

August 1, 1973

W. G. Reinmuth, Chief, Technical Support Branch  
Directorate of Regulatory Operations, Headquarters

JERSEY CENTRAL POWER & LIGHT COMPANY, DOCKET NO. 50-219  
PRELIMINARY SUMMARY REPORT ON SNUBBER REPAIR ACTIVITIES DURING  
SPRING 1973 REFUELING OUTAGE

The enclosed preliminary report is forwarded for your information. The licensee has stated that the final report will be officially transmitted to the AEC. It is suggested that this report be treated as proprietary information pending receipt of the final report.

E. Morris Howard, Chief  
Facility Construction and  
Engineering Support Branch

Enclosure:  
Preliminary Rpt

cc: F. A. Dreher, w/o enc.

3/192

OFFICE ▶	REG OPS					
SURNAME ▶	HEISHMAN:maz	HOWARD	Tilton			
DATE ▶	8/1/73					

Form AEC-318 (Rev. 9-53) AECM 0240

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# Jersey Central Power & Light Company



MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 539-6111

July 31, 1973

Mr. A. Giambusso  
Deputy Director for Reactor Projects  
Directorate of Licensing  
United States Atomic Energy Commission  
Washington, D. C. 20545



Dear Mr. Giambusso:

Subject: Oyster Creek Station  
Docket No. 50-219  
Reactor Coolant System Leakage

The purpose of this letter is to report a violation of the Technical Specifications, Paragraph 3.3.D., "Reactor Coolant System Leakage". Operation of the reactor at power continued when it was not recognized that an increasing absorption pool level, combined with the rate of leakage into the drywell sump, originated from the same source and thereby resulted in an "unexplained" leak rate in excess of 5.0 gpm. This event is considered to be an abnormal occurrence as defined in the Technical Specifications, Paragraph 1.15.F. Notification of this event as required by the Technical Specifications, Paragraph 6.6.2.a., was made to AEC Region I, Directorate of Regulatory Operations on Monday, July 23, 1973.

As indicated in Figure 1, attached, an increasing rate of leakage into the drywell sump began to occur on July 1, 1973 and continued through July 19, 1973, reaching a peak of approximately 3.02 gpm when averaged over 24 hours. As shown in Figure 2, attached, a plot of absorption pool (torus) water level developed on July 23, 1973 over the same period indicated the level to be increasing starting about July 11, 1973. It is now estimated that the unexplained leak rate increased to >5.0 gpm at some time during July 17, 1973 and continued to be above the 5.0 gpm limit until the plant was shut down and depressurized on July 21, 1973.

The source of this leakage was found to be a feedwater check valve hinge pin seal plug, which due to its position and the manner in which the water was spraying out, resulted in leakage to both the drywell floor and the torus. Valve data is as follows:

Manufacturer: Anchor Valve Company  
Type: 18" - 60C# Swing Check Valve  
Material: Cast Carbon Steel - Stallite Trim BW Ends

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July 31, 1973

In order to repair this seal, the erosion of the seating surface on the valve body was machined out and the plug adapted to fit using a procedure developed by MPR Associates and concurred in by the valve manufacturer and the PORC. In addition, a calculation was performed which verified that after machining, the valve wall thickness was still satisfactory. A successful leak test was conducted on July 24, 1973 and the plant returned to service.

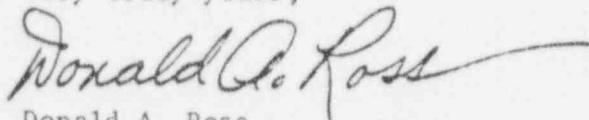
The allowable leakage rates of coolant from the reactor system are based in part on predicted and experimentally observed behavior of cracks in pipes. As noted in the basis of the Technical Specifications, "...evidence suggests that the leakage somewhat greater than the limit specified or unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly." The Technical Specifications limit referred to in the above is 5.0 gpm; whereas, in this instance, the maximum leak rate approach is 6.75 gpm.

Since the source of leakage in this case was a gasketed seal, no undo safety significance need be associated with this event. The possibility of the drywell sump under unusual circumstances not identifying the total unidentified drywell leakage must be recognized.

To prevent a reoccurrence of this type event, Procedure 515.3, "Small Piping Leaks in Drywell" will be revised to recognize that in cases where the torus water level is increasing and the leakage source cannot be identified, this inleakage will be added to the drywell unidentified leakage.

