GPU NUCLEAR

GPU Nuclear, Inc.
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Tel 609-971-4000

December 22, 1999
1940-99-20677

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen,

Subject: Oyster Creek Nuclear Generating Station, (OCNGS)
Docket No. 50-219
Technical Specification Change Request (TSCR) No. 267
Changes Arising From NRC Engineering Inspection
Supplement 1 – Revised Pages

By letter dated April 15, 1999 GPU Nuclear submitted TSCR 267. The changes requested in that submittal affected some pages which were modified by TSCR 266. Amendment 208, the response to TSCR 266, was issued in June, 1999. Therefore, the pages in TSCR 267 that were common to 266, did not reflect the changes as issued in Amendment 208. The revised pages are attached with change bars indicating all changes including changes in pagination. Also marked is a sentence in the first paragraph on page 2.3-7 that was included in the original submittal but not noted in the application.

Also enclosed is a Certificate of Service for this request, certifying service to the chief executives of the township and county in which the facilities are located, as well as the designated official of the state of New Jersey, Bureau of Nuclear Engineering.

If additional information is required, please contact Dennis Kelly of my staff at (609) 971-4246.

Sincerely,

Sander Levin

Sander Levin
Acting Director
Oyster Creek

cc: Region I Administrator
Oyster Creek Project Manager
Oyster Creek Senior Resident Inspector

A001

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December 22, 1999
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The Honorable William J. Boehm
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Dear Mayor:

Subject: Oyster Creek Nuclear Generating Station
Operating License No. DPR-16
Technical Specification Change Request No. 267
Supplement 1 – Revised Pages

Enclosed is one copy of the Technical Specification Change Request No. 267, Supplement 1 – Revised Pages for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the U.S. Nuclear Regulatory Commission on December 22, 1999.

Very truly yours,

Sander Levin

Sander Levin
Acting Director
Oyster Creek

SL/DPK
Enclosure

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December 22, 1999
1940-99-20677

Mr. Kent Tosch, Director
Bureau of Nuclear Engineering
Department of Environmental Protection
CN 415
Trenton, NJ 08628

Dear Mr. Tosch:

Subject: Oyster Creek Nuclear Generating Station
Operating License No. DPR-16
Technical Specification Change Request No. 267
Supplement 1 – Revised Pages

Enclosed is one copy of the Technical Specification Change Request No. 267, Supplement 1 – Revised Pages for the Oyster Creek Nuclear Generating Station Operating License.

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Very truly yours,

Sander Levin
Acting Director
Oyster Creek

SL/DPK
Enclosure

OYSTER CREEK NUCLEAR GENERATING STATION

OPERATING LICENSE
NO. DPR-16

TECHNICAL SPECIFICATION
CHANGE REQUEST NO. 267, Supplement 1
DOCKET NO. 50-219

Applicant submits by this Technical Specification Change Request No. 267, Supplement 1 to the Oyster Creek Nuclear Generating Station Technical Specifications, modified pages 2.3-4, 2.3-5, 2.3-6, 2.3-7 and 2.3-8

By: Sander Levin

Sander Levin
ActingDirector
Oyster Creek

Sworn to and Subscribed before me this 22nd day of Dec. 1999.

Betty Goodheart

Notary Public BETTY GOODHEART
A Notary Public of New Jersey
My Commission Expires Mar. 22, 2004

NOTARY PUBLIC

2.3 LIMITING SAFETY SYSTEM SETTINGS

Bases:

Safety limits have been established in Specifications 2.1 and 2.2 to protect the integrity of the fuel cladding and reactor coolant system barriers, respectively. Automatic protective devices have been provided in the plant design for corrective actions to prevent the safety limits from being exceeded in normal operation or operational transients caused by reasonably expected single operator error or equipment malfunction. This Specification establishes the trip settings for these automatic protection devices.

The Average Power Range Monitor, APRM[1], trip setting has been established to assure never reaching the fuel cladding integrity safety limit. The APRM system responds to changes in neutron flux. However, near the rated thermal power, the APRM is calibrated using a plant heat balance, so that the neutron flux that is sensed is read out as percent of the rated thermal power. For slow maneuvers, such as those where core thermal power, surface heat flux, and the power transferred to the water follow the neutron flux, the APRM will read reactor thermal power. For fast transients, the neutron flux will lead the power transferred from the cladding to the water due to the effect of the fuel time constant. Therefore, when the neuron flux increases to the scram setting, the percent increase in heat flux and power transferred to the water will be less than the percent increase in neutron flux.

The APRM trip setting will be varied automatically with recirculation flow, with the trip setting at the rated flow of 61.0×10^6 lb/hr of greater being 115.7% of rated neutron flux. Based on a complete evaluation of the reactor dynamic performance during normal operation as well as expected maneuvers and the various mechanical failures, it was concluded that sufficient protection is provided by the simple fixed scram setting (2,3). However, in response to expressed beliefs (4) that variation of APRM flux scram with recirculation flow is a prudent measure to ensure safe plant operation, the scram setting will be varied with recirculation flow.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which could have an adverse effect on reactor safety because they unnecessarily challenge the operators. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit and yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.3.A, when the MFLPD is greater than the fraction of the rated power (FRP). the adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

The low pressure isolation of the main steam line at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cool-down of the vessel. The low-pressure isolation protection is enabled with entry into IRM range 10 or the RUN mode. In addition, a scram on 10% main steam isolation valve (MSIV) closure anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. Bypass of the MSIV closure scram function below 600 psig is permitted to provide sealing steam and allow the establishment of condenser vacuum. Advantage is taken of the MSIV scram feature to provide protection for the low-pressure portion of the fuel cladding integrity safety limit. To continue operation beyond 12% of rated power, the IRM's must be transferred into range 10. Reactor pressure must be above 825 psig to successfully transfer the IRM's into range 10. Entry into range 10 at less than 825 psig will result in main steam line isolation valve closure and MSIV closure scram. This provides automatic scram protection for the fuel cladding integrity safety limit which allows a maximum power of 25% of rated at pressures below 800 psia. Below 600 psig, when the MSIV closure scram is bypassed, scram protection is provided by the IRMs.

