March 21, 1988

Docket No. 50-219

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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: ISSUANCE OF AMENDMENT (TAC NO. 67019)

The Commission has issued the enclosed Amendment No. 120 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated January 19, 1988.

The amendment changes the following sections of the Technical Specifications:

Section 1.7
Table 3.1.1 (Footnote 5)
Section 1.39 (New Section)
Section 3.3-A(iv) and Bases
References
Figure 3.3.1
Section 3.4.A(i)
Section 3.4.A.10(c)
Section 3.4.D.1
Section 3.4.D.3 (New Section)
Section 3.5.A.3
Section 3.8.A
Section 4.3.A and Bases

The extent of changes to the Technical Specifications is to incorporate: (1) new pressure-temperature (P-T) operating curves for operation beyond 10 effective full power years (EFPY) and (2) requirements of material surveillance specimens and neutron flux monitors. A definition of Reactor Vessel Pressure Testing and the testing conditions will also be added to the Technical Specifications.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Original signed by

Alexander W. Dromerick, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 120to DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDI-4 SNorris 03/7/88 | PM:PDI-4 ADromerick:bd 03/7/88 | D:PDI-4 JStolz 03/07/88 | OGC 03/16/88

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

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 120
License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation and, et al., (the licensee), dated January 19, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 120, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: March 21, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 120

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
Page 1.0-2	Page 1.0-2
--	Page 1.0-7
Page 3.1-18	Page 3.1-17
Page 3.3-1	Page 3.3-1
Page 3.3-5	Page 3.3-5
Page 3.3-8	Page 3.3-8
--	Page 3.3-9a
--	Page 3.3-9b
--	Page 3.3-9c
Page 3.4-2	Page 3.4-2
Page 3.4-4	Page 3.4-4
Page 3.4-5	Page 3.4-5
Page 3.5-2	Page 3.5-2
Page 3.8-1	Page 3.8-1
Page 4.3-1	Page 4.3-1
Page 4.3-2	Page 4.3-2

1.7 COLD SHUTDOWN

The reactor is at cold shutdown when the mode switch is in the shutdown mode position, there is fuel in the reactor vessel, all operable control rods are fully inserted, and (except during reactor vessel pressure testing), the reactor coolant system maintained at less than 212°F and vented.

1.8 PLACE IN SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the shutdown condition is met.

1.9 PLACE IN COLD SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the cold shutdown condition is met.

1.10 PLACE IN ISOLATED CONDITION

Proceed with and maintain an uninterrupted normal isolation of the reactor from the turbine condenser system including closure of the main steam isolation valves.

1.11 REFUEL MODE

The reactor is in the refuel mode when the reactor mode switch is in the refuel mode position and there is fuel in the reactor vessel. In this mode the refueling platform interlocks are in operation.

1.12 REFUELING OUTAGE

For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the end of the previous refueling outage, the test or surveillance need not be performed until the next regularly scheduled outage. Following the first refueling outage, the time between successive tests or surveillance shall not exceed 20 months.*

1.13 PRIMARY CONTAINMENT INTEGRITY

Primary containment integrity means that the drywell and adsorption chamber are closed and all of the following conditions are satisfied:

A. All non-automatic primary containment isolation valves which are not required to be open for plant operation are closed.

B. At least one door in the airlock is closed and sealed.

C. All automatic containment isolation valves specified in Table 3.5.2 are operable or are secured in the closed position.

D. All blind flanges and manways are closed.

---

*The time between successive tests or surveillances shall not exceed 30 months prior to the cycle 10 refueling outage only.

1.39 REACTOR VESSEL PRESSURE TESTING

System pressure testing required by ASME Code Section XI, Article IWA-5000, including system leakage and hydrostatic tests, with reactor vessel completely water solid, core not critical and Section 3.2.A satisfied.

TABLE 3.1.1 (CONTD)

k. All four (4) drywell pressure instrument channels may be made inoperable during the integrated primary containment leakage rate test (See Specification 4.5), provided that the plant is in the cold shutdown condition and that no work is performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel.
l. Bypass in IRM Ranges 8, 9, and 10.
m. There is one time delay relay associated with each of two pumps.
n. One time delay relay per pump must be operable.
o. There are two time delay relays associated with each of two pumps. One timer per pump is for sequence starting (SK1A, SK2A) and one timer per pump is for tripping the pump circuit breaker (SK7A, SK8A).
p. Two time delay relays per pump must be operable.
q. Manual initiation of affected component can be accomplished after the automatic load sequencing is completed.
r. Time delay starts after closing of containment spray pump circuit breaker.
s. These functions not required to be operable with the reactor temperature less than 212°F and the vessel head removed or vented or during reactor vessel pressure testing.
t. These functions may be operable or bypassed when corresponding portions in the same core spray system logic train are inoperable per Specification 3.4.A.
u. These functions not required to be operable when primary containment integrity is not required to be maintained.
v. These functions not required to be operable when the ADS is not required to be operable.
w. These functions must be operable only when irradiated fuel is in the fuel pool or reactor vessel and secondary containment integrity is required per specification 3.5.B.
y. The number of operable channels may be reduced to 2 per Specification 3.9-E and F.
z. The bypass function to permit scram reset in the shutdown or refuel mode with control rod block must be operable in this mode.
aa. Pump circuit breakers will be tripped in 10 seconds ± 15% during a LOCA by relays SK7A and SK8A.
bb. Pump circuit breakers will trip instantaneously during a LOCA.
cc. Only applicable during startup mode while operating in IRM range 10.
dd. If an isolation condenser inlet (steam side) isolation valve becomes or is made inoperable in the open position during the run mode comply with Specification 3.8.E. If an AC motor-operated outlet (condensate return) isolation valve becomes or is made inoperable in the open position during the run mode comply with Specification 3.8.F.
ee. With the number of operable channels one less than the Min. No. of Operable Instrument Channels per Operable Trip Systems, operation may proceed until performance of the next required Channel Functional Test provided the inoperable channel is placed in the tripped condition within 1 hour.
ff. This function is not required to be operable when the associated safety bus is not required to be energized or fully operable as per applicable sections of these technical specifications.

