Licensee submitted 14 day Licensee Event Report NP-32-77-16 on October 7, 1977.

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rupture of the rupture disc on the pressurizer quench tank that resulted in damage to the mirror insulation on the No. 2 steam generator. An NRC Region III inspector was dispatched to the site September 26, 1977 to investigate the results of the incident, the action being taken by the licensee and corrective action planned.

Based upon the inspectors review and upon telephone conversations between Region III and representatives of Toledo Edison Company, an immediate action letter was issued to the licensee on September 30, 1977. This letter designated the corrective action required before the reactor could be returned to operation.

In the initial review of the incident, the inspector reviewed current status of the plant, proposed corrective action, details of the event, its safety significance, operation of engineered safety features during the event, conformance of the limiting conditions of operations, possible generic aspects, and possible radioactive releases or contaminations.

In followup, the inspectors reviewed the licensee evaluations of the incident, the results of testing and completed and/or planned corrective action.

The findings are as follows:

a. Transient Chronology

Initial Conditions

The reactor was at the 15% plateau in the startup test program and had accumulated approximately one effective full power day (EFPD) history. A high pressure turbine pressure tap between the turbine and the governor valve was found to be cracked. Power was reduced to approximately 9% and the turbine was shutdown to repair the leak. Main feed pump turbine 1-2 was receiving steam from steam generator (OTSG) No. 2, and feeding both OTSG's through their respective startup feed control valves.

Sequence of Events

21:34:20 - A spurious half trip actuated the Steam-Feedwater Rupture Control System (SFRCS). The half trip closed the startup feed control valve to OTSG No. 2. The operator has

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only valve position demand signal indication and thus was unaware of the feed isolation. Other valves that would move on the half trip were already in the tripped position and thus gave no alarm. The alarms received by the operator were main steam lines 1 and 2 isolation valve (MSIV) solenoid trouble alarms which alarm only on the computer. (This was a partial arming of the control circuit that actuates the MSIV to close on a full SFRCS trip).

21:34:44 - Low level alarm OTSG No. 2. (Setpoint 24" startup range). MFP turbine 1-2 is still steaming off the generator but the feed is isolated causing the level to decrease rapidly.

21:34:56 - High temperature alarm loop 2. (Setpoint 560.6° F wide range cold leg). Decreasing OTSG No. 2 level reduces heat transfer capability.

21:35:16 - High pressurizer level alarm. (Setpoint 220"). Coolant is expanding into pressurizer from increasing loop temperature.

21:35:18 - OTSG No. 2 low level trip alarm. (Setpoint 17" startup range). This signal combined with the spurious half trip completes the logic for a full SFRCS trip. The full trip closed the MSIV's and lined up both OTSG's to the auxiliary feed system.

21:35:26-:49 - A reconstruction of data indicates that the pressurizer power relief valve actuated and cycled nine times before failing in the open position. The relief valve cycled about the setpoint (2255 psig) because a close relay which provides a 50 psi deadband was physically missing from the system. The rapid cycling apparently caused some deformation of the pilot valve stem. However, pilot valve failed open due to galling of the stem. This resulted in the electromatic relief valve failing open and led to a continued depressurization of the reactor coolant system (RCS).

21:35:36-:38 - AFPT's 1 and 2 discharge valves were open. AFPT 2 only came up to 2600 RPM (normal is 3600 RPM) because of binding in the woodward governor linkage. This corresponds to a shutoff head of approximately 700 psid. Thus, no water was fed into OTSG No. 2 as this pressure was considerably below OTSG No. 2 steam pressure until 11 to 15 minutes into the transient.

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21:35:55 - Pressurizer power relief temperature high. (Setpoint 200°F). Control room indication that relief valve had opened.

21:36:04 - AFP's 1 and 2 discharge valves open and water is being fed into OTSG No. 1.

21:36:07 - The operator manually tripped the reactor because pressurizer level was approximately 300" and rising. About two seconds after the trip, level reached 303" and started decreasing. Loop 2 hot leg temperature reached a maximum of approximately 584°F six seconds after the trip and started decreasing. Loop 2 cold leg temperature reached a maximum of approximately 579°F 14 seconds after the trip. RCS pressure continued to blow down and various Reactor Protection System (RPS) trips occurred from low pressure as designed.

21:37:17 - The Safety Features Actuation System (SFAS), incident level 1 initiated at 1600 psi. The pressurizer quench tank vent isolation valves closed on containment isolation due to SFAS actuation.

21:37:49 - At this time it was noted that HPI flow indicator FYIHP3A was not indicating flow into the RCS, however, it was later determined that the initial flow was blocked by two higher head makeup pumps injecting 140 GPM through this line.

21:40:22 - The containment normal sump pump came on indicating the quench tank rupture disc had blown. HPI pumps were shutdown at this time as pressurizer level was normal.

21:41:50 - Saturation pressure was reached in the reactor coolant system. Steam formation was probably occurring in the Reactor Coolant Pump (RCP) suctions.

21:43:41 - RCP's 1-1 and 2-2 were tripped. At this time the transient was essentially terminated with the exception of the subsequent recovery actions. The block valve for the failed electromatic pressurizer relief valve was closed approximately 20 minutes after the start of the incident.

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b. Analysis of the Failures and Corrective Action

(1) SFRCS Half Trip

The half trip logic in the SFRCS is used on certain valves in the steam and feedwater systems to meet the single failure criteria for isolation of the atmospheric steam line power vent, the MSIV bypass, main steam warmup drain and startup feed control valves. A one of four input will close these valves. (It should be noted that all except the main steam atmospheric power vent valve are normally in the closed position when operating above 15% power). A two of four logic is used on other valves in the Steam-Feedwater System.

Failure to see this spurious trip that initiated the incident was due to two reasons. The logic circuit requires only a signal duration of 25-35 milliseconds to lock in. The computer which is used to show the alarm condition is scanning at a one second interval. Therefore, trip signals of less than one second may not be seen on the computer. Other visual or alarm indications (such as annunicators) were not available in the control room.

Corrective action by the licensee was divided into two phases to rectify this problem. The first and immediate action was to connect six channel Brush Recorders on SFRCS input channels to provide detection of short term spurious signals. Followup action involves the installation of annunicator windows for half trips on steam generator low level, loss of reactor coolant pumps and steam-feedwater delta pressure. An annunicator presently exists on low steamline pressure. A feasibility study will be made to determine whether a time delay circuit can be used on the trip signal to make sure that the computer will also see short term signals (i.e. less than one second)

(2) Auxiliary Feedwater Pump

On a full SFRCS trip the two auxiliary feedwater pumps (AFP) are aligned and started to feed the two steam

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generators. The alignment is one AFP to one steam generator except for the main steamline break accident. For the steamline break (detected by low steamline pressure) the AFP to the affected steam generator is aligned to receive steam and feed the unaffected steam generator (the one without the line break).

When the spurious half trip isolated feedwater flow to No. 2 steam generator, a full SFRCS trip was initiated on low steam generator level (see Transient Chronology). The AFP's were properly aligned and started. However, No. 2 AFP only reached approximately 2600 rpm (vs desired 3600 rpm) due to a binding in the Woodward governor on the Terry Turbine driving the AFP. At 2600 rpm the maximum pump discharge pressure is approximately 700 psig. This head was not adequate to provide feedwater to the No. 2 steam generator until pressure in the steam generator decreased to this pressure (approximately 11 to 15 minutes into the event).

The loss of one steam generator for a controlled cooldown of the reactor is within the accident analysis which assumes only one steam generator available. However, the failure of the governor on the AFP turbine presents a generic or common mode failure problem. (This was reported by the licensee in a letter from L. E. Roe to J. G. Keppler dated October 11, 1977 in accordance with 10 CFR, Part 21.21(b)).

The licensee through the manufacturer analyzed the failure mechanism of the governor. It was concluded that under certain conditions the servomotor control driving the turbine speed control against the high speed stop places a misalignment force on the T-bar of the governor linkage. This misalignment force creates a potential for the governor to bind at a speed position less than design speed upon turbine startup. The governors for both AFP's were modified to correct this problem.

The failure of the No. 2 AFP to come up to speed and feed the No. 2 steam generator also resulted in the steam generator "boiling dry." This was concluded based upon the rapid rate of pressure decay inside the steam generator. Although this condition is not desirable, the incident is within the design analysis for the

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steam generator as supplied by the manufacturer. The analysis placed a 20 cycle limitation on the generator over its lifetime. The licensee has an administrative system for tracking operational transients over the lifetime of the plant to insure cycle limitations are not exceeded.

(3) Reactor Coolant System Depressurization

The transient on the secondary side caused a corresponding operational transient in the Reactor Coolant System (RCS). This transient, while not desirable, would normally be within the design capability of the sytem without damage to equipment. However, due to the failure of the electromatic relief valve on the pressurizer, to properly reset after relieving there was damage to the mirror insulations on the No. 2 steam generator, minor damage to a ventilating duct and spillage of reactor coolant inside the containment vessel.

The licensee's investigation into the failure of the electromatic relief valve revealed that the close relay was missing from its control circuit. The missing relay caused the relief valve to cycle around its set point of 2255 psig until the pilot valve steam stuck in the open position. This failure of the pilot valve caused the relief valve to remain open continuously relieving the pressurizer to the Quench Tank. The relief valve remained until approximately 20 minutes into the event when the operator closed the block valve to the relief valve. Lack of earlier recognition that the relief valve had failed open was due to the fact that the operator did not have positive indications of the valve position on the control board. To correct this problem, the licensee installed a position light on the control board to indicate the position of the pilot valve solenoid. (Usual indications that the relief valve opens initially is by a temperature monitor in the valve line).

The licensee's inspection of the pilot valve revealed that the stem stuck in the open position due to galling.

