CP&L

Carolina Power & Light Company
PO Box 10429
Southport NC 28461

ENCLOSURE 6
Exempt From Public Disclosure
Per 10 CFR 2.790

John Paul Cowan
Director - Site Operations
Brunswick Nuclear Plant
910 457-2869

OCT 28 1994

SERIAL: BSEP 94-0413
10 CFR 50.90
TSC 94TSB04

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 & 50-324/LICENSE NOS. DPR-71 & DPR-62
REQUEST FOR LICENSE AMENDMENTS
INCREASED INSTRUMENT SURVEILLANCE TEST INTERVALS AND ALLOWABLE OUT-OF-SERVICE TIMES

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light Company hereby requests amendments to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed amendments would increase the surveillance test intervals (STIs) and allowable out-service times (AOTs) for selected instrumentation addressed in Section 3/4.3 of the Technical Specifications. The proposed STI and AOT changes are in accordance with NRC approved General Electric Company Licensing Topical Reports (LTRs) and NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4". The proposed Technical Specification changes permit specified channel functional tests to be conducted quarterly rather than weekly or monthly. This amendment enhances operational safety by reducing 1) the potential for inadvertent plant scrams, 2) excessive test cycles on equipment, and 3) the diversion of plant personnel and resources on unnecessary testing.

Enclosure 1 provides a detailed description of the proposed changes and the basis for the changes.

Enclosure 2 details the basis for the Company's determination that the proposed changes do not involve a significant hazards consideration.

Enclosure 3 provides an environmental evaluation which demonstrates that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment needs to be prepared in connection with the issuance of the amendment.

A001 1/1 Change: NRC PDR Ltr 1 Encl w/att Prop

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ENCLOSURE 6

Exempt From Public Disclosure
Per 10 CFR 2.790

Enclosures 4 and 5 provide the marked-up Technical Specifications for Units 1 and 2 respectively. Typed Technical Specification pages will be submitted at a later date.

Enclosures 6, 7, and 8 contain plant-specific evaluations by the General Electric Company (GE) in support of the proposed changes. Some of the information contained in the Enclosure 6 report is considered by GE to be proprietary and an affidavit to that effect is included with the enclosure. Pursuant to 10 CFR 2.790, it is requested that the proprietary information contained in that report be withheld from public disclosure.

Pursuant to 10 CFR 50.91(b)(1), Carolina Power & Light Company is providing a copy of this license amendment to Mr. Dayne H. Brown of the State of North Carolina.

Upon approval, CP&L requests that the NRC staff issue the License Amendments with an effective date no later than 90 days from the date of issuance. This latitude permits appropriate procedural/program revisions and training necessary to implement the proposed changes.

CP&L considers this amendment to represent a significant Cost-Beneficial Licensing Action (CBLA). Savings from this amendment are expected to be in excess of $130,000 per cycle ($2.6 million over the life of the plant assuming 10 additional cycles per unit); therefore, CP&L requests that appropriate review priority be provided for these amendments. CP&L further requests that the amendments be issued no later than the Spring, 1995 Unit 1 refuel outage (B110R1), currently scheduled to begin April 1, 1995.

Please refer any questions regarding this submittal to Mr. R. P. Lopriore at (910) 457-2212.

Sincerely,

John Paul Cowan

John Paul Cowan

JDF/kah

Enclosures

John Paul Cowan, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Gwendolyn R. Whitlatch
Notary (Seal)

My commission expires: August 12, 1996

cc: Mr. D. H. Brown, State of North Carolina
Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. P. D. Milano, NRR senior Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick Senior Resident Inspector
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1 AND 2
NRC DOCKETS 50-325 & 50-324
OPERATING LICENSES DPR-71 & DPR-62
REQUEST FOR LICENSE AMENDMENT
INCREASED INSTRUMENT SURVEILLANCE TEST INTERVALS AND
ALLOWABLE OUT-OF-SERVICE TIMES

BASIS FOR CHANGE REQUEST

I. INTRODUCTION

The proposed Technical Specification changes contained herein represent revisions to Technical Specification Section 3/4.3 - Instrumentation (and the associated Bases) to increase the surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for specified instrumentation. The proposed changes will permit specified channel functional tests to be conducted quarterly rather than weekly or monthly. AOTs for repairs will be increased from 1 hour to 12 or 24 hours, and AOTs for required surveillance tests will be increased from 2 hours to 6 hours. The proposed changes also include revisions to the wording of the Action Statements which implement the AOTs. The revised wording clarifies the required actions and precludes extended, uncompensated "loss-of-function" situations during instrument repair and surveillance testing. Proposed STI and AOT changes are in accordance with General Electric Company (GE) Licensing Topical Reports (LTRs) which have been reviewed and approved by the NRC staff. The proposed revisions to the Action Statements provide compensatory actions consistent with those set forth in supplemental letters from the BWR Owners' Group (BWROG) to the NRC and incorporated in the new Standard Technical Specifications (NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4").