3.3 REACTOR COOLANT

Applicability: Applies to the operating status of the reactor coolant system.

Objective: To assure the structure integrity of the reactor coolant system.

Specification: A. Pressure Temperature Relationships

(i) Reactor Vessel Pressure Tests - the minimum reactor vessel temperature at a given pressure shall be in excess of that indicated by the curve in Figure 3.3.1(a). The maximum temperature for Reactor Vessel Pressure Testing is 250°F.

(ii) Heatup and Cooldown Operations: Reactor noncritical--the minimum reactor vessel temperature for heatup and cooldown operations at a given pressure when the reactor is not critical shall be in excess of that indicated by the curve in Figure 3.3.1(b).

(iii) Power operations--The minimum reactor vessel temperature for power operations at a given pressure shall be in excess of that indicated by the curve in Figure 3.3.1(c).
Note: Figures 3.3.1(a), (b) and (c) apply when the closure head is on the reactor vessel and studs are fully tensioned.

(iv) Appropriate new pressure temperature limits must be approved as part of this Technical Specification when the reactor system has reached fifteen effective full power years of reactor operation.

B. Reactor Vessel Closure Head Boltdown

The reactor vessel closure head studs may be elongated .020" (1/3 design preload) with no restrictions on reactor vessel temperature as long as the reactor vessel is at atmospheric pressure. Full tensioning of the studs is not permitted unless the temperature of the reactor vessel flange and closure head flange is in excess of 100°F.

C. Thermal Transients

1. The average rate of reactor coolant temperature change during normal heatup and cooldown shall not exceed 100°F in any one hour period.

2. The pump in an idle recirculation loop shall not be started unless the temperature of the coolant within the idle recirculation loop is within 50°F of the reactor coolant temperature.

Transformation temperature. The minimum temperature for pressurization at any time in life as to account for the toughness properties in the most limiting regions of the reactor vessel, as well as the effects of fast neutron embrittlement.

Figures 3.3.1(a), (b) and (c) are derived from an evaluation of the fracture toughness properties performed on the specimens contained in Reactor Vessel Materials Surveillance Program Capsule No. 2 (Reference 14). The results of dosimeter wire analyses (Reference 14) indicated that the neutron fluence (E>1.0 MeV) at the end of 15 effective full power years of operation is 1.11×10^{18} n/cm^2 at the 1/4T (T=vessel wall thickness) location. This value was used in the calculation of the adjusted reference nil-ductility temperature which, in turn, was used to generate the pressure-temperature curves in Figures 3.3.1(a), (b) and (c). The 250°F maximum pressure test temperature provides ample margin against violation of the minimum required temperature. Secondary containment is not jeopardized by a steam leak during pressure testing, and the Standby Gas Treatment system is adequate to prevent unfiltered release to the stack.

Stud tensioning is considered significant from the standpoint of brittle fracture only when the preload exceed approximately 1/3 of the final design value. No vessel or closure stud minimum temperature requirements are considered necessary for preload values below 1/3 of the design preload with the vessel depressurized since preloads below 1/3 of the design preload result in vessel closure and average bolt stresses which are less than 20% of the yield strengths of the vessel and bolting materials. Extensive service experience with these materials has confirmed that the probability of brittle fracture is extremely remote at these low stress levels, irrespective of the metal temperature.

The reactor vessel head flange and the vessel flange in combination with the double "O" ring type seal are designed to provide a leak tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flange. Both the head and the head flange have an NDT temperature of 40°F, and they are not subject to any appreciable neutron radiation exposure. Therefore, the minimum vessel head and head flange temperature for bolting the head flange and vessel flange is established as 40°F + 60°F or 100°F.

