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Company

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United States Nuclear Regulatory Commission
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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
RSCS ELIMINATION/RWM LOW POWER SETPOINT REDUCTION
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Gentlemen:

In accordance with 10CFR50.90, the Public Service Electric & Gas (PSE&G) Company requests a revision to the Technical Specifications (TSs) for the Hope Creek Generating Station. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed TS changes contained herein represent changes to the TS Index, the Bases for TS 2.2.1.2, "Average Power Range Monitor", TS 4.1.3.1.2, "Control Rod Operability", TS 3.1.3.6, "Control Rod Drive Coupling", TS 3.1.3.7, "Control Rod Position Indication", TS 3.1.4.1, "Rod Worth Minimizer", TS 3/4.1.4.2, "Rod Sequence Control System", TS 3/4.10.2, "Special Test Exceptions - Rod Sequence Control System", the Bases for TS 3/4.1.4, "Control Rod Program Controls", and the Bases for TS 3/4.10.2, "Rod Sequence Control System." The changes eliminate the Rod Sequence Control System (RSCS) Limiting Condition for Operation and Surveillance Requirements from the TSs and reduce the Rod Worth Minimizer (RWM) low power setpoint from 20% to 10%. Changes to other sections of the TSs are also proposed as necessary to delete reference to the RSCS from the TSs and to incorporate additional requirements necessary to support the elimination. Approval of these changes is requested to support implementation during the next refueling outage in September 1997.

The changes are consistent with those approved in a March 22, 1989 Safety Evaluation Report issued for the Limerick Generating Station and have been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c). A determination has been made that this request involves no significant hazards considerations. Changes similar to the proposed changes have also been incorporated into the improved standard TSs.

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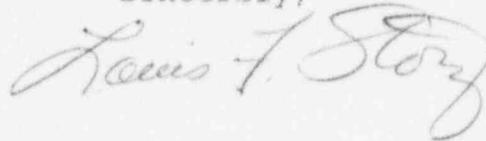
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The basis for the requested change is provided in Attachment 1. A 10CFR50.92 evaluation with a determination of no significant hazards consideration is provided in Attachment 2. The marked up TS pages affected by the proposed changes are provided in Attachment 3.

Upon NRC approval of this proposed change, PSE&G requests that the amendment be made effective on the date of issuance, but allow an implementation period of sixty days to provide sufficient time for associated administrative activities.

Should you have any questions regarding this request, we will be pleased to discuss them with you.

Sincerely,



Affidavit
Attachments (3)

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HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
CHANGE TO TECHNICAL SPECIFICATIONS

BASIS FOR REQUESTED CHANGE

The basis for the proposed changes are described in this attachment. The content includes a discussion of the requested changes and their purpose, relevant background information, the justification for the proposed changes, and a conclusion.

REQUESTED CHANGE AND PURPOSE:

The changes proposed by this amendment application delete references to and requirements associated with the Rod Sequence Control System (RSCS) from the following Technical Specifications (TSs):

TS Bases 2.2.1.2	Reactor Protection System Instrumentation Setpoints, Average Power Range Monitor
TS 4.1.3.1.2	Control Rod Operability
TS 3.1.3.6 (Action a)	Control Rod Drive Coupling
TS 3.1.3.7 (Action a)	Control Rod Position Indication
TS 3/4.1.4.2	Rod Sequence Control System
TS 3/4.10.2	Special Test Exception - Rod Sequence Control System
TS Bases 3/4.1.4	Control Rod Program Controls
TS Bases 3/4.10.2	Special Test Exception - Rod Sequence Control System

In order to assure that operation without active use of the RWM is minimized, TS 3.1.4.1, "Rod Worth Minimizer", is being revised to limit the extent of RWM inoperability.

In addition, the RWM minimum allowable low power setpoint (LPSP) is being reduced from 20% to 10% of rated thermal power.

BACKGROUND:

The NRC staff review and approval of the generic Technical Specification changes was reviewed for applicability to Hope Creek Generating Station. The methodologies described in the Hope Creek UFSAR for the Control Rod Drop Accident (RDA), the Rod Withdrawal Error (RWE) from the Startup Condition and the RWE at Power are directly referenced to the General Electric (GE) Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR-II)", and the accompanying US Supplement NEDE-24011-P-A-US. These methodologies are generic for all BWRs, and the RDA analysis is performed generically for each fuel product line which GE produces.

