GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION

Provisional Operating
License No. DPR-16

Technical Specification
Change Request No. 181, Revision 1
Docket No. 50-219

Applicant submits, by this Technical Specification Change Request No. 181, Revision 1 to the Oyster Creek Nuclear Generating Station Technical Specifications, a change to pages 2.3-2, 2.3-6, and 4.3-1.

By E E Fitzpatrick
E. E. Fitzpatrick
Vice President and Director
Oyster Creek

Sworn and Subscribe to before me this 16th day of October, 1990.

Judith M. Crowe
A Notary Public of NJ

JUDITH M. CROWE
Notary Public of New Jersey
My Commission Expires 1-25-95

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	Docket No. 50-219
GPU Nuclear Corporation	)	

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 181, Revision 1 for Oyster Creek Nuclear Generating Station Technical Specifications, filed with the U.S. Nuclear Regulatory Commission on October 16, 1990 has this day of October 16, 1990, been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the United States mail, addressed as follows:

The Honorable Debra Madensky
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

By E E Fitzpatrick
E. E. Fitzpatrick
Vice President and Director
Oyster Creek

OYSTER CREEK NUCLEAR GENERATING STATION
PROVISIONAL OPERATING LICENSE NO. DPR-16
DOCKET NO. 50-219
TECHNICAL SPECIFICATION CHANGE REQUEST NO. 181
REVISION 1

Applicant hereby requests the Commission to change Appendix A to the above captioned license as below, and pursuant to 10CFR50.91, an analysis concerning the determination of no significant hazards consideration is also presented:

1.0 SECTIONS TO BE CHANGED

Sections 2.3 and 4.3.

2.0 EXTENT OF CHANGE

Eliminate seven main steam safety valves (safety valves) by taking credit for high flux reactor scram in the safety analysis.

Sections 2.3.F and 4.3.E are revised to delete seven safety valves with the two highest setpoints. The bases for Section 2.3 are revised to incorporate credit for reactor scram and no credit for recirculation pump trip for safety valve sizing and change total number of safety valves from sixteen to nine.

3.0 CHANGES REQUESTED

The requested changes are shown on attached Technical Specification pages 2.3-2, 2.3-6 and 4.3-1.

4.0 DISCUSSION

The purpose of this Technical Specification Change Request is to propose the elimination of seven safety valves with the two highest setpoints. Appropriate safety analyses have been performed to demonstrate the acceptability of the reduction in the number of safety valves. A reduction in safety valves would result in significant cost savings in maintenance and surveillance testing. In addition, it is estimated that the deletion of seven safety valves would reduce exposure by 20 man-rem per outage.

The reactor pressure vessel (RPV) and the pressure relief system were designed in accordance with Section I, 1962 edition of the American Society of Mechanical Engineers (ASME) "Boiler and Pressure Vessel Code". Under the provision of Section I, code qualified safety valves must limit the rise in the RPV pressure to less than the ASME code limit. Previous analyses performed to demonstrate compliance with the code requirements did not take credit for reactor scram, electromatic relief valves (EMRVs), turbine bypass valves and the isolation condensers. To satisfy this requirement, Oyster Creek currently employs 16 steam safety valves.

The current version of the ASME code, Section I, allows credit for independent sensing devices that stop the flow of fuel to the boiler. Since the code is for fossil boilers, the analogy for a nuclear plant is that credit for an independent or diverse shutdown system such as flux scram, would perform the same function of fuel stoppage, i.e. boiler shutdown. Thus, credit could be taken for its functioning in the overpressure protection analysis consistent with the current interpretation of the ASME code.

In addition, NUREG-0800, "Standard Review Plan", indicates that the safety valves should be designed with sufficient capacity to limit the pressure to less than 110% of the reactor coolant pressure boundary (RCPB) design pressure (as specified by ASME Boiler and Pressure Vessel Code, Section III) during the most severe abnormal operational transient with credit for a reactor scram. All BWR plants designed in accordance with Section III of the ASME Code currently take credit for high neutron flux scram for safety valve sizing.

The appropriate code limits are observed for the new configuration. This system has no function during normal operation, and it is anticipated that there is a low probability of safety valve actuation since overpressure is relieved by the isolation condensers, the turbine bypass valves and the EMRVs.

The safety analysis requirements for Oyster Creek have been reviewed in order to establish the analyses that are potentially affected by the reduction in the number of safety valves. In the safety analysis process, no credit is taken for the operation of the safety valves except for the ASME code overpressure protection analysis and the evaluation of anticipated transients without scram (ATWS). These events have been reanalyzed using the NRC approved methodology for Oyster Creek with the exception below, to demonstrate compliance with the appropriate event acceptance limits.

License Basis Analyses

The safety valves at Oyster Creek are required to protect the primary coolant pressure boundary against overpressure. Overpressure protection is provided by limiting peak pressure in the reactor vessel to 110% of design pressure and to 115% of design pressure for the recirculation piping. The RPV design pressure is 1250 psig which requires the limit to be 1375 psig (1390 psia). The recirculation piping design pressure is 1200 psig, which results in a limit of 1380 psig (1395 psia).