Detailed stress analyses(4) were made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these analyses are presented and compared to allowable stress limits in Reference (4). The specific conditions analyzed included 120 cycles of normal startup and shutdown with a heating and cooling rate of 100°F per hour applied continuously over a temperature range of 100°F to 546°F and for 10 cycles of emergency cooldown at a rate of 300°F per hour applied over the same range. Thermal stresses from this analysis combined with the primary load stresses fall within ASME Code Section III allowable stress intensities. Although the Oyster Creek Unit 1 reactor vessel was built in accordance with Section I of the ASME Code, the design criteria included in the reactor vessel specifications were in essential agreement with the criteria subsequently incorporated into Section III of the Code.[6]

pH, chloride, and other chemical parameters are made to determine the cause of the unusual conductivity and instigate proper corrective action. These can be done before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Several techniques are available to correct off-standard reactor water quality conditions including removal of impurities from reactor water by the cleanup system, reducing input of impurities causing off-standard conditions by reducing power and reducing the reactor coolant temperature to less than 212°F. The major benefit of reducing the reactor coolant temperature to less than 212°F is to reduce the temperature dependent corrosion rates and thereby provide time for the cleanup system to re-establish proper water quality.

References

(1) FDSAR, Volume I, Section IV-2
(2) (Deleted)
(3) (Deleted)
(4) Licensing Application Amendment 16, Design Requirements Section
(5) (Deleted)
(6) FDSAR, Volume I, Section IV-2.3.3 and Volume II, Appendix H
(7) FDSAR, Volume I, Table IV-2-1
(8) Licensing Application Amendment 34, Question 14
(9) Licensing Application Amendment 28, Item III-B-2
(10) Licensing Application Amendment 32, Question 15
(11) (Deleted)
(12) (Deleted)
(13) Licensing Application Amendment 16, Page 1
(14) GPUN TDR 725 Rev. 0: Testing and Evaluation of Irradiated Reactor Vessel Materials Surveillance Program Specimens.

Fig. 3.3.1 (a)