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There is no explanation as to why the reset relay was missing from the control circuit. A review of the pre-operational test procedure performed prior to and during hot functional testing shows that the electromatic valve functioned properly. Prior to the preoperational test, the licensee performed a yellow line (circuit checkout) of the control scheme. Both of these items indicate that the control relay had been in the circuit. It should be noted that the electromatic relief valve control circuits are not classified safety related and therefore do not fall within the normal quality control purview.

After inspection of the relief valve and replacement of the pilot valve, the licensee tested the valve by manually cycling the valve six times with the RCS at approximately 600 psig. The pilot valve stuck in the open position again after the sixth cycle.

Inspection of the pilot valve stem revealed some scratches on the stem and that the outside diameter of the stem was .0005 inches oversize. (Normal annulus clearance is .001 inches). At the recommendation of the valve manufacturer, the stroke of the solenoid for the pilot valve was reduced from 3/8 inches to 1/8 inches. This change in stroke still allows the pilot to function as designed and considerably reduces the stem surface area exposed to the steam, dirt, boric acid, etc. when the valve operates.

Reduction in surface area exposed to the steam flow prevents possible accumulation of contaminates on the bearing surface of the stem.

After the correction to the stroke travel and stem diameter, the electromatic relief valve was retested. The valve was manually cycled ten times at approximately 600 psig RCS pressure and once at approximately 2200 psig. The valve functioned as designed.

The RCS components are designed for forty cycles of a generalized depressurization transient in which the pressure drops 1400 psi and the temperature drops 62°F in fifteen minutes. In the actual transient on the RCS side, the pressure dropped approximately

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1300 psi in eight minutes and the cold and hot leg temperature in loop 2 dropped approximately 41.5 and 45°F respectively in 7.5 minutes.

An analysis of the transient was performed by B&W for the licensee. Based upon this analysis, B&W concluded that this transient was within the scope of the generalized depressurization transient previously analyzed.

(4) Reactor Coolant Pumps

Because the Reactor Coolant pumps (RCP) operated at or near the saturation pressure during portions of the transient, there was some concern of possible damage to pump shaft seal bearings and impellers. The operating condition was reviewed by the pump manufacturer. The manufacturer concluded that there was small risk of any damage. To provide assurance, the licensee performed instrumented tests in Mode 5 and 3 to verify normal operating parameters.

(5) Reactor Fuel

B&W also evaluated the possible effects on fuel performance as a result of the transient and concluded that there were no safety concerns with respect to the reactor fuel. This conclusion was based upon:

- Core burnup on September 24 was approximately 1 EFPD (no significant fission product inventory).

- Because of the low operating power history (15% and less) there was no significant decay heat source as compared to the source from the RCP's.

- Conservative estimate that the maximum ΔP between the internal fuel rod pressure and RCS pressure was 300 psi and a maximum clad temperature of 550°F at this ΔP.

(6) Containment Contamination from RCS

The spillage of RCS water did not constitute an airborne release problem inside containment. This was due in part

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because of the short operating history of the fuel assemblies. No detectable fission gases were present in the RCS. There was low level contaminates due to normal activation products in the RCS (activation of normal corrosion products). This contamination was controlled and cleaned up by the licensee using standard radiation control procedures.

(7) Training and Retraining

To insure that operating personnel understood the sequence of events, the licensee conducted retraining on the SFRCS. This training involved two four hour sessions. The first session involved a description and analysis of the event. The second session covered a detailed description of SFRCS. The training of the various shift teams plus other personnel was completed October 22, 1977.

4. Second Loss of Feedwater Transient

On October 22, 1977, a spurious half trip from SFRCS closed the startup feedwater valve to the steam generator. During this transient, all plant operating equipment performed as designed. Both AFP's started and reached 3600 rpm. The pressurizer electromatic relief valve actuated twice and reset as designed.

Although the 6 channel Brush recorders installed to record the source of the spurious signal did not record the event, the licensee was able by a process of elimination to isolate the possible sources. Two buffer amplifiers and three integrated circuit clips were replaced as probable causes for the spurious signal. The licensee is continuing his efforts to positively identify the source of spurious signal.

5. Electrical Grid Stability Modification

Per Condition 2.C.(3)(q) of Operating License NPF-3, the licensee submitted an evaluation and proposed modifications to electrical grid protective system to NRR for review. The purpose of these modifications is to insure adequate breaker coordination, alarm and isolation of the onsite electrical system in sufficient time to permit the required Class 1E equipment to operate in the event of offsite grid degradation.

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The inspector reviewed and examined the implementation of the subject modification as designated by the licensee's letter of July 18, 1977 (Serial No. 293) to J. E. Stoltz from L. E. Roe and Facility Change Request 77-217 (original and supplement No. 1). No deficiencies were identified by this review.

The review effort included review and examination of procurement records, certification for Class 1E equipment, work orders used for installation, setpoint changes and setting, safety review, SRB review of the facility change and procedures used, and discussions with members of the engineering staff and operating staff involved in the design and installation of the modifications.

6. Plant Operations - General

The inspector reviewed general plant operations including an examination of selected operating logs, jumper-lift wire logs, deviation reports for the period of July 1977 through October, 1977. This review was made to determine compliance with technical specifications and administrative procedure requirements.

a. Deviation Reports

While reviewing the deviation reports, the inspector noted that 38 reports (for the period July 11, 1977 to August 16, 1977) had not been filed in the master file. Seven of these were on the SRB agenda for final review and closeout. The others were still outstanding for final resolution. It was noted that although copies of these reports were not in the master file, the reports are logged and are being tracked by the technical section. For the reports found in the master file, many still remain open and require final closeout.

In response to a previous item of noncompliance (Inspection Report 77-16), the licensee initially revised Administrative Procedure AD 1807.00 on July 12, 1977 and approved it for implementation on September 9, 1977 to improve and clarify the review process for deviation reports. It was noted there has been an apparent improvement in the tracking and review of DVR's issued since the procedure revision.

However, even though the DVR's are used by the licensee as a control document that does not get closed out until all

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corrective action is completed (including corrective action covered by facility change request or action item record), there appears to be a large number of open DVR's in the files. Many of these can and should be closed in a more orderly matter. This need to reduce the large backlog of open DVR's was discussed with the licensee during an exit interview.

b. Jumpers and Lift Wire Log

The inspector reviewed the jumper and lift wire log and the implementation of Administrative Procedure AD 1823.00, Jumper and Lift Wire Control. From a previous inspection, (Inspection Report 77-16), it had been noted that a large number of jumper and lift wire tags were outstanding. The status of the licensee's effort to reevaluate the need of these jumpers and lift wires was reviewed. Considerable progress had been made and the effort is still in progress as noted by the internal review presently being conducted by the operations section.

The inspector selected several jumper-lift wire - tags numbers at random and verified that they do exist.

The need to expedite this effort was discussed in the exit interview.

AD 1823.00 jumper-lift wire log sheets requires the persons placing the tag to reference the work order request number and the reason for the tag on the log sheets. A review of the log sheets indicated that this was not being consistently done. Examples were:

- Tag No. 4073 through 4084, no reason was given for tags.
- Tag No. 4163 through 4166, no work order number was referenced.
- Tag No. 4123 through 4126, no reason was given for tags.

In the exit interview the inspector discussed how failure to provide this information is considered an inadequate review by the tagging supervisor (DBTS), the Shift Foreman who is suppose to review the current status each shift and the Operations Engineer (or his representative) monthly review. This failure to properly implement AD 1823.00 is considered an item of noncompliance.

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Also discussed in the exit interview was a possible inconsistency in how individual DBTS handled critical tags (tags with safety implications). The current procedure does not require documentation to show how and what the DBTS considered in the placement of critical tags.

7. Plant Tour

The inspector toured various areas of the plant to observe operations and activities in progress. This included general state of housekeeping, proper alignment of valves in the high pressure injection of SFAS, status of EVS boundary, leaks, pipe vibrations, radiation controls, shift manning, discussion with operating personnel concerning lighted annunciators, and review of a startup procedure currently being implemented.

No items of noncompliance were specifically identified. However, two items were left unresolved and there was one item of major concern.

As discussed in an exit interview, the plant is currently operating with a large number of lighted annunciators. The inspector stated that there appears to be a need for the licensee to review the current status of lighted annunciators for the purpose of eliminating nuisance alarms, alarms that apparently have logic problems or in which the setpoint span may be too tight. Examples are:

- Panel 1, window 2-8, Emerg D/G FOS T 1/2, Hi/Low. If tank is overfilled, light is lit (alarmed condition). The operators concern is low level since he has no other means to determine what the level is. A sudden change in the status of tank could go undetected (from Hi alarm to Low alarm).

- Panel 1, window 1-5 and 1-6, ESSEN Bus E-1 and F-1, Breaker not normal. Window always lit because there is no normal position.

- Panel 5, windows 5-1, SFAS CTMT Rad Low-Fail. Window remains always lit because of detector location. There is not sufficient background radiation to make the meter read above zero.

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On October 27, 1977, at approximately 1500 hours, the water tight door separating the two auxiliary feedwater pumps was found open during a tour. The door was immediately closed. This item is unresolved pending further investigation.

On October 27, 1977, a black oily material was found dripping from a ceiling cable penetration in the cable spreading room. This item is unresolved pending further investigation by the licensee.

8. Onsite Review of Inspection Process

The inspector was accompanied by two people from Battelle-Columbus and one person from the Office of Nuclear Regulatory Research on October 19 and 20, 1977. Battelle is currently under contract to the NRC to evaluate methods for applying WASH-1400 methodologies to the inspection process. Observations were made of the current inspection process.

9. Review of Nonroutine Events Reported by the Licensee

The inspector reviewed licensee actions with respect to the following listed nonroutine events reports to verify that the events were reviewed and evaluated by the licensee as required by Technical Specifications, that corrective action was taken by the licensee, and that safety limits, limiting safety system settings, and limiting conditions for operation were not exceeded. The inspector examined selected Operations Committee minutes, licensee investigation reports, logs, and records, and inspected equipment and interviewed selected personnel.