Operation of the reactor at pressure lower than 825 psig requires that the mode switch be in the STARTUP position and the IRMs be in range 9 or lower. The protection for the fuel clad integrity safety limit is provided by the IRM high neutron flux scram in each IRM range. The IRM range 9 high flux scram setting at 12% of rated power provides adequate thermal margin to the safety limit of 25% of rated power. There are few possible significant sources of rapid reactivity input to the system through IRM range 9: effects of increasing pressure at zero and low void content are minor; reactivity excursions from colder makeup water, will cause an IRM high flux trip; and the control rod sequences are constrained by operating procedures backed up by the rod worth minimizer. In the unlikely event of a rapid or uncontrolled increase in reactivity, the IRM system would be more than adequate to ensure a scram before power could exceed the safety limit. Furthermore, a mechanical stop on the IRM range switch requires an operator to pull up on the switch handle to pass through the stop and enter range 10. This provides protection against an inadvertent entry into range 10 at low pressures. The IRM scram remains active until the mode switch is placed in the RUN position at which time the trip becomes a coincident IRM upscale, APRM downscale scram.

The adequacy of the IRM scram was determined by comparing the scram level on the IRM range 10 to the scram level on the APRMs at 30% of rated flow. The IRM scram is at 38.4% of rated power while the APRM scram is at 52.7% of rated power. The minimum flow for Oyster Creek is at 30% of rated power and this would be the lowest APRM scram point. The increased recirculation flow to 65% of flow will provide additional margin to CPR Limits. The APRM scram at 65% of rate flow is 87.1% of rated power, while the IRM range 10 scram remains at 38.4% of rated power. Therefore, transients requiring a scram based on flux excursion will be terminated sooner with a IRM range 10 scram than with an APRM scram. The transients requiring a scram by nuclear instrumentation are the loss of feedwater heating and the improper startup of an idle recirculation loop. The loss of feedwater heating transient is not affected by the range 10 IRM since the feedwater heaters will not be put into service until after the LPRM downscales have cleared, thus insuring the operability of the APRM system. This will be administratively controlled. The improper startup of an idle recirculation loop becomes less severe at lower power level and the IRM scram would be adequate to terminate the flux excursion.

The Rod Worth Minimizer is not required beyond 10% of rated power. The ability of the IRMs to terminate a rod withdrawal transient is limited due to the number and location of IRM detectors. An evaluation was performed that showed by maintaining a minimum recirculation flow of 39.65×10^6 lb/hr in range 10 a complete rod withdrawal initiated at 35% of rated power or less would not result in violating the fuel cladding safety limit. Therefore, a rod block on the IRMs at less than 35% of rated power would be adequate protection against a rod withdrawal transient.

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR Fuel Cladding Integrity Safety Limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship. Therefore, the worst-case MCPR, which could occur during steady-state operation, is at 108% of the rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of the rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gains.

The settings on the reactor high pressure scram, anticipatory scrams, reactor coolant system relief valves and isolation condenser have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition, the APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits, e.g., turbine trip and loss of electrical load transients (5). In addition to preventing power operation above 1060 psig, the pressure scram backs up the other scrams for these transients and other steam line isolation type transients. Actuation of the isolation condenser during these transients removes the reactor decay heat without further loss of reactor coolant thus protecting the reactor water level safety limit.

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 the 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 level 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). The TAF definition of 353.3 inches from vessel zero is based on a fuel length of 144 inches and it is applicable to the current fuel length of 145.24 inches.

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 valves to a load rejection and failure of the turbine bypass system. This scram is initiated by the loss of turbine acceleration relay oil pressure. The timing for this scram is almost identical to the turbine trip.

The undervoltage protection system includes a 2 out of 3 coincident logic relay designed to shift emergency buses to on-site power should normal power be degraded to an unacceptable level. There is a separate relay system designed to shift emergency buses C and D to on-site power should normal power be lost. The trip points and time delay settings have been selected to assure an adequate power source to emergency safeguards systems in the event of a total loss of normal power or degraded conditions which would adversely affect the functioning of engineered safety features connected to the plant emergency power distribution system.

The APRM downscale signal insures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal coincident with an associate IRM Upscale (High-High) or Inoperative signal generates a trip signal. This function is not specifically credited in the accident analyses but it is retained for overall redundancy and diversity of the RPS.

<u>References</u>

(1) FDSAR, Volume 1, Section VII-4.2.4.2

(2) FDSAR, Amendment 28, Item III.A-12

(3) FDSAR, Amendment 32, Question 13

(4) Letters, Peter A. Morris, Director, Division of Reaction Licensing, USAEC, to John E. Logan, Vice President, Jersey Central Power and Light Company

(5) FDSAR, Amendment 65, Section B.XI

(6) FDSAR, Amendment 65, Section B.IX