HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. NPF-57 are affected by this change request:

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6.9.1.9	6-21

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than ~~1.10~~ 1.09 with two recirculation loop operation and shall not be less than ~~1.12~~ 1.11 with single recirculation loop operation, in both cases with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.*

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than ~~1.10~~ 1.09 with two recirculation loop operation or less than ~~1.12~~ 1.11 with single recirculation loop operation and in both cases with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

*Values applicable to Cycle 8 operation only.

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 ~~1.10~~ 1.09 for two recirculation loop operation and ~~1.12~~ 1.11 for single recirculation loop operation. MCPR greater than ~~1.10~~ 1.09 for two recirculation loop operation and ~~1.12~~ 1.11 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. hese conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the applicable NRC-approved critical power correlation is not valid for all critical power calculations performed at reduced 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 28×10^3 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 28×10^3 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.

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in NEDE-24011-P-A (the latest approved revision)*, General Electric Standard Application for Reactor Fuel (GESTAR II).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, via the Licensee Event Report System within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

*For Cycle ~~8~~ 9, as evaluated in the Safety Evaluation dated ~~11/4/97~~ to support License Amendment No. ~~107~~.

NON-PROPRIETARY VERSION OF ATTACHMENT 1 TO LCR H98-06

HOPE CREEK GENERATING STATION
REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

BASIS FOR REQUESTED CHANGE:

The changes proposed in this request implement an appropriately conservative Safety Limit Minimum Critical Power Ratio (SLMCPR) for the Hope Creek Cycle 9 core and fuel designs. These changes are required to address SLMCPR issues identified in a 10CFR21 notification made by General Electric on May 24, 1996 (Reference 1). That 10CFR21 notification discussed non-conservative SLMCPR calculation methodologies that impacted Hope Creek. As a result of the issues discussed in that 10CFR21 notification, Hope Creek issued Licensee Event Report (LER) 96-014-00, dated May 14, 1996 (which was supplemented by LER 96-014-01, dated September 30, 1996). As described in the corrective actions in that LER, Hope Creek conservatively controlled the minimum critical power ratio (MCPR) at a value that bounds initial accident conditions. On November 4, 1997, the NRC issued a Safety Evaluation Report (SER) for Hope Creek TS Amendment No. 107, which implemented a SLMCPR for the current Cycle 8 operation. The justification for the SLMCPR values proposed in this request is similar to that contained in the PSE&G submittals referenced in the November 4, 1997, SER.

REQUESTED CHANGE, PURPOSE AND BACKGROUND:

As shown in Attachment 3 of this letter, TS 2.1.2 is being modified to: 1) replace the 1.10 MCPR limit for two recirculation loop operation with a 1.09 MCPR limit (for Cycle 9); and 2) replace the 1.12 MCPR limit for single recirculation loop operation with a 1.11 MCPR limit (for Cycle 9). In addition, the Bases for TS 2.1, "Safety Limits", will be revised to reflect the new 1.09 MCPR limit for two recirculation loop operation and 1.11 MCPR limit for single recirculation loop operation. The bases changes are for information only and do not require NRC approval. An additional administrative change is also being made to TS 6.9.1.9 to reflect the new Cycle 9 SLMCPR.

In the course of calculating a cycle-specific SLMCPR for another utility, General Electric Company (General Electric) determined that the GESTAR II (General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-11[1], and U. S. Supplement NEDE-

[1] Revision 11 has since been superseded by Revision 13, dated August, 1996. The Revision 13 material pertinent to this application is unchanged from Revision 11. For purposes related to evaluation of this application, Revisions 11 and 13 may be considered equivalent and used interchangeably.

24011-P-A-11-US[1], November 17, 1995) generic SLMCPR may be non-conservative when applied to some core and fuel designs. The NRC was informed of this condition in a telephone call by General Electric on March 27, 1996, which became the subject of a 10 CFR Part 21 notification from General Electric dated May 24, 1996 (Reference 1).