By letter dated August 15, 1986, the Boiling Water Reactors Owners Group (BWROG) submitted a proposed amendment to General Electric Topical Report NEDE-24011-P-A. Attached to the proposed amendment was a BWROG and General Electric (GE) report, "Modification to the Requirements for Control Rod Drop Accident Mitigating Systems" that justified the changes. The proposed changes included: 1) a request to eliminate the required use of the RSCS while retaining the RWM to provide backup to the operator for control rod pattern control and 2) a request to lower the RWM low power setpoint from 20% to 10%. The proposed changes were applied to GE's generic methodologies described in GESTAR-II and referenced in the Hope Creek UFSAR.

The NRC staff reviewed the referenced report and approved the generic TS changes in a letter from A. S. Thadani (NRC) to J. S. Charnley (GE) dated December 27, 1987. The NRC staff provided a Safety Evaluation Report (SER) with their approval which defined the basis for their acceptance and delineated the limitations under which licensees could reference the report in plant specific license applications. The NRC staff approval of the GESTAR-II amendment was issued in a subsequent revision of the document. PSE&G has reviewed the NRC staff safety evaluation for the GESTAR-II amendment, and has determined that it is applicable to Hope Creek.

The basis for acceptance of the changes, as documented in the NRC SER, is summarized as follows:

The SER Sections 2.0 and 3.0 indicate that, when the requirements for the RSCS were promulgated in the early 1970s, there were a number of "perceived" problems or unknown factors relating to the rod drop accident (RDA) that were of significance in the decision to require the RSCS. The first of these problems was the fact that new analyses showed that single errors in rod patterns could, in some cases, lead to rod worths for which an RDA could exceed limits. The second of the concerns was an RWM system that was, at the time, a poorly maintained, frequently bypassed system. Second operator substitution for the RWM was used routinely and was providing

minimal protection since procedures and quality control were frequently poor. The RWM, being a computer-related protection system, was also, at the time, an inherently distrusted system. The final factors involved in the decision were the fact that no trusted study of the probability of exceeding enthalpy limits as a result of an RDA existed and the fact that improved calculations providing more realistic modeling of an event were in only the early stages of development and use. As a result, nothing existed to alleviate concerns arising from the above perceptions.

Since the original decision was made, activities have been completed which alleviate the original concerns. Section 3.1 of the SER discusses a backfit study, completed in 1975, that concluded that no less than 13 separate events must occur in order for the peak fuel pellet enthalpy to exceed the NRC staff acceptance criteria of 280 calories/gram. The study calculated a conservative probability of $10E-12$ for these occurrences and their combinations, using the assumption that neither RSCS nor RWM were in use. Section 3.2 of the SER discusses continued studies of RDA methodology conducted by Brookhaven National Laboratories (BNL). The results of those studies have indicated a substantial reduction in enthalpy for a given rod worth. A conservative conclusion from the results is that there is a large likelihood that error patterns would not lead to a rod worth that could exceed limits in the zero power RDA. The study also clearly showed that, in the 10% power range and above, peak enthalpies would always be well below limits. Section 3.5 of the SER states that these analyses confirmed that the effects of an RDA are not significant above 10% Rated Thermal Power since the fuel enthalpies are well below the required limits even for significant error patterns and that the original Low Power Setpoint (LPSP) of 20% of Rated Thermal Power was required as an extreme bound for the analyses that were, at the time, uncertain.

Operating plants were not backfit with an RSCS but were required to implement TS changes to implement more mandatory use of the RWM. SER Section 3.3 indicates that the changes appear to have resulted in greatly improving RWM availability and use and in reducing the routine use of the second operator. Section 3.3 also indicates that computer controlled LCO surveillance and even protection systems have become recognized as being acceptable by the staff.