P/T Limits

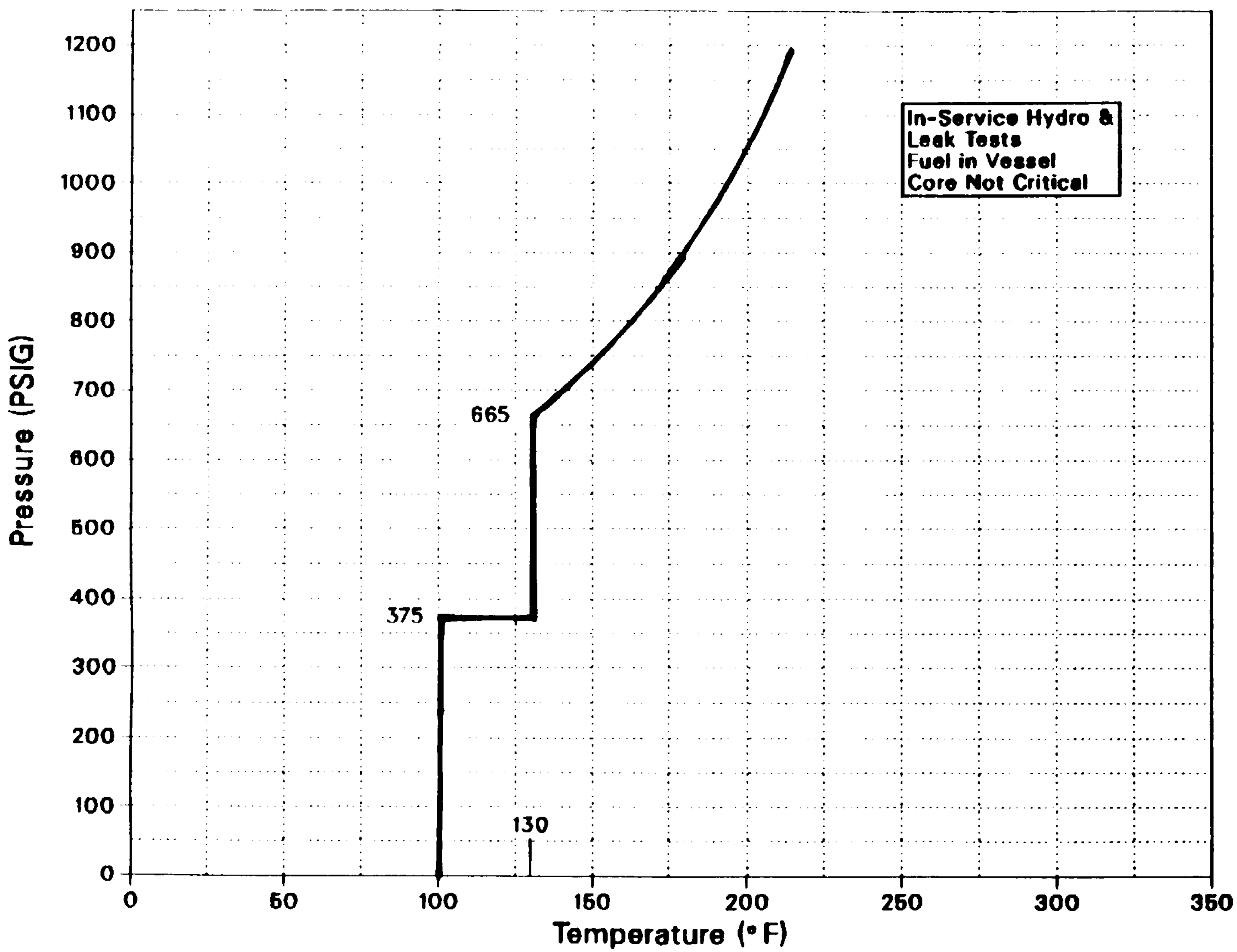
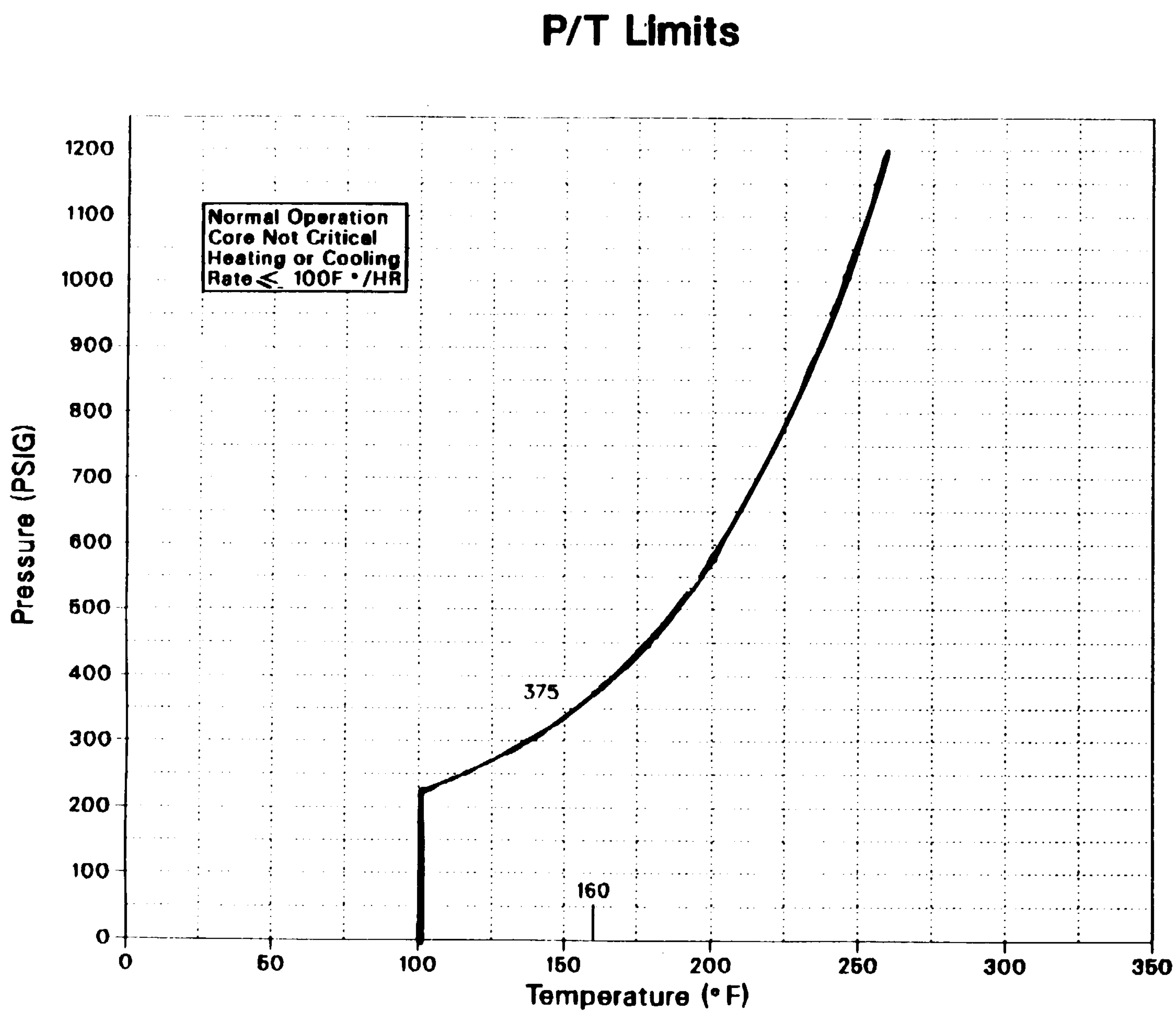


