GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION

Facility Operating
License No. DPR-16

Technical Specification Change Request
Request No. 218
Docket No. 50-219

Applicant submits, by this Technical Specification Change Request No. 218 to the Oyster Creek Nuclear Generating Station Operating License, changes to Sections 3.1 and 4.1.

By J J Barton
J. J. Barton
Vice President and Director
Oyster Creek

Sworn and Subscribed to before me this 12th day of May 1994.

Judith M. Crowe
A Notary Public of NJ

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

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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. 218, for Oyster Creek Nuclear Generating Station Technical Specifications, filed with the U.S. Nuclear Regulatory Commission on May 12, 1994, has this day of May 12 1994, been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the United States mail, addressed as follows:

The Honorable Theodore J. Hutler
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

By J J Barton
J. J. Barton
Vice President and Director
Oyster Creek

OYSTER CREEK NUCLEAR GENERATING STATION
FACILITY OPERATING LICENSE NO. DPR-16
DOCKET NO. 50-219
TECHNICAL SPECIFICATION CHANGE REQUEST NO. 218

Applicant hereby requests the Commission to change Appendix A to the above captioned license as below, and pursuant to 10 CFR 50.92, an evaluation concerning the determination of no significant hazards considerations is also presented:

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

SECTIONS TO BE CHANGED

Sections 3.1 and 4.1

EXTENT OF CHANGE

Sections 3.1 and 4.1 are to be replaced in their entirety due to the number of changes, including editorial, and for pagination purposes. The technical changes regarding proposed allowed out-of-service times (AOT) and surveillance test intervals (STI) are contained in the reformatted Tables 3.1.1 and 4.1.1, respectively. The changes are indicated by vertical marks in the right-hand margin.

SUPPORTING INFORMATION AND NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

I. INTRODUCTION

The proposed changes reflect Standard Technical Specification (STS) revisions contained in General Electric Company (GE) Licensing Topical Reports (LTR) which, based upon reliability analyses, support increases in surveillance test intervals (STI) and allowed out-of-service times (AOT) for surveillance and repair. These changes are beneficial in reducing: i) potential unnecessary plant scrams, ii) excessive equipment test cycles, and iii) the diversion of personnel and resources on unnecessary testing. The NRC staff has reviewed and approved these LTRs in Safety Evaluation Reports (SER). Technical changes regarding Channel Calibration requirements for APRM Scram, High Drywell Pressure (for Core Cooling) and Turbine Trip Scram instrumentation are also proposed. Editorial changes are proposed to correct and clarify Technical Specifications.

II. DESCRIPTION OF THE PROPOSED CHANGES

The Technical Specification requirements regarding STIs (Table 4.1.1) and AOTs (Table 3.1.1) are to be revised for Reactor Protection System (RPS), Emergency Core Cooling System (ECCS) Actuation, Isolation Actuation and Control Rod Block Instrumentation in accordance with the following GE LTRs:

- NEDC-30851P-A "Technical Specification Improvement Analyses for BWR Reactor Protection System" dated March 1988
- NEDC-30936P-A (Parts 1 and 2) "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)" dated December 1988

- NEDC-30851P-A (Supplement 2) "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation" dated March 1989

- NEDC-31677P-A "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation" dated July 1990

- NEDC-30851P-A (Supplement 1) "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation" dated October 1988

- GENE-770-06-1-A "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications" dated December 1992.

The NRC staff SERs documenting the review and approval of these LTRs have been incorporated into the LTRs. In addition, proposed AOT changes are based on the most recent approved BWR Owners' Group letters which were used in establishing AOTs for the new STS. The requested changes are as follows (editorial changes are discussed separately below):

EDITORIAL CHANGES

1. Capitalization of Definitions - Throughout Sections 3.1 and 4.1, the definitions, where they appear in Specifications, bases, tables and table notes, are capitalized to highlight the fact they are terms with specific meanings. This is consistent with STS convention. The definitions are contained in TS Section 1.0.

2. Table 3.1.1 Heading - The table column which currently reads Min. No. of Instrument Channels Per Operable Trip Systems is incorrect in that the word Systems should not be plural. A separate column specifies the required number of Trip Systems. The subject column specifies the number of instrument channels for each Trip System and, therefore, the word System in this column is properly singular.

3. Table 3.1.1, Sections A.7, B.6 and L.1 - Minor grammatical changes are made to ensure this instrumentation (High Radiation in Main Steamline Tunnel) is consistently identified where it appears in its functions of scram (Section A.7) and isolation (Sections B.6 and L.1). Since this instrumentation is common to both scram and isolation functions, it needs to be consistently displayed to preclude confusion when new note oo to Table 3.1.1 is applied.

4. Table 3.1.1, Section C - The word <u>plant</u> is deleted from the Action Required since it is superfluous. This allows the Action Required to refer directly to the TS definition for PLACE IN COLD SHUTDOWN CONDITION. Also, in order to clarify the function achieved, the heading is revised to Isolation Condenser <u>Initiation</u>.

5. Table 3.1.1, Section C.2 - The word <u>level</u> is added to the C.2 description of the Isolation Condenser Initiation variable on Low-Low Reactor Water <u>Level</u>. This is for clarification purposes and makes this description consistent with other reactor water level instrument descriptions in the table.

6. Table 3.1.1, Section D - The description of the function in this section should be clarified to indicate that instrumentation under D initiates Core Spray. Therefore, the function description is revised to read Core Spray Initiation.

7. Table 3.1.1, Section J.1 - The requirement for operability in the Startup mode was inadvertently omitted in License Amendment 72. The table is revised to restore that omission.

8. Table 3.1.1, Section J.4, Notes and Associated Footnote - Note gg is deleted since it concerns a 1985 licensing condition which is no longer in effect. This note expired at the end of the Cycle 10M outage. Oyster Creek is currently operating in Cycle 14.

9. Table 3.1.1, Note t - A typographical error in this note currently allows sensors to be "...operable or bypassed..." when the systems they are associated with are inoperable. The note should correctly read "...inoperable or bypassed..." since this note would be unnecessary to allow operability. This clarifies that Core Spray instrumentation operability is not required if its associated and supported Core Spray System is inoperable.

10. Table 3.1.1, Note y - All references to note y in Table 3.1.1 were deleted by License Amendment 75. This note is no longer applicable and should be deleted.

11. Table 3.1.1, Sections M.1-3 and N.a and b and Table Notes - Note kk is added to indicate that the allowable out-of-service time for surveillance of the Diesel Generator Load Sequence Timers and Loss of Power instrumentation does not change and remains two hours. The GE LTRs do not address this instrumentation. In addition, Section M.3 is revised to clarify that the load sequence timer is associated with the Reactor Building Closed Cooling Water Pumps.