Two inoperable relative position indicators in Group 6 of CRD System (NP-32-77-15).

Loss of Reactor Coolant pressure due to failure of pressurizer power operated relief valve (NP-32-77-16). (See Paragraph 3 for details)

No items of noncompliance or deviations were identified.

The following licensee event reports were reviewed and closed out on the basis of an in-office review and evaluation:

a. Hydro test on High Pressure Injection line (NP-33-77-28).

b. Steam generator level limit exceeded (NP-33-77-30).

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c. Chlorine detector AE 5358A inoperable (NP-33-77-31).

d. Loss of Shield Building integrity for H_2 Purge System 18 month test (NP-33-77-33).

e. Planned replacement of overload heaters in Control Room Emergency Vent condensing unit (NP-33-77-37).

f. Loss of DC power to speed control switches for SS 815 and SS 816 for testing (NP-33-77-39).

g. Removal of Auxiliary Feedwater pump 1-2 from service to implement wiring change (NP-33-77-41).

h. Main steam supply line check valve bonnet leak (NP-33-77-42).

i. Loss of shield building integrity due to unlatched door (NP-33-77-47).

j. Containment normal sump total flow instrument string inoperable (NP-33-77-49).

k. AFP turbine 1-2 inoperable due to loss of speed control (NP-33-77-51).

l. AFP turbine 1-1 inoperable due to loss of speed control (NP-33-77-52).

m. AFP 1-2 inoperable due to ground in speed control switch (NP-33-77-53).

n. Flow path from boric acid storage system operable (NP-33-77-54).

o. Main steam isolation valve M 5100 failed closed due to loss of control air (NP-33-77-55).

p. NI-3 inoperable to connect to reactimeter (NP-33-77-59).

q. API for control rod 4 of group 4 inoperable (NP-33-77-63).

r. DH pump 1-2 made inoperable to install union in cooling water line (NP-33-77-66).

s. Channel 3 of RPS inoperable (NP-33-77-67).

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t. Hydraulic snubber EBD-19-H144 inoperable (NP-33-77-69).

u. Main steam line hydraulic snubber SR 17 and SR 11 inoperable (NP-33-77-70).

v. AFP 1-1 steam supply isolated for maintenance (NP-33-77-71).

w. Reactor coolant system T avg less than 525°F (NP-33-77-75).

10. Unresolved Item

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. Two unresolved items disclosed during the inspection are discussed in Paragraph 7.

11. Exit Interview

The inspectors met with licensee representatives (denoted in Paragraph 1) on October 7, 21 and briefly on October 27, 1977 to summarize the findings of the inspection. The licensee representatives made the following remarks in response to certain of the items discussed by the inspector.

October 7, 1977

Stated that in addition to the corrective action taken to date on the September 24, 1977 incident they would (Paragraph 3):

a. Instrument SFRCS inputs to help detect spurious signals.

b. Add SFRCS annunciator windows.

c. Study the feasibility of a time delay mechanism so that the computer can log short term spurious signals.

d. Complete training on the SFRCS by October 22, 1977.

e. Test the modified AFP turbine governors in place in Mode 3.

f. Test the electromatic pressurizer relief valve cold and hot (at approximately 600 psig). During a telecon on October 14, 1977, when the relief valve failed on the sixth hot cycle test, the licensee stated that they would retest the valve ten times at 600 psig and once at approximately 2200 psig.

g. Complete the testing of the reactor coolant pumps at a pressure equal to or above 1300 psig.

Acknowledge that the return to power operation was predicated upon the successfull completion of the above tests. Also stated that they would keep the inspector informed of the progress of the testing.

Acknowledge the inspectors request for a detailed followup report on the September 24, 1977 incident including a detailed analysis of the long term effect of the transients.

<u>October 21, 1977</u>

Acknowledge the inspectors concern about the jumper-lift wire logs and acknowledged the inspectors statement with respect to the apparent item of noncompliance (Paragraph 6.b.).

Acknowledge the inspectors concern about the status of control room annunciators and stated that they have been pursuing the problem. (Paragraph 7)

Acknowledge the inspectors statements about the number of outstanding deviation reports. (Paragraph 6.a.)

<u>October 27, 1977</u>

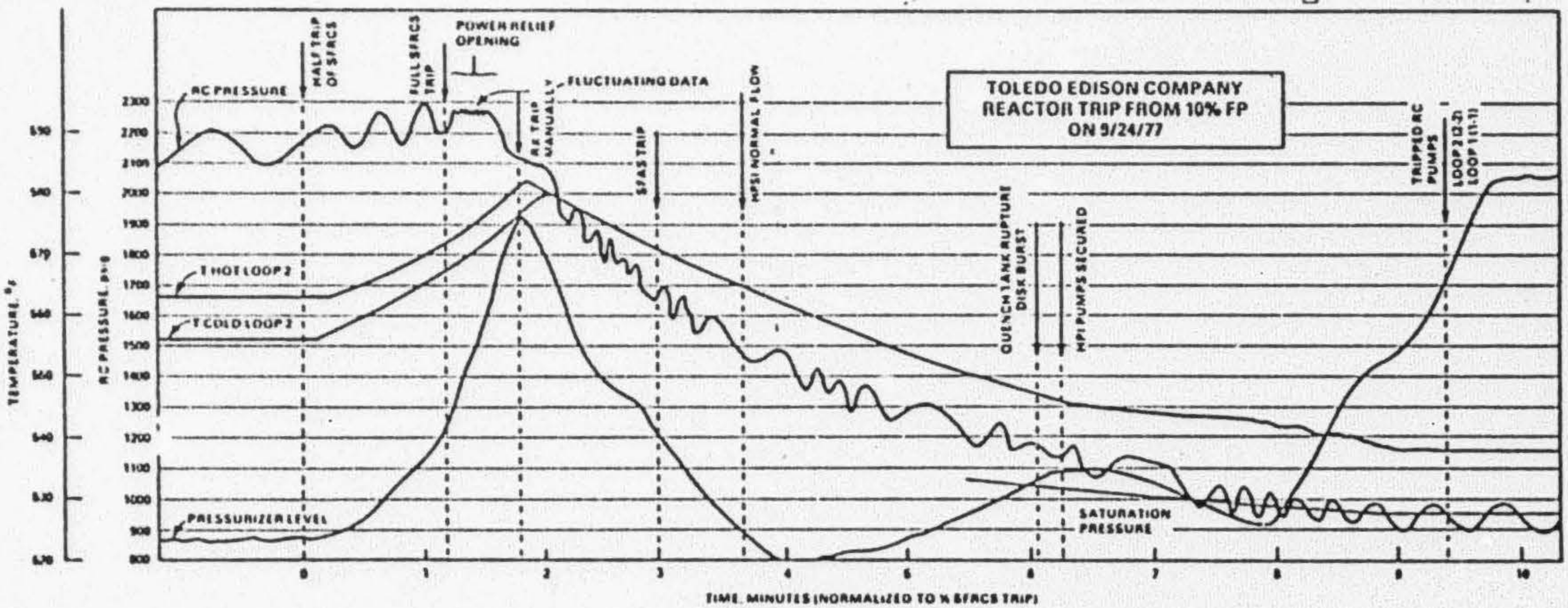
Acknowledge the inspectors review of the modification of the electrical grid system. (Paragraph 5)

Acknowledge the inspectors findings concerning the open water tight door and stated that they would check their procedures to determine why the door was left open. (Paragraph 7)

Acknowledge the inspectors findings concerning the black oily material dripping from a penetration in the cable spreading room and that they would immediately investigate and determine the extent of the problem. Informed the inspector on October 28, via telephone that the problem appeared to be confined to the one penetration and that they were pursuing it further with both Dow Chemical Company and BISCO. (Paragraph 7)