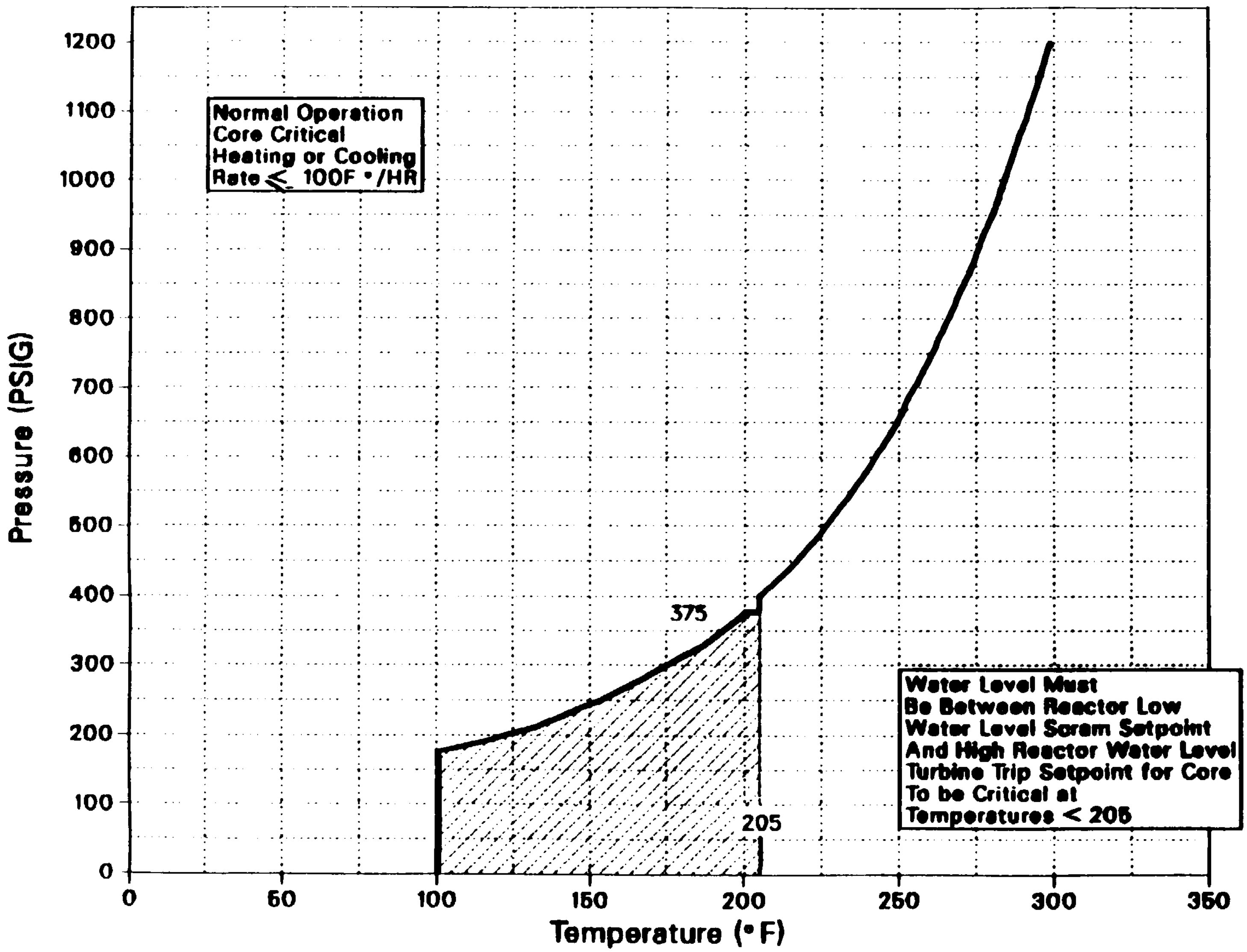
Fig. 3.3.1 (b)



P/T Limits
Normal Operation
Core Not Critical
Heating or Cooling
Rate ≤ 100F °/HR
Pressure (PSIG)
Temperature (°F)
375
160
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
50
150
250
350

Fig. 3.3.1 (c)

P/T Limits



a. At least one core spray pump, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations.

b. The fire protection system is operable, and

c. These systems are demonstrated to be operable on a weekly basis.

8. If necessary to accomplish maintenance or modifications to the core spray systems, their power supplies or water supplies, reduced system availability is permitted when the reactor is in the refuel mode with the reactor coolant system maintained at less than 212°F or in the startup mode for the purposes of low power physics testing. Reduced core spray system availability is defined as follows:

   a. At least one core spray pump in each loop, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves in each loop can be started or operated from the control room or from local control stations.

   b. The fire protection system is operable and,

   c. Each core spray pump and all components in 3.4.A.8a are demonstrated to be operable every 72 hours.

9. If Specifications 3.4.A.7 and 3.4.A.8 cannot be met, the requirements of Specification 3.4.A.6 will be met and work will be initiated to meet minimum operability requirements of 3.4.A.7 and 3.4.A.8.

10. The core spray system is not required to be operable when the following conditions are met:

   a. The reactor mode switch is locked in the "refuel" or "shutdown" position.

   b. (1) There is an operable flow path capable of taking suction from the condensate storage tank and transferring water to the reactor vessel, and

      (2) The fire protection system is operable.

   c. The reactor coolant system is maintained at less than 212°F and vented (except during reactor vessel pressure testing).

   d. At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, 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 provided the two pumps are not in the same loop. If more than two pumps become inoperable, the limits of Specification 3.4.C.3 shall apply.

5. During the period when one diesel is inoperable, the containment spray loop and emergency service water system loop connected to the operable diesel shall have no inoperable components.

6. If primary containment integrity is not required (see Specification 3.5.A), the containment spray system may be made inoperable.

7. If Specifications 3.4.C.3, 3.4.C.4, 3.4.C.5 or 3.4.C.6 are not met, the reactor shall be placed in cold shutdown condition. If the containment spray system or the emergency service water system becomes inoperable, the reactor shall be placed in the cold shutdown condition and no work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel.

