Jersey Central Power & Light Company

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

January 18, 1973

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

Dear Mr. Giambusso:

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION
DOCKET 50-219
FUEL DENSIFICATION EFFECTS

As indicated in my letter on the same subject dated December 29, 1972, we have received and reviewed the General Electric Company Topical Report entitled "Densification Considerations in BWR Fuel Design and Performance" dated December 1972. Based on this review we have concluded that this report applies to all of the fuel presently in the Oyster Creek Station reactor and fulfills the requirements of your November 20, 1972 request for information concerning the effects of densification on the Oyster Creek reactor.

The majority of the fuel in the present core (556 out of 560 bundles) is General Electric fuel, identified as Oyster Creek Types I and II. The four remaining fuel bundles (Type III) were manufactured by the Exxon Nuclear Company. Oyster Creek Facility Change Request No. 3 dated April 19, 1972, compares Type III fuel with Type II fuel. Of significance from the standpoint of any potential densification effects are the following minor differences:

a) The Type III fuel has a nominal pellet density of 93.5% compared to a value for the Type II fuel of 94.3%. However, the tolerances on these values (0 values for Type III and Type II fuels of .0052 and .0084 respectively) are such that minimum pellet densities are essentially the same in both fuel types.

b) The nominal pellet-to-clad diametral gap is 11 mils in the Type III fuel and 12 mils in the Type II fuel, which would tend to make the gap conductivity somewhat higher in the Type III fuel.

c) The minimum nominal clad thickness on Type III fuel is 35.5 mils vs. 32 mils on the Type II fuel. Hence, the Type III fuel clad will be somewhat more resistant to creep deformation than the Type II fuel, and hence, more resistant to the formation of axial gaps between pellets.

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Based on the above minor differences, it is concluded that any potential effects of densification on Type III fuel are no worse, and are perhaps less severe than the effects on Type II fuel.

The possibility of densification and its effects on the fuel to be loaded in the next reload are still being reviewed and will be reported in conjunction with Jersey Central's application to load and operate the next core reload.

Very truly yours,

R H Sims

R. H. Sims
Vice President

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Jersey Central Power & Light Company

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

January 17, 1973

JAN 22 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
Valve Malfunctions

On December 29, 1972, an operator opened a cabinet door carrying protective fuses for generator monitoring relays. As a result, a turbine trip was initiated immediately followed by an anticipatory reactor scram. At the time of this event, the reactor was operating at 1830 MWt.

As expected, the associated increase in primary system pressure was terminated by actuation of the electromatic relief valves. One (1) electromatic relief valve failed to reseat causing a primary system blowdown into the vapor suppression chamber (torus). As a result of this event, three (3) different valves failed to operate in their intended manner. Each of these valve failures constitutes an abnormal occurrence and are included as Attachments I, II, and III.

These failures can be summarized briefly as follows:

1) Failure of the "D" electromatic relief valve to reseat caused by a piece of material from the associated disc retainer becoming trapped under the pilot seat.

2) Failure of main steam isolation valve NS04B to close upon receiving an isolation signal due to a sticking pilot operated power valve.

3) Failure of isolation condenser NE01B condensate return valve V-14-35 to open due to excessive valve breakaway torque causing its motor operator to burn out.

Each of these situations were thoroughly investigated and corrected. In addition, due to AEC concern over valve operability problems currently being experienced by the industry, we conducted a complete surveillance test program on all valves related to reactor safety.

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Valves in thirteen (13) different systems (approximately 115 valves in all) were judged to fall in this category and all but one (1) were satisfactorily tested prior to plant restart. One (1) torus to drywell vacuum breaker valve did not meet our normal opening force criterion and required some maintenance prior to plant start-up. This deficiency is discussed in detail in Attachment IV to this letter.

Since more reliable valve operation is our desire at Oyster Creek and in response to Directorate of Regulatory Operations request, we intend to give the matter of valve surveillance frequency and surveillance testing adequacy, additional review. All valves related to reactor safety will be included in our review. Consideration is being given to a meaningful increase in surveillance frequency. We will evaluate the method of surveillance testing to assure it is the best possible means available to determine valve operability.

Discussions with appropriate manufacturers and valve users will be pursued to benefit from their experience in these areas. We will continue to keep your office advised of our efforts in this area in a timely manner.