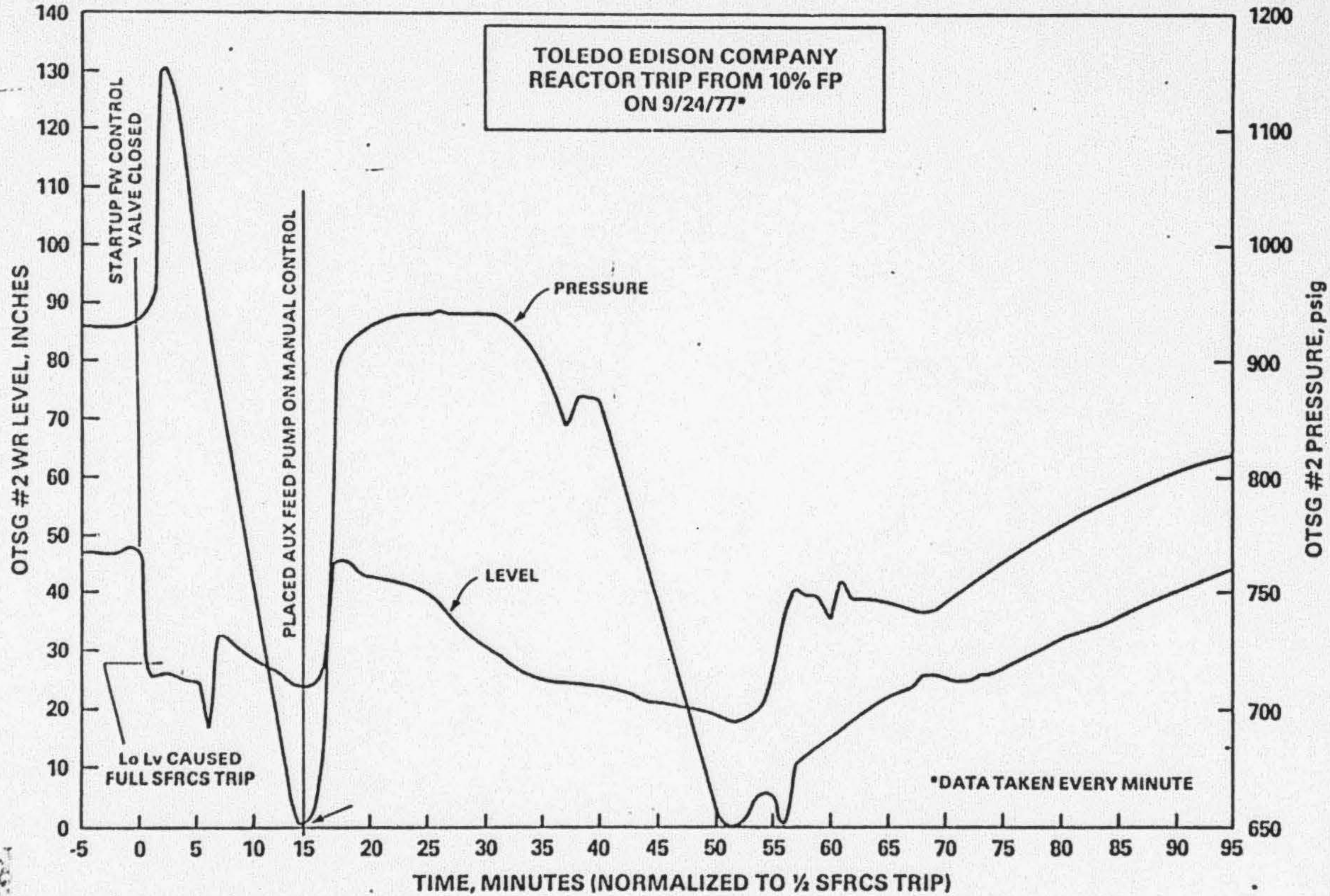
791144

POOR ORIGINAL

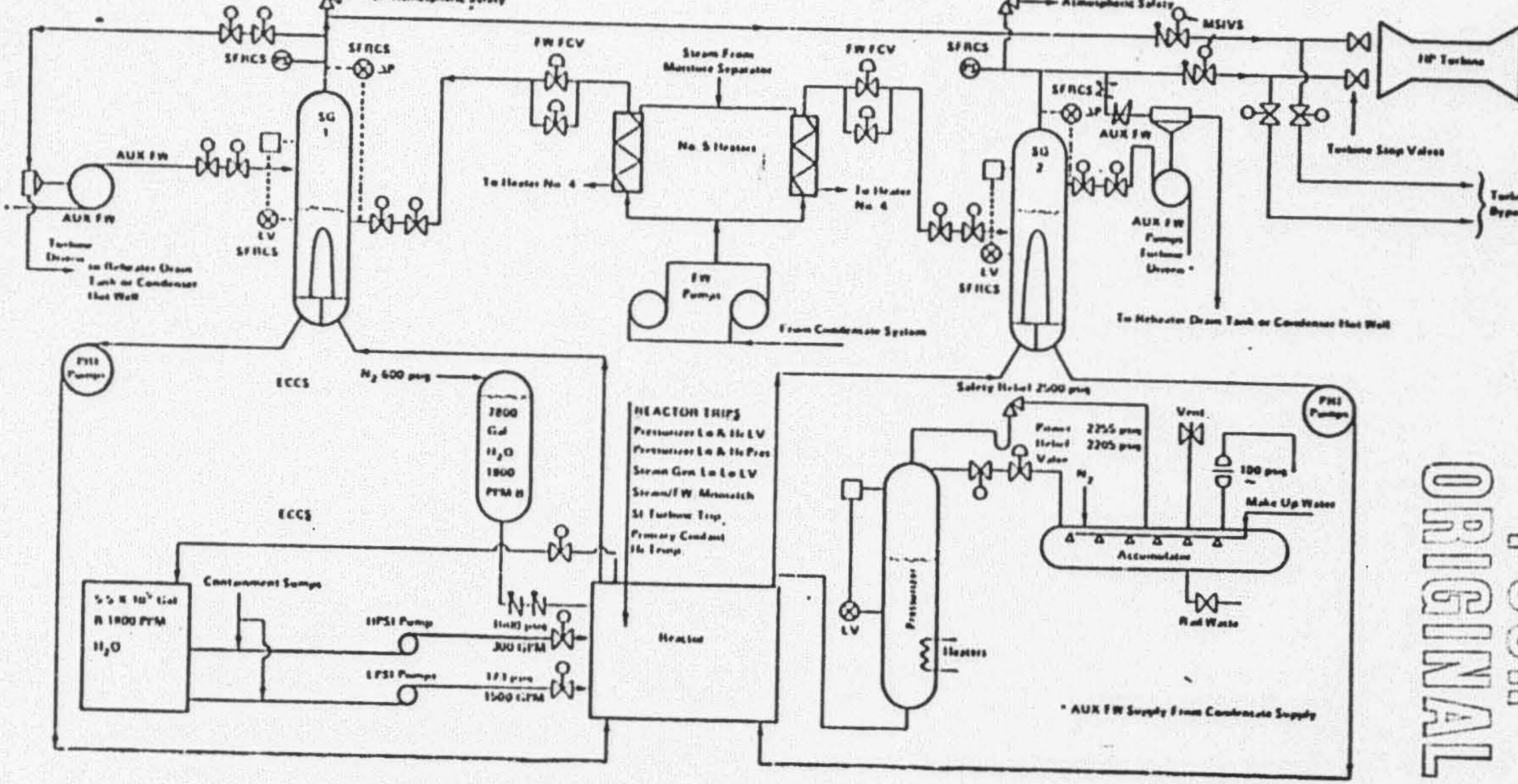


NOTE: THE PRIMARY SYSTEM WAS OPENED TO ATMOSPHERE THRU PRESSURIZER ACCUMULATOR FOR 16 MINUTES

DAVIS-BESSE UNIT 1 PRIMARY SYSTEM OPERATING CONDITIONS DURING OCCURRENCE ON 9/27/77


TOLEDO EDISON COMPANY
REACTOR TRIP FROM 10% FP
ON 9/24/77*
STARTUP FW CONTROL VALVE CLOSED
PLACED AUX FEED PUMP ON MANUAL CONTROL
PRESSURE
LEVEL
Lo Lv CAUSED FULL SFRCS TRIP
*DATA TAKEN EVERY MINUTE
OTSG #2 WR LEVEL, INCHES
OTSG #2 PRESSURE, psig
TIME, MINUTES (NORMALIZED TO ½ SFRCS TRIP)
140
130
120
110
100
90
80
70
60
50
40
30
20
10
0
1200
1100
1000
900
800
750
700
650
-5
0
5
10
15
20
25
30
35
40
45
50
55
60
65
70
75
80
85
90
95