For Oyster Creek, a main steam isolation valve (MSIV) closure without scram or credit for operation of the EMRVs, also known as the safety valve sizing transient, is analyzed in the updated FSAR Chapter 15 to determine the adequacy of the safety valves to prevent vessel overpressurization. This event has been demonstrated as being limiting using the NRC approved Oyster Creek safety analysis methodology. In previous license basis analyses, this event analysis was used to cover both the code overpressure protection and ATWS analysis requirements. For this analysis, the MSIVs are assumed to close in 3 seconds, all scram activations fail, and all solenoid-operated relief valves (EMRVs), bypass valves and isolation condenser isolation valves are assumed to fail. Credit is taken for the RPT. The void collapse results in a power increase followed by a pressure increase which is limited only by the safety valves and the nuclear characteristics of the core design. This transient was analyzed as part of the Cycle 12 reload and resulted in a peak pressure at the bottom of the vessel of 1305 psia which is well below the vessel pressure safety limit of 1390 psia (1375 psig).

In order to evaluate the impact of the reduction in the safety valves on the license basis analysis requirements, it is necessary to evaluate two events in place of the safety valve sizing transient. These two events are the MSIV closure with high flux scram with no credit taken for the non-safety grade RPT and the MSIV closure ATWS with RPT. Previous evaluations have demonstrated that the use of the MSIV closure as the initiating event bounds the spectrum of potential initiating events for Oyster Creek. The analysis results for these two events are described in more detail below.

MSIV Closure with High Flux Scram (9 Safety Valves) and No RPT

The licensed cycle 12 reload model was used for this analysis with the NRC approved RETRAN-02 Mod4 code. A single change to the model was made to assure conservative results for peak pressure. This involved increasing the rainout velocity in the upper downcomer volume from 3 feet per second to 1000 feet per second. It was observed for this transient that when the level in the upper downcomer dropped below the separator drains, liquid was entrained in the steam region of the upper downcomer and subsequently, some of this liquid was carried over by RETRAN into the upper plenum volume. The use of a large rainout velocity in the upper downcomer volume prevented carry-over of liquid and resulted in higher peak pressures.

For the reload transients, the level does not drop below the separators' drains and the 3 feet per second rainout velocity is conservative.

The same assumptions as previously listed were used with the exception of allowing a high flux scram with no RPT. For this analysis, a scram would normally occur on MSIV closure of 10%. This anticipates the pressure and neutron flux transients which occur during normal or inadvertent valve closure. However, no credit is taken for this scram signal in the analyses. The reactor is assumed to be scrammed by the high flux scram signal at a conservative setpoint of 120% as compared to the actual setpoint of 115.7%.

For this analysis, the nine safety valves (Banks 1 and 2) are assumed to open on a high steam line pressure of 1240 psia (Bank 1-4 valves), 1249 psia (Bank 2-4 valves) and 1257 psia (Bank 3-1 valve), and close on a low steam line pressure of 1190 psia (Bank 1-4 valves), 1199 psia (Bank 2-4 valves) and 1207 psia (Bank 3-1 valves). It was determined that nine safety valves (as opposed to 16 for the license basis analysis) would limit the peak pressure at the bottom of the vessel to 1375.5 psia, below the code limit of 1390 psia. The pressure in the recirculation piping is 1382 psia which is within the 1395 psia code requirement for the piping.

MSIV Closure ATWS (8 Safety Valves) with RPT

A MSIV closure ATWS with RPT was evaluated to demonstrate that the results for the postulated ATWS event were acceptable considering the reduction of safety valves. This transient is analyzed with the same conditions as described above, using eight safety valves, but credit is taken for the EMRVs and for recirculation pump trip. The peak pressure in the reactor vessel was determined to be 1297 psia. Based on these results, the ATWS analysis is less limiting than the above flux scram case and does not have to be reanalyzed for future reloads. This analysis only addresses the overpressurization limits associated with an ATWS since the effects of other limits remain the same with eight or 16 safety valves.

5.0 DETERMINATION

The proposed Technical Specification Change Request does not involve a significant hazards consideration for the reasons as stated below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated:

   The removal of seven safety valves will not increase the probability of occurrence of an accident previously evaluated in the SAR since the remaining safety valves remain unchanged. The only event initiator that involves a safety valve is a spurious valve opening. The proposed reduction in valves will slightly reduce the probability of a spurious valve opening. Thus, the probability of a valve opening is not increased.

   In the safety analysis process, credit for the operation of the safety valves is only taken in the code overpressure protection and ATWS events. These events have been reanalyzed using the approved Oyster Creek license analysis methodology. With the reduced number of safety valves and no credit for the high flux scram, the peak calculated pressure due to these events previously reported in the Safety Analysis report would be increased. However, with the proposed change to the design basis to take credit for the high flux scram and no creidt for RPT, the appropriate event acceptance limits are satisfied.