We are enclosing forty copies of this report.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

DAR/mp
Attachments (I, II, III, and IV)
Enclosures (40)

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

ATTACHEMNT I

Failure of the "D" Electromatic Relief Valve to reseat after performing its protective function is considered to be an abnormal occurrence as defined in the Technical Specifications, Paragraph 1.15.F.

On December 29, 1972, with the reactor operating at 1830 MWt thermal, a turbine trip was initiated which caused a reactor anticipatory scram. The turbine trip caused an immediate increase in reactor pressure which was properly terminated by automatic actuation of the Electromatic Relief Valves upon reaching the 1070 psig setpoint. The "D" Electromatic Relief Valve failed to reseat, however, after the design blowback of 22 psi was achieved. Upon recognizing that the blowdown had not terminated when the main steam isolation valves closed, checks of the Electromatic Relief Valve temperature sensors were made. These temperature indications verified that the relief valves had operated. The fact that one valve had not reseated was confirmed by hearing steam collapsing in the vapor suppression chamber. In an attempt to determine which relief valve had not reseated, each valve was individually cycled. When the "D" Relief Valve was selected to open, no change in noise level was noted, thus indicating it was already open. Several unsuccessful attempts were made to reseat the "D" Electromatic Relief Valve by operating its control switch.

This event resulted in a primary system cooldown rate of 158°F per hour for the first hour following the scram and rates of 27°F per hour and 37°F per hour for the second and third hours, respectively. Also, as a result of this event, approximately 50,000 gallons of reactor water in the form of steam was blowndown into the vapor suppression chamber.

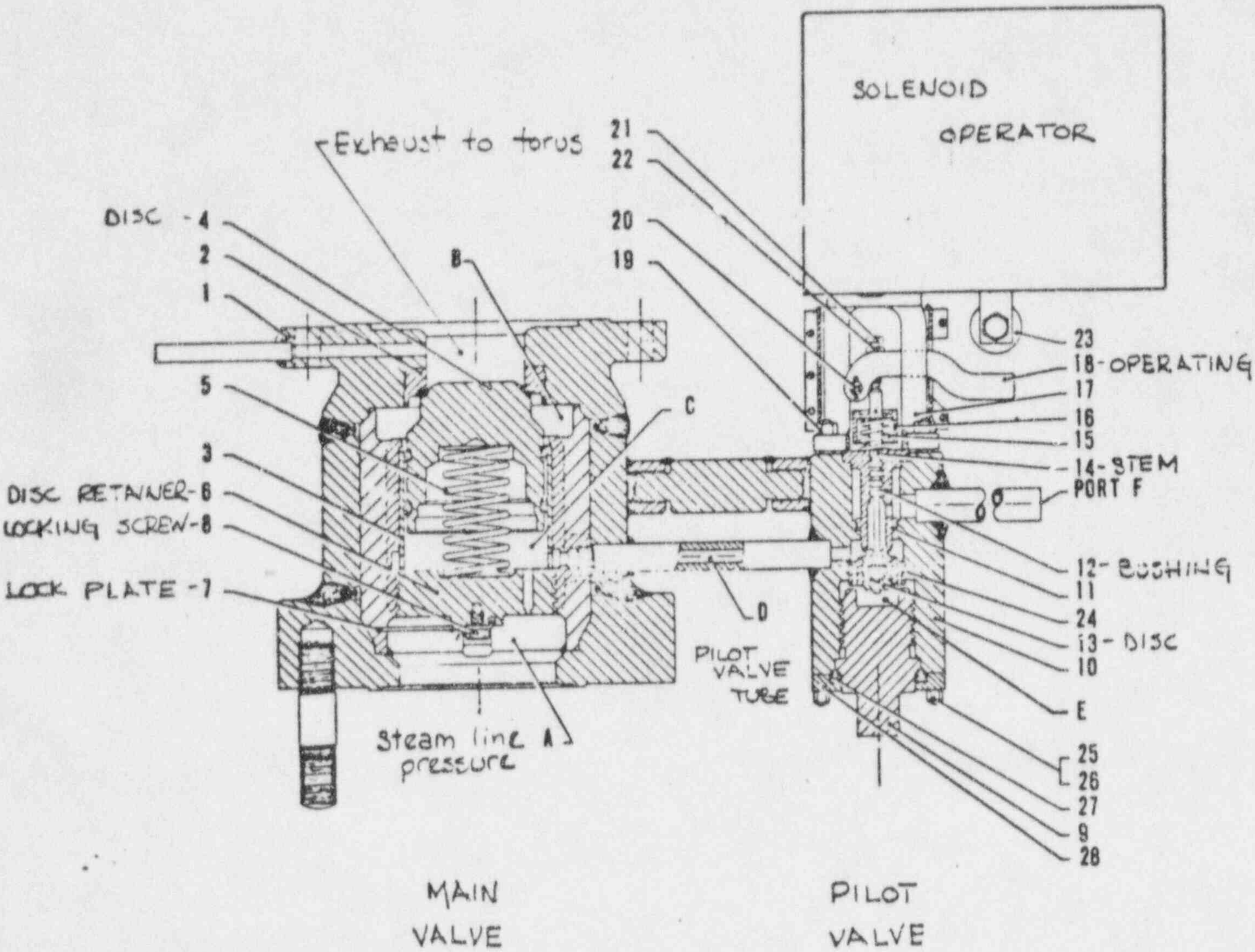
Upon investigation, the pilot valve for the "D" Electromatic Relief Valve was found to be open, thus causing the main valve to remain open as long as pressure remained on the main steam line.

