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Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960
(201) 455-8200

March 1, 1979

Mr. Boyce H. Grier, Director
Office of Inspection and Enforcement
Region I
United States Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. Grier:

Subject: Oyster Creek Nuclear Generating Station
Docket No. DPR-16
Annual Report of
Station Modifications - 1978

Enclosed are two (2) copies of the annual report of the modifications completed at the Oyster Creek Nuclear Generating Station for the year 1978. This report is submitted in accordance with 10 CFR 50.59 (b).

Very truly yours,

Donald A. Ross

Donald A. Ross, Manager
Generating Stations-Nuclear

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Enclosures

cc: Director (40 copies)
Office of Inspection and Enforcement
United States Nuclear Regulatory Commission
Washington, DC 20555
c/o Distribution Services Branch, DDC, ADM

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OYSTER CREEK NUCLEAR GENERATING STATION

OYSTER CREEK NUCLEAR GENERATING STATION

PROVISIONAL OPERATING LICENSE DPR-16

ANNUAL REPORT

STATION MODIFICATIONS

1978

- PREFACE -

In accordance with 10 CFR 50.59 (b), this report includes those facility changes, tests, and experiments conducted at the Oyster Creek Nuclear Generating Station without prior NRC approval during the period January 1, 1978 through December 31, 1978.

OYSTER CREEK NUCLEAR GENERATING STATION

STATION MODIFICATIONS

1978

1. Containment Spray Loop Seal

While replacing the containment spray heat exchangers during the 1978 refueling outage, the emergency service water discharge piping on System I was rerouted to produce a loop seal, thereby preventing the heat exchangers from draining while the system is in standby. The change was made to reduce water hammer previously experienced when starting the system for test.

In addition, 150 psig and 250 psig relief valves were installed on the containment spray and emergency service water sides of the new heat exchanger respectively, replacing the lower rated valves installed to protect the degraded heat exchangers.

The safety evaluation concluded that the loop seal will reduce water hammer and reduce air-water interface corrosion. Precautions to maintain secondary containment during the installation were included in the installation procedures. The safety evaluation concluded that this modification does not constitute an unresolved safety question.

2. High Purity Condensate for Control Rod Drive System

In order to reduce the probability of cracking in control rod drive (CRD) components, a modification was installed to change the CRD water supply from the condensate storage tank to a high purity, deaerated source. The modification was accomplished by injecting water from the discharge of the condensate demineralizers into the 12-inch pipe from the condensate storage tank to the CRD pump suction. This 12-inch line also serves as an alternate water supply for both core spray systems. In operation, the modification will cause flow into the condensate storage tank through the 12-inch pipe, assuring high quality water which is also deaerated when condenser vacuum is maintained.

The safety evaluation concluded that the modification does not affect the safety analysis or Technical Specifications or the margin of nuclear safety and, therefore, does not constitute an unreviewed safety question.

3. Automatic Recirculation Pump Trip (RPT)

In response to the NRC position that a recirculation pump trip would significantly limit the consequences of an anticipated transient without scram (ATWS) event, a circuit modification was installed during the 1978 refueling outage. An additional trip coil was installed on each of the five recirculation pump M-G set drive

motor 4160-V circuit breakers so that each motor could be tripped by either the A/B or C 125-V DC power supply. Initiation logic uses instrumentation and circuitry associated with the isolation condenser initiation signals. The RPT will trip all five recirculation pumps on reactor high pressure or reactor low-low water level, each of which is preceded by a reactor scram. The RPT circuit meets single failure criteria and is testable. Minor changes to the isolation condenser initiation circuit were included.

The safety evaluation included a review of applicable transient analyses and performance of a transient analysis of a turbine trip without bypass transient with the RPT. The safety evaluation concluded that installation of the RPT does not involve any unreviewed safety questions.

4. Acid and Caustic Modification

Severe corrosion of the acid and caustic piping to the condensate demineralizer and makeup demineralizer regeneration systems has occurred. In order to alleviate the cause of corrosion and also to assure proper chemical feed during regeneration, a system of acid and caustic day tanks with individual metering pumps was installed in the turbine building basement.

The safety evaluation concluded that this modification does not involve any unreviewed safety questions.

5. 125-V DC Separation

In order to provide increased physical separation of 125-V DC power supplies, the 125-V DC system was extensively modified during the 1978 refueling outage. The modification consists of an additional 1200 amp-hour 125-V battery, two (2) chargers, a new distribution center (DC-C), a new distribution panel (DC-F) and a new motor control center (DC-2). A majority of the equipment was installed in the 4160-V switchgear room, a considerable distance from existing A/B batteries. Increased separation was ensured by running the "C" battery loads in new conduit.

