Correction Letter to Adm't. 75 to DPR-16

December 24, 1984

Docket No. 50-219
LS05-84-12-019

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Mr. P. B. Fiedler
Vice President and Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fiedler:

SUBJECT: CORRECTION TO AMENDMENT NO. 75 TO THE TECHNICAL SPECIFICATIONS

Re: Oyster Creek Nuclear Generating Station

By letter dated August 27, 1984, the Commission issued Amendment No. 75 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. We have discovered that eight (8) pages in Amendment No. 75 were issued on out-of-date Technical Specification pages. These eight (8) TS pages did not include changes that were issued in previous amendments and the less than or equal symbol needed on page 3.10-2 for the required local linear heat generation rate. Please replace the previously issued TS pages 3.1-11, 3.4-5, 3.5-6, 3.5-7, 3.10-2, 4.2-1, 4.2-1a and 4.2-2 with the enclosed corrected pages which now have the previous amendments indicated on the pages.

Sincerely,

Original signed by

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Corrected TS pages
to Amendment No. 75
to License No. DPR-16

cc w/enclosure:
See next page

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CJamerson:jb	JDonohew	JZwolinski		DCrutchfield
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Function	Trip Setting	Reactor Modes in which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating [tripped] Trip systems	Min. No. of Instrument Channels Per Operable Trip Systems	Action Required*
K. Rod Block								No control rod withdrawals permitted
1. SRM Upscale	≤ 5 x 10^5 cps		X	X(1)		1	2	
2. SRM Downscale	≥ 100 cps(f)		X	X(1)		1	2	
3. IRM Downscale	≥ 5/125 fullscale(g)		X	X		2	3	
4. APRM Upscale	**		X(s)	X	X	2	3(c)	
5. APRM Downscale	≥ 2/150 fullscale				X	2	3(c)	
6. IRM Upscale	≤ 108/125 fullscale		X	X		2	3	
7. a) water level high scram discharge volume North	≤ 14 gallons		X(z)	X(z)	X(z)	1	1 per instrum. volume	
b) water level high scram discharge volume South	≤ 14 gallons		X(z)	X(z)	X(z)	1	1 per instrum. volume.	
L. Condenser Vacuum Pump Isolation								Insert Control Rods
1. High Radiation in Main Steam Tunnel	≤ 10 x Normal background		During Startup and Run when vacuum pump 1 operating			2	2	

makeup valves in both isolation condensers are demonstrated daily to be operable.

3. If Specifications 3.4.B.1 and 3.4.B.2 are not met; reactor pressure shall be reduced to 110 psig or less, within 24 hours.

4. The time delay set point for initiation after coincidence of low-low-low reactor water level and high drywell pressure shall be set not to exceed two minutes.

C. <u>Containment Spray System and Emergency Service Water System</u>

1. The containment spray system and the emergency service water system shall be operable at all times with irradiated fuel in the reactor vessel, except as specified in Specifications 3.4.C.3, 3.4.C.4, 3.4.C.6 and 3.4.C.8.

2. The absorption chamber water volume shall not be less than 82,000 ft^3 in order for the containment spray and emergency service water system to be considered operable.

3. If one emergency service water system loop becomes inoperable, its associated containment spray system loop shall be considered inoperable. If one containment spray system loop and/or its associated emergency service water system loop becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 7 days provided the remaining containment spray system loop and its associated emergency service water system loop each have no inoperable components and are demonstrated daily to be operable.

4. If a pump in the containment spray system or emergency service water system becomes inoperable, the reactor may remain in operation for a period not to exceed 15 days provided the other similar pump is demonstrated daily to be operable. A maximum of two pumps may be inoperable

inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories (Reference 11) has shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to define precisely an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.5.A.7.d prohibits startup with inoperable snubbers.

Secondary containment(5) is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service and provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the overall containment system, it is required at all times that primary containment is required. Moreover, secondary containment is required during fuel handling operations and whenever work is being performed on the reactor or its connected systems in the reactor building since their operation could result in inadvertent release of radioactive material.

When secondary containment is not maintained, the additional restrictions on operation and maintenance give assurance that the probability of inadvertent releases of radioactive material will be minimized. Maintenance will not be performed on systems which connect to the reactor vessel lower than the top of the active fuel unless the system is isolated by at least one locked closed isolation valve.

The standby gas treatment system (6) filters and exhausts the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs.