The changes proposed by this Request for License Amendment are the result of an extensive effort between the BWROG, GE, and the NRC on determining proper STIs and AOTs for various instrumentation Technical Specifications. The proposed changes will enhance operational safety by: (1) reducing the potential for inadvertent plant scrams, (2) reducing excessive test cycles on equipment, (3) reducing occupational exposure, and (4) reducing the diversion of plant personnel and resources on unnecessary testing. This amendment will not authorize any change in the types of effluents or in the authorized power level of the facility.

This enclosure provides a detailed description of the proposed changes and the basis for the changes. Enclosure 2 details the basis for CP&L's determination that the proposed changes do not involve a significant hazards consideration, pursuant to 10 CFR 50.92. Enclosure 3 provides an environmental evaluation which demonstrates that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment needs to be prepared in connection with the issuance of the amendment. Specific changes proposed by this amendment are set forth in the marked-up Technical Specification pages provided in Enclosure 4 (Unit 1) and Enclosure 5 (Unit 2). Although not a formal part of the Technical

Specifications (as described in 10 CFR 50.36), Bases Section changes are provided in Enclosures 4 and 5. Typed Technical Specifications will be submitted at a later date. Enclosures 6, 7 and 8 contain supporting information for the proposed changes.

II. DESCRIPTION OF PROPOSED CHANGES

The proposed changes revise the Reactor Protection System (RPS), Isolation Actuation, ECCS Actuation, Control Rod Withdrawal Block, Control Room Emergency Ventilation System, ATWS-RPT, EOC-RPT, and RCIC Actuation Instrumentation Technical Specification (TS) requirements regarding the surveillance test intervals (STIs) and allowable out-of-service times (AOTs) in accordance with the following approved GE LTRs:

- NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

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

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

- NEDC-30936P-A, Part 1, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)," December 1988.

- NEDC-30936P-A, Part 2, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)," December 1988.

- NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," 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," December 1992.

- GENE-770-06-2P-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

The NRC staff SERs, documenting the review and approval of these LTRs, have been incorporated into the LTRs.

Subsequent to the issuance of the SERs for the topical reports, the NRC staff questioned whether, in some cases, the example Technical Specifications provided in the topical reports would allow plant operation for an extended period of time with a loss of automatic trip

capability (i.e., loss-of-function). This issue was particularly important with respect to RPS operability. Therefore, the BWROG worked closely with the NRC staff to rewrite the RPS Action Statements in a manner that addressed this "loss-of-function" issue. The revised RPS actions were documented in the following supplemental letter from the BWROG to the NRC staff. These RPS actions, as well as similar actions to address loss-of-function for other instrumentation, were incorporated in the new Standard Technical Specifications (NUREG-1433):

•BWROG-92102, letter from C. L. Tully (BWROG) 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)," November 4, 1992.

Additional enhancements, related to the implementation of AOTs for Isolation and ECCS instrumentation, were provided in the following supplemental letters and were incorporated in NUREG-1433:

•OG90-579-32A, letter from W. P. Sullivan and J. F. Klapproth (GE) to Millard L. Wohl (NRC), "Implementation Enhancements To Technical Specifications on Changes Given In Isolation Actuation Instrumentation Analysis", June 25, 1990.

•OG90-319-32D, letter from W. P. Sullivan and J. F. Klapproth (GE) to Millard L. Wohl (NRC), "Clarification of Technical Specification Changes Given in ECCS Actuation Instrumentation Analysis", March 22, 1990.

This amendment request includes proposed changes to the Actions that, except for editorial differences (e.g., format and wording), are consistent with the Actions set forth in the above letters and NUREG-1433. Format and wording differences were necessary to: (1) maintain consistency within the BSEP Technical Specifications, (2) minimize the number of changes, (3) accommodate editorial preferences, and (4) enhance operator understanding.