12. Section 4.1, Specification - A minor editorial change to the Specification revises "...as per definitions..." to "...using the definitions... ." This change does not alter the meaning or intent of the Specification and is purely grammatical.

13. Table 4.1.1, NOTE 2 - This note refers to "...Section 2.3 Specifications (1) (a) and (2) (a)... ." These specifications are not currently identified in this fashion. The correct identification of these specifications is "...A.1 and A.2..." and NOTE 2 is revised accordingly. This change is a correction and ensures consistency between NOTE 2 and Section 2.3.

14. Table 4.1.1, Instrument Channels 18, 20 and 25 - The surveillance interval is currently displayed as "1/20" which means once every 20 months. This is revised to 1/20 mo to clarify that the interval is 20 months and to ensure consistency with the nomenclature used in Table 4.1.1.

15. Table 4.1.1, Instrument Channels 23, 24 and 26 - The surveillance interval is currently indicated as "Every 3 months." Since the nomenclature used in Table 4.1.1 is 1/3 mo for this interval, the

surveillance frequency description for the subject instrument channels is revised to conform. This change does not alter the interval and is editorial.

16. Table 4.1.1, Instrument Channels 28a and 28b - The Channel Check interval for these instrument channels is specified as "daily." The nomenclature indicated in the table legend shows "1/d" as the description of this interval. To ensure consistency, the Channel Check interval for the subject instrument channels is revised to conform with the legend. This change does not alter the interval and is editorial.

17. Table 4.1.1, NOTES - The table notes are moved from the first page of the table to a more appropriate location at the end of the table.

18. Table 4.1.1, Legend - The table legend establishes nomenclature used in Table 4.1.1 to specify surveillance intervals. Revisions are required to ensure the legend contains definitions for intervals which appear in the table. Currently, Table 4.1.1 contains the conventions 1/mo, 1/20 mo (as corrected by editorial change No. 14 above) and 1/24 mo. However, the table legend does not currently address them. The following definitions are added to the legend: <u>1/mo = Once per month</u>, <u>1/20 mo = Once every 20 months</u>, and <u>1/24 mo = Once every 24 months</u>. These definitions were inadvertently omitted in previous license amendments and their inclusion now serves to correct that omission. In addition, the table legend defining the once every 18 month interval should be deleted since this interval no longer appears in Table 4.1.1.

19. Sections 3.1 and 4.1, Amendment Nos. on Pages - As part of the pagination process for this change request, the previous license amendment numbers displayed on the bottom of each page (where applicable since not all pages have amendment numbers) have been revised to ensure that the numbers accurately reflect associated changes to the pages.

<u>NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION FOR EDITORIAL CHANGES</u>

The above nineteen proposed changes are editorial in nature and are typical of example I.c.2.e.i in 51FR7744. Therefore, they do not:

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

The editorial changes described above do not change the design or operation of any structure, system or component relied upon to prevent or mitigate the consequences of any accident evaluated. These editorial changes also do not add new structures, systems or components which may have an effect on existing elements of the facility. The changes proposed correct, clarify and/or retain existing requirements.

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

Since neither physical changes to the facility nor changes in its operation are involved in the proposed editorial changes to the Technical Specifications, there is no possibility for creation of a new or different kind of accident.

(3) Involve a significant reduction in the margin of safety.

Facility configuration and operation are unaffected by the proposed editorial changes. As a result no changes in margin of safety occur.

The editorial changes described and evaluated above are purely administrative to achieve consistency or correct an error in the Technical Specifications.

TECHNICAL CHANGES (LTR SPECIFIC)

NEDC-30851P-A dated March 1988/MDE-98-0485,Rev. 1 dated July 1985

1. Table 3.1.1, Section A.2-A.12 and Table Notes - Note nn is added which provides allowable out-of-service times for repair for the specified scram parameters. Note nn is clarified in accordance with BWROG letter 92102 from C. L. Tully to B. K. Grimes (NRC), "BWR Owners' Group (BWROG) Topical Reports on Technical Specification Improvement Analysis for BWR Reactor Protection Systems - Use for Relay and Solid State Plants (NEDC-30884 and NEDC-30851P)," dated November 4, 1992. Note nn will apply to the following scram parameters:

   a. High Reactor Pressure - Parameter 2
   b. High Drywell Pressure - Parameter 3
   c. Low Reactor Water Level - Parameter 4
   d. High Water Level in Scram Discharge Volume - Parameter 5.a & 5.b
   e. Low Condenser Vacuum - Parameter 6
   f. High Radiation in Main Steamline Tunnel - Parameter 7
   g. Average Power Range Monitor (APRM) - Parameter 8
   h. Intermediate Range Monitor (IRM) - Parameter 9
   i. Main Steamline Isolation Valve Closure - Parameter 10
   j. Turbine Trip Scram - Parameter 11
   h. Generator Load Rejection Scram - Parameter 12

2. Table 3.1.1, Note c - Note c is revised to allow any one APRM to be removed from service for up to six hours for surveillance without tripping its trip system.

3. Table 4.1.1, Instrument Channel Nos. 1 (Scram Function), 2, 3, 5.b, 11 (APRM Scram Trips) and 13.a - The Channel Test interval is revised to quarterly from weekly or monthly for the scram instrumentation identified below:

   a. High Reactor Pressure - Instrument Channel 1
   b. High Drywell Pressure (Scram) - Instrument Channel 2
   c. Low Reactor Water Level - Instrument Channel 3
   d. High Water Level in Scram Discharge Volume - Instrument Channel 5.b (analog)
   e. APRM Scram Trips - Instrument Channel 11
   f. High Radiation in Main Steamline - Instrument Channel 13.a

4. Table 4.1.1, Instrument Channel No. 30 and NOTE 1 - This is a new requirement added to ensure that the automatic scram contactors are

tested on a weekly basis. The test of the automatic scram contactors using the subchannel test switches does not have to be performed at each weekly interval if the automatic scram contactors are tested by other means, i.e. by performance of a different required Channel Calibration or Test (e.g., Low Reactor Water Level). The current NOTE 1 is deleted as discussed below in Item 12 and replaced with the note concerning the weekly test of the automatic scram contactors.