Two separate filter trains are provided each having 100% capacity. (6) If one filter train becomes inoperable, there is no immediate threat to secondary containment and reactor operation may continue while repairs are being made. Since the test interval for this system is one month (Specification 4.5), the time out-of-service allowance of 7 days is based on considerations presented in the Bases in Specification 3.2 for a one-out-of-two system.

References:
(1) FDSAR, Volume I, Section V-1
(2) FDSAR, Volume I, Section V-1.4.1
(3) FDSAR, Volume I, Section V-1.7
(4) Licensing Application, Amendment 11, Question III-25
(5) FDSAR, Volume I, Section V-2
(6) FDSAR, Volume I, Section V-2.4
(7) Licensing Application, Amendment 42
(8) Licensing Applciation, Amendment 32, Question 3
(9) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.
(10) Bodega Bay Preliminary Hazards Summary Report, Appendix I, Docket 50-205, December 28, 1962.
(11) Report H. R. Erickson, Bergen-Paterson to K. R. Goller, NRC, October 7, 1974. Subject: Hydraulic Shock Sway Arrestors.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed on August 2, 1976, which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system. The maintenance of a drywell-suppression chamber differential pressure within the range shown on Figure 3.5-1 with a suppression chamber water level corresponding to a downcomer submergence range of 3.0 to 5.3 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

B. Local LHGR

During power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly, at any axial location shall not exceed the maximum allowable LHGR:

B.1 Fuel Types V and VB

As calculated by the following equation;

$$LHGR \leq LHGR_d \left[1 - \frac{\Delta P}{P} \max \left(\frac{L}{LT} \right) \right]$$

Where: $LHGR_d$ = Limiting LHGR (=14.5)

$\frac{\Delta P}{P}$ = Maximum Power Spiking Penalty (=0.033 and 0.039 for Fuel Types V and VB respectively)

LT = Total Core Length - 144 inches
L = Axial position above bottom of core

B.2 Fuel Type P8x8R

$LHGR \leq 13.4$ KW/ft.

B.3 If at any time during operation it is determined by normal surveillance that the limiting value of LHGR is being exceeded, action shall be initiated to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two [2] hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

4.2 REACTIVITY CONTROL

Applicability: Applies to the surveillance requirements for reactivity control.

Objective: To verify the capability for controlling reactivity.

Specification:

A. Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25% Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

B. The control rod drive housing support system shall be inspected after reassembly.

C. 1. After each major refueling outage and prior to resuming power operation, all operable control rods shall be scram time tested from the fully withdrawn position with reactor pressure above 800 psig.

2. Following each reactor scram from rated pressure, the mean 90% insertion time shall be determined for eight selected rods. If the mean 90% insertion time of the selected control rod drives does not fall within the range of 2.4 to 3.1 seconds or the measured scram time of any one drive for 90% insertion does not fall within the range of 1.9 to 3.6 seconds, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is maintained.

3. Following any outage not initiated by a reactor scram, eight rods shall be scram tested with reactor pressure above 800 psig provided these have not been measured in six months. The same criteria of 4.2.C(2) shall apply.

D. Each partially or fully withdrawn control rod shall be exercised at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

E. Surveillance of the standby liquid control system shall be as follows:

1. Pump operability	Once/month
2. Boron concentration determination	Once/month
3. Functional test	Each refueling outage
4. Solution volume and temperature check	Once/ day

F. At specific power operation conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison shall be made every equivalent full power month. The initial rod inventory measurement performed when equilibrium conditions are established after a refueling or major core alteration will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

G. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days, except in shutdown mode*, and shall be cycled at least one complete cycle of full travel at least quarterly.

H. All withdrawn contol rods shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE. This will be done at least once per refueling cycle by placing the mode switch in shutdown and by verifying that:

a. The drain and vent valves close within 30 seconds after receipt of a signal for control rods to scram, and

b. The scram signal can be reset and the drain and vent valves open when the scram discharge volume trip is bypassed.

Basis: The core reactivity limitation (Specification 3.2.A) requires that core reactivity be limited such that the core could be made subcritical at any time during the operating cycle, with the strongest operable contol rod fully withdrawn and all other operable rods fully inserted. Compliance with this requirement can be demonstrated conveniently only at the time of refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration is performed with the reactor core in the cold, xenon-free condition and will show that the reactor is sub-critical at that time by at least R + 0.25% Δk with the highest worth operable control rod fully withdrawn.

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* These valves may be closed intermittently for testing under administrative control.