Specific changes requested for each of the affected Technical Specifications are identified in Enclosures 4 and 5. For convenience in reviewing this request, the following descriptions of those changes are divided into eight subsections corresponding to each of the eight affected specifications. The applicable LTRs and supporting documents for the changes to each specification are identified in the list of references at the beginning of each subsection.

Specification 3/4.3.1 - Reactor Protection System Instrumentation

References:	NEDC-30851P-A	(GE LTR)
	MDE-81-0485, Rev. 1	(Enclosure 6)
	BWROG-92102	(BWROG letter to NRC)

1. Revise Actions a & b and Footnotes * & ** to incorporate a check for trip capability (loss of function) and to provide an AOT for repair of either 1, 6, or 12 hours, depending on the degree of redundancy remaining for the Functional Unit.

2. Change the description for Functional Unit 2.b in Tables 3.3.1-1 and 4.3.1-1, from

"Flow Biased Neutron Flux - High" to "Flow Biased Simulated Thermal Power - High." (This is an editorial change to make this description consistent with the item 2.b description in Table 2.2.1-1 for this Functional Unit.)

3. Revise Table 3.3.1-1, Note (a) to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability based on the terminology used in NUREG-1433.

4. Increase the Channel Functional Test intervals in Table 4.3.1-1 from W (weekly) to Q (quarterly) for the following Functional Units:

   Average Power Range Monitor:
   a. item 2.b, Flow Biased Simulated Thermal Power - High
      [Also, Note (f) in Table 4.3.1-1 is revised as follows: The word "setpoint" is changed to "channel" to conform to the intended surveillance required for this design feature. This change was made to the original Standard Tech Specs (NUREG-0123) subsequent to the adoption of these specifications by the Brunswick plants, which represented the initial application of the standard. This change brings the Brunswick plants into line with the surveillance practice established for similar plants.]
   b. item 2.c, Fixed Neutron Flux - High, 120%
   c. item 2.d, Inoperative
   d. item 2.e, Downscale

5. Increase the Channel Functional Test intervals and, where applicable, Channel Calibration intervals and corresponding Note (l) in Table 4.3.1-1 from M (monthly) to Q (quarterly) for the following Functional Units:

   a. item 3, Reactor Vessel Steam Dome Pressure - High
   b. item 4, Reactor Vessel Water Level - Low, Level 1
   c. item 5, Main Steam Line Isolation Valve - Closure (Channel Functional Test only)
   d. item 6, Main Steam Line Radiation - High (Channel Functional Test only)
   e. item 7, Drywell Pressure - High
   f. item 9, Turbine Stop Valve - Closure (Channel Functional Test only)
   g. item 10, Turbine Control Valve Fast Closure, Control Oil Pressure - Low (Channel Functional Test only)

   CP&L submitted, on September 30, 1994, a license amendment request to delete the MSLRM scram and isolation functions. Markups of the Technical Specifications and discussions of the basis for the changes for the Main Steam Line Radiation Monitor (MSLRM) functions are being provided, therefore, for consistency.

6. Add new Functional Unit Number 13, "Automatic Scram Contactors" to Tables 3.3.1-1 and 4.3.1-1, providing weekly testing requirements for the automatic scram contactors. The automatic scram contactors are currently tested at least once every seven days as part of the APRM sensor channel tests. As discussed in Sections 5.7.1 & 5.7.3 of NEDC-30851P-A and Sections 5.3 & 5.6 of the NRC SER,

the APRM sensor channel tests can be increased to quarterly, provided the automatic scram contactors continue to be tested on a weekly basis. For later plant designs (as in the generic model) this can be accomplished by actuating the manual scram circuits on a weekly basis. For earlier plant designs, such as BSEP, the manual scram circuits are separate from the automatic scram circuits. For these designs, weekly functional testing of the automatic scram contactors can be assured by explicitly including them as a separate line-item in the RPS Instrumentation Tables.

7. Modify Bases Section 3/4.3.1 to reference the generic LTR NEDC-30851P-A and Plant-Specific Report MDE-81-0485, Rev. 1 which provide the detailed basis for the above proposed changes.