PLANT SYSTEM INTERACTIONS



POOR ORIGINAL

792147

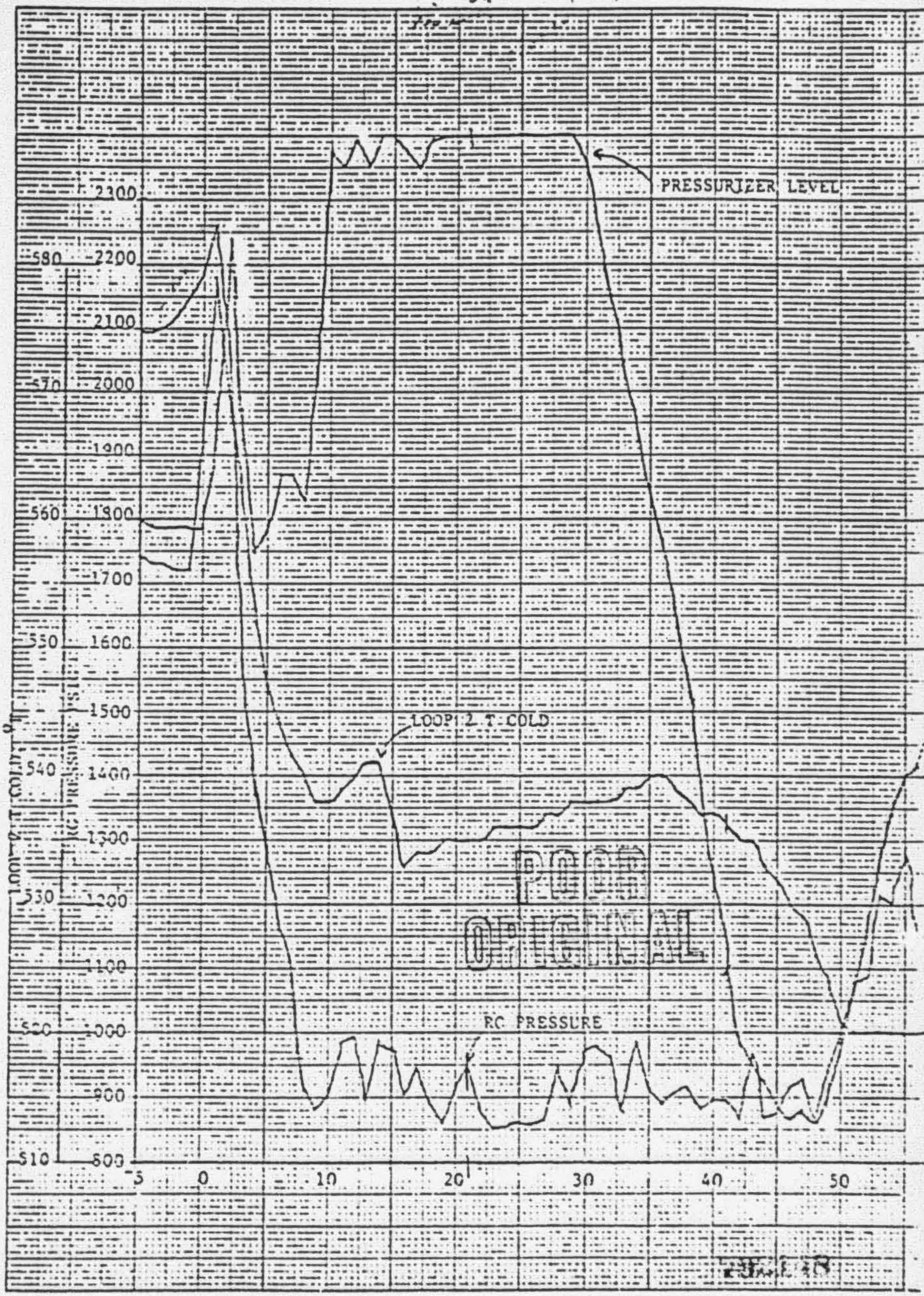
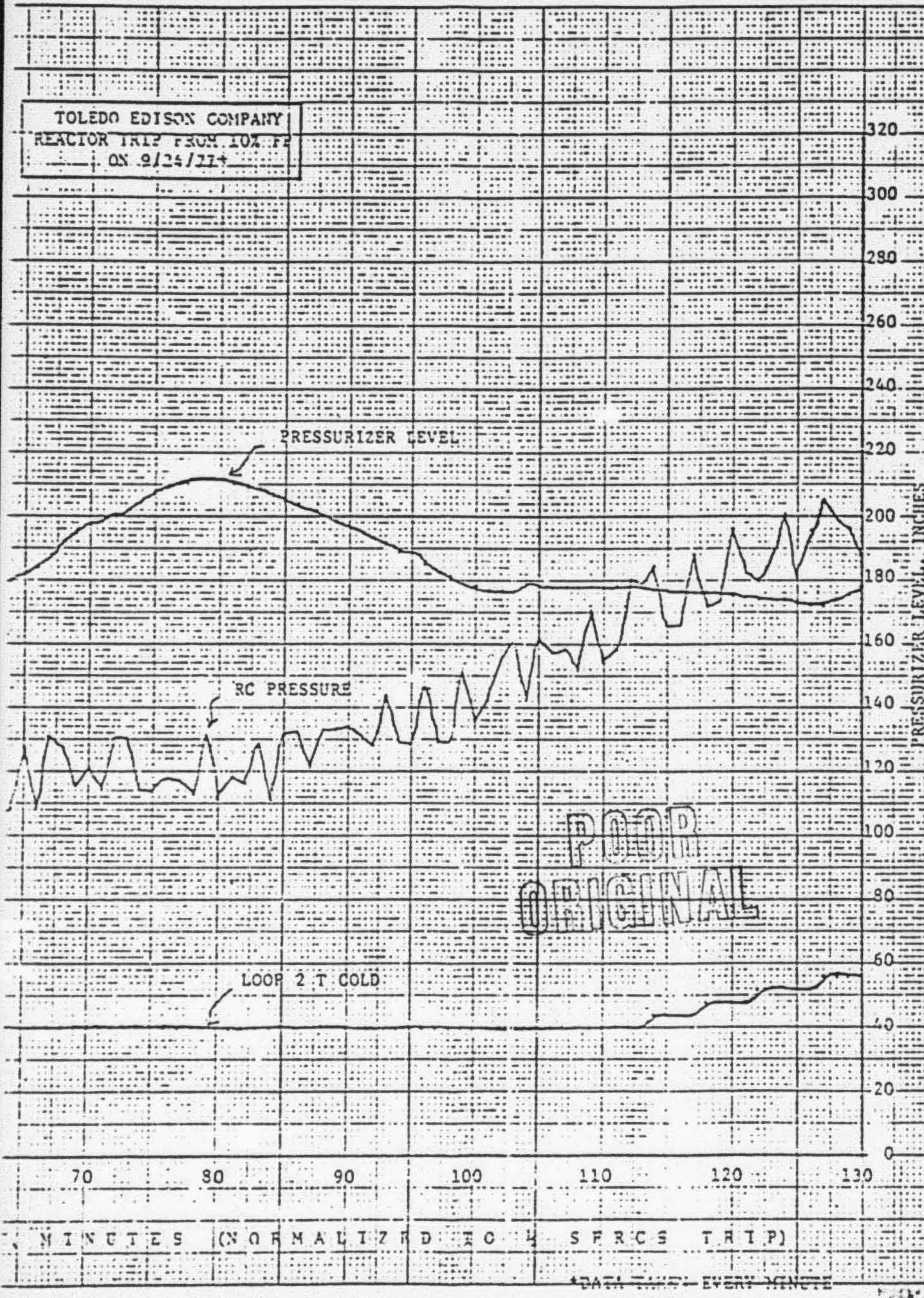
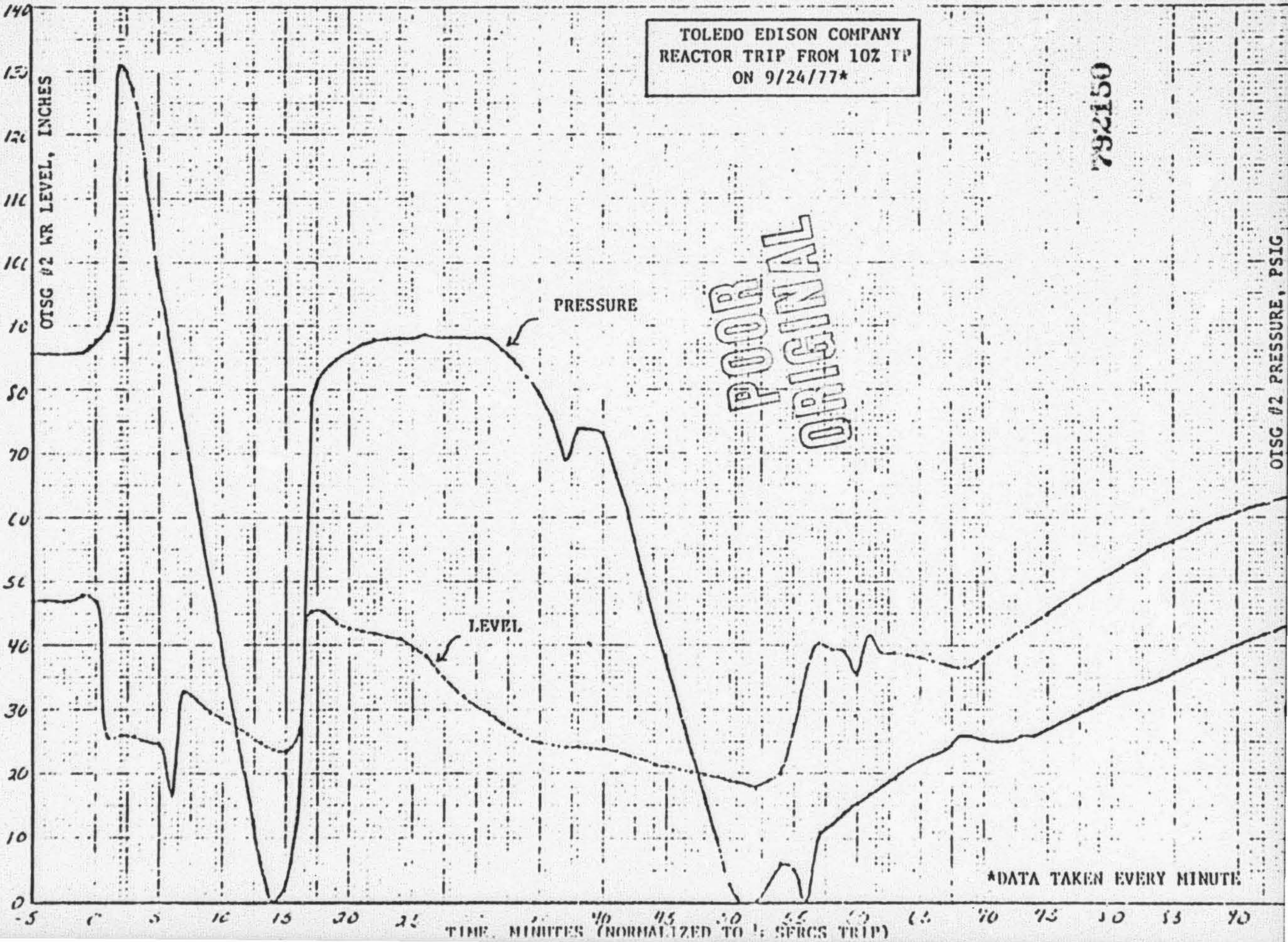

PRESSURIZER LEVEL
LOOP 2 T COLD
RC PRESSURE
LOOP 2 T COLD °F
RC PRESSURE PSIG
580
570
560
550
540
530
520
510
2300
2200
2100
2000
1900
1800
1700
1600
1500
1400
1300
1200
1100
1000
900
800
-5
0
10
20
30
40
50
POOR ORIGINAL