Figure 1 attached identifies the valve parts under discussion.

Further investigation revealed that a piece of thread material from the disc retainer of the main valve had blown through the pilot valve tube. During the valve actuation it became lodged between the pilot valve seat and disc, thus the pilot valve was not able to fully close which, in turn, maintained the main valve in an open position.

It was also noted that the tie wire, lock screw and lock arm were missing from the disc retainer. Since the lock arm and screw were missing from the "D" valve, the remaining 4 relief valves were removed for inspection. As a result of this inspection, it was found that a small tab from the lock arm of the "B" relief valve had also broken off. Similar parts for the other 3 valves were found to be in place and secure. In addition, it was observed that the valve operator's spring guide for the "E" relief valve was misaligned, thus causing the solenoid plunger to be restricted in its upward travel.

The safety significance associated with this event centers around the blowdown event. The attached tabulation develops the actual primary system temperature/time relationship.



ELECTROMATIC RELIEF VALVE

FIGURE 1

Primary System Temperature/Time Relationship

Time	Time	Pressure Psig	Sat. Time °F	Cooldown Rate @ Timed Interval	Cooldown Rate Hourly
608	-	1000	546		
				402°F/hr	
610	2 min	660	499		
					158°F/
618	10	550	479		
				132	
638	.5 hr	355	437		
				98	
708	1	200	388		
				26	
738	1.5	170	375		27
				28	
808	2	140	361		
				28	
838	2.5	115	347		37
				46	
908	3	80	324		
					50
1000	4	30	274		
					15
1008	5	20	259		
					0.
1208	6	20	259		

This event nearly approximates the blowdown case analyzed in Report Number CENC-1143, "Analytical Report for Jersey Central Reactor Vessel". The actual temperature transient analyzed is as follows:

(1)	1056°F/hr. for .17 hours	(546°F to 366°F)
(2)	C°F/hr. for 1.62 hours	(366°F)
(3)	100°F/hr. for 2.77 hours	(366°F to 90°F)
	Average - 100°F/hr. for 4.56 hours	(546°F to 90°F)

According to this report, which is the basis for Amendment No. 16 of the FDSAR, the reactor head studs are the most limiting components for the blowdown event, being allowed 308 blowdown cycles. The next most limiting component (non-replaceable) is the support skirt which is allowed 780 cycles. The most limiting reactor components considering a 300°F/hr. cooldown from 546°F to 100°F are again the head studs (280 cycles) and the support skirt (780 cycles - based on the more severe stresses caused by the 100°F/hr. cooldown from 546°F to 100°F). Therefore, no unreviewed safety significance is associated with this event other than counting it as a thermal cycle due to a blowdown. Concerning the failure of the "D" Electromatic Relief Valve, the safety significance of this event has been previously analyzed in Amendment No. 15, Section II of the FDSAR (Primary Containment Design Report).

In order to prevent recurrence of lock arm failures, these components were removed from all relief valve assemblies. A new method of preventing disc retainer rotation was installed. This method consists of pinning the disc retainer to the cage. Dresser Company, manufacturer of the valves, was contacted concerning this repair and approved same, taking into condideration that the installation of a set screw can in no way cause the clearance between the threads to increase. All other lead threads on each disc retainer were examined and feathered edges removed as necessary. The spring guides for the "E" valve were realigned and the solenoid plunger was checked for freedom of movement.