<u>NEDC-30936P-A, (Parts 1 and 2) dated December 1988/RE-004 dated January 1987</u>

5. Table 3.1.1, Sections C.1-2, D.1-3, G.1-3 and Table Notes - Note pp is added providing an allowable out-of-service time for repair for the specified parameters as identified below:

   a. <u>ISOLATION CONDENSER INITIATION (Section C)</u>

      1) High Reactor Pressure - Parameter 1
      2) Low-Low Reactor Water Level - Parameter 2

   b. <u>CORE SPRAY INITIATION (Section D)</u>

      1) Low-Low Reactor Water Level - Parameter 1
      2) High Drywell Pressure - Parameter 2
      3) Low Reactor Pressure (valve permissive) - Parameter 3

   c. <u>AUTOMATIC DEPRESSURIZATION (Section G)</u>

      1) High Drywell Pressure - Parameter 1
      2) Low-Low-Low Reactor Water Level - Parameter 2
      3) AC Voltage - Parameter 3

6. Table 4.1.1, Instrument Channel Nos. 1 (Isolation Condenser Actuation Function), 4 (Isolation Condenser Actuation and Core Spray Actuation Functions), 6 and 9 (Core Spray Actuation and Automatic Depressurization Actuation Functions) - The Channel Test interval is revised to quarterly from monthly for the ECCS Actuation instrumentation indicated below:

   a. High Reactor Pressure - Instrument Channel 1
   b. Low-Low Water Level - Instrument Channel 4
   c. Low-Low-Low Water Level - Instrument Channel 6
   d. High Drywell Pressure (Core Cooling) - Instrument Channel 9

<u>NEDC-30851P-A, Supplement 2 dated March 1989/NEDC-31677P-A dated July 1990</u>

7. Table 3.1.1, Sections B.1-6, F.1-2, H.1-2 and L and Table Notes - Note oo is added which provides allowable out-of-service times for repair for the specified isolation actuation parameters. Note oo is clarified in accordance with GE letter OG90-579-32A from W. P. Sullivan and J. F. Klapproth (GE) to M. L. Wohl (NRC), "Implementation Enhancements to Technical Specification Changes Given in Isolation Actuation Instrumentation Analysis," dated June 25, 1990. Note oo will apply to following isolation actuation instrumentation:

   a. <u>REACTOR ISOLATION (Section B)</u>

1) Low-Low Reactor Water Level - Parameter 1
2) High Flow in Main Steamline A - Parameter 2
3) High Flow in Main Steamline B - Parameter 3
4) High Temperature in Main Steamline Tunnel - Parameter 4
5) Low Pressure in Main Steamline - Parameter 5
6) High Radiation in Main Steamline Tunnel - Parameter 6

b. PRIMARY CONTAINMENT ISOLATION (Section F)

1) High Drywell Pressure - Parameter 1
2) Low-Low Reactor Water Level - Parameter 2

c. ISOLATION CONDENSER ISOLATION (Section H)

1) High Flow Steam Line - Parameter 1
2) High Flow Condensate Line - Parameter 2

d. CONDENSER VACUUM PUMP ISOLATION (Section L)

1) High Radiation in Main Steamline Tunnel - Parameter 1

8. Table 4.1.1, Instrument Channel Nos. 2 (Primary Containment Isolation Function Common to Scram), 4 (Reactor Isolation and Primary Containment Isolation Functions Common to ECCS), 7, 8, 13.a (Reactor Isolation and Condenser Vacuum Pump Isolation Functions Common to Scram), 14 and 15 - The Channel Test interval is revised to quarterly from weekly or monthly for the following isolation actuation instrumentation:

a. High Drywell Pressure (Scram) - Instrument Channel 2
b. Low-Low Water Level - Instrument Channel 4
c. High Flow in Main Steamline - Instrument Channel 7
d. Low Pressure in Main Steamline - Instrument Channel 8
e. High Radiation in Main Steamline - Instrument Channel 13.a
f. High Radiation in Reactor Building - Instrument Channel 14
g. High Radiation on Air Ejector Off-Gas - Instrument Channel 15

NEDC-30851P-A, Supplement 1 dated October 1988/GENE-770-06-1-A dated December 1992

9. Table 4.1.1, Instrument Channel No. 12 - The Channel Test frequency is revised to quarterly from monthly for the Control Rod Block instrumentation below:

a. APRM Rod Blocks - Instrument Channel 12

TECHNICAL CHANGES (GLOBAL)

10. Sections 3.1 and 4.1, Bases - The bases for the two TS Instrumentation sections are revised to reflect the GE LTRs as the basis for AOT and STI changes and include them as references.

11. Table 3.1.1, Global Note * - This note, which appears at the end of the table and just before the indicated Notes section, describes the Action Required column and provides an allowed out-of-service time for performing surveillance. The note is revised to allow six hours instead

of two hours for surveillance of instrument channels provided at least one operable channel in the same trip system is monitoring the parameter. The six hour allowance is supported by the LTRs for all instrumentation in Table 3.1.1 except the Diesel Generator Load Sequence Timers and Loss of Power instruments. Note kk (see Editorial Change No. 10) was established for this instrumentation and retains the two hour allowance.

12. Table 4.1.1, NOTE 1 and Figure 4.1.1 - Current NOTE 1 and Figure 4.1.1 are deleted since surveillance intervals will be based on the reliability analyses contained in the GE LTRs and not on the methodology described in the current TS bases using Figure 4.1.1. The methodology currently allows Channel Test intervals for specific instruments to be adjusted according to the failure rate of the larger population of similar instrumentation. This methodology, as described in the bases, is no longer applicable because the number and type of instruments has changed while the methodology was not revised. Figure 4.1.1 would allow surveillance Channel Test intervals to be adjusted between monthly and quarterly. Since Oyster Creek has not used this method of adjusting Channel Test intervals, Tests have been performed on a monthly basis.

ADDITIONAL TECHNICAL CHANGES

13. Table 4.1.1, Instrument Channel 11 (APRM Scram Trips) - The Channel Calibration interval for the APRM Scram Trips is currently weekly. This is identical to the current Channel Test interval for this instrumentation. The changes proposed in this license amendment request and supported by the LTRs will revise the Channel Test interval to quarterly. There will be no benefit resulting from the change in the Channel Test interval if the Channel Calibration interval remains the same. Therefore, as a result of a plant-specific review of a change to a quarterly Channel Calibration requirement for this instrumentation, a revision to quarterly is requested. Instrument setpoint drift performance has been evaluated and found acceptable for the revised interval.