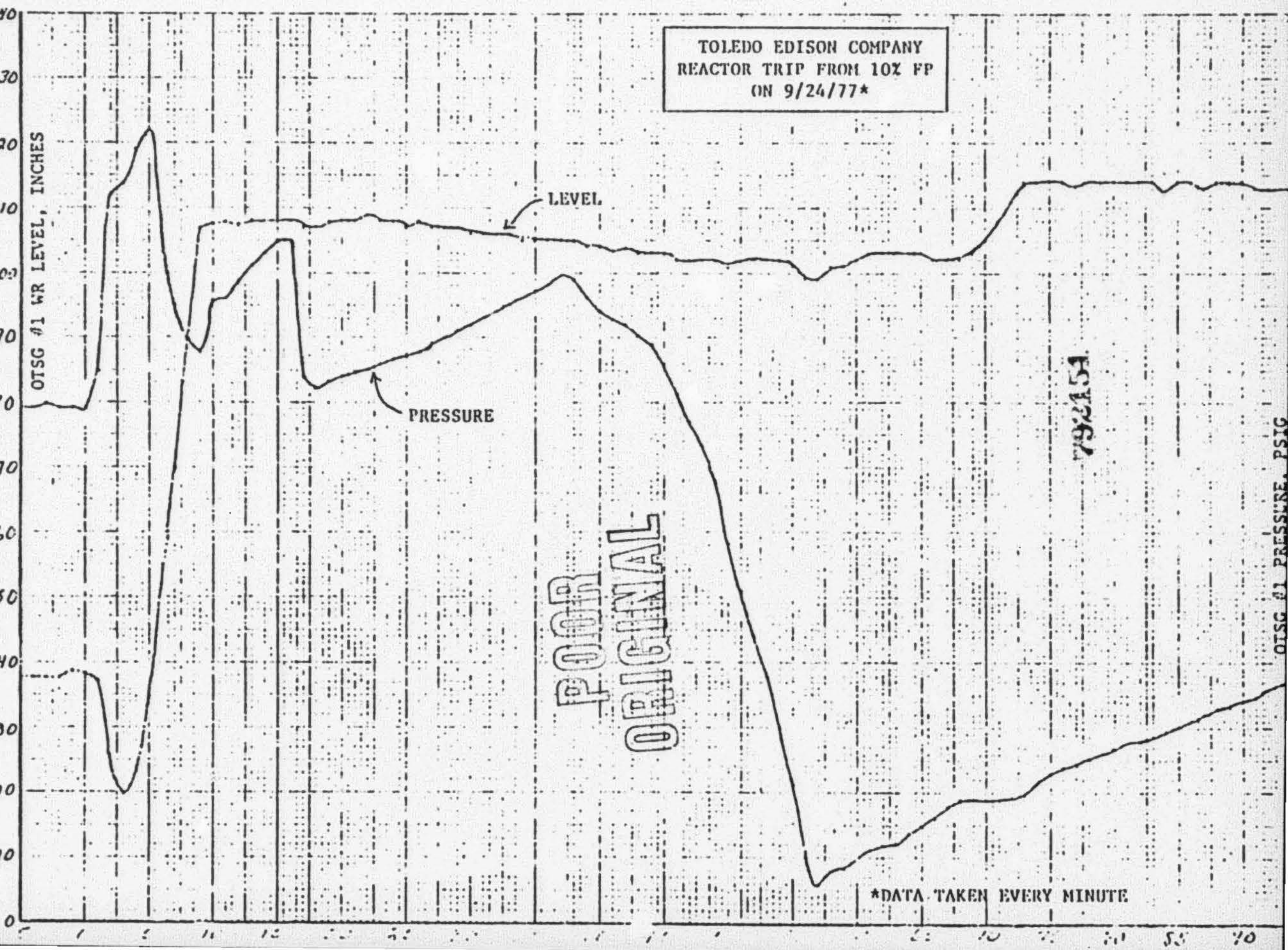
FIGURE 7-2



TOLEDO EDISON COMPANY
REACTOR TRIP FROM 10% FP
ON 9/24/77*
PRESSURIZER LEVEL
RC PRESSURE
LOOP 2 T COLD
POOR ORIGINAL
PRESSURIZER LEVEL, INCHES
0
20
40
60
80
100
120
140
160
180
200
220
240
260
280
300
320
70
80
90
100
110
120
130
MINUTES (NORMALIZED TO 4 SFRCS TRIP)
*DATA TAKEN EVERY MINUTE

792149


TOLEDO EDISON COMPANY
REACTOR TRIP FROM 10% FP
ON 9/24/77*
792150
PRESSURE
LEVEL
POOR ORIGINAL
OTSG #2 WR LEVEL, INCHES
OTSG #2 PRESSURE, PSIG
*DATA TAKEN EVERY MINUTE
TIME, MINUTES (NORMALIZED TO SFRCS TRIP)


TOLEDO EDISON COMPANY
REACTOR TRIP FROM 10% FP
ON 9/24/77*
LEVEL
PRESSURE
OTSG #1 WR LEVEL, INCHES
OTSG #1 PRESSURE, PSIG
*DATA TAKEN EVERY MINUTE
POOR ORIGINAL
140
130
120
110
100
90
80
70
60
50
40
30
20
10
0

9.3.3 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

The auxiliary building drainage system includes the following: equipment drains, floor drains, sumps, miscellaneous waste drain tank, and detergent waste drain tank. Radioactive equipment and floor drains discharge either directly into the miscellaneous drain tank or into sumps and are pumped into the drain tank for radioactive waste processing. Low-radioactivity wastes from designated laboratory sinks, laundry washing machine, lavatories, and shower room drains are collected in the detergent waste drains tank for radioactive waste processing. Nonradioactive areas, such as the electrical penetration rooms, electrical switchgear rooms, cable spreading room, and control room, discharge to the station external drain system. Oil interceptors are provided for the emergency diesel generators and diesel fuel day tank room drains. Acid-neutralizing tanks are provided for all battery rooms drains.

Turbine building equipment and floor drains are collected in low-point sumps and pumped through oil separators before discharging into the station exter-
nal drain system. In addition to the normal sump pumps, there are two high- 10
capacity pumps capable of handling a total of 1200 gpm of fire sprinkler flow. These pumps operate on high-high sump level, and the discharge is directly to the main transformer pit which has the capacity for the high flow and provisions for oil separation. In the event of radioactivity in the secondary system, these drains can be pumped to the condensate demineralizer holdup tanks.

The normal sump in the containment vessel discharges directly into the miscellaneous drain tank or the clean waste receiving. The pumps are of the submersible type.

All sumps are provided with a duplex pump system with an alternator to automatically start, stop, and alternate each of the pumps and to start the second pump any time the one in operation is unable to carry the load. All sumps are provided with high-level alarms.

The submersible sump pumps in the ECCS pump rooms, which normally discharge to the miscellaneous drain tank, are provided with a bypass to discharge directly to the clean waste receivers should it become necessary if a room is flooded. These pumps and piping are designed to Seismic Class I.

Applicable design codes and standards are listed in Table 9-1.

A flow diagram of this system is shown in Figures 9-9 and 9-9A. 4

Disposal of collected radioactive drains is discussed in Chapter 11.

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9.3.4 MAKEUP AND PURIFICATION SYSTEM

9.3.4.1 Design Bases

The makeup and purification system is designed to accommodate the following functions during normal reactor operation:

a. Supply the reactor coolant system with preoperational fill and operational makeup water.

b. Provide seal injection water for the reactor coolant pumps.

c. Provide for purification of the reactor coolant to remove corrosion and fission products.

d. Control the boric acid concentration in the reactor coolant in order to control reactivity.

e. In conjunction with the pressurizer, accommodate temporary changes in reactor coolant volume due to small temperature changes.

f. Maintain the proper concentration of hydrogen and corrosion inhibiting chemicals in the reactor coolant.

g. Provide makeup to the reactor coolant system for protection against small breaks in the reactor coolant pressure boundary.

9.3.4.2 System Description

9.3.4.2.1 General

The makeup and purification system is shown schematically in the process and instrumentation drawing, figure 9-21. tables 9-10 and 9-11 list the system performance and component design characteristics.

Letdown Cooler

The letdown cooler reduces the temperature of the letdown flow from the reactor coolant system to a temperature suitable for demineralization and injection to the reactor coolant pump seals. The letdown coolers are two shell and tube heat exchangers arranged in parallel. Each is designed to handle half the maximum letdown flow. Heat in the letdown cooler is rejected to the component cooling water passed through the shell side of the coolers.

Letdown Flow Control

The normal letdown flow rate at reactor operating pressures is controlled by a fixed block orifice sized for the normal purification rate. A parallel, normally closed, remotely operated valve can be opened to obtain flow rates up to the maximum letdown capability, e.g., reactor coolant boron concentration adjustment. This valve is also used to maintain the desired letdown rate at reduced reactor coolant pressure, i.e., during startup and shutdown. In addition, there is a second parallel, normally closed valve which may be manually positioned for flow control.

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Table 9-10

Makeup and Purification System Performance Data

Normal letdown flow, gpm	45
Maximum letdown flow, gpm	140
Total flow to each reactor coolant pump seal, gpm	8
Seal inleakage to reactor coolant system per reactor coolant pump, gpm	7
Temperature to reactor coolant pump seals, F	120
Purification letdown fluid temperature, F	120
Makeup tank normal operating pressure range, psig	15 - 35
Makeup tank nominal water volume, ft^3	400

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Table 9-11

Makeup and Purification System Component Data

Makeup Pump

Quantity	2
Type	Centrifugal, Mechanical Seal
Rated capacity, gpm	150
Rated head, ft at sp gr = 1	5800
Motor horsepower	450
Pump material	SS Wetted Parts
Design pressure, psig	3050
Design temperature, F	200
Seismic Class	I
Code	Draft ASME P&V, Class 2

Letdown Cooler

Quantity	2 Half-Capacity (Maximum Letdown)
Type	Shell and Tube
Heat transferred, Btu/h	16.2×10^6
Letdown flow lb/h	3.5×10^4
Letdown temperature change, F	557 to 120
Material, shell/tube	CS/SS
Design pressure, shell/tube, psig	200/2500
Design temperature, shell/tube, F	350/600
Cooling water temperature change, F	95-176
Seismic Class	I (shell and anchors) 10
Code	ASME Section III-C and VIII

Seal Return Cooler

Quantity	2 Full Capacity
Type	Shell and Tube
Heat transferred, Btu/h	40×10^4
Seal return cooler tubeside flow, lb/h	2.122×10^4
Material, shell/tube	CS/SS

Table 9-11 (Cont'd)

Design pressure, shell/tube, psig	150/150
Design temperature, shell/tube, F	250/200
Cooling water flow, lb/h	1.9628×10^4
Cooling water temperature change, F	95-115
Seismic Class	I
Code	ASME Section III-C and VIII

Makeup Tank

Quantity	1
Volume, ft3	600
Design pressure, psig	100
Design temperature, F	200
Material	SS
Seismic Class	I
Code	ASME Section III-C