The cap screw from the "D" Valve was found lodged in the seat of the 30" header Main Steam Drain Valve, V-1-65. Attempts made to locate the remaining missing parts were unsuccessful. Discussions were held with General Electric, I&SE Representatives, which indicated that the strainers provided ahead of the turbine main steam stop valves would trap a piece the size of the missing locking plate. The decision has been made to inspect the main steam stop valve strainers for the missing lock plate during the next refueling outage presently scheduled for this Spring.

Prior to putting the plant back on the line, a hot functional test of each of the five Electromatic Relief Valves was performed. No difficulties were experienced with the operation of these valves during this test.

In addition to those measures already taken to prevent a repetition of this event, a review of the operating procedures associated with an inadvertent blowdown is being made. This procedure will be revised as required, if found inadequate.

We intend to investigate with the manufacturer the history of problems associated with this type of Relief Valve and obtain his recommendations for possible modifications which could improve this valves operation and reliability.

ATTACHMENT II

Failure of NS04B, Main Steam Isolation Valve to close upon receiving an isolation signal is an abnormal occurrence as defined in Section 1.15.F.

The reactor mode switch key was broken off in trying to switch to the "Start-Up" mode following the reactor scram. [illegible]s allowed reactor pressure to reach 850 psig with the switch in the "RUN" mode and caused closure of the MSIVs. After the isolation, it was discovered that NS04B, the south outside isolation valve, was not closed. Its switch was placed in the "CLOSED" position and the valve still did not close. The switch was then cycled "OPEN" and "CLOSED", whereupon the valve closed.

The valve closure problem was determined to be caused by a sticking pilot operated power valve. This power valve is manufactured by Numatics, Inc. (reference Numatics print 91746) and modified by General Electric Company, FDI 322.

The AC and DC Solenoid Valves were checked and operated satisfactorily. The pilot operated power valve was found to be sticking. A new pilot operated power valve was installed and tested satisfactorily. Similar valves were inspected on the remaining three (3) main steam isolation valves and found to be in satisfactory condition. The sticking valve was dismantled and examined. A small amount of fine red dust was found on the sleeve "O" rings. The piston was removed and cleaned, and although it showed no visible sign of wear, it still exhibited a tendency to stick at one end of its travel.

Each of the two main steam lines are provided with two (2) isolation valves in series, one inside and one outside the drywell. As noted in Amendment 65 to the FDSAR, a modification was made to each of the pilot operated power valves during the Fall 1971 outage, increasing the piston and cylinder clearances. No difficulties had been experienced with any of the valves, since that time,until a closure problem occurred with NS04B on December 3, 1972 prior to a reactor startup.

This difficulty was determined at that time to be a result of a malfunctioning control switch which was replaced. The valve subsequently cycled properly. As discussed with our DRO I inspector,subsequent review of this earlier event in view of recent findings would indicate the switch did not contribute to the problem;however,it was a sign of difficulty with the pilot operated power valve.

To prevent reoccurrence of this event we intend to investigate a modification which would permit daily exercise tests to be accomplished by using the main power operated pilot valve. The valve manufacturer will be contacted with regard to instituting a preventative maintenance program on these valves.

The main steam isolation valve full closure test program will continue to be followed, i.e., testing the valves whenever sufficient load reductions on plant occur, but with a minimum frequency of six weeks between tests rather than the present Technical Specification frequency of three months if at power We will factor into the present study on the long-term suitability of the main steam isolation valves, the problems associated with power operated pilot valves which have now been experienced.

ATTACHMENT III

The failure of Isolation Condenser NE01B Condensate Return Valve V-14-35, to open when manually actuated during a plant cooldown is considered to be an abnormal occurrence as defined in the Technical Specifications, Paragraph 1.15.E.

Following a partial depressurization of the primary system, it was desired to place the "B" Isolation Condenser in service to assist with the final reactor system cooldown. The control switch for the Condensate Return Valve, V-14-35, was placed in the "OPEN" position and left there for approximately 15-20 seconds. However, when no opening indication was evident, the control switch was returned to "AUTOMATIC". At this time, the green "CLOSED" indicating light extinguished indicating that the valve motor supply breaker had opened.