14. Table 4.1.1, Instrument Channels 9 (High Drywell Pressure for Core Cooling) and 23 (Turbine Trip Scram) - A quarterly Channel Calibration interval is added for instrument channel 9. Currently, a Channel Calibration interval is not specified for the High Drywell Pressure instruments. Since they are calibrated on a quarterly interval, it is appropriate to include this surveillance requirement in the Technical Specifications. The quarterly Channel Calibration interval for the High Drywell Pressure instruments is consistent with the interval currently specified for a majority of the instruments in the table. In addition, there is no Channel Calibration interval specified for instrument channel 23 (Turbine Trip Scram). This is appropriate since this trip parameter senses turbine stop valve position via limit switches which are fixed in position and adjusted during valve maintenance. This trip parameter and its switch adjustment methods are similar to Main Steamline Isolation Valve Scram instrumentation (Instrument Channel 10) for which the Technical Specifications require only a Channel Test. The Turbine Trip Scram Channel Calibration column is revised to indicate not applicable (N/A).

III. JUSTIFICATION FOR THE PROPOSED CHANGES

GPU Nuclear (GPUN) has determined that the generic analyses performed by GE for the BWR Owners' Group to revise AOTs and STIs for RPS, ECCS Actuation, Isolation Actuation and Rod Block instrumentation are applicable to the Oyster Creek Nuclear Generating Station. GPUN has completed plant-specific evaluations required by the NRC SERs which approved the LTRs for use by individual facilities. The following discussion provides the information required by the NRC staff for a plant-specific submittal. As stated in the SERs, three issues must be addressed to apply the RPS LTR (NEDC-30851P-A) and two issues must be addressed to apply the other LTRs to an individual facility when specific Technical Specifications are considered for revision. These issues are addressed as follows:

Issue 1. Confirm the applicability of the generic analyses to the specific facility. (This issue applies to all LTRs)

<u>RESPONSE</u>

a. Licensing Topical Report NEDC-30851P-A, Appendix L identifies GPUN's Oyster Creek (OC), a BWR-2 plant, as a participant in the BWR Owners' Group Technical Specification Improvement Program which managed the development of this LTR. Section 7.4, "Conclusions of Plant Specific Applications," states that:

   "The evaluation found various differences between the RPS configuration of various plants and the generic plant. These differences include HFA relays, four scram contactors for BWR/2, sensor differences, scram parameter differences, and SDV sensor diversity differences. The assessment of these differences shows that while the HFA relays and the four scram contactors for BWR/2 would result in a higher overall RPS failure frequency, the improved technical specification intervals and allowable out-of-service times based on the generic plant would result in a net improvement of plant safety for plants with such differences. The effect of other differences on the RPS failure frequency is insignificant. Therefore, the generic results can be applied to plants in the BWROG Technical Specification Improvement Program"

   Enclosed with this Technical Specification Change Request is the plant-specific evaluation report MDE-98-0485 prepared by GE titled "Technical Specification Improvement Analysis for the Reactor Protection System for Oyster Creek Nuclear Generating Station" which concludes in Section 4 that "the generic analysis in Reference 1 (NEDC-30851P-A) is applicable to OC." GPUN has reviewed NEDC-30851P-A and MDE-98-0485 and verified that the generic analysis is applicable to OC. For a discussion of the differences between the RPS at OC and the generic plant analyzed in NEDC-30851P-A, see the response to Issue 3 below.

b. Licensing Topical Report NEDC-30936P-A, Appendix N (Part 1) and Appendix B (Part 2) identifies GPUN as a participant in the BWR ECCS Actuation Instrumentation evaluation. Section 5.3 (Part 2) specifically analyzes BWR-2 plants. Enclosed with this Technical Specification Change Request is the plant-specific evaluation performed by GE contained in report RE-004 titled "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Oyster Creek Nuclear Generating Station." GPUN has reviewed the LTR and the plant-specific report and verified that the generic analysis is applicable to OC. The review identified two minor discrepancies in the plant-specific report which have no impact upon its conclusions. These discrepancies are both contained in report RE-004, Section II.E. They are as follows:

1) The Reactor Pressure Low instrumentation relating to Core Spray System actuation at OC is currently tested quarterly versus monthly as indicated in the table. This has no impact on the analysis and no change is being requested for this STI.

2) The Core Spray Pump Discharge Pressure input to the ADS actuation logic needs clarification. At OC this is performed by pump differential pressure instrumentation which is functionally equivalent to pump discharge pressure. Our evaluation has verified that the generic analysis remains applicable to OC.

c. Licensing Topical Report NEDC-30851P-A, Supplement 2, Appendix A identifies GPUN as a participant in the BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation evaluation. Section 3.4 analyzes the BWR-2 plants. GPUN has reviewed this LTR and verified that the generic analysis is applicable to OC.

d. Licensing Topical Report NEDC-31677P-A, Appendix E identifies GPUN as a participant in the BWR Isolation Actuation Instrumentation evaluation. Section 5.4 and Appendix C3 analyzes BWR-2 plants. GPUN has reviewed this LTR and verified that the generic analysis is applicable to OC.

e. Licensing Topical Report NEDC-30851P-A, Supplement 1, Appendix B identifies GPUN as a participant in the BWR Control Rod Block Instrumentation evaluation. Section 4.0 addresses the BWR-2 plants. GPUN has reviewed the LTR and verified that the generic analysis is applicable to OC.

f. Licensing Topical Report GENE-770-06-1-A identifies application of changes to STIs and AOTs for Selected Instrumentation Technical Specifications to all BWR plants. GPUN has reviewed the LTR and verified that it is applicable to OC. This LTR was found applicable only to the limited extent that it identified AOTs for Control Rod Block Instrumentation which NEDC-30851P-A, Supplement 1 did not explicitly address.

Issue 2 **Demonstrate, by use of current drift information provided by the equipment vendor or plant-specific data, that the drift characteristics for instrumentation used in RPS, ECCS, Isolation and Rod Block instrument channels in the plant are bounded by the assumptions used in the LTRs when the functional test interval is extended from weekly or monthly to quarterly.** (This issue applies to all LTRs)

RESPONSE

Since the LTRs do not contain quantitative instrument drift assumptions, the BWR Owners' Group and NRC staff concluded that additional clarification to address this issue was appropriate. As a result, the NRC staff provided additional guidance in a letter from C. E. Rossi (NRC) to R. F. Janecek (BWROG) dated April 27, 1988 which states that:

> "...licensees need only confirm that the setpoint drift which could be expected under the extended STIs has been studied and either (1) has been shown to remain within the existing allowance in the RPS and ESFAS instrument setpoint calculation or (2) that the allowance and setpoint have been adjusted to account for the additional expected drift."