Specification 3/4.3.2 - Isolation Actuation Instrumentation

References:	NEDC-30851P-A, Suppl 2	(GE LTR)
	NEDC-31677P-A	(GE LTR)
	OG90-579-32A	(GE letter to NRC)
	GENE-A31-00001-02	(Enclosure 8)

8. Delete the phrase "and place the inoperable channel in the tripped condition" from TS 3.3.2, Action a. (This is an editorial change only. It does not change the intent or any existing requirement. Required Actions for inoperable channels are addressed in Actions b & c, and Table 3.3.2-1, therefore this phrase is redundant and unnecessary.)

9. Revise Actions b & c and delete Footnotes * & ** to include a check for loss-of-function and provide an AOT for repair of 12 hours for Trip Functions that are common to RPS and 24 hours for Trip Functions that are not common to RPS.

10. Change "logic chains" to "logic trains" in the second sentence of Surveillance Requirement 4.3.2.3. (This is an editorial change to make the terminology consistent within this sentence and consistent with Surveillance Requirements 4.3.1.3 and 4.3.3.3.)

11. Revise Table 3.3.2-1, Notes (b) and (c) into one new Note (b) which retains the existing 2 hour AOT for Trip Functions designed such that it is not possible to maintain trip capability during required surveillance testing and to increase the AOT for required surveillance testing from 2 hours to 6 hours for Trip Functions that can be tested while maintaining trip capability.

12. Increase the Channel Functional Test intervals and, where applicable, the Channel Calibration intervals and corresponding Note (b) in Table 4.3.2-1 from M (monthly) or W (weekly) to Q (quarterly) for the following Trip Functions:

    Primary Containment Isolation

    a. item 1.a.1, Reactor Vessel Water Level - Low, Level 1
    b. item 1.a.2, Reactor Vessel Water Level - Low, Level 3
    c. item 1.b, Drywell Pressure - High
    d. item 1.c.1, Main Steam Line Radiation - High (Channel Functional Test only)

e. item 1.c.2, Main Steam Line - Pressure - Low
f. item 1.c.3, Main Steam Line Flow - High
g. item 1.d, Main Steam Line Tunnel Temperature - High (Channel Functional Test only)
h. item 1.e, Condenser Vacuum - Low
i. item 1.f, Turbine Building Area Temperature - High (Channel Functional Test only)
j. item 1.h, Reactor Building Exhaust Radiation - High (Channel Functional Test only)

Secondary Containment Isolation

k. item 2.a, Reactor Building Exhaust Radiation - High (Channel Functional Test only)
l. item 2.b, Drywell Pressure - High
m. item 2.c, Reactor Vessel Water Level - Low, Level 2

Reactor Water Cleanup System Isolation

n. item 3.c, Reactor Vessel Water Level - Low, Level 2

Core Standby Cooling Systems Isolation

o. item 4.a.1, HPCI Steam Line Flow - High
p. item 4.a.3, HPCI Steam Supply Pressure - Low (Channel Functional Test only)
q. item 4.a.6, HPCI Turbine Exhaust Diaphragm Pressure - High (Channel Functional Test only)
r. item 4.a.10, Drywell Pressure - High
s. item 4.b.1, RCIC Steam Line Flow - High
t. item 4.b.3, RCIC Steam Supply Pressure - Low (Channel Functional Test only)
u. item 4.b.6, RCIC Turbine Exhaust Diaphragm Pressure - High (Channel Functional Test only)
v. item 4.b.12, Drywell Pressure - High

Shutdown Cooling System Isolation

w. item 5.a, Reactor Vessel Water Level - Low, Level 1
x. item 5.b, Reactor Steam Dome Pressure - High (Channel Functional Test only) [Note: In comparison to other BWR/4s, there is nothing unique about the BSEP design for this Trip Function. Therefore, the S/U functional test requirement and corresponding Note (c) for this Trip Function are deleted, consistent with the surveillance requirements established for similar plants and with revisions to the Standard Tech Specs (including NUREG-0123 and NUREG-1433) that were issued subsequent to the original BSEP Tech Specs].

13. Modify Bases Section 3/4.3.2 to reference generic LTRs NEDC-30851P-A, Supplement 2 and NEDC-31677P-A which provide the detailed basis for the above proposed changes.

Specification 3/4.3.3 - Emergency Core Cooling System Actuation Instrumentation

References:	NEDC-30936P-A, Parts 1 and 2	(GE LTR)
	OG90-319-32D	(GE letter to NRC)
	RE-011, Rev. 1	(Enclosure 7)

14. In specification 3.3.3, insert the word "channels" between "instrumentation" and "shown". (This is an editorial change to make the wording of this specification consistent with the wording of specifications 3.3.1 and 3.3.2.)

