Ref: LCR 90-09

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATIONS AND BASES CHANGE

PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NPF-57

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

ref: LCR 90-09

DESCRIPTION OF THE CHANGE

As shown on the marked-up Technical Specifications (TS) and BASES pages in Attachment 2, PSE&G requests that DEFINITION 1.10, CRITICAL POWER RATIO, and BASES Sections B2.1.1, B2.1.2, B3/4.2.3, Bases Table B2.1.2-2, and certain References in those Bases Sectic›s be revised.

REASON FOR THE CHANGE

The proposed changes are administrative in nature, making the Hope Creek Generating Station (HCGS) TS definition for CRITICAL POWER RATIO more generic with regard to the critical power correlation. This will permit the use of new NRC-approved fuel designs and their associated NRC-approved correlations without requiring amendment of this section each time. Additionally, the BASES for THERMAL POWER and MINIMUM CRITICAL POWER RATIO are modified to reference the latest approved version of the GE Standard Application for Reactor Fuel (GESTAR II) and to more clearly define how certain MCPR factors are determined.

JUSTIFICATION FOR THE CHANGE

Since the fuel design and its supporting analysis methodologies have to be previously reviewed and approved by NRC before use in the HCGS reactor, this change to a DEFINITION is essentially administrative in nature. This proposed amendment will preclude TS DEFINITION evisions every time there are minor changes in the fuel ma facturer's critical power correlations to support their new fuel design features. Provided those changes are reviewed and approved by the NRC, the more generic reference, "**applicable NRC-approved critical power correlation**", is appropriate.

The changes to the BASES, which deviate from the standard TS language are clarifications provided to PSE&G by General Electric Company for inclusion in this amendment request.

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

PSE&G has, pursuant to 10CFR50.92, reviewed the proposed amendment to determine whether our request involves a significant hazards consideration. We have determined that:

The operation of Hope Creek Generating Station (HCGS) in accordance with the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment does not involve a physical or procedural change for any structure, component or system that affects the probability or consequences of any accident or malfunction of equipment important to safety previously evaluated in the Updated Final Safety Analysis Report (UFSAR). In order to install any new fuel design in the HCGS reactor, the change in fuel design and supporting correlations will have been previously reviewed and approved by the NRC and the limiting transients previously evaluated in the SAR will have been re-analyzed for each reload design. New core operating limits will have been generated and documented in the CORE OPERATING LIMITS REPORT (referenced in the Technical Specifications) to ensure that all safety criteria were met for all analyzed accidents and limiting transients. Therefore, the CRITICAL POWER RATIO definition will always be correct - in that the CPR correlation being used will have been approved by the NRC as part of any new fuel design approval.

The operation of Hope Creek Generating Station (HCGS) in accordance with the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

There are no physical changes to the plant or to the manner in which the plant is operated involved in the proposed revision. The proposed change will define CRITICAL POWER RATIO as the ratio of that power in an assembly which is calculated by application of the "**applicable NRC-approved critical power correlation**" to cause some point in the assembly to experience boiling transition, divided by the actual assembly power. The previous definition specified General Electric's "GEXL" correlation which has been modified to include considerations for a high performance spacer (ferrule type) design in the GE9 fuel. The new correlation used is termed, "GEXL-plus". This proposed amendment will preclude TS DEFINITION revisions every time there are minor changes in the fuel manufacturer's critical power correlations to support their new fuel design features. Provided those changes are reviewed and approved by the NRC, the more generic reference, "**applicable NRC-approved critical power correlation**", no new or different accident, from any previously evaluated, is created by this broader definition.

The operation of Hope Creek Generating Station (HCGS) in accordance with the proposed change does not involve a significant reduction in a margin of safety.

For each core loading, chapters 4 and 15, which contain information about the fuel design and the results of safety analyses, are re-evaluated. This process ensures that the fuel system design, nuclear design, thermal/hydraulic design and the conclusions of the original core analysis remain valid for the accidents and limiting transients previously evaluated in the FSAR. The proposed revision will merely redefine, in broader terms, the definition of critical power ratio and will not cause a change in any margin of safety.