When this issue was identified to Hope Creek, LER 96-014-00 was transmitted to the NRC to document this issue. Since that time, General Electric has calculated a revised plant-specific SLMCPR value for Hope Creek Cycle 7 and Cycle 8 as part of the Reload Licensing Analyses. The calculated SLMCPR values for Hope Creek Cycle 7 and Cycle 8 were based upon NRC approved methods *(General Electric Standard Application for Reactor Fuel,* NEDE-24011-P-A-11, and U. S. Supplement NEDE-24011-P-A-11-US, November 17, 1995), which have been discussed between General Electric and the NRC during meetings held on April 17, 1996 and May 6 through 10, 1996. The implementing procedures are identical to those used for similar recent analyses for other facilities and described in General Electric's proposed Amendment 25 to GESTAR II (R. J. Reda (GE) to T. E. Collins (NRC), *Proposed Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle Specific Safety Limit MCPR,* December 13, 1996). These procedures incorporate cycle specific parameters into the analysis which include: 1) the reference core loading; 2) conservative variations of projected control blade patterns; 3) the actual bundle parameters; and 4) the full cycle exposure range. This calculation resulted in the current Cycle 8 SLMCPR values of 1.10 for two loop operation and 1.12 for single loop operation. On November 4, 1997, the NRC issued an SER for the Hope Creek Cycle 8 SLMCPR values.

Subsequently, General Electric has performed analysis for the Hope Creek Cycle 9 core and fuel design. The method used to analyze Cycle 9 and determine the new SLMCPR values is provided in the following section. PSE&G proposes that the Hope Creek Technical Specifications be revised as indicated in Attachment 3 of this submittal to incorporate these new SLMCPR values for Cycle 9 operation.

JUSTIFICATION OF REQUESTED CHANGES:

The proposed changes contained in this submittal will revise the Technical Specifications to reflect the new SLMCPR values calculated by General Electric for Hope Creek. As stated previously, these plant specific evaluations were performed by General Electric for Hope Creek, Reload 8, Cycle 9 and were calculated using NRC approved methods.

Introduction

For Hope Creek, the Fuel Cladding Integrity Safety Limit is set such that no mechanistic 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 the onset of transition boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that the onset of transition 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. The new SLMCPRs for Cycle 9 at Hope Creek are 1.09 for two loop operation and 1.11 for single loop operation.

Control Rod Pattern Development for the Hope Creek Cycle 9 SLMCPR Analysis

Projected control blade patterns for the rodded burn through the cycle were used to deplete the core to the cycle exposures to be analyzed. At the desired cycle exposures the bundle exposure distributions and their associated R-factors were utilized for the SLMCPR cases to be analyzed. The use of different rod patterns to achieve the desired cycle exposure has been shown to have a negligible impact on the actual calculated SLMCPR. An estimated SLMCPR was obtained for an exposure point near beginning of cycle (BOC), middle of cycle (MOC), and the end of cycle (EOC) in order to establish which exposure points would produce the highest (most conservative) calculated SLMCPR.

The Safety Limit MCPR is analyzed with radial power distributions that maximize the number of bundles at or near the Operating Limit MCPR during rated power operation. This approach satisfies the stipulation in Reference 2 that the number of rods susceptible to boiling transition be maximized. General Electric has established criteria to determine if the control rod patterns and resulting radial power distributions are acceptable. These criteria were discussed with the NRC inspection team during May 6-10, 1996 meetings and have since been incorporated into the General Electric technical design procedures. These criteria include no gross violations of technical specification operating limits (e.g., MCPR, MAPLHGR, LHGR), criticality (calculated, normalized k_{eff} near one) and total number of bundles [[]] of the MCPR of the core.

Different rod patterns were analyzed until the criteria on the above parameters were met. The rod pattern search was narrowed by starting from a defined set of patterns known from prior experience to yield the flattest possible MCPR distributions. This was done for three exposure points in the cycle. A Monte Carlo analysis was then performed for the [[]] to establish the maximum SLMCPR for the cycle. The maximum SLMCPR occurred at the [[]].

Comparison of Hope Creek Cycle 9 SLMCPR value to the Generic GE9B Value

Table 1 summarizes the relevant input parameters and results of the SLMCPR determination for both the generic GE9B core and the Hope Creek Cycle 9 core. GESTAR II (Reference 3) specifies that the SLMCPR analysis for a new fuel design shall be performed for a large high power density plant assuming a bounding equilibrium core. The C-lattice GE9B product line generic SLMCPR (1.07) was determined according to this specification. Hope Creek Cycle 9 core is a C-lattice equilibrium core of GE9B fuel.

In general, the calculated safety limit is dominated by two key parameters: (1) flatness of the bundle pin-by-pin power/R-factor distributions; and (2) flatness of the core bundle-by-bundle MCPR distributions. Greater flatness in either parameter yields more rods susceptible to boiling transition and thus a higher calculated SLMCPR. Hope Creek has a bundle R-factor distribution more peaked than the generic GE9B equilibrium core. The core MCPR distributions for Hope Creek Cycle 9 is flatter than for the generic GE9B core.