Setpoint drift is monitored during Channel Calibration when setpoints are required to be verified, not during performance of Channel Tests. If the Channel Calibration interval is shorter than the proposed quarterly Channel Test, the Channel Calibration interval will require a change to quarterly as well, since a Channel Calibration includes the same test elements as a Channel Test. There would be no benefit in revising the Channel Test interval without a commensurate change in the Channel Calibration interval. The change in Channel Calibration interval would then require consideration of the effects on setpoint drift. The Channel Calibration intervals for OC instrumentation addressed by the LTRs are all equal to or greater than once per quarter except for the calibration of the APRM Scram Trips (Table 4.1.1, Instrument Channel No. 11). The calibration interval given for these instrument channels is weekly. In this case, GPUN has evaluated the impact of revising the Channel Calibration to quarterly and concluded that setpoint drift for the longer interval is acceptable.

Although the Channel Calibration interval for all other instrumentation included in this Technical Specification Change Request is quarterly or longer, GPUN has been verifying setpoints for some additional instrumentation during the monthly Channel Tests. These additional instruments have also been evaluated to determine any impact of a longer interval between setpoint verifications. The results of the evaluation indicate that it is acceptable to revise these setpoint verification intervals as well, although no change is needed for the Technical Specification Channel Calibration interval.

Issue 3 Confirm that the differences between the parts of the RPS that perform the trip functions in the plant and those of the base case plant were included in the plant-specific analysis done using the procedures of Appendix K of NEDC-30851P or present plant-specific analyses to demonstrate no appreciable change in RPS availability or public risk. (This issue applies only to NEDC-30851P-A)

RESPONSE

Enclosed as part of this Technical Specification Change Request, GE Report MDE-98-0485, Revision 1, "Technical Specification Improvement Analysis for the Reactor Protection System for Oyster Creek Nuclear Generating Station" provides a plant-specific evaluation to determine whether the generic study contained in LTR NEDC-30851P-A is applicable to OC to revise STIs and add AOTs in Technical Specifications for the RPS instrumentation. This report utilizes the procedures in Appendix K of NEDC-30851P-A to identify and evaluate the differences between the parts of the RPS that perform the trip functions at OC and those analyzed in the generic study. The report was completed in 1985 and does not take into consideration changes made to RPS instrumentation since then. It does comprise the primary assessment of the differences in the RPS between OC and the generic plant, therefore it is being submitted with the recognition that further explanation is necessary to correct discrepancies contained therein. GPUN performed an evaluation of NEDC-30851P-A and MDE-98-0485 to verify that the generic analysis is applicable to OC and concluded that it remains applicable with discrepancies addressed as follows:

1) MDE-98-0485, Appendix A, Section II, Part B.1 - This section lists the RPS sensors and identifies type, total number and number per RPS channel. The number given for main steam isolation valve (MSIV) position is 4 total and 1/RPS channel. The correct values are 8 total and 2/RPS channel. Oyster Creek has two separate limit switches which initiate scram at 10% closure for each MSIV. This brings Oyster Creek more in line with the generic model since the generic model contains four instrument channels per trip system as shown in NEDC-30851P-A, Table 7.4. With two sensors per channel and two channels per trip system, the number of sensors per trip system for MSIV position at OC is four.

2) MDE-98-0485, Section 3, Item k; Appendix A, Section II, Part B.1 and Section III, Part B.1 - The plant-specific report indicates bi-stable switches are used to monitor reactor pressure and reactor water level. Subsequent to the preparation of the report, these sensors were replaced with analog trip units and transmitters. Since the generic model assumed use of analog trip units/transmitters there is no longer a difference between the generic model and Oyster Creek.

3) MDE-98-0485, Section 3, Items e and i; Appendix A, Section II, Parts B.1, B.8, C.2, C.3, and G.8 (Functional Unit 14 in Table) and Section III, Parts B.12 and C.2 - The low scram air header pressure switches have been removed from the scram logic. This scram was not considered in the generic model, therefore, no difference exists between the generic model and Oyster Creek.

4) MDE-98-0485, Appendix A, Section II, Part B.8 lists available RPS trip bypasses. While MDE-98-0485 indicates the Scram Discharge Volume High Water Level Trip Bypass is not available at Oyster Creek, this is not the case. Oyster Creek does have this bypass which is consistent with the generic model as described in NEDC-30851P-A, Appendix E, Part B.6. OC conforms to the generic model in this respect.

5) MDE-98-0485, Appendix A, Section II, Part B.8 indicates an available RPS trip bypass on Low Air Header Pressure. As discussed in 3 above in this section, the low scram air header pressure scram feature has been removed. The bypass for this feature has also been removed. This is consistent with the generic model as no RPS trip bypass is indicated for this feature in NEDC-30851P-A, Appendix E.

6) MDE-98-0485, Appendix A, Section II, Parts C.3, D.1 and F.1 and Section III, Part D.1 - Oyster Creek uses GE Type CR205 automatic scram contactors while one GE Type CR205 and one GE Type CR305 are used for the manual scram contactors. The generic model used GE Type CR105 for automatic and manual scram contactors. This is a newly-identified plant-specific difference between OC and the generic model. There is no significant reliability effect because the common cause failure rate of GE Type CR105, CR205 and CR305 contactors are the same and the use of diverse contactor models tends to improve reliability.

7) MDE-98-0485, Section 3, Item m; Appendix A, Section II, Part G.2 and Section III, Part G.1 - Calibration frequency requirements for analog trip units were incorporated into the Technical Specifications after preparation of the plant-specific report. The quarterly calibration interval for analog trip units and annual calibration interval for transmitters is within the range of the sensitivity study documented in NEDC-30851P-A.

8) MDE-98-0485, Section 3, Item m; Appendix A, Section II, Parts G.4. and G.6 - Although Oyster Creek Technical Specifications do not provide specific AOTs for inoperable instrument channels or trip systems, the bases in Technical Specification Section 3.1 indicate that prompt action is taken to trip the channel or trip system to compensate for the inoperable condition when it involves one trip system. This is done immediately. However, when both trip systems are involved the Action Required is initiated immediately since tripping both trip systems will initiate a trip which is undesirable.

9) MDE-98-0485, Section 3, Item 3; Appendix A, Section II, Part G.7 and Section III, Part G.1 - Technical Specifications currently allow a channel to be inoperable for up to 2 hours for required surveillance without placing the trip system in the tripped condition. This had been changed from a one hour allowance, is consistent with the old GE STS (NUREG 0123) and is the nominal original AOT for surveillance considered in NEDC-30851P-A.