Section 3.3 of the SER provides requirements for adopting the generic TS changes. Specifically, TS changes to remove the RSCS should provide for minimizing operation without the active use of the RWM and should provide strong incentives for RWM maintenance and use without creating excessive operational restrictions. In addition, the occasional necessary use of a second operator RWM replacement should be strengthened by a utility review of relevant procedures, related forms, and quality control to assure that the second operator provides an effective and truly independent monitoring process. Hope Creek has addressed these requirements as discussed in the justification section below.

As a result of the above discussions and conclusions, on December 14, 1988, a license amendment application was submitted to the NRC for the Limerick Generating Station that requested TS changes to permit removal of the RSCS and to reduce the RWM LPSP to 10%. This request was subsequently approved by the NRC in a Safety Evaluation Report dated March 22, 1989. Justification used to support that change cited the August 15, 1986 submittal by the BWROG and GE. The primary justification for the change was that the RSCS is redundant to the RWM and is therefore not needed to mitigate the control rod drop accident (RDA). Further justification was provided which indicated that existing NRC sponsored RDA analysis methodology improvements showed significantly less severe peak fuel enthalpy values for a given dropped rod reactivity worth and that an existing NRC probability study demonstrated an extremely low probability for an RDA event exceeding fuel damage criteria.

JUSTIFICATION FOR REQUESTED CHANGES

The proposed changes to the HCGS TSs are justified based on the same conclusions contained in the December 27, 1987 NRC SER for the generic TS changes. In the SER, the NRC staff accepted the proposal to delete the RSCS TS requirements and to reduce the LPSP to 10%. However, the SER noted that the following additional specific requirements as being necessary and required that each be discussed in or provided with proposed TS changes, as appropriate:

ISSUE 1

Operation should be minimized without the active use of the RWM system by revising the RWM system Technical Specification to reduce second operator substitution. This requirement would provide a strong incentive for RWM maintenance without creating any excessive operational restrictions.

RESPONSE

The Technical Specification changes being proposed in this amendment application allow only one reactor startup per calendar year with the RWM unavailable prior to or during withdrawal of the first twelve control rods.

ISSUE 2

The relevant procedures, forms and quality control associated with the occasional use of the second operator substitution should be reviewed to strengthen the actions required when the RWM system is not available. This review would assure that the second operator provides an effective and truly independent monitoring process.

RESPONSE

Hope Creek Operations Procedure OP-SO.SF-0003(Q), "Rod Worth Minimizer Operation," requires the following actions:

- (a) Reactor Engineering must be notified prior to bypassing the RWM when continued rod motion is planned;
- (b) Rod pull listings must be available, updated, and ready for use whenever the RWM is bypassed;
- (c) If the RWM is bypassed and continued rod withdrawal is required, rod withdrawal must be performed in accordance with Reactor Engineering Procedure RE-FM.ZZ-0001(Q), "Guidelines for Control Rod Movement - Power Operations" or Reactor Engineering Procedure RE-FM.ZZ-0002(Q), "Guidelines for Control Rod Movement - Special Test and Operations"

Hope Creek Operations Procedure OP-AP.ZZ-0005(Q), "Department Operating Practices", defines a technically qualified member of the unit staff as a person who has completed one of the following:

- (a) Senior Reactor Operator License Training Program
- (b) Senior Reactor Operator Certification
- (c) Station Nuclear Engineering Training and meets the requirements of ANSI/ANS 3.1-1981.

Reactor Engineering Procedure RE-FM.ZZ-0001(Q) establishes guidelines for development and implementation of sequencing constraints governing movement of control rods during normal power operation from the all-rods in condition to the target pattern. This procedure requires the following actions:

- (a) A second licensed operator or technically qualified member of the unit technical staff must be present at the reactor control console to verify compliance with the rod pull listing when the RWM is bypassed during rod withdrawals below the RWM low power setpoint.
- (b) The second licensed operator or technically qualified member of the unit technical staff must initial the appropriate space on the rod pull listing next to the operator's initials.

In addition, procedures HC.RE-ST.ZZ-0009(Q) and HC.RE-SO.RJ-0008(Q) provide guidance for RWM sequence verification and program operation.

These procedures provide adequate controls to ensure that the second operator provides an effective and truly independent monitoring process.

ISSUE 3

The rod patterns used should be at least equivalent to Banked Pattern Withdrawal Systems (BPWS) or an improved version such as the Reduced Notch Worth Procedure. This would assure that potential rod worths are reduced.

