

UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

Docket No. 50-219

JUL 16 1975

Jersey Central Power & Light Company  
ATTN: Mr. I. R. Finfrock, Jr.  
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Gentlemen:

Our letter to you of February 15, 1975, discussed the Steam Vent Clearing Phenomenon and the Steam Quenching Vibration Phenomenon at various BWR plants with Mark I Containments. We also requested that you initiate action in accordance with a prescribed schedule of major events set forth in this letter. The first action to be accomplished was your submittal of proposed Technical Specifications to revise the suppression pool water temperature limits. Your letter of April 1, 1975, proposed no such changes to the Technical Specifications.

Because of the potential adverse effects on public health and safety of continued plant operation in accordance with existing Technical Specifications related to this matter, we believe that appropriate changes to these Technical Specifications are needed to assure that the integrity of the pressure suppression pool of your facility continues to be maintained. Accordingly, unless you inform us in writing within 20 days of the date of this letter that you do not agree with this course of action, including your reasons, we plan to initiate steps to issue the enclosed change to the Technical Specifications of the Oyster Creek Nuclear Generation Station. A copy of the related Safety Evaluation is enclosed.

Sincerely,

*Karl R. Goller*

Karl R. Goller, Assistant Director  
for Operating Reactors  
Division of Reactor Licensing

Enclosures:

1. Proposed Changes to Technical Specifications
2. Safety Evaluation

cc: See next page



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Jersey Central Power & Light Company

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PROPOSED CHANGES TO  
TECHNICAL SPECIFICATIONS  
PROVISIONAL OPERATING LICENSE NO. DPR-16  
DOCKET NO. 50-219

The proposed changes to the Technical Specifications are shown on the attached pages and are identified by a vertical line in the margin.

### 3.5 CONTAINMENT

**Applicability:** Applies to the operating status of the primary and secondary containment systems.

**Objective:** To assure the integrity of the primary and secondary containment systems.

**Specifications:** A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained within the following limits.
  - a. Maximum water volume - 92,000 ft<sup>3</sup>
  - b. Minimum water volume - 82,000 ft<sup>3</sup>
  - c. Maximum water temperature
    - (1) During normal power operation - 95°F
    - (2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.
    - (3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.
    - (4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120F.
2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

### 3. Reactor Building to Suppression Chamber Vacuum Breaker System

- a. Except as specified in Specification 3.5.A.3.b below, two reactor building to suppression chamber vacuum breakers in each line shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the air-operated vacuum breakers shall not exceed 0.5 psid. The vacuum breakers shall move from closed to fully open when subjected to a force equivalent to not greater than 0.5 psid acting on the vacuum breaker disc.
- b. From the time that one of the reactor building to suppression chamber vacuum breakers is made or found to be inoperable, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is made operable sooner, provided that the procedure does not violate primary containment integrity.
- c. If the limits of Specification 3.5.A.3.a are exceeded, reactor shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

### 4. Pressure Suppression Chamber - Drywell Vacuum Breakers

- a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specification 3.5.A.4.b and c, below. Suppression chamber - drywell vacuum breakers shall be considered operable if:
  - (1) The valve is demonstrated to open from closed to fully open with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.
  - (2) The valve disk will close by gravity to within not greater than 0.10 inch of any point on the seal surface of the disk when released after being opened by remote or manual means.
  - (3) The position alarm system will announce in the control room if the valve is open more than 0.10 inch at any point along the seal surface of the disk.

- b. Two of the fourteen suppression chamber - drywell vacuum breakers may be inoperable provided that they are secured in the closed position.
  - c. One position alarm circuit for each operable vacuum breaker may be inoperable for up to 15 days provided that each operable suppression chamber - drywell vacuum breaker with one defective alarm circuit is physically verified to be closed immediately and daily during this period.
5. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 5.0% O<sub>2</sub> with nitrogen gas within 24 hours after the reactor mode selector switch is placed in the run mode. Primary containment deinerting may commence 24 hours prior to a scheduled shutdown.
  6. If specifications 3.5.A.1.a, b, c(1) and 3.5.A.2 through 3.5.A.5 cannot be met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

B. Secondary Containment

1. Secondary containment integrity shall be maintained at all times unless all of the following conditions are met.
  - a. The reactor is subcritical and Specification 3.2.A is met.
  - b. The reactor is in the cold shutdown condition.
  - c. The reactor vessel head or the drywell head are in place.
  - d. No work is being performed on the reactor or its connected systems in the reactor building.
  - e. No operations are being performed in, above, or around the spent fuel storage pool that could cause release of radioactive materials.
2. The standby gas treatment system shall be operable at all times when secondary containment integrity is required except as specified by Specification 3.5.B.3.

3. One standby gas treatment filter circuit may be inoperable for 7 days, when standby gas treatment system operability is required, except during reactor startup, provided the remaining filter circuit is proved operable daily.
4. If Specifications 3.5.B.2 and 3.5.B.3 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the conditions of Specification 3.5.B.1 shall be met.