The uncontrolled bundle pin-by-pin power distributions were compared between the Cycle 9 GE9B bundle, which dominates the contribution to fuel pins in boiling transition, and the GE9B bundle used in the generic SLMCPR analysis. For Hope Creek Cycle 9, the distribution of uncontrolled R-factors for the highest power rods in each bundle is not as flat as the bundle used in the generic analysis. For example, for Hope Creek Cycle 9, the bundles which contribute [[]] of the pins undergoing boiling transition (out of the 0.100% of all pins in the core in boiling transition) have [[]], while the generic GE9B bundle has [[]]. A difference of [[]] was selected as this roughly corresponds to 0.01 in MCPR for GE9B fuel.

By keeping the limiting bundles uncontrolled, it is assured that the flattest possible pin-by-pin R-factors are used in the SLMCPR calculation. By design, the R-factor distributions are optimized for their uncontrolled state, and control blade insertion causes the distributions to become more peaked. Therefore, the most conservative approach is to perform the SLMCPR calculation where the "base" rod pattern places all the potentially limiting bundles in an uncontrolled state. The Hope Creek Cycle 9 SLMCPR analysis has all of the bundles [[]]. The generic GE9B analysis [[]] of the core MCPR in an uncontrolled state [[]].

Hope Creek Cycle 9 has [[]] for the generic GE9B core. These bundles near the core MCPR are far more important to the determination of the SLMCPR than the bundles within [[]] of the core MCPR. The core MCPR distribution for Hope Creek Cycle 9 is thus seen to be considerably flatter than the distribution for the generic GE9B core. It is concluded that the greater flatness of the Hope Creek Cycle 9 core MCPR distribution is enough to overcome the greater flatness of the generic GE9B pin-by-pin R-factors and is the primary reason the calculated SLMCPR for the Hope Creek Cycle 9 core is 0.02 higher than the calculated SLMCPR for the generic GE9B equilibrium core. However, it should be pointed out that no specific sensitivity studies have been performed for Hope Creek to quantify the relationship between SLMCPR and flatness as described in terms of percent of bundles within a set delta CPR of the core MCPR or the number of pins within a set delta R-factor of the limiting R-factor within a bundle.

A specific single loop SLMCPR calculation was performed for Hope Creek Cycle 9. The calculation uses the same procedure described above for cycle specific dual loop calculation, except that it applies the larger uncertainties specified by reference 4 for single loop operation conditions. From the results of these calculations, it was determined that the single loop operation adder is 0.02 for Cycle 9 (for a single loop operation SLMCPR of 1.11). This is consistent with previous studies, which have linearly correlated the single loop operation adder to the dual loop SLMCPR.

Table 1: Comparison of Generic GE9B and Hope Creek Cycle 9 Cores

Quantity, Description	GE9B Generic	Hope Creek Cycle 9
Number of bundles in core	764	764*
Limiting cycle exposure point [[		
		]]
Calculated Safety Limit MCPR	1.07	1.09

* The Cycle 9 SLMCPR calculations are based upon a core consisting of 100% GE9B fuel: 196 fresh bundles (2.80% enriched), 236 once burnt bundles (176 at 3.27% enriched and 60 at 2.98% enriched), 232 twice burnt bundles (88 at 3.25% and 144 at 3.24% enriched), 72 thrice burnt bundles (24 at 3.25% and 48 at 3.24% enriched) and 28 bundles burned four cycles (all at 3.25% enriched).

CONCLUSIONS:

Based on all of the facts, observations and arguments presented above, Hope Creek concludes that the calculated dual loop SLMCPR value of 1.09 for the Hope Creek Cycle 9 core is justified and that this value, which is 0.02 higher than the 1.07 value calculated for the generic C-lattice GE9B equilibrium core, is appropriate. In addition, Hope Creek concludes that the calculated single locp SLMCPR adder value of 0.02 for the Hope Creek Cycle 9 core is also justified and that this value, which is 0.01 higher than the 0.01 value calculated for the generic C-lattice GE9B equilibrium core, is also appropriate.

REFERENCES:

1. General Electric letter to NRC, 10CFR Part 21, Reportable Condition, SLMCPR Evaluations, dated May 24, 1996.

2. Licensing Topical Report, *General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application*, NEDO-10958-A, January 1977.

3. *General Electric Standard Application for Reactor Fuel*, NEDE-24011-P-A-13-US, August 1996.

4. *General Electric Fuel Bundle Designs*, NEDE-31152P, Revision 6, April 1997.