8. The containment spray system may be made inoperable during the integrated primary containment leakage rate test required by Specification 4.5, provided that the reactor is maintained in the cold shutdown condition and that no work is performed on the reactor or its connected systems which could result in lowering the reactor level to less than 4'8" above the top of the active fuel.

D. Control Rod Drive Hydraulic System

1. The control rod drive (CRD) hydraulic system shall be operable when the reactor water temperature is above 212°F except as specified in 3.4.D.2 and 3.4.D.3 below.

2. If one CRD hydraulic pump becomes inoperable when the reactor water temperature is above 212°F, the reactor may remain in operation for a period not to exceed 7 days provided the second CRD hydraulic pump is operating and is checked at least once every 8 hours. If this condition cannot be met, the reactor water temperature shall be reduced to $<$ 212°F.

3. During reactor vessel pressure testing, at least one CRD pump shall be operable.

E. Core Spray and Containment Spray Pump Compartments Doors

The core spray and containment spray pump compartments doors shall be closed at all times except during passage in order to consider the core spray system and the containment spray system operable.

F. Fire Protection System

1. The fire protection system shall be operable at all times with fuel in the reactor vessel except as specified in Specification 3.4.F.2.

2. If the fir protection system becomes inoperable during the run mode, the reactor may remain in operation provided both core spray system loops are operable with no inoperable components.

Bases:

This specification assures that adequate emergency core cooling capability is available when the core spray system is required. Based on the loss-of-coolant analysis for the worst line break, a core spray of at least 3400 gpm is required with 35 seconds to assure effective core cooling.*[1] Thus, if one loop becomes inoperable, the operable loop is capable of providing cooling to the core and the reactor may remain in operation for a period of 7 days provided repairs can be completed within that time. The 7 days is based upon the consideration discussed in the bases of Specification 3.2 and the pump operability tests of Specification 4.4. If repairs cannot be made, the reactor is depressurized and vented to prevent pressure buildup and no work is allowed to be performed on the reactor which could result in lowering the water level below 4'8" above the top of active fuel.

Each core spray loop contains redundant active components. Therefore, with the loss of one of these components the system is still capable of supplying rated flow and the system as a whole (both loops) can tolerate an additional single failure of one of its active components and still perform the intended function and prevent clad melt. Therefore, if a redundant active component fails, a longer repair period is justified based on the consideration given in the bases of Specification 3.2. The consideration indicates that for a one out of 4 requirement the time out of service would be

$$\frac{}{1.71} = \frac{30 \text{ days}}{1.71} = 17.5 \text{ days}$$

*Core Spray System 2 is required to deliver 3640 gpm.

b. (1) There is an operable flow path capable of taking suction from the condensate storage tank and transferring water to the reactor vessel, and

(2) The fire protection system is operable.

c. The reactor coolant system is maintained at less than 212°F and vented.

d. At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations, and the torus is mechanically intact.

e. (1) No work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel and the condensate storage tank level is greater than thirty (30) feet (360,000 gallons). At least two redundant systems including core spray pumps and system components must remain operable as defined in d. above.

or

(2) The reactor vessel head, fuel pool gate, and separator-dryer pool gates are removed and the water level is above elevation 117 feet.

NOTE: When filling the reactor cavity from the condensate storage tank and draining the reactor cavity to the condensate storage tank, the 30 foot limit does not apply provided there is a sufficient amount of water to complete the flooding operation.

3. 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 or during reactor vessel pressure testing.

a. With one or more of the containment isolation valves shown in Table 3.5.2 inoperable:

3.8 ISOLATION CONDENSER

Applicability: Applies to operating status of the isolation condenser.

Objective: To assure heat removal capability under conditions of reactor vessel isolation from its normal heat sink.

Specification:

A. The two isolation condenser loops shall be operable during power operation and whenever the reactor coolant temperature is greater than 212°F except as specified in C, below or during reactor vessel pressure testing.

B. The shell side of each condenser shall contain a minimum water volume of 22, 730 gallons. If the minimum volume cannot be maintained or if a source of makeup water is not available to the condenser, the condenser shall be considered inoperable.

C. If one isolation condenser becomes inoperable during the run mode the reactor may remain in operation for a period not to exceed 7 days provided the motor operated isolation and condensate makeup valves in the operable isolation condenser are demonstrated daily to be operable.

D. If Specification 3.8.A and 3.8.B are not met, or if an inoperable isolation condenser cannot be repaired within 7 days, the reactor shall be placed in the cold shutdown condition.

E. If an isolation condenser inlet (steam side) isolation valve (V-14-30, 31, 32 or 33) becomes or is made inoperable, in the open position during the run mode, the redundant inlet isolation valve shall be demonstrated operable. If the inoperable valve is not returned to service within 4 hours declare the affected isolation condenser inoperable, isolate it and comply with Specification 3.8.C.

F. If an AC motor-operated isolation condenser outlet (condensate return) isolation valve (V-14-36 or 37) becomes or is made inoperable in the open position in the run mode, return the valve to service within 4 hours or declare the affected isolation condenser inoperable, isolate it and comply with Specification 3.8.C.