RESPONSE

The Hope Creek Generating Station maintains procedures to ensure that rod patterns are in accordance with BPWS.

CONCLUSIONS:

Based on the above, PSE&G has concluded that the proposed changes to the Technical Specifications are justified. As described in Attachment 2 of this letter, PSE&G also concludes that this submittal does not involve any significant hazards considerations.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSES NPF-57
DOCKET NO. 50-354
CHANGE TO THE TECHNICAL SPECIFICATIONS
RSCS ELIMINATION/RWM LOW POWER SETPOINT REDUCTION

10CFR50.92 EVALUATION

Public Service Electric & Gas (PSE&G) has concluded that the proposed changes to the Hope Creek Generating Station Technical Specifications (TSS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGE

As indicated in Attachment 3, these proposed changes revise the Technical Specifications to eliminate the references to the Rod Sequence Control System (RSCS) and to reduce the RWM low power setpoint (LPSP) from 20% to 10%.

BASIS

The proposed changes to the HCGS Technical Specifications:

1. Do not involve a significant increase in the probability or consequences of an accident previously evaluated.

A. RSCS Deletion

The RSCS system restricts the pattern of control rods prior to a postulated control rod drop accident (RDA) so as to minimize the reactivity worth of the dropped rod. The RSCS provides no mitigation following the postulated RDA. The ability to restrict the pattern of control rods also allows the RSCS to be able to reduce the probability of a Continuous Rod Withdrawal During Reactor Startup, as described in the Hope Creek UFSAR Section 15.4.1.2 and Appendix 15B. However, to determine the consequence of such a rod withdrawal event, the RSCS is not credited, and the rod is assumed to be fully withdrawn from the core at its maximum rate. The RDA is therefore the only analyzed accident impacted by the proposed deletion of the RSCS system. Since the RSCS system plays no role in preventing a RDA, it therefore does not affect the probability of occurrence of this postulated accident.

As stated in an NRC Safety Evaluation Report dated December 27, 1987, the RSCS system is the result of requirements promulgated by the NRC staff in the early 1970's in response to unknowns and

perceived problems relating to the RDA. The GE calculational methodology being used at that time produced results showing that, even without pattern errors, calculated enthalpies for the RDA approached limiting values. In addition, the Rod Worth Minimizer (RWM) Technical Specifications were not effective in ensuring RWM availability and use, and the system was poorly maintained and frequently bypassed thus providing no significant protection. Second operator substitution for the RWM was used routinely and was providing minimal protection. Finally, no reliable study existed to address the probability of exceeding enthalpy limits as a result of an RDA.

Information associated with the above concerns has been significantly expanded or modified. Studies using improved methodologies have proven significantly lower peak fuel enthalpy values compared with methodologies in use when the RSCS was originally developed. In addition, a reliable probability study has been completed showing that the probability of an RDA exceeding NRC limits is very low. As a result, NRC review of the RSCS requirements has concluded that the RSCS system is not needed and operation without it is acceptable provided: 1) TSSs are modified to minimize the use of the second operator option, 2) procedures and quality control associated with the second operator option are reviewed to ensure that this option provides an effective and truly independent monitoring process; and 3) rod patterns used are at least equivalent to Banked Pattern Withdrawal System (BPWS) patterns. Each of these items has been addressed for the Hope Creek Generating Station.

As a result of the resolution of the original concerns associated with the RDA, the RWM system and limited use of the second operator option, when properly instituted, are now deemed to provide adequate protection to maintain the consequences of the RDA at an acceptable level. The remaining concerns regarding operation without the RSCS system and proper use of the second operator substitution option have been addressed for the Hope Creek Generating Station. We therefore conclude that the redundant RSCS system is no longer necessary and its deletion from the Technical Specifications will not significantly increase the probability or consequences of an RDA.

