ATTACHMENT

PROPOSED CHANGE

PRESSURE-TEMPERATURE RELATIONSHIP

APPENDIX A

TO

PROVISIONAL OPERATING LICENSE DPR-16
OYSTER CREEK NUCLEAR GENERATING STATION
GPU NUCLEAR CORPORATION
DOCKET NO. 50-219

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 (a) in Figure 3.3.1. 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 (b) in Figure 3.3.1.

(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 (c) in Figure 3.3.1.

Note: Curves (a), (b) and (c) in Figure 3.3.1 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 seventeen (17) 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 85°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 has to account for the toughness properties in the most limiting regions of the reactor vessel, as well as the effects of fast neutron embrittlement.

Curves (a),(b) and (c) on Figure 3.3.1 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 17 effective full power years of operation is 2.44×10^{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 (a),(b), and (c) on Figure 3.3.1 (Reference 15). 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 surface adjacent to the "O" rings of the head and vessel flange. The original Code requirement was that boltup be done at qualification temperatures (T3OL) plus 60°F. Current Code requirements state (Ref. 16) that for application of full bolt preload and reactor pressure up to 20% of hydrostatic test pressure, the RPV metal temperature must be at RT_{NDT} or greater. The boltup temperature of 85°F was derived by determining the highest value of (T3OL + 60) and the highest value of RT_{NDT}, and by choosing the more conservative value of the two. Calculated values of (T3OL + 60) and RT_{NDT} of the RPV metal temperature were 85°F and 36°F, respectively (Ref. 15). Therefore, selecting the boltup temperature to be 85°F provides 49°F margin over the current Code requirement based on RT_{NDT}.

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)

The expected number of normal heatup and cooldown cycles to which the vessel will be subjected is 80(7). Although no heatup or cooldown rates of 300°F per hour are expected over the life the vessel and the vessel design did not consider such events(6), stress analyses have been made which showed the allowable number of such events is 22,000 on the basis of ASME Section III alternating stress limits.

During reactor operation, the temperature of the coolant in an idle recirculation loop is expected to remain at reactor coolant temperature unless it is valved out of service. Requiring the coolant temperature in an idle loop to be within 50°F of the reactor coolant temperature before the sump is started assures that the change in coolant temperature at the reactor vessel nozzles and bottom head region are within the conditions analyzed for the reactor vessel as discussed above.

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite AC power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work(8) utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in the 3.3-D on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm as specified in 3.3-D, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage of the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the plant should be shut down to allow further investigation and corrective action.

The drywell floor drain sump and equipment drain tank provide the primary means of leak detection(9,10). Identified leakage is that from valves and pumps in the reactor system and from the reactor vessel head flange gasket. Leakage through the seals of this equipment is piped to the drywell equipment drain tank. Leakage from other sources is classified as unidentified leakage and is collected in the drywell floor drain sump. Leakage which does not flash in a vapor will drain in the sump. The vapor will be condensed in the drywell ventilation system and routed to the sump.

Condensate cannot leave the sump or the drywell equipment drain tank unless the respective pumps are running. The sump and the drain tank are provided with two pumps each. Alarms are provided for the sump that will actuate on a predetermined pumpout rate(10) and will be set to actuate at a leakage that is less than the unidentified leakage limit of 5 gpm.

Additional qualitative information(10) is available to the operator via the monitored drywell atmospheric condition. However, this information is not quantitative since fluctuation in atmospheric conditions are normally expected, and quantitative measurements are not possible. The temperature of the closed cooling water which serves as coolant for the drywell ventilation system is monitored and also provides information which can be related to reactor coolant system leakage(9). Additional protection is provided by the drywell high pressure scram which would be expected to be reached within 30 minutes of a steam leak of about 12 gpm(10).

During a loss of offsite AC power, the control rod drive hydraulic pumps, which are powered by the diesels, each can supply 110 gpm water makeup to the reactor vessel. A 25 gpm limit for total leakage, identified and unidentified, was established to be less than the 110 gpm makeup of a single rod drive hydraulic pump to avoid the use of the emergency core cooling system in the event of a loss of normal AC power.