Enclosed are forty (40) copies of this report.

Very truly yours,



Donald A. Ross  
Manager, Nuclear Generating Stations

DAR:cs  
Enclosures

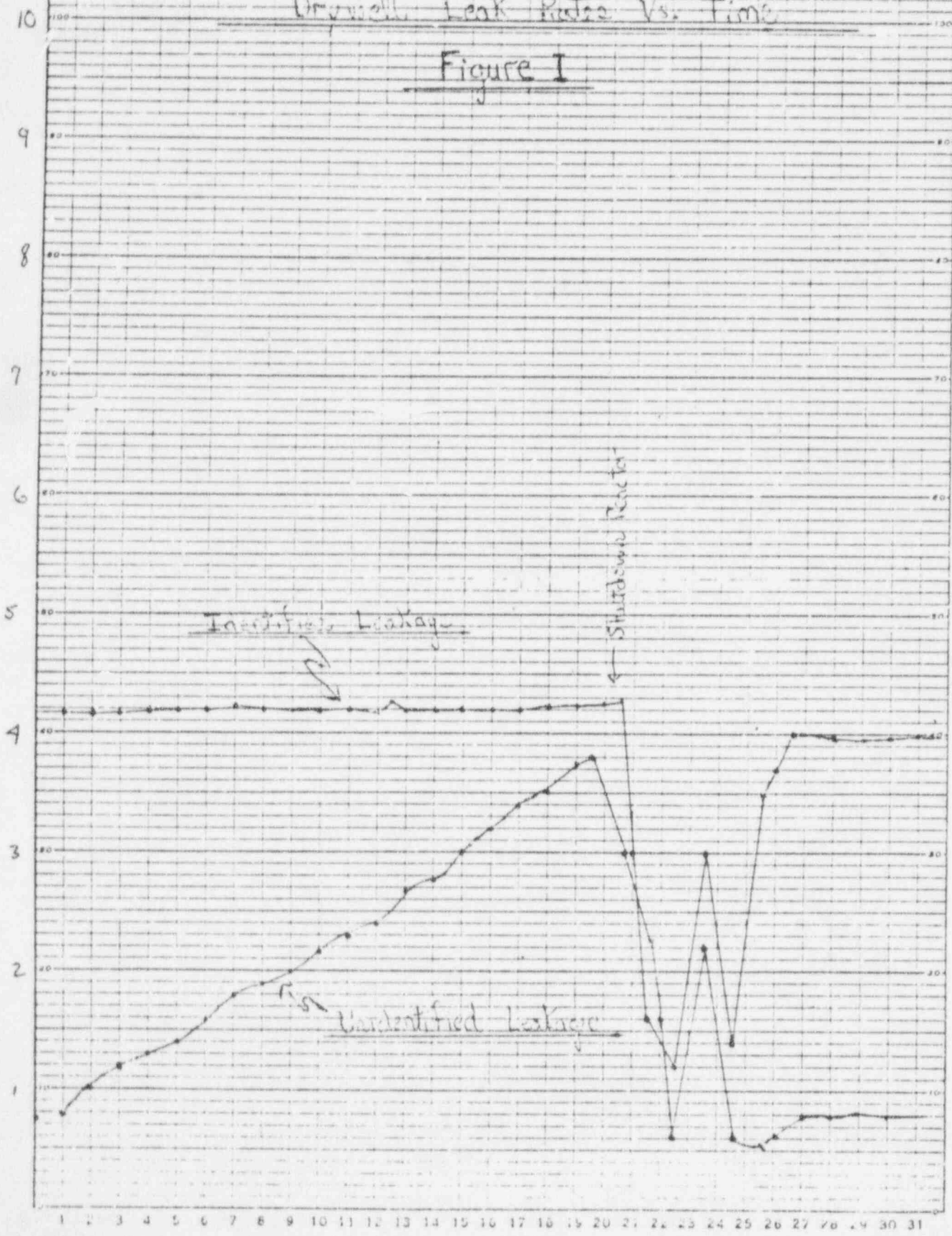
cc: Mr. J. P. O'Reilly, Director  
Directorate of Regulatory Operations, Region I

1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31

# Drywell Leak Rates Vs. Time

## Figure I

Leak Rates (gpm)