Basis:

The purpose of the isolation condenser is to depressurize the reactor and to remove reactor decay heat in the event that the turbine generator and main condenser is unavailable as a heat sink.[1] Since the shell side of the isolation condensers operate at atmospheric pressure, they can accomplish their purpose when the reactor temperature is sufficiently above 212°F to provide for the heat transfer corresponding to reactor decay heat. The tube side of the isolation condensers form a closed loop with the reactor vessel and can operate without reducing the reactor coolant water inventory.

4.3 REACTOR COOLANT

Applicability: Applies to the surveillance requirements for the reactor coolant system.

Objective: To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Materials surveillance specimens and neutron flux monitors shall be installed in the reactor vessel adjacent to the wall at the midplane of the active core. Specimens and monitors shall be periodically removed, tested, and evaluated to determine the effects of neutron fluence on the fracture toughness of the vessel shell materials. The results of these evaluations shall be used to assess the adequacy of the P-T curves of Figures 3.3.1(a), (b) and (c). New curves shall be generated as required.

B. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 systems and components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(g)(6)(i).

C. Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(g)(6)(i).

D. A visual examination for leaks shall be made with the reactor coolant system at pressure during each scheduled refueling outage or after major repairs have been made to the reactor coolant system in accordance with Article 5000, Section XI. The requirements of specification 3.3.A shall be met during the test.

E. Each replacement safety valve or valve that has been repaired shall be tested in accordance with subsection IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code. Setpoints shall be as follows:

Number of Valves	Set Points (psig)
4	1212 ± 12
4	1221 ± 12
4	1230 ± 12
4	1239 ± 12

F. A sample of reactor coolant shall be analyzed at least every 72 hours for the purpose of determining the content of chloride ion and to check the conductivity.

*G. Primary Coolant System Pressure Isolation Valves
Specification:

1. Periodic leakage testing [(a)] on each valve listed in table 4.3.1 shall be accomplished prior to exceeding 600 psig reactor pressure every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceeding 9 months, whenever the valve is moved whether by manual actuation or due to flow conditions, and after returning the valve to service after maintenance, repair or replacement work is performed.

H. Reactor Coolant System Leakage

1. Unidentified leakage rate shall be calculated at least once every 4 hours.

2. Total leakage rate (identified and unidentified) shall be calculated at least once every 8 hours.

3. A channel calibration of the primary containment sump flow integrator and the primary containment equipment drain tank flow integrator shall be conducted at least once per 18 months.

Bases:

Data is available relating neutron fluence (E>1.0 MeV) and the change in the Reference Nil-Ductility Transition Temperature (RT_{NDT}). The pressure-temperature (P-T) operating curves of Figures 3.3.1(a), (b) and (c) were developed based on the results of testing and evaluation of specimens removed from the vessel after 8.38 EFPY of operation. Similar testing and analysis will be performed throughout vessel life to monitor the effects of neutron irradiation on the reactor vessel shell materials.

The inspection program will reveal problem areas should they occur, before a leak develops. In addition, extensive visual inspection for leaks will be made on critical systems. Oyster Creek was designed and constructed prior to

[(a)] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

*NRC Order dated April 20, 1981.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 120

TO PROVISIONAL OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND
JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

INTRODUCTION

By letter dated January 19, 1988, the GPU Nuclear Corporation (the licensee) proposed to revise the pressure-temperature limits in the Oyster Creek Nuclear Generating Station Technical Specifications through 15 effective-full-power years (EFPY). The proposed pressure-temperature limits were developed from the licensee's submittal, "Testing and Evaluation of Irradiated Reactor Vessel Materials Surveillance Program Specimens," TDR-725. The limits consist of three curves that set minimum pressure and temperature for three operating conditions - hydrostatic and leakage test, heatup or cooldown (core not critical), and heatup or cooldown (core critical). Presently, the plant is about to reach 10 EFPY which is the current technical specification limit for the pressure-temperature curves. The proposed new curves will allow the operator to operate the reactor continuously through 15 EFPY without violating the Technical Specifications.

DISCUSSION

Part of the NRC's effort to ensure integrity of the reactor vessel is to periodically evaluate the reduction in fracture toughness of the vessel material due to neutron irradiation embrittlement. The effort consists of three steps.

First, the licensee is required to establish a surveillance program in accordance with Appendix H of 10 CFR 50, which requires periodic withdrawal of surveillance capsules from the reactor vessel. The capsules are installed in the vessel prior to startup and they should contain test specimens that were made from the plate, weld, and heat affected zone materials of the reactor beltline.

Secondly, the licensee is required to perform Charpy impact tests, tensile tests, and neutron fluence measurements of the specimens. These tests define the condition of vessel embrittlement at the time of capsule withdrawal in terms of the shift of the reference temperature, RT_{NDT}, and upper shelf energy. The licensee should also predict the future vessel embrittlement by calculating the adjusted RT_{NDT} and upper shelf energy at a specific EFPY. The licensee may use either Revision 1 or draft Revision 2 of Regulatory Guide 1.99 to calculate the adjusted RT_{NDT}. The upper shelf energy is the average energy value for all specimens whose test temperature is above the upper end of the transition temperature region. The licensee is required by 10 CFR 50 Appendix G to assure that the adjusted RT_{NDT} will not exceed 200°F and that the upper shelf energy will not be below 50 ft-lb at the end of plant life.