10) MDE-98-0485, Section 3, Item d; Appendix A, Section II, Parts B.1, B.6, C.3 and G.8 (Functional Unit 9 in Table) and Section III, Part B.9 - In addition to Type 2 switches, Oyster Creek currently utilizes an analog transmitter loop for the Scram Discharge Volume (SDV) High Water Level (Type 1) sensor. The addition of a Type 1 SDV water level sensor (analog transmitter) brings Oyster Creek in line with the generic model establishing SDV level sensor diversity, i.e., transmitters and switches (Type 2). There is no longer a difference between OC and the generic model. The analog channels are functionally tested monthly and calibrated quarterly with test current and yearly with test pressure. The number of channels per trip system is two.

11) MDE-98-0485, Appendix A, Section II, Part H.1 - Flux sensors, radiation sensors, and analog sensors are generally not included in the Channel Tests at Oyster Creek as instrument loop test switches provide the means for logic testing. These differences are

consistent with the Technical Specification definition for Channel Test and have no effect with respect to the generic model since they are within the range of the sensitivity study performed in NEDC-30851P-A. These are newly-identified differences which were not included in the plant-specific report.

12) MDE-98-0485, Appendix A, Section II, Parts H.2 and H.3 - When an individual sensor channel is in repair, the logic channel is tripped, and if the individual sensor channel is in test, the sensor is temporarily inoperable but the logic channel is not necessarily tripped. These conditions are permitted by current Technical Specifications and have no effect when comparing OC to the generic model since they are within the range of the sensitivity study performed in NEDC-30851P-A.

13) Oyster Creek utilizes a High Recirculation Flow Scram which is not identified in MDE-98-0485. Two recirculation flow converters, one for each RPS trip system initiate a scram when recirculation flow exceeds the trip setpoint. This parameter does not serve as a scram sensor for any of the more severe initiating events as defined in NEDC-30851P-A and consequently the addition of another scram initiation signal will have no significant impact on RPS failure frequency. These sensors utilize the APRM sensor HFA relays. A recent calculation has shown that due to recirculation pump flow limitations, the high recirculation flow setpoint cannot physically be achieved. As a result, GPUN is considering a request, which would be forwarded under separate cover in the future, to remove this parameter as a scram initiator. Elimination of this scram parameter would have no effect on OC relationship to the generic model.

14) MDE-98-0485, Section 3, Item p; Appendix A, Section II, Part H.4 and Section III, Part G.2 - The number of scram contactor actuations currently experienced during Channel Tests at Oyster Creek differs from those assumed in the generic model and identified in the plant-specific report. The differences are described as follows:

    a) APRM Channel Test results in 6 actuations per scram contactor in each automatic trip logic channel.
    b) MSIV Closure Channel Test causes 2 actuations per scram contactor in each automatic trip logic channel.
    c) Turbine Control Valve fast closure Channel Test consists of 2 actuations per scram contactor in each automatic trip logic channel since the <40% power Turbine Trip Scram Bypass switches are also tested during this surveillance.
    d) The High Recirculation Flow Scram test is performed quarterly and consists of 2 actuations per scram contactor in each automatic trip logic channel.
    e) IRM Front Panel test is performed weekly and initiates 2 actuations per scram contactor in each automatic trip logic channel whenever the reactor is not in the RUN MODE.

The above differences have no significant effect since they remain within the range of the sensitivity study performed in NEDC-30851P-A. The total number of automatic scram contactor actuations is estimated to be approximately 406 actuations per contactor per year. This exceeds the estimate in NEDC-30851P-A of 272 (+\-65) actuations per year but has no significant impact on RPS reliability since the variation in the number of scram contactor actuations has been assessed to be an insignificant contributor to wear out.

15) MDE-98-0485, Appendix A, Section III, Part F.2 - The backup scram valves are de-energized to trip. There is no significant effect on RPS reliability because the operation of the backup scram valves is controlled by the scram contactors.

The differences between the RPS at Oyster Creek and the generic model, described in the plant-specific evaluation (MDE-98-0485) as updated above, are bounded by the analysis contained in the RPS LTR (NEDC-30851P-A). GPUN concludes that the generic analysis remains applicable to Oyster Creek. General Electric Company has reviewed GPUN's evaluation and confirms the generic analysis is applicable.

IV. CONCLUSION

As discussed in Section III above, GPU Nuclear has addressed the three issues for the RPS LTR (NEDC-30851P-A) and the two issues for the other GE LTRs which the NRC staff in the associated SERs has indicated are necessary to implement the generic Technical Specification changes identified in the LTRs on a plant-specific basis. The first issue, which applies to all LTRs, required confirmation that the generic analyses apply to Oyster Creek. The two required plant-specific reports regarding RPS and ECCS instrumentation for Oyster Creek concluded that the generic analyses are applicable to Oyster Creek. These reports (MDE-98-0485 and RE-004) are included with this change request. The information they provide addresses the differences between OC and the generic analyses and, when applied with the conclusions contained in NEDC-30851P-A, NEDC-30936P-A and the update information given in response to Issues 1 and 2 in Section III above, justifies the proposed changes. GPUN has reviewed the LTRs and the plant-specific reports and verified the generic analyses are applicable to Oyster Creek. The second issue required demonstrating that setpoint drift will not be a concern when the Channel Test interval is increased to quarterly. For instruments with Channel Calibration intervals shorter than the proposed quarterly Channel Tests, extending the Channel Test interval would require consideration of the effects on setpoint drift since setpoint verification is required only during Channel Calibrations. This is the case with only one instrument. The APRM Scram STI for both Channel Calibration and Test has been evaluated and found acceptable to be revised to quarterly. In addition, instruments whose setpoints have been verified on a monthly schedule, have also been evaluated to ensure setpoint drift can be accommodated for a longer interval. The third issue applies only to RPS LTR NEDC-30851P-A and requires confirmation that differences between the trip functions in the base case plant and OC are bounded by the LTR analyses. GE report MDE-98-0485 titled "Technical Specification Improvement Analysis for the Reactor Protection System for Oyster Creek Nuclear Generating Station" documents differences and their impact. GPUN has provided an update to this report in the discussion regarding this issue in Section III above. GPUN's evaluation concluded that the differences and their impact do not significantly affect the Technical Specification improvement provided by LTR NEDC-30851P-A and the RPS configuration at OC is bounded by the generic analysis.

Approved clarifications used in the development of NUREG 1433 "Standard Technical Specifications, General Electric Plants, BWR/4" were utilized in the proposed allowable out-of-service time notes in Table 3.1.1 to ensure a "loss of function" does not occur.