Purification Demineralizer Prefilter

Quantity	1	
Capacity, gpm	140	
Vessel material	SS	
Vessel design pressure, psig	150	
Vessel temperature, F	200	
Nominal rating, microns	3-5	
Vessel seismic Class	II	10
Vessel code	ASME III-Class 3	

Purification Demineralizer

Quantity	2
Type	Mixed Bed, boric acid saturated
Vessel material	SS
Usable resin volume, ft3	50
Design flow, gpm	70
Vessel design pressure, psig	150
Vessel design temperature, F	200

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Table 9-11 (Cont'd)

Vessel seismic Class	I
Vessel design code	ASME Section III-C
Purification Filter	
Quantity	2
Capacity, gpm	140
Vessel material	SS
Vessel design pressure, psig	150
Vessel design temperature, F	200
Nominal rating, microns	3-5
Vessel seismic Class	II
Vessel design code	ASME III-Class 3
Seal Injection Filter	
Quantity	2
Capacity, gpm	50
Vessel material	SS
Vessel design pressure, psig	3050
Vessel design temperature, F	200
Nominal rating, microns	3
Vessel seismic Class	I
Vessel design Code	ASME III-Class 2
Purification Demineralizer	
Quantity	1
Type	Cation, Non-regenerative
Design flow, gpm	140
Usable resin volume, ft3	51
Vessel design pressure, psig	150
Vessel design temperature, F	200
Vessel material	304 SS
Vessel seismic Class	II
Vessel design Code	ASME III, Class 3

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Letdown Flow Radiation Monitor

The letdown flow radiation monitor is discussed in subsection 11.4.2.2.1.

Purification Demineralizer Prefilter

The purification demineralizer prefilter is designed to remove particulate matter from the letdown stream prior to entering the purification demineralizer. This filter minimizes any accumulation of radioactive crud in the purification resin and on the downstream piping of the purification system and waste disposal system. The prefilter is of the backflushable type and is sized to handle the maximum letdown flow rate.

Purification Demineralizers (Mixed-bed)

Two mixed-bed demineralizers are supplied. These are boric acid saturated and are used to remove reactor coolant impurities other than boron. Since the reactor coolant may be contaminated with fission and corrosion products, the resins will remove certain radioactive impurities. Chapter 11 describes coolant activities, coolant handling and storage, and expected limits on activity discharge. The two mixed-bed demineralizers are used for normal purification with one unit in operation and the other on standby. Each demineralizer is sized to handle one-half the maximum letdown flow rate.

Purification Demineralizer (Cation)

The cation purification demineralizer is used primarily to keep the concentrations of Cs-137 and Li-7 in the reactor coolant within acceptable limits. It is normally bypassed and is put into service only when it is needed. This demineralizer is sized to handle the maximum letdown flow and can be used instead of, or in series with, the two mixed-bed purification demineralizers.

Purification Filters

Two makeup filters are installed in parallel to remove particulates from the effluent of the purification demineralizers. This prevents solids from entering the makeup tank and thus from entering the reactor coolant pump seals and the reactor coolant system. Each filter is sized for maximum letdown flow.

Makeup Tank

The makeup tank serves as a receiver for letdown, seal return, chemical addition, and makeup. The tank also accommodates temporary changes in system coolant volume. The volume of the tank is such that the useful tank volume, in conjunction with the pressurizer, accommodates the expected expansion and contraction of the reactor coolant system during normal power transients. (between 15% and 100% power)

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Makeup Pumps

The makeup pumps are designed to return the purified letdown flow to the reactor coolant system and supply the seal water flow to the reactor coolant pumps. One pump provides normal makeup to the reactor coolant system plus the required seal water flow. Both pumps are driven by motors receiving power from redundant essential buses. 13

Seal Return Coolers

The seal return coolers are sized to remove the heat added by the makeup pump recirculation and the heat picked up in passage through the reactor coolant pump seals. The two shell and tube coolers are arranged in parallel and each is sized for full flow. Heat from these coolers is rejected to the component cooling water passing through the shell side of the coolers.

Seal Injection Filters

Two filters of the disposable cartridge type are installed in parallel in the seal injection line. These filters remove particulates which could enter the reactor coolant pump seals and result in increased seal wear. Each filter is sized to handle normal seal injection flow.

9.3.4.2.2 Mode of Operation

The makeup and purification system is operated during all phases of the nuclear steam supply systems (NSSS) operating life, including startup, power operation, and shutdown. The system may also be operated during refueling by employing the purification equipment through interconnections to the decay heat removal system. During normal NSSS operation, one makeup pump continuously supplies high pressure water from the makeup tank to the seals of the reactor coolant pumps, and to a makeup line which is connected to the reactor inlet by a high pressure injection line. This line is the only interconnection between the makeup system and the high pressure injection system. Makeup flow to the reactor coolant system is regulated by the makeup control valve, which operates on signals from the liquid level controller of the reactor coolant system pressurizer.

A control valve in the RC pump seal injection line automatically maintains the desired flow rate to the seals. Needle valves in the individual seal injection lines are used to manually throttle flow to the seals of each pump. A part of the water supplied to the seals leaks into the reactor coolant system. The remainder returns to the makeup tank after passing through one of the two seal return coolers.

Seal water inleakage to the reactor coolant system requires a continuous letdown of reactor coolant to maintain the desired coolant inventory. In addition, letdown of reactor coolant is required for removal of impurities and boric acid from the reactor coolant and to accommodate volume changes in the reactor coolant system during changes in power level. Reactor coolant is removed from one of the reactor inlet lines, cooled during passage through one of the letdown coolers, passed from the containment vessel through a containment isolation valve, reduced in pressure during flow through the letdown flow control station, passed through the purification

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demineralizer prefilter, and then passed through a purification demineralizer (and/or the cesium removal demineralizer as required) to a three-way valve which directs the coolant either through the purification makeup filter to the makeup tank or to the clean radioactive waste disposal system. The normal let- 6
down flow rate is 45 gpm. This permits recirculation of one reactor coolant system volume through the purification train during a 24 hour period. The maximum letdown flow rate allowed at full reactor pressure is 140 gpm. This flow rate permits changing boron concentration by bleeding coolant from the reactor coolant system during xenon peaking following a 50 percent power change. During this period, non-borated reactor grade water is added to the reactor coolant system to dilute the boric acid concentration in the reactor coolant system. This is done to compensate for the negative reactivity addition resulting from the xenon peaking.

Normally, the three-way valve is positioned to direct the letdown flow to the makeup tank. If the boric acid concentration in the reactor coolant is to be reduced, the three-way valve is positioned to direct the letdown flow to the clean radioactive waste disposal system. Boric acid removal is accomplished 6
in the clean radioactive waste disposal system either by directing the letdown flow through a deborating demineralizer with the effluent returned directly to the makeup tank, or by directing the letdown flow to a clean waste receiver tank. The level in the makeup tank is maintained with deborated water from storage or with demineralized water from the station demineralized water storage tank. The flow of demineralized water is measured and totaled by an inline flow integrator and associated instrumentation. The flow of demineralized water to the makeup tank is controlled remotely by the makeup tank feed flow valve. During normal operation the flow integrator 1
(batch controller), the integrated control system interlock, or the operator will terminate dilution. The above procedure for reducing the reactor coolant system boric acid concentration is the feed and bleed method.

The makeup tank also receives chemicals for addition to the reactor coolant. Chemicals in solution are injected into the letdown flow upstream of the purification filters and then pass into the makeup tank which serves as a final mixing location. A hydrogen overpressure is maintained in the tank to ensure that a predetermined amount of dissolved hydrogen remains in the reactor coolant.

System control is accomplished remotely from the control room with the exception of the seal return coolers, and the cesium removal demineralizer for operation in series with the other purification demineralizers. The letdown flow rate is established by the block orifice during normal operation, but may be increased by opening the letdown control valve. The spare purification demineralizer can be placed in service by remote positioning of the demineralizer isolation valves. The cesium removal demineralizer can be placed in service in parallel with one of the purification demineralizers by remote positioning of the isolation valve. To place the cesium removal demineralizer in service in series with one of the purification demineralizers requires using manual valves. Diverting the letdown flow to the clean radioactive 6
waste treatment system is accomplished by remote positioning of the three-way valve and the valves in the clean radioactive waste disposal system. 6
The control valve in the injection line to the reactor coolant pump seals is automatically set by a flow controller to maintain the desired total flow rate to the seals.

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The reactor coolant makeup control valve is automatically controlled by the pressurizer level controller. The makeup pumps are remotely controlled from the control room. Switch-over from one purification makeup filter to the other is remotely controlled from the control room. The purification demineralizer prefilter can be bypassed by remote operation from the control room. Operation of the seal return coolers requires manual positioning of valves.

Coolant at the refueling boron concentration is supplied to the reactor coolant system for preoperational fill by using the boric acid pumps and the clean waste receiver transfer pumps or the demineralized water supply pumps. The fill line bypasses the makeup tank and makeup pumps and connects into the RC System through the normal makeup control valve. When the fill operation is completed, the auxiliary fill line is secured; makeup and inventory control is then continued by operation of a makeup pump.

The makeup and purification system provides makeup to the reactor coolant system to replenish inventory lost due to a small rupture in the reactor coolant system pressure boundary. The makeup control valve senses a decrease in pressurizer level and positions itself to maintain level. A high flow alarm is associated with the increase in additional makeup. Following this alarm will be a low makeup tank level alarm. Additional makeup is provided by sources of reactor coolant grade water upstream of the makeup tank.

The makeup and purification system serves no emergency function.

9.3.4.3 Safety Evaluation

9.3.4.3.1 Reliability Consideration

This system provides important functions for the normal operation of the unit. |6
Redundant components and alternate flow paths have been provided to improve system reliability.

In addition to the letdown orifice, the system has two full-capacity control valves in parallel with the orifice. One of these control valves is manually operated and one is remotely operated.

Two mixed bed purification demineralizers are provided. One is normally in use while the other is a spare.