Knowing that the control switch was in the "OPEN" position for 15-20 seconds and then receiving indications that the 70 amp motor supply breaker opened, verifies that an opening signal was applied to the valve motor. Apparently, the motor could not supply sufficient torque to break the valve away from its seat. Upon investigation the valve motor was found to be burned out. The valve was then operated manually at which time some difficulty was experienced in breaking the disc free of its seat. Once broken off the seat, the valve stroked fully open and fully closed freely by hand. A new motor was installed and operation of the valve checked with no adjustments made to the torque or limit switch settings. The motor operator preformed properly. The valve was then completely disassambled and thoroughly inspected, along with its associated limit torque operator. The only flaw observable during this inspection was a small burr on the stem which was considered to be of no significance. The valve was then completely reassembled with new motor thermal overloads and motor power supply breakers being installed. The valve was then operated satisfactorily. A complete set of motor currents were obtained for future reference.

The requirement for at least one of the isolation condensers to act as a means for heat removal is detailed in Amendment #67 to FDSAR. In this case, however, the "B" isolation condenser would not have been able to fulfull that function and, therefore, the safety significance of this event is that the isolation condenser system did not have its intended redundancy.

To prevent reoccurrence of this event we plan to institute a means to obtain more accurate motor amperage data while performing the valve operability test and also to check the repeatability of the associated torque switch settings. This will be done on an increased testing frequency of once per week in order to obtain the additional motor current data. Presently the surveillance testing of this component has been required on a monthly basis.

ATTACHMENT IV

The failure of one drywell-absorption chamber vacuum breaker to satisfy our opening test criteria is considered an abnormal occurrence as defined in Technical Specification 1.15.E. This event was reported to the Directorate of Regulatory Operations - Region I by telephone on January 8, 1973.

The fourteen (14) drywell-absorption chamber vacuum breakers which are normally tested during each refueling outage, were part of a list of valves being tested prior to plant start-up. Testing is accomplished by the use of a spring scale to measure the amount of force necessary to open the vacuum breakers and the force necessary to hold them open. Valve V-26-8 failed to meet the maximum allowable opening force criteria and also exhibited a tendency to stick partially open. Another valve V-26-9 also showed a tendency to stick partially open when allowed to close. The packing was removed from both valves, inspected, and reinstalled. Adjustments were made to both valve packing glands allowing both valves to operate satisfactorily.

The safety significance associated with this situation is as follows. The Drywell Absorption Chamber Vacuum Breaker System is required to prevent water oscillations in the downcomers due to low steam flow rates in the downcomers and to provide protection against negative pressure conditions in the Drywell portion of the containment vessel. The vacuum relief area to vent area ratio used to size the vacuum relief valves for Oyster Creek was found to limit water oscillations in the downcomers to about two (2) feet from the Bodega Bay tests. Allowing one valve to be inoperative reduces the total vacuum relief area by only 7% based on interpolation of Bodega data and such a reduction would increase the oscillations by only about one-half foot, which is negligible. Only about 25% of the available vacuum relief capacity is required to protect against negative pressure in the containment.

To prevent reoccurrence of this event we intend to investigate the suitability and application of the present packing material for low pressure, high humidity environment.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

JAN 1 7 1973

Docket No. 50-219

Jersey Central Power and Light Company
ATTN: R. H. Sims, Vice President
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

Your letter dated October 6, 1972, reported an incident in which it was discovered that the two pumps in the standby liquid control system were inoperable at the same time, when this system was required to be operable. Your letter dated April 20, 1972, reported an incident in which the supply dampers for the Reactor Building Ventilation System failed to close during a surveillance test. Both incidents involved the inadvertent disabling of one component by racking out the circuit breaker for a different component.

We understand through discussions with your staff that redesign of the control circuits that caused the above incidents would be implemented such that this type of malfunction would be eliminated. We also understand that you conducted a review of the control circuits of all safety related equipment at the plant to assure that disabling of one component does not, through incorporation in other interlocking or sequencing controls, render other components inoperable. It appears that in the cases cited above, the racked out position of breakers had not been included in the failure mode analysis of those control circuits.

We request that you submit, by April 1, 1973, a description of the modifications made as corrective action for the above incidents as well as the status of the implementation, the results of your review of the control circuits of all other safety related equipment, and proposed changes to the Technical Specifications that will require the testing of a redundant system immediately after its counterpart is rendered inoperable or found to be inoperable and then at the frequency presently given in your Technical Specifications.

Sincerely,

Robert J. Schemel

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

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cc: See next page

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cc: GPU Service Corporation
ATTN: Mr. Thomas M. Crimmins, Jr.
Safety and Licensing Manager
260 Cherry Hill Road
Parsippany, New Jersey 07054

G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006