Conclusion:

Based upon the foregoing evaluation, we have determined that this proposed change does not involve a Significant Hazards Consideration.

Ref: LCR 90-09

ATTACHMENT 2

INSERTS AND MARKED-UP PAGES

INSERTS FOR PROPOSED CHANGES

INSERT 1

applicable NRC-approved critical power

INSERT 2

performed at reduced

INSERT 3

a statistical model that combines all of the uncertainties in operating parameters and in the procedures used to calculate critical power. Calculation of the Safety Limit MCPR is defined in Reference 1. The required inputs to the statistical model are the uncertainties listed in Bases Table B2.1.2-1.

INSERT 4

Reference:

1. General Electric Standard Applicationfor Reactor Fuel, NEDE-24011-P-A (latest approved revision).

INSERT 5

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INSERTS - Cont'd

INSERT 6

The codes used to evaluate transients are discussed in Reference 2.

INSERT 7

operating limit

INSERT 8

The K_f factors are determined in the following manner: The change in CPR is determined as a function of core flow along the rated power flow control line. Then, for a given scoop tube setpoint in the manual flow control operating mode, the MCPR at reduced flow is established that would give the Safety Limit MCPR if the core flow was increased to the scoop tube setpoint. The ratio of the MCPR at reduced flow to the operating limit MCPR is the K_f factor at that reduced flow.

INSERT 9

is employed except the MCPR at low flow is established such that the MCPR is equal to the operating limit MCPR at RATED THERMAL POWER and rated core flow.

INSERT 10

are equal to or greater than

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, TIPs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be highest value of the FLPD which exists in the core.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating ~~reload~~ cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

INSERT 1

1.10 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.13 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

INSERT 2

INSERT 1

The use of the correlation is not valid for all critical power calculations pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

SAFETY LIMITS

BASES

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

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The Safety Limit MCPR is determined using ~~the General Electric Thermal Analysis Basis, GETAB[a], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), (GEXL), correlation.~~

~~The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.~~

~~The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2.~~

~~The bases for the uncertainties in the core parameters are given in NEDO-20340[b] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958-A[a]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.~~

a. ~~"General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.~~

b. ~~General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Amendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.~~

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DELETE

~~Bases Table B2.1.2-2~~

~~NOMINAL VALUES OF PARAMETERS USED IN~~

~~THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT~~

~~THERMAL POWER~~	~~3323 MWt~~
~~Core Flow~~	~~108.5 Mlb/hr~~
~~Dome Pressure~~	~~1010.4 psig~~
~~Channel Flow Area~~	~~0.1089 ft^2~~
~~R-Factor~~	~~High enrichment - 1.043~~ ~~Medium enrichment - 1.039~~ ~~Low enrichment - 1.030~~

INSERT 5

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

INSERT 6

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-3 that are input to a GE-core dynamic behavior transient computer program. ~~The code used to evaluate pressurization events is described in NEDO-24154(3) and the program used in non-pressurization events is described in NEDO-10802(2). The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149(4). The principal result of this evaluation is the reduction in MCPR caused by the transient.~~

The purpose of the K_f factor specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The K_f factors may be applied to both manual and automatic flow control modes.

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The K_f factors values specified in the CORE OPERATING LIMITS REPORT were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.

INSERT 8

~~For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube set point and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.~~

MINIMUM CRITICAL POWER RATIO (Continued)

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For operation in the automatic flow control mode, the same procedure ~~was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and rated thermal flow.~~

The K_f factors specified in the CORE OPERATING LIMITS REPORT are conservative ~~for the General Electric plant operation~~ because the operating limit MCPRs of Specification 3.2.3 ~~is the same as~~ the original 1.20 operating limit MCPR used for the generic derivation of K_f.

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At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

2. ~~R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, NEDO-10802, February 1973.~~

INSERT 4

~~3. Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, October 1978.~~

~~4. TASC 01-A Computer Program for the Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.~~