Thirdly, the licensee is required to develop a set of pressure-temperature curves based on the adjusted RT_{NDT} of the limiting vessel material. The curves should satisfy the recommended methods and requirements described in 10 CFR 50, Appendix G and Standard Review Plan 5.3.2.

EVALUATION

The Oyster Creek Nuclear Station is a boiling water reactor which has an inside diameter of 213 inches and mean wall thickness of 7.125 inches. The reactor vessel was fabricated from ASTM A302, Grade B plate material. The submerged arc weld materials were RACO #3 bare wire and ARCO B-5 flux. Manual metal arc welding used 8018 covered electrodes.

General Electric installed three specimen capsules as a part of the reactor vessel surveillance program. The withdrawal of the first capsule in 1971 was unsuccessful. Capsule No. 2 was withdrawn in March 1984 at 8.38 EFPY. After examining specimens in capsule No. 2, the licensee found several material discrepancies and that the program does not meet requirements of 10 CFR 50 Appendix H. For example, the limiting material and the beltline welds were not included in the capsule. The exact copper and nickel contents of several plates and welds were unavailable. These discrepancies were partly due to the vintage of the plant and partly because the surveillance program was initiated before the issuance of Appendix H. Nevertheless, the staff had reviewed the surveillance program under the Systematic Evaluation Program guidelines in the early 1980's and found it acceptable. In this evaluation, the staff concentrated not on review of the program itself but on the pressure-temperature curves.

The specimen capsule data showed that the G-308-1 plate had a RT_{NDT} shift of 72°F measured at 30 ft-lb transition temperature and had received a neutron fluence of 7.46×10^{17} n/cm^2. Since the G-308-1 plate showed a higher RT_{NDT} shift than that of the weld and heat-affected-zone materials in the capsule, the data of the G-308-1 plate were used in the RT_{NDT} calculation of the limiting material.

The licensee used Regulatory Guide 1.99, draft Rev. 2 to calculate the adjusted RT_{NDT} because the Rev. 1. calculation showed a lower RT_{NDT} shift.

To calculate the highest adjusted RT_{NDT}, the licensee compared the copper and nickel contents of the G-308-1 plate to those unirradiated specimens of five other beltline plates not placed in the capsule. The licensee conservatively applied the chemistry factor, neutron fluence and measured RT_{NDT} of the G-308-1 plate to one of the five plate specimens that had the worst combination of copper content, nickel content, and initial RT_{NDT}. The calculation showed that the G-8-6 plate had the highest adjusted RT_{NDT} of 125°F at the neutron fluence of 1.11×10^{18} n/cm^2, 15EFPY, and 1/4T (vessel thickness) location. The G-8-6 plate was selected as the limiting material.

The licensee also predicted the end-of-life adjusted RT_{NDT} of 142°F and the upper shelf energy of 61.5 ft-lb at a neutron fluence of 2.38×10^{18} n/cm^2 for the G-8-6 plate. These values satisfy the 10 CFR 50 Appendix G requirements.

To construct the pressure-temperature curves, the licensee followed closely the method described in NRC's Standard Review Plan 5.3.2 and ASME Section III Appendix G except in the membrane stress calculation. To calculate the membrane stress, the licensee used the "vessel radius-thickness" relationship whereas SRP 5.3.2 prescribed the "allowable stress-design pressure" relationship. The former gives a lower temperature profile than that of the latter; but, the former method is not necessarily incorrect. The staff determined that the licensee's method was acceptable based on the stress analysis of a cylindrical container having a large radius-to-thickness ratio. (Ref. Roark, R.J., "Formulas for Stress and Strain," 4th edition, page 308). The lower part of the pressure-temperature curves also has to satisfy the specific requirements of 10 CFR 50 Appendix G for boiling water reactors because the boiling water reactor vessel has an inherent pressure-temperature limitation when the reactor water level is within the normal range for power operation and the reactor pressure is less than 10 percent of the preservice system hydrostatic test pressure. The pressure-temperature curve is limited by the closure flange regions that are highly stressed by the bolt preload. The minimum permissible temperature should be 60°F above the initial RT_{NDT} of the flange and when the test pressure is above 20% of the hydrotest pressure, the permissible temperature should be 90°F above the initial RT_{NDT}. Based on an initial RT_{NDT} of 40°F for the Oyster Creek reactor flange, the minimum temperature should be 100°F and the permissible test temperature should be 130°F. Examining the lower part of the pressure-temperature curves, the staff determines that the curves satisfy the 10 CFR 50 Appendix G requirements.

The staff has reviewed the proposed pressure-temperature curves and corresponding paragraphs in the Technical Specifications. The licensee has applied appropriately Regulatory Guide 1.99, draft Rev. 2, 10 CFR 50 Appendix G, and Standard Review Plan 5.3.2 to calculate the adjusted RT_{NDT} and to develop the pressure-temperature curves. The staff concludes that the proposed pressure-temperature curves are valid through 15 EFPY and may be incorporated into the Oyster Creek Nuclear Station Technical Specifications.

ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security nor to the health and safety of the public.

Dated: March 21, 1988

Principal Contributor:

J. Tsao