Additional technical changes are proposed to include a quarterly Channel Calibration requirement for the High Drywell Pressure for Core Cooling instruments and a clarification to indicate that a Channel Calibration is not applicable for Turbine Trip Scram instruments in Table 4.1.1 where there currently are no calibration requirements shown for these instruments.

Finally, editorial changes are proposed which include 1) corrections to ensure consistent use of nomenclature, 2) correction of typographical errors, 3) reformatting of the instrumentation tables and, 4) deletion of a note which is no longer applicable relating to a 1985 licensing action.

V. NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION OF TECHNICAL CHANGES

1. <u>The operation of the Oyster Creek Nuclear Generating Station, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

The generic analysis contained in LTR NEDC-30851P-A assessed the impact of changing RPS STIs and adding AOTs on RPS failure frequency, scram frequency and equipment cycling. Specifically, Section 5.7.4, "Significant Hazards Assessment," of NEDC-30851P-A states that:

> "Fewer challenges to the safeguards system, due to less frequent testing of the RPS, conservatively results in a decrease of approximately one percent in core damage frequency. This decrease is based upon the following:
>
> - Based on the plant-specific experience presented in Appendix J, the estimated reduction in scram frequency (0.3 scrams/yr) represents a 1 to 2 percent decrease in core damage frequency based on the BWR plant-specific Probabilistic Risk Assessments (PRAs) listed in Table 5-8.
> - The increase in core damage frequency due to less frequent testing is less than one percent. This increase is even lower (less than 0.01 percent) when the changes resulting from the implementation of the Anticipated Transients Without Scram (ATWS) rule are considered. Therefore, this increase is more than offset by the decrease in CDF due to fewer scrams.
> - The effect of reducing unnecessary cycles on RPS equipment, although not easily quantifiable also results in a decrease in core damage frequency.
> - The overall impact on core damage frequency of the changes in allowable out-of-service times is negligible."

The BWR Owners' Group concluded that the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated since the increase in probability of a scram failure due to RPS unavailability is insignificant. The overall probability of an accident is decreased as the time RPS logic operates as designed is increased resulting in less inadvertent scrams during testing and repair. The plant-specific evaluation performed by GPUN and GE demonstrates that while the Oyster Creek RPS differs from the generic model analyzed in the RPS LTR (NEDC-30851P-A), the net effect of the differences do not alter the generic conclusions. The AOTs proposed for RPS instrumentation are based on improved wording developed for use in NUREG 1433, "Standard Technical Specifications, General Electric Plants, BWR/4," which ensures a loss of function does not occur. In addition, the change to the APRM Scram Channel Calibration surveillance interval from weekly to quarterly has been evaluated by GPUN to determine the effect on setpoint drift. The results of the evaluation show acceptable performance of this scram parameter ensuring that the safety analysis remains valid. The clarification that a Channel Calibration is not applicable to Turbine Trip Scram instrumentation is appropriate since this trip parameter senses turbine stop valve position via limit switches which are fixed in position and adjusted, as necessary, during valve maintenance. This trip parameter and its switch adjustment methods are similar to the Main Steamline Isolation Valve Scram for which the Technical Specifications require only a Channel Test.

LTR NEDC-30936P-A (Parts 1 and 2) contains an assessment of the impact of changing STIs and AOTs for BWR ECCS Actuation Instrumentation. Section 4.0, "Technical Assessment of Changes," of NEDC-30936P-A (Part 2) states that:

> "The results indicate an insignificant (less than 5E-7 per year) increase in water injection function failure frequency when STIs are increased from 31 days to 92 days, AOTs for repair of the ECCS actuation instrumentation are increased from one hour to 24 hours, and AOTs for surveillance testing are increased from two to six hours. For all four BWR models the increase represents less than 4% increase in failure frequency. However, when other factors which influence the overall plant safety are considered, the net result is judged to be an improvement in plant safety."

From this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated since the increase in probability of a water injection failure due to ECCS instrumentation unavailability is insignificant and the net result is judged to be an improvement in plant safety.
The plant-specific evaluation performed by GPUN and GE demonstrates that while the Oyster Creek ECCS differs from the generic model analyzed in LTR NEDC-30936P-A, the net effect of the plant-specific differences do not alter the generic conclusions. The addition of a quarterly Channel Calibration STI for the High Drywell Pressure ECCS initiation parameter is consistent with the calibration interval requirement for other similar instrumentation at Oyster Creek and ensures the regular performance of calibrations. This is a new requirement not currently contained in the Technical Specifications and experience performing the High Drywell Pressure (Core Cooling) instrument calibration at a quarterly interval has proven adequate for instrument performance monitoring.

LTRs NEDC-30851P-A, Supplement 2 and NEDC-31677P-A contain generic analyses assessing the impact of changing the STIs and AOTs for BWR Isolation Actuation

Instrumentation which are common or not common to RPS and ECCS instrumentation. Section 4.0, "Summary of Results", of NEDC-30851P-A, Supplement 2 states that:

> "The results indicate that the effects on probability of failure to initiate isolation are very small and the effects on probability or frequency of failure to isolate are negligible in nearly every case. In addition, the results indicate that increasing the AOT to 24 hours for tests and repairs has a negligible effect on the probability of failure of the isolation function. These combined with changes to the testing intervals and allowed out-of-service times for RPS and ECCS instrumentation provide a net improvement to plant safety and operations."

and Section 5.6, "Assessment of Net Effect of Changes," of NEDC-31677P-A states that:

> "A reduction in core damage frequency (CDF) of at least as much as estimated in the ECCS instrumentation analysis can be expected when the isolation actuation instrumentation STIs are changed from one month to three months. The chief contributor to this reduction is the channel functional tests for the MSIVs. Inadvertent closure of the MSIVs will cause an unnecessary plant scram. This reduction in CDF more than compensates for any small incremental increase (10% or 1.0E-07/year) in calculated isolation function failure frequency when the STI is extended to three months."

Based on this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the consequences of an accident previously evaluated since the increase in probability of an isolation failure due to isolation instrumentation unavailability is insignificant. The proposed wording of the AOTs is based on the clarifications used in the development of NUREG 1433, "Standard Technical Specifications, General Electric Plants, BWR/4," which ensures a loss of function does not occur where applied to isolation actuation instrumentation.

LTR NEDC-30851P-A, Supplement 1 contains a generic analysis assessing the impact of changing Control Rod Block STIs on Rod Block failure frequency. Section 5 (Brookhaven National Laboratory Technical Evaluation Report - Attachment 2 to the NRC SER) of NEDC-30851P-A, Supplement 1 states that:

> "The BWROG proposed changes to the Technical Specifications concerning the test requirements for BWR control rod block instrumentation. The changes consist of increasing the surveillance test intervals from one to three months. These test interval extensions are consistent with the already approved changes to STIs for the reactor protection system. The technical analysis reviewed and verified as documented herein indicates that there will be no significant changes in the availability of the control rod block function if these changes are implemented. In addition, there will be a negligible impact on the plant core melt frequency due to the decreased testing."