The safety evaluation for the 125-V DC system modification addressed the use of non-class 1E equipment until it can be replaced with 1E qualified equipment. The equipment installed meets or exceeds the quality of comparable equipment within the original 125-V DC system. It was noted that the NRC was previously informed of this modification and of deviations from class 1E equipment. The safety evaluation concluded that the modification does not involve any unreviewed safety questions. A Technical Specification change identifying the new DC system was submitted to the NRC.

6. Control Air Dryer

An additional control air dryer of increased capacity was installed during the refueling outage to improve the quality of control air. Considerable maintenance problems and increased demand on the

original dryer caused control air quality to be consistently less than specifications. Redundant pre and post filters were also installed. In addition, a service air cut off bypass switch was installed to prevent automatic isolation of the service air system when using service air for respirators.

The safety evaluation concluded that the modification does not involve any unreviewed safety questions.

7. Security System

The Oyster Creek security system was upgraded to meet the requirements of 10 CFR 73.55. The modifications included erection of two (2) new access buildings, site lighting, a vital area access control system, perimeter monitoring, upgraded control room wall, and improved control room doors.

The safety evaluation considered the effect of additional loads added to the standby diesel generators in great detail. It was concluded that these modifications do not involve any unreviewed safety questions.

8. A/B Battery Room Fire Protection

An automatic fire suppression system was installed to protect the "A" and "B" 125-V DC station batteries, battery room switchgear and M-G sets, the electric tray room and the tunnel connecting the battery room with the electric tray room. The system uses Halon 1301 total flooding and is actuated by products of combustion sensors. Automatic damper isolation was included. The ventilation system to the battery room was modified to detect loss of air flow.

The safety evaluation concluded that the modification does not involve any unreviewed safety questions.

9. Reactor Building Fire Water Connection

Connections were made into the fire water supply lines for Core Spray System I and Core Spray System II in anticipation of future automatic and manual fire suppression systems in the reactor building. The addition consisted of a tee, a valve, short piping runs, and a blanked flange on each system.

The safety evaluation considered the potential for flooding of engineered safeguards equipment in the reactor building. It was concluded that the modification does not involve an unreviewed safety question.

10. Torus to Drywell Differential Pressure Instrumentation

Two (2) digital (LED) differential pressure instruments were installed on RK03 in the reactor building to indicate torus to drywell differential pressure. The instruments share common pressure taps from both the drywell and torus but have independent power supplies fed by

redundant emergency buses. These instruments are used to satisfy Technical Specification requirements. Previously, pressure indication on the drywell and on the torus were used to determine the pressure differential.

The safety evaluation concluded that this modification does not involve any unreviewed safety questions.

11. Additional Resin Storage Tank

An additional resin storage tank was installed in the condensate demineralizer regeneration system. This tank will permit operation with all seven (7) condensate demineralizers in service (previously only six (6) demineralizers and one (1) as a spare) while regenerating or cleaning a spare bed. It is anticipated that this modification will reduce the generation lost due to fouled condensate demineralizers and aid in maintaining feedwater quality.

The safety evaluation concluded that this modification does not involve an unreviewed safety question.

12. Condensate Demineralizer Resin Cleaner

An automatic bump and rinse cycle (ABRO) was added to the condensate demineralizer resin regeneration system which cleans the resins by multiple air scrub and rinse cycles. The system was added to remove oxide and resin fines from the demineralizer, thereby reducing the demineralizer pressure drop and permitting more effective chemical regeneration.

The safety evaluation concluded that this modification does not involve any unreviewed safety questions.

13. Temporary Torus Water Treatment System

During the 1978 refueling outage, the torus water was removed from the torus and processed through a temporary treatment system. This was done to remove chloride ions from the water, reducing corrosion potential of the suppression pool structure. The temporary treatment system included anion and cation resins and charcoal filters loaded in disposable radwaste liners as well as associated piping and hoses.

The safety evaluation included assessment of the consequences of spills involving the radioactive, chromated water and the conditions required for reactor safety when the torus is empty. The safety evaluation concluded there were no unreviewed safety questions.