Materials in the primary system are primarily 304 stainless steel and zircaloy fuel cladding. The reactor water chemistry limits are placed upon conductivity and chloride concentration since conductivity is measured continuously and gives an indication of abnormal conditions or the presence of unusual materials in the coolant, while chloride limits are specified to prevent stress corrosion cracking of stainless steel.

Chlorides are known to (1) promote intergranular stress corrosion cracking of sensitized steels, (2) induce transgranular cracking of non-sensitized stainless steels, (3) promote pitting and (4) promote crevice attack in most RCS materials (BWR Water Chemistry Guidelines, EPRI, April 1, 1984). The higher the concentration, the faster the attack. Therefore, the level of chloride in the reactor water should be kept as low as is practically achievable. The limits are therefore set to be consistent with Regulatory Guide 1.56 (Rev. 1).

In the case of BWR's where no additives are used in the primary coolant, and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. When the conductivity is within its proper normal range, pH, chloride, and other impurities affecting conductivity and water quality must also be within their normal ranges. Significant changes in conductivity provide the operator with a warning mechanism so that he can investigate and remedy the conditions causing the change.

Measurements of 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.

Specifications 3.3.F.1 and 3.3.F.2 require a minimum of four OPERATING recirculation loops during reactor POWER OPERATION. Core parameters have not been established for POWER OPERATION with less than four OPERATING loops. Therefore, Specification 3.3.F.3 requires reactor POWER OPERATION to be terminated and the reactor placed in the REFUEL MODE or SHUTDOWN CONDITION within 12 hours.

During four loop POWER OPERATION the idle loop, when it is not isolated, is required to have its discharge valve closed and its discharge bypass and suction valves open. This provides and limits reactor coolant backflow through an idle loop and thus minimizes the occurrence of a severe cold water addition transient during startup of an idle loop. In addition, with the discharge bypass and suction valves in an idle loop open the coolant inventory in the loop is available during LOCA blowdown.