Bases contained in GE Topical Report GENE-770-06-1-A assessed the impact of changing STIs and AOTs on failure frequency for selected systems. Section 2.0, "Summary," of GENE-770-06-1-A states that:

"Technical bases are provided for selected proposed changes to the instrumentation STIs and AOTs that were identified in the BWROG Improved BWR Technical Specification activity. These STI and AOT changes are consistent with approved changes to the RPS, ECCS, and isolation actuation instrumentation. These proposed changes do not result in a degradation to overall plant safety."

The BWR Owners' Group concluded from the generic analysis in NEDC-30851P-A, Supplement 1 and the bases in GENE-770-06-1-A that the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated. GPUN's utilization of GENE-770-06-1-A is limited to the identified AOTs for Control Rod Block instrumentation analyzed in NEDC-30851P-A since the Control Rod Block LTR did not explicitly address AOTs.

2. <u>The operation of Oyster Creek Nuclear Generating Station, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

The addition of allowable out-of-service times (AOTs) consistent with wording developed for use in Improved Standard Technical Specifications to ensure no loss of function and the revision of surveillance test intervals (STIs) does not alter the function of RPS, ECCS, Isolation or Rod Block instrumentation nor involve any type of plant modification. No new modes of plant operation are involved with the changes.

Adding a quarterly Channel Calibration STI for High Drywell Pressure instrumentation (for Core Cooling) establishes a requirement in the Technical Specifications which is not currently incorporated. This is an additional requirement beyond that already in place for this instrumentation and will not alter its operation since by their nature STIs ensure proper instrument performance. The clarification that a Channel Calibration is not applicable to Turbine Trip Scram instrumentation is appropriate since this trip parameter senses turbine stop valve position via limit switches which are fixed in position and adjusted during valve maintenance. This trip parameter and its switch adjustment methods are similar to the Main Steamline Isolation Valve Scram for which the Technical Specifications require only a Channel Test. Revising the Channel Calibration STI for APRM Scram instruments from weekly to quarterly allows these instruments to benefit from the Channel Test STI change provided by the generic analysis in the RPS LTR. The benefits include a significant reduction in the number of half-scram states the plant will undergo reducing the potential for inadvertent plant trips. The effect of setpoint drift over the longer interval has been evaluated and found acceptable.

The proposed changes will not alter the physical characteristics of any plant systems or components and all safety-related systems and components remain within their applicable design limits. Thus, system and component performance is not adversely affected by these changes, thereby assuring that the design capabilities of those systems and components are not challenged in a manner not previously assessed so as to create the possibility of a new or different kind of accident.

3. <u>The operation of the Oyster Creek Nuclear Generating Station, in accordance with the proposed amendment, will not involve a significant</u>

reduction in a margin of safety.

The NRC staff has reviewed and approved the generic studies contained in the GE Licensing Topical Reports and has concurred with the BWR Owners' Group that the proposed changes do not significantly affect the availability of RPS, ECCS Actuation, Isolation Actuation and Control Rod Block instrumentation. The proposed addition of allowable out-of-service times for instruments addressed by the LTRs provides reasonable times for making repairs and performing tests. The lack of sufficient out-of-service time provided in current Technical Specifications can create an urgency during repairs and tests which could cause an increased risk of error. In addition, placing an individual channel or trip system in a tripped condition because no AOT exists, as in current Technical Specifications, increases the potential for an inadvertent scram or equipment actuation. The proposed AOTs provide realistic times to complete required actions without increasing overall instrument failure frequency and ensure that no loss of function occurs, therefore, there is no significant reduction in the margin of safety.

The LTRs demonstrate that extending surveillance test intervals does not result in significant changes in the probability of instrument failure. Where Channel Calibration frequency has not changed, assurance exists that setpoints will not be affected by drift. In the case of the APRM Scram Channel Calibration, the proposed change to quarterly from weekly has been evaluated and found acceptable. Expected instrument performance over the extended interval will assure that applicable safety analyses will continue to be met. In addition, other instrumentation was evaluated for drift effects of setpoints and was found acceptable. The addition of a quarterly Channel Calibration interval for High Drywell Pressure (for Core Cooling) is consistent with Channel Calibration STIs for most other instrumentation at Oyster Creek and has been the interval used to achieve an adequate level of instrument performance monitoring. The clarification that a Channel Calibration is not applicable to Turbine Trip Scram instrumentation ensures consistency in the establishment of surveillance requirements. This trip parameter senses turbine stop valve position via limit switches which are fixed in position and adjusted during valve maintenance. This trip parameter and its switch adjustment methods are similar to the Main Steamline Isolation Valve Scram for which the Technical Specifications require only a Channel Test. These proposed changes, when coupled with the reduced probability of test-induced plant transients and equipment failures, do not result in a reduction in the margin of safety.

General Electric Company Reports
MDE-98-0485 and RE-004
Technical Specification Improvement Analyses for the
Reactor Protection System and ECCS Actuation Instrumentation
Oyster Creek Nuclear Generating Station

NOTE: Portions of the Reactor Protection System plant-specific report (MDE-98-0485) are proprietary to General Electric Company and should be withheld from public disclosure pursuant to 10 CFR 2.790 and not reproduced.

General Electric Company

AFFIDAVIT

I, **Robert C. Mitchell,** being duly sworn, depose and state as follows:

(1) I am Project Manager, Safety Evaluation Programs, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report MDE-98-0485, Rev. 1, *Technical Specification Improvement Analysis for the Reactor Protection System for Oyster Creek Nuclear Generating Station,* dated July 1985. This information is delineated by brackets around the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs 4.b and 4.d, above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it would provide other parties, including competitors, with a valuable interpretive information regarding the application of reliability based methodology to BWR instrumentation. A substantial effort has been expended by General Electric to develop this information in support of the BWR Owners' Group Technical Specifications Improvement Program.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing the expertise to determine and apply the appropriate evaluation process.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

Robert C. Mitchell, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 8th day of APRIL 1994.

Robert C. Mitchell

Robert C. Mitchell
General Electric Company

Subscribed and sworn before me this 8th day of April 1994

Mary L. Kendall

Notary Public, State of California

MARY L. KENDALL
COMM. # 987864
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires MAR 26, 1997