Bases:

Specifications are placed on the operating status of the containment systems to assure their availability to control the release of any radioactive material from irradiated fuel in the event of an accident condition. The primary containment system (i) provides a barrier against uncontrolled release of fission products to the environs in the event of a break in the reactor coolant systems.

Whenever the reactor coolant water temperature is above 212°F, failure of the reactor coolant system would cause rapid expulsion of the coolant from the reactor with an associated pressure rise in the primary containment. Primary containment is required, therefore, to contain the thermal energy of the expelled coolant and fission products which could be released from any fuel failures resulting from the accident. If the reactor coolant is not above 212°F, there would be no pressure rise in the containment. In addition, the coolant cannot be expelled at a rate which could cause fuel failure to occur before the core spray system restores cooling to the core. Primary containment is not needed while performing low power physics tests since the rod worth minimizer would limit the worst case rod drop accident to 1.5%Δk. This amount of reactivity addition is insufficient to cause fuel damage.

The absorption chamber water volume provides the heat sink for the reactor coolant system energy released following the loss-of-coolant accident. The core spray pumps and containment spray pumps are located in the corner rooms and due to their proximity to the torus, the ambient temperature in those rooms could rise during the design basis accident. Calculations<sup>(7)</sup> made, assuming an initial torus water temperature of 100°F and a minimum water volume of 82,000 ft<sup>3</sup>, indicate that the corner room ambient temperature would not exceed the core spray and containment spray pump motor operating temperature limits, and, therefore, would not adversely affect the long

term core cooling capability. The maximum water volume limit allows for an operating range without significantly affecting the accident analyses with respect to free air volume in the absorption chamber. For example, the containment capability<sup>(8)</sup> with a maximum water volume of 92,000 ft<sup>3</sup> is reduced by not more than 3.5% metal-water reaction below the capability with 82,000 ft<sup>3</sup>.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and suppression chamber and reactor building so that the containment external design pressure limits are not exceeded.

The vacuum relief system from the reactor building to the pressure suppression chamber consists of two 100% vacuum relief breaker subsystems (2 parallel sets of 2 valves in series). Operation of either subsystem will maintain the containment external pressure less than the external design pressure; the external design pressure of the drywell is 2 psi; the external design pressure of the suppression chamber is 1 psi (FDSAR Amendment 15, Section II).



P. Suppression Chamber Surveillance

1. At least once per day the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.
2. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.
3. Whenever heat from relief valve operation is being added to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.
4. Whenever operation of a relief valve is indicated and the suppression pool temperature reaches 160F or above while the reactor primary coolant system pressure is greater than 200 psig, an external visual examination of the suppression chamber shall be made before resuming normal power operation.

Basis:

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be 38 psig which would rapidly reduce to 20 psig within 100 seconds following the pipe break. The total time the drywell pressure would be above 35 psig is calculated to be about 7 seconds. Following the pipe break absorption chamber pressure rises to 20 psig within 8 seconds, equalizes with drywell pressure at 25 psig within 60 seconds and thereafter rapidly decays with the drywell pressure decay. (1)

The design pressures of the drywell and absorption chamber are 62 psig and 35 psig, respectively. (2) The design leak rate is 0.5%/day at a pressure of 35 psig. As pointed out above, the pressure response of the drywell and absorption chamber following an accident would be the same after about 60 seconds. Based on the calculated primary containment pressure response discussed above and the absorption chamber design pressure, primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and absorption chamber function as a unit, the primary containment will be tested as a unit rather than testing the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.0%/day at 35 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90 percent for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 10 rem and the maximum total

After the containment oxygen concentration has been reduced to meet the specification initially, the containment atmosphere is maintained above atmospheric pressure by the primary containment inerting system. This system supplies nitrogen makeup to the containment so that the very slight leakage from the containment is replaced by nitrogen, further reducing the oxygen concentration. In addition, the oxygen concentration is continuously recorded and high oxygen concentration is annunciated. Therefore, a weekly check of oxygen concentration is adequate. This system also provides capability for determining if there is gross leakage from the containment.

The drywell exterior was coated with Firebar D prior to concrete pouring during construction. The Firebar D separated the drywell steel plate from the concrete. After installation, the drywell liner was heated and expanded to compress the Firebar D to supply a gap between the steel drywell and the concrete. The gap prevents contact of the drywell wall with the concrete which might cause excessive local stresses during drywell expansion in a loss-of-coolant accident. The surveillance program is being conducted to demonstrate that the Firebar D will maintain its integrity and not deteriorate throughout plant life. The surveillance frequency is adequate to detect any deterioration tendency of the material. (8)

The operability of the instrument line flow check valves are demonstrated to assure isolation capability for excess flow and to assure the operability of the instrument sensor when required.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.