   The activity will not significantly increase the probability of occurrence or consequence of a malfunction of equipment important to safety previously evaluated in the SAR based on a reliability analysis of RPT, EMRVs and remaining safety valves (9) which shows that the likelihood of reactor vessel overpressure due to an ATWS remains very small. Also, since there will be seven fewer safety valves, the likelihood of an initiating event involving spurious opening of a safety valve is reduced.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed activity does not create a possibility for an accident or malfunction of a different type than any previously identified in the SAR since existing safety valves remain unchanged, and no systems are affected by this modification. Analyses demonstrate that all of the appropriate event acceptance limits have been satisfied for the proposed new configuration.

   The seven safety valves removed will be replaced with blind flanges to maintain the reactor coolant pressure boundary (RCPB). After installation, initial service leak test will be performed, thus assuring the integrity of the RCPB.

3. Involve a significant reduction in a margin of safety.

The margin of safety as presently defined in the basis for the Technical Specifications does not take credit for high flux scram. This Technical Specification Change Request proposes to take credit for high flux scram with no credit for RPT and then require only nine safety valves to mitigate the consequences of a MSIV closure transient.

For the purposes of this evaluation, the margin of safety is defined as the margin between the safety limit and fission product barrier failure. Because the event does not exceed the event limit (1375 psig), the margin of safety is not reduced.

6.0 IMPLEMENTATION

It is requested that the amendment authorizing this change become effective for operating Cycle 13.

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS	
B. Neutron Flux, Control Rod Block	The Rod Block setting shall be $S \leq [(0.90 \times 10^{-6}) W + 53.1] \left[\frac{FRP}{MFLPD}\right]$ with a maximum setpoint of 108% for core flow equal to 61 x 10^{6} lb/hr and greater. The definitions of S, W, FRP and MFLPD used above for the APRM scram trip apply. The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used. This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM rod block curve by the reciprocal of the APRM gain change.	
C. Reactor High, Pressure, Scram	$\leq$1060 psig	
D. Reactor High Pressure, Relief Valves Initiation	2 @ $\leq$ 1070 psig 3 @ $\leq$ 1090 psig	
E. Reactor High Pressure, Isolation Condenser Initiation	$\leq$1060 psig with time delay $\leq$3 seconds	
F. Reactor High Pressure, Safety Valve Initiation	4 @ 1212 psig 4 @ 1221 psig 1 @ 1230 psig	$\pm$12 psi $\pm$12 psi $\pm$12 psi
G. Low Pressure Main Steam Line, MSIV Closure	$\geq$825 psig (initiated in IRM range 10)	
H. Main Steam Line Isolation Valve Closure, Scram	$\leq$10% Valve Closure from full open	
I. Reactor Low Water Level, Scram	$\geq$11'5" above the top of the active fuel as indicated under normal operating conditions	
J. Reactor Low-Low Water Level, Main Steam Line Isolation Valve Closure	$\geq$7'2" above the top of the active fuel as indicated under normal operating conditions	

The reactor coolant system safety valves offer yet another protective feature for the reactor coolant system pressure safety limit since these valves are sized assuming no credit for other pressure relieving devices. In compliance with Section I of the ASME Boiler and Pressure Vessel Code, the safety valve must be set to open at a pressure no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. The safety valves are sized according to the Code for a condition of main steam isolation valve closure while operating at 1930 MWt, followed by [1] a reactor scram on high neutron flux, [2] failure of recirculation pump trip on high pressure, [3] failure of the turbine bypass valves to open, and [4] failure of the isolation condensers and relief valves to operate. Under these conditions, a total of 9 safety valves are required to turn the pressure transient. The ASME B&PV Code allows a ±1% of working pressure (1250 psig) variation in the lift point of the valves. This variation is recognized in Specification 4.3.

The low pressure isolation of the main stream lines at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cool-down of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the STARTUP position and the IRMs be in the range 9, or lower, where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valves closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure.

The low water level trip setting of 11'5" above the top of the active fuel has been established to assure that the reactor is not operated at a water level below that for which the fuel cladding integrity safety limit is applicable. With the scram set at this point, the generation of steam, and thus the loss of inventory, is stopped. For example, for a loss of feedwater flow a reactor scram at the value indicated and isolation valve closure at the low-low water level set point results in more than 4 feet of water remaining above the core after isolation (6).

During periods when the reactor is shut down, decay heat is present and adequate water level must be maintained to provide core cooling. Thus, the low-low level trip point of 7'2" above the core is provided to actuate the core spray system (when the core spray system is required as identified in Section 3.4) to provide cooling water should the level drop to this point.*

The turbine stop valve(s) scram is provided to anticipate the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valve(s) and failure of the turbine bypass system.

The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control

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
1	1230 ± 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.