15. Delete the phrase "and place the inoperable channel in the tripped condition" from TS 3.3.3, Action a. (This is an editorial change only. It does not change the intent or any existing requirement. Required Actions for inoperable channels are addressed in Action b and Table 3.3.3-1, therefore this phrase is redundant and unnecessary.)

16. Revise Table 3.3.3-1, including Actions 30-33, to address "Minimum Operable Channels" on a "per Trip Function" basis instead of the current "per Trip System" basis. (This is an editorial change to achieve consistency with Standard Tech Specs, including NUREG-0123 and NUREG-1433, and to simplify application of the topical reports to the Brunswick Units.)

17. Revise Table 3.3.3-1, Action 33 to reference HPCI instead of HPCS. (This is an editorial change to correct a typographical error. BSEP has a High Pressure Coolant Injection system not High Pressure Core Spray).

18. Revise and clarify Table 3.3.3-1 Actions to include, depending on degree of redundancy, a check for loss-of-function and an AOT for repair of 24 hours.

19. Revise Table 3.3.3-1, Note (a) to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability based on the terminology used in NUREG-1433.

20. Delete temporary Note (f). As indicated in this note, this was a temporary exception to ACTION 31 (proposed ACTION 30) and is no longer applicable.

21. Increase the Channel Functional Test interval and, where applicable, the Channel Calibration intervals, and corresponding Note (b) in Table 4.3.3-1 from M (monthly) to Q (quarterly) for the following Trip Functions:

    Core Spray System

    a. item 1.a, Reactor Vessel Water Level - Low, Level 3
    b. item 1.b, Reactor Steam Dome Pressure - Low
    c. item 1.c, Drywell Pressure - High

Low Pressure Coolant Injection Mode of RHR System

d. item 2.a, Drywell Pressure - High
e. item 2.b, Reactor Vessel Water Level - Low, Level 3
f. item 2.c, Reactor Vessel Shroud Level
h. item 2.d.1, Reactor Steam Dome Pressure - Low, RHR Pump Start and LPCI Injection Valve Actuation
i. item 2.d.2, Reactor Steam Dome Pressure - Low, Recirculation Loop Pump Discharge Valve Actuation

High Pressure Coolant Injection System

j. item 3.a, Reactor Vessel Water Level - Low, Level 2
k. item 3.b, Drywell Pressure - High
l. item 3.c, Condensate Storage Tank Level - Low (Channel Functional Test only)
m. item 3.d, Suppression Chamber Water Level - High (Channel Functional Test only)

Automatic Depressurization System

n. item 4.b, Reactor Vessel Water Level - Low, Level 3
o. item 4.c, Reactor Vessel Water Level - Low, Level 1
p. item 4.e, Core Spray Pump Discharge Pump Pressure - High (Channel Functional Test only)
q. item 4.f, RHR (LPCI MODE) Pump Discharge Pressure - High (Channel Functional Test only)

22. Modify Bases Section 3/4.3.3 to reference generic LTR NEDC-30936P-A and plant-specific report RE-011, Rev. 1 which provide the detailed basis for the above proposed changes.

Specification 3/4.3.4 - Control Rod Withdrawal Block Instrumentation

References:	NEDC-30851P-A, Supplement 1	(GE LTR)
	GENE-770-06-1-A	(GE LTR)

23. Revise Table 3.3.4-1, Note (a) to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability based on the terminology used in NUREG-1433.

24. Increase the Channel Functional Test interval requirement in Table 4.3.4-1 from M (monthly) to Q (quarterly) for the following Trip Functions:

APRM

a. item 1.a, Upscale (Flow Biased)
b. item 1.c, Downscale

Rod Block Monitor

c. item 2.a, Upscale
d. item 2.c, Downscale

25. Modify Bases Section 3/4.3.4 to reference generic LTRs NEDC-30851P-A, Supplement 1 and GENE-770-06-1-A which provide the detailed basis for the above proposed changes.

Specification 3/4.3.5.5 - Control Room Emergency Ventilation System Instrumentation

Reference: GENE-770-06-1-A (GE LTR)

26. Increase the Channel Functional Test interval for Table 4.3.5.5-1, Radiation Protection, item 1, Control Building Air Intake, from M (monthly) to Q (quarterly).

27. Modify Bases Section 3/4.3.5.5 to reference generic LTR GENE-770-06-1-A, which provides the detailed basis for the above proposed changes.