14. Oyster Creek Cycle VIII Reload

The Oyster Creek core was refueled with 168 Exxon Nuclear Company, Inc. Type V-B fuel assemblies. This fuel is essentially the same as the Cycle V reload fuel with the exception of several minor

mechanical design improvements. These improvements did not affect the mechanical, thermal-hydraulic, or nuclear performance of the fuel design. The characteristics of the Cycle VIII reload fuel design (Type V-B) were described in the Cycle V Licensing data, Amendment 76 to the Oyster Creek FSAR. Use of this fuel design was approved by the NRC on May 22, 1975.

Figure 1 shows the Cycle VIII reload pattern which consists of the following:

Fuel Type	Loaded at BOC	Number of Assemblies	Fuel** Geometry
III-E	3	64	7x7 (ENC)
III-F	4	32	7x7 (ENC)
IV-F	5	36	7x7 (ENC)
V	5	4	8x8 (ENC)
V-B	5	72	8x8 (ENC)
V-B	6	56	8x8 (ENC)
V-B	7	128	8x8 (ENC)
V-B	8	168	8x8 (ENC)

The "Reload Information and Safety Evaluation Report for Oyster Creek Cycle VIII Reload" was reviewed by the Station Superintendent, the Plant Operations Review Committee, and the Independent Safety Review Group. Based on the reviews, it was determined that neither the refueling nor subsequent operation for Cycle VIII core involved an unreviewed safety question as defined in 10 CFR 50.59 (a) (2). It was further determined that no Technical Specification changes were required to implement the refueling and subsequent operation. The core was, therefore, reloaded, surveilled, and operated in accordance with the provisions of 10 CFR 50.59 (a) (1) and the existing Operating License.

Oyster Creek Nuclear Generating Station

V 19.5	VB 14.1	VB 8.2	VB 16.2	VB 8.8	VB 16.4	VB 8.8	VB 16.2	VB 8.3	VB 16.3	VB 6.8	IIIF 17.0	IIIF 15.1
VB 14.0	VB 0.0	VB 16.0	VB 0.0	VB 18.7	VB 0.0	VB 8.8	VB 0.0	VB 17.7	VB 0.0	VB 8.7	VB 0.0	IIIE 15.8
VB 8.2	VB 15.6	VB 7.1	VB 18.4	VB 0.0	VB 19.0	VB 9.0	VB 19.0	VB 0.0	IIIF 18.4	VB 0.0	IIIE 16.2	IIIF 15.7
VB 16.0	VB 0.0	VB 18.7	VB 0.0	VB 19.0	VB 0.0	VB 9.0	VB 0.0	IIIF 17.8	VB 0.0	VB 6.8	VB 0.0	IIIE 19.4
VB 8.7	VB 19.0	VB 0.0	VB 19.1	VB 7.2	VB 18.4	VB 8.8	VB 17.5	VB 0.0	IIIF 17.3	VB 0.0	IIIE 18.0	IIIF 16.2
VB 16.5	VB 0.0	VB 19.0	VB 0.0	VB 18.3	VB 0.0	VB 16.6	VB 0.0	VB 8.7	VB 0.0	VB 8.3	IIIE 21.6	
VB 8.8	VB 8.7	VB 9.0	VB 9.1	VB 8.8	VB 16.6	VB 0.0	VB 17.6	VB 0.0	VB 9.0	IIIF 18.5		
VB 16.3	VB 0.0	VB 19.1	VB 0.0	VB 17.5	VB 0.0	VB 17.3	VB 0.0	IIIE 19.1	VB 0.0	IIIE 18.9		
VB 8.3	VB 17.6	VB 0.0	IIIF 17.2	VB 0.0	VB 8.6	VB 0.0	IIIE 18.9	VB 0.0	VB 8.8	IIIE 22.5		
VB 16.3	VB 0.0	IIIF 18.5	VB 0.0	IIIF 17.6	VB 0.0	VB 8.7	VB 0.0	VB 8.8	IIIF 22.1			
VB 6.8	VB 8.5	VB 0.0	VB 6.7	VB 0.0	VB 8.4	IIIF 17.0	IIIE 19.0	IIIE 22.4				
IIIF 17.1	VB 0.0	IIIE 16.1	VB 0.0	IIIE 18.1	IIIE 21.4							
IIIF 15.2	IIIE 16.8	IIIF 15.7	IIIE 19.5	IIIF 16.1								

Cycle Exposures

Cycle 7 ΔE = 683.1 GWD
Cycle 8 ΔE = 540 GWD (Projected)

xx	Fuel Type
yy	BOC8 Exposure GWD/MTU

Figure 1 Oyster Creek - Cycle 8 Reload, 168 Fresh Assemblies (one quarter of symmetric core loading)