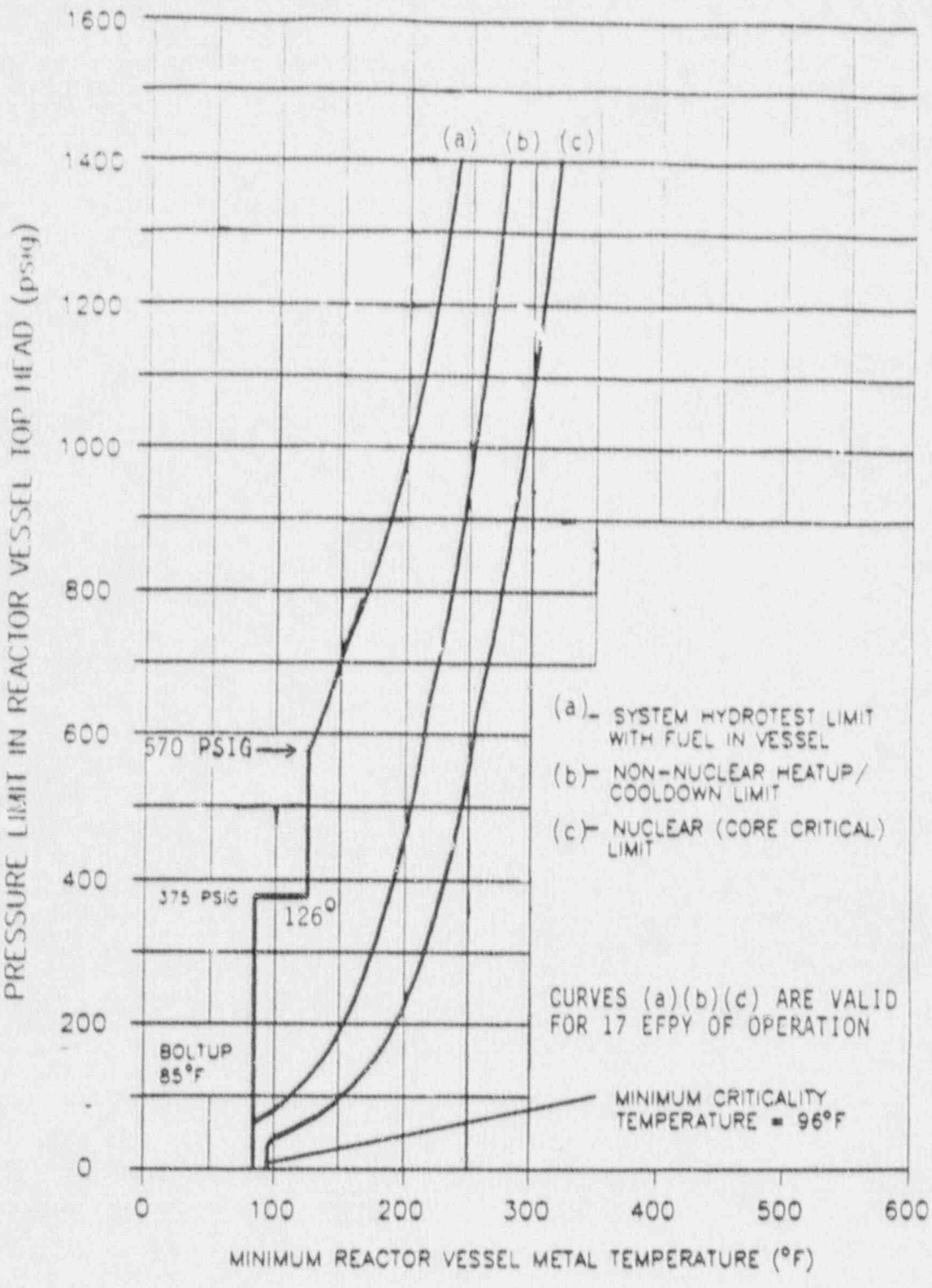
The requirements of Specification 3.3.F.2 for partial loop operation in which the idle loop is isolated, preclude the inadvertent startup of a recirculation pump with a cold leg thus avoiding any reactivity addition transient or reactor vessel nozzle thermal stress concerns.

Specifications 3.3.F.4 and 3.3.F.6 assure that an adequate flow path exists from the annular space, between the pressure vessel wall and the core shroud, to the core region. This provides sufficient hydraulic communication between these areas, thus assuring that reactor water instrument readings are indicative of the level in the core region. For the bounding loss of feedwater transient(2), a single fully open recirculation loop transfers coolant from the annulus to the core region at approximately five times the boiloff rate with no forced circulation(3). With the reactor vessel flooded to a level above 185 inches TAF or when the steam separator and dryer are removed, the core region and all recirculation loops can therefore be isolated. When the steam separator and dryer are removed, safety limit 2.1.D ensures water level is maintained above the core shroud.

References: (1) FDSAR, Volume I, Section IV-2
(2) Letter to NRC dated May 19, 1979, "Transient of May 2, 1979"
(3) General Electric Co. Letter G-EN-9-55, "Revised Natural Circulation Flow Calculation", dated May 29, 1979
(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. 3: Testing and Evaluation of Irradiated Reactor Vessel Materials Surveillance Program Specimens
(15) SASR 90-89 (GE Nuclear Energy): Pressure-Temperature Curves Per Regulatory Guide 1.99, Revision 2 for Oyster Creek Nuclear Generating Station.
(16) Paragraph G-2222(C), Appendix G, Section XI, ASME Boiler and Pressure Vessel Code, 1989 Edition with 1989 Addenda, "Fracture Toughness Criteria for Protection Against Failure."

FIGURE 3.3.1

OYSTER CREEK P/T LIMITS



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 (a),(b) and (c) in Figure 3.3.1. 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 preceding 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>1.0MeV) and the change in the Reference Nil-Ductility Transition Temperature (RT_{NDT}). The pressure-temperature (P-T) operating curves (a),(b) and (c) in Figure 3.3.1 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

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(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.