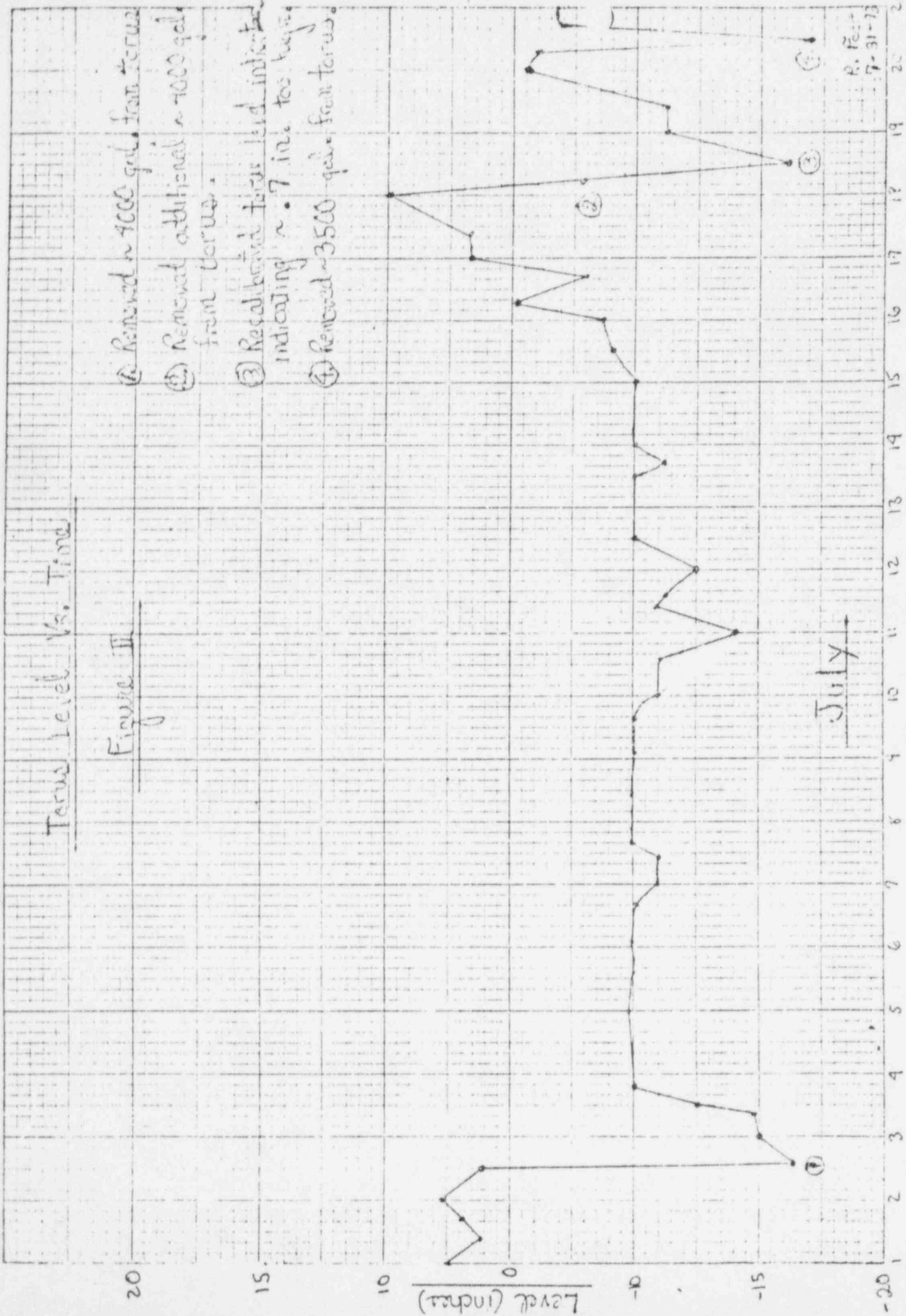
MONTH OF July 19 73  
Time (days)

E. Inc.  
7-31-73

Torus Level Vs. Time

Figure III

- ① Removed ~ 4000 gal. from torus.
- ② Removed additional ~ 4000 gal. from torus.
- ③ Re-calibrated torus lid indicator indicating ~ 7 in. too high.
- ④ Removed ~ 3500 gal. from torus.



July