B. RWM Setpoint Reduction

The RWM system restricts the pattern of control rods prior to a postulated control rod drop accident (RDA) so as to minimize the reactivity worth of the dropped rod. The RWM provides no mitigation following the postulated RDA. The ability to restrict the pattern of control rods also allows the RWM to be able to reduce the probability of a Continuous Rod Withdrawal During Reactor Startup, as described in the Hope Creek UFSAR Section 15.4.1.2 and Appendix 15B. However, to determine the consequence

of such a rod withdrawal event, the RWM is not credited, and the rod is assumed to be fully withdrawn from the core at its maximum rate. The RDA is therefore the only analyzed accident impacted by the proposed reduction in the RWM setpoint. Since the RWM system plays no role in preventing a RDA, it therefore does not affect the probability of occurrence of this postulated accident.

Existing calculations have demonstrated that no significant RDA can occur above 10% power. Calculations by both General Electric and the Brookhaven National Laboratory indicate that, even with significant error patterns, peak fuel enthalpy is reduced well below required limits at 10% power. The 20% limit was originally required as an extreme bound because of the then existing uncertainties in the analyses. Based on the current analyses, the 10% level is now acceptable and deemed to provide adequate protection to maintain the consequences of an RDA at an acceptable level. Changing the RWM setpoint from 20% to 10% will therefore not significantly increase the consequences of any previously analyzed accident.

2. Do not create the possibility of a new or different kind of accident from any accident previously evaluated.

A. RSCS Deletion

Operation of the RSCS cannot cause or prevent an accident; this system functions to minimize the consequences of an RDA. The Bank Position Withdrawal Sequence (BPWS) will still be used to ensure that rod pull pattern are constrained to those assumed in the RDA. The RSCS has no impact on the operation of any other system, and therefore its deletion will not contribute to a malfunction in any other equipment nor create the possibility of a new or different accident from any accident previously evaluated.

B. RWM Setpoint Reduction

Operation of the RWM cannot cause or prevent an accident; this system functions to minimize the consequences of an RDA. The RWM has no impact on the operation of any other system, and therefore changing its setpoint from 20% to 10% will not contribute to a malfunction in any other equipment nor create the possibility of a new or different accident from any accident previously evaluated.

3. Do not involve a significant reduction in a margin of safety.

A. RSCS Deletion

When the original decisions were made regarding the need for the RSCS system, numerous perceived problems in the RDA analysis

existed. As noted in the discussion of the consequences of previously analyzed accidents in Item 1 above: 1) the perceived RDA problems have been resolved; 2) reviews of the RDA have concluded that the RSCS is not needed to mitigate the consequences of an RDA; and 3) operation without the RSCS is acceptable. The RWM and limited use of second operator substitution, when properly instituted, are now deemed adequate to ensure that peak fuel enthalpies remain below NRC limits. Therefore, the deletion of the redundant RSCS system will not significantly decrease any margin of safety.

B. RWM Setpoint Reduction

The Bases for the HCGS TSS state that when thermal power is greater than 20%, there is no possible rod worth that, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 calories per gram. Existing calculations demonstrate that the RDA is not a significant concern above 10% power, and therefore, a mitigation system is not needed for higher power level operation. Calculations by both General Electric and the Brookhaven National Laboratory indicate that, even with significant error patterns, peak fuel enthalpy is reduced well below required limits (280 calories per gram) at 10% power. The 20% limit was originally required as an extreme bound because of the then existing uncertainties in the analyses. Based on the current analyses, the 10% level is now acceptable and deemed to provide adequate assurance that the peak fuel enthalpy will remain below the NRC limits during a postulated RDA. Changing the RWM setpoint from 20% to 10% will therefore not significantly reduce any margin of safety.

CONCLUSION

Based on the above, PSE&G has determined that the proposed changes do not involve a significant hazards consideration.

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FACILITY OPERATING LICENSES 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:

<u>Technical Specification</u>	<u>Page</u>
Index 3/4.1.4	v
Index 3/4.10.2	xv
Index B 3/4.10.2	xxi
Bases 2.2.1.2	B 2-6
4.1.3.1.2	3/4 1-4
3.1.3.6	3/4 1-11
3.1.3.7	3/4 1-13
3.1.4.1	3/4 1-16
3/4.1.4.2	3/4 1-17
3/4.10.2	3/4 10-2
Bases 3/4.1.4	B 3/4 1-3
	B 3/4 1-5
Bases 3/4.10.2	B 3/4 10-1