There are two makeup pumps, each capable of supplying the required reactor coolant pump seal and makeup flow. One is normally in operation while the other, kept on standby status, is used as needed.

There are two letdown coolers and two seal return coolers. One of each is in operation normally, while the other is a spare. Similarly, there are two purification makeup filters and two seal injection filters provided.

The purification prefilter and cesium removal purification demineralizer are provided for use as required. Redundancy in these components is not necessary.

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9.3.4.3.2 Malfunction Analysis

A malfunction analysis of the makeup and purification system is given in table 9-12, it demonstrates that, in the event of any credible single active failure, the system can still meet its normal operating design requirements.

9.3.4.3.3 System Isolation

The letdown line and the reactor coolant pump controlled bleedoff line are outflow lines which penetrate the containment vessel. These lines contain electric motor operated isolation valves inside the containment vessel and solenoid operated isolation valves outside the containment vessel which are automatically closed by a safety features actuation signal.

The injection lines to the reactor coolant pump seals are inflow lines penetrating the containment vessel. Each of the four seal injection lines contains a stop check valve inside the containment vessel and a solenoid operated valve outside the containment vessel. The solenoid valve outside the containment vessel is actuated by a safety features actuation signal and is designed to fail close on loss of its air supply. An air accumulator is provided to keep this valve open in the event of a failure in the instrument air supply system. This prevents unnecessary loss of seal injection flow and the possible reactor coolant pump damage which may result. The line providing makeup flow to the reactor coolant system does so via one of the high pressure injection lines. Isolation for the makeup line is accomplished by a stop check valve in the high pressure injection line inside the containment vessel and a solenoid operated isolation valve outside the containment vessel which is automatically closed by a safety features actuation signal. Check valves in the discharge of each makeup pump provide further backup for containment vessel isolation. Also, the solenoid operated isolation valves outside the containment vessel in the letdown, reactor coolant pump controlled bleedoff, and makeup lines are designed to fail closed on loss of air supply.

9.3.4.3.4 Leakage Considerations

Design and installation of the components and piping in the makeup and purification system consider radioactive service. Except where flanged connections have been installed for ease of maintenance, the system is of all-welded construction. Principal valves have double packing with provisions for leak-off connections.

The consequences of a pipe failure in the makeup and purification system depend upon the location of the rupture. If the rupture were to occur between the reactor coolant loop and the first isolation valve or check valve, it would lead to uncontrolled loss of coolant from the reactor coolant system. This accident is included in Chapter 6. If the rupture were to occur beyond the first isolation valve or outside the containment vessel the release of radioactivity would be limited by the small line sizes and by closing of the isolation or check valve. Leakage detection in the makeup and purification system is achieved by monitoring of the makeup tank level as a function of time.

752162

Table 9-12

Malfunction Analysis of Makeup and Purification System

Component	Malfunction	Comment
1. Letdown Cooler	Tube rupture in one cooler.	Redundant pressure switches are installed on the shell side of the letdown coolers. These switches will detect pressurization in the component cooling water system due to a tube rupture. Upon detecting pressurization the pressure switches close the redundant valves on the inlet to the affected cooler. After the affected cooler has been isolated, letdown is switched to the redundant cooler. The spare letdown cooler is sufficient to meet normal makeup operations. 6
2. Letdown Coolers	Loss of cooling water flow due to failure of component cooling water system downstream of containment vessel isolation valve.	This malfunction results in loss of the capability for feed and bleed. However, cold shutdown can still be achieved. Boric acid may be added in combination with the required demineralized water so that the total added quantity injected will produce the required soluble poison concentration level as well as the required makeup for contraction (582F to 140F)and is approximately equal 3250 ft^3. Required volume of 7 wt % boric acid solution (assuming the CRA of highest worth stuck out of the core) is approximately 460 ft^3.
3. Block orifice	Fails	Either of the two full flow Control valves in parallel with the block orifice have capability of maintaining normal letdown flow.

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Table 9-12 (Cont'd)

Component	Malfunction	Comment
4. Makeup Pump	Fails while operating	Adequate makeup and seal injection flow is provided by the redundant pump. 6
5. Makeup Pump	Fails to start	Adequate makeup and seal injection flow is provided by the redundant pump.
6. Seal Return Cooler	Tube rupture in one Cooler	The failed cooler can be manually isolated, and the spare cooler can be manually brought on line to provide sufficient seal return and makeup pump recirculation cooling.
7. Isolation Valves outside Containment vessel	Loss of air supply	Air accumulators are provided on the air inlet to the valves so that the valves will remain in the same position as they were 6 prior to loss of air. Enough air is available so that the valves will close upon receipt of SFAS signal. The isolation valves in the seal injection lines are equipped with air accumulators to ensure seal injection flow is maintained to the Reactor Coolant Pumps. Injection of boric acid to the reactor coolant system can be manually maintained through the use of handjacks on the isolation valves and manual throttle valves.

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Possible leakage of hydrogen from the makeup tank has been taken into consideration. The tank and connected piping will be hydrostatically tested to demonstrate leak tightness during preoperational testing. Thereafter, the only likely source of leakage will be the safety valve, and it will be piped to the radioactive waste system.

Control of leakage for major loss of coolant accident is described in Chapter 6. The capability of the makeup and purification system to handle small breaks as required by AEC Criterion 33 is also discussed in Chapter 6.

9.3.4.4 Tests and Inspections

Active and passive components of the makeup and purification system will be examined periodically to determine their operating condition. Periodic visual inspections and preventive maintenance will be conducted according to sound maintenance practice.

9.3.4.5 Instrumentation Applications

9.3.4.5.1 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that might affect system safety. The variables of conditions of operation which are limited are as follows:

a. Makeup Tank Level

High and low water level in the makeup tank are alarmed at 86 and 55 inches respectively. A loss of level in this tank could cause a loss of NPSH to the makeup pumps, thus causing loss of makeup flow which could have a serious affect on the reactor coolant system if it occurred during cooldown or when makeup for contraction of the reactor coolant system is required. A low level interlock with the three-way valve is set at 18 inches and signals the valve to switch from bleed operation to makeup operation to prevent draining of the tank. Another low level interlock is supplied to trip the makeup pumps and is set at 5 inches to prevent going below the required net positive suction head and causing cavitation.

The makeup tank level is measured by redundant, manually selectable transmitters. The selected signal is transmitted to the control room to be recorded and actuate alarms.

b. Letdown Line Temperature

A high temperature alarm is set at 135F and signals a high temperature in the letdown line downstream of the letdown coolers. The alarm also signals the isolation valve outside the containment vessel to close, thus protecting the purification demineralizers' resins. Hot reactor coolant, if allowed to pass through the demineralizers, could damage the demineralizer resins.

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c. Dilution Control

Initiation of the dilution cycle must be by the operator. Several safeguards are incorporated into the design to prevent inadvertent excessive dilution of the reactor coolant boric acid concentration. These safeguards are as follows:

1. The process of normal deboration of the reactor coolant cannot start unless specific control rod groups are withdrawn to a certain point which allows for deboration. This control rod group position interlock through the integrated control system either permits or prohibits continuous dilution depending upon the control rod group position. Because of this interlock, the demineralized water makeup valve and the three-way valve can be operated simultaneously only when the control rod group is withdrawn to a preset position. The demineralized water makeup valve is automatically closed, and the three-way valve position is automatically changed when the rods have inserted to a present position.

2. The dilution valves are interlocked so that the operator must preset the desired dilution batch size before initiating the dilution cycle. The dilution cycle will automatically terminate when the dilution flow has integrated to the preset batch size.

3. The operator is able to manually terminate the dilution cycle at any time.

9.3.4.5.2 Instrumentation Application

The instrumentation in the makeup and purification system provides measurements which are used to indicate, record, alarm, interlock and control process variables such as level and flow as follows:

a. The following process variables are measured and a signal is transmitted that provides indication in the control room.

1. Letdown flow.
2. Makeup pump discharge header pressure and temperature. 6
3. Makeup flow (low range)

b. The following process variables are measured and a signal is transmitted that will actuate alarms and provide indication in the control room.

1. Seal injection filter differential pressure.
2. Makeup tank pressure.
3. Makeup flow (wide range).
4. Purification filter differential pressure.

c. The letdown temperature is measured and signals are transmitted that will actuate alarms and provide indication in the control room. 6

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Temperature switches are provided at the discharge of the letdown coolers, and one is provided further downstream on the letdown line. The switches on the discharge of the coolers will close the inlet valves to the coolers upon detection of high temperatures. High temperatures at the switch further downstream will close the containment vessel exterior isolation valve. 6

d. The following process variables are measured and locally indicated.

1. Makeup pump discharge pressure.
2. MU flow control valve bypass flow.
3. Purification Deminerilizer filter differential pressure.

e. The RC pump seal bleedoff flow is measured and a signal is transmitted that will actuate low flow alarms in the control room.

f. The RC pump seal inlet flow is measured and a signal is transmitted that will actuate alarms and provide indication in the control room. An analog signal is also provided for positioning the RC pump seal bleedoff flow control valve.

g. The following process control valves are manual/electric positioned from the control room.

1. Letdown flow control valve.
2. Makeup system three-way feed & bleed control valve.

h. Signals from the following process variables are transmitted to the plant computer for indication and/or alarm.

1. Letdown temperature.
2. Letdown flow.
3. Letdown pressure.
4. Makeup tank temperature.
5. Makeup tank level.
6. Letdown radiation.
7. Letdown boron analysis.
8. Makeup tank pressure.
9. Makeup pump discharge pressure.
10. Makeup pump discharge temperature. 6
11. Seal injection flow.
12. Makeup flow.
13. Seal return flow.

i. The makeup system feed and bleed controls (batch controller) is a device that measures the amount of boric acid or demineralized water added to the reactor coolant system. The feed and bleed controls (batch controller) will automatically terminate the addition when the quantity preselected by the operator is reached.

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