Specification 3/4.3.6.1 - ATWS Recirculation Pump Trip (RPT) System Instrumentation

Reference: GENE-770-06-1-A (GE LTR)

28. In specification 3.3.6.1, replace "trip systems" with "channels" and revise Table 3.3.6.1-1 column heading from "Minimum Number Operable Trip Systems Per Operating Pump" to "Minimum Operable Channels Per Trip System". (This is an editorial change to clarify intent; provide consistency with the language used in the Action statements and to make the wording of this specification consistent with the wording of Specifications 3.3.1, 3.3.2 and 3.3.3.)

29. Replace TS 3.3.6.1, Actions b, c, d and e with new Actions b, c, and d to more clearly reflect the system design and intent of the Actions, to provide appropriate compensatory actions when a loss-of-function condition exists, and to incorporate appropriate AOTs for repairing inoperable channels and trip systems consistent with NUREG-1433.

30. Revise Table 3.3.6.1-1, Footnote (a) to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability based on the terminology used in NUREG-1433.

31. Increase the ATWS-RPT Channel Functional Test and Calibration intervals, and corresponding Note (b) in Table 4.3.6.1-1 from M (monthly) to Q (quarterly) for the following Trip Functions:

    a. item 1, Reactor Vessel Water Level - Low, Level 2
    b. item 2, Reactor Vessel Pressure - High

32. For Unit 2 only, change the Channel Functional Test interval from M (monthly) to NA (Not Applicable) for the Reactor Vessel Pressure - High, Transmitter in Table 4.3.6.1-1. (This is an editorial change to correct a typographical error. The Channel Functional Test is performed at the analog trip unit and, as correctly indicated in the corresponding Unit 1 specification, is not applicable to the transmitter.)

33. Modify Bases Section 3/4.3.6 to reference generic LTR GENE-770-06-1-A which provides the detailed basis for the above proposed changes.

Specification 3/4.3.6.2 - End-of-Cycle Recirculation Pump Trip (RPT) System Instrumentation (Unit 2 only)

Reference: GENE-770-06-1-A (GE LTR)

34. Revise TS 3.3.6.2, Actions b and c.1 from "within 1 hour" to "within 12 hours" to provide an AOT for repair of 12 hours.

35. Revise Table 3.3.6.2-1, Footnote (a) to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability based on the terminology used in NUREG-1433.

36. Increase the EOC-RPT Channel Functional Test intervals in Table 4.3.6.2.1-1 from M (monthly) to Q (quarterly) for the following Trip Functions:

    a. item 1, Turbine Stop Valve - Closure
    b. item 2, Turbine Control Valve - Fast Closure

37. Modify Bases Section 3/4.3.6 to reference generic LTR GENE-770-06-1-A which provides the detailed basis for the above proposed changes.

Specification 3/4.3.7 - Reactor Core Isolation Cooling System Actuation Instrumentation

References: GENE-770-06-2P-A (GE LTR)
OG90-319-32D (GE letter to NRC)

38. Correct typographical error in LCO statement 3.3.7 (change "ther" to "their").

39. Revise Table 3.3.7-1, Footnote (a) to increase the AOT for required surveillance testing from 2 hours to 6 hours and, except for Functional Unit 2 (Reactor Vessel Water Level - High), to incorporate a check to ensure that trip capability is maintained. As indicated in Table 3.3.7-1, Footnote (b), Functional Unit 2 consists of a single logic train (one trip system) with a two-out-of-two logic. Therefore, during surveillance testing of the associated instrument channels, it is not feasible to maintain the automatic RCIC <u>shutoff</u> capability provided by Functional Unit 2. Applying the 6 hour AOT to this Functional Unit (even though automatic trip capability is not maintained) is acceptable, since it will not prevent automatic <u>initiation</u> of RCIC by Functional Unit 1 (Reactor Vessel Water Level - Low, Level 2). This exception to the requirement for maintaining trip capability during surveillance testing is also included in NUREG-1433.

40. Revise Table 3.3.7-1, Actions 50.a, 51, and 52 to provide 24 hours to repair an inoperable channel and to provide appropriate compensatory actions when a loss-of-function condition exists.

41. Increase the Channel Functional Test intervals and, where applicable, Channel Calibration intervals and corresponding Note (b) in Table 4.3.7.1-1 from M (monthly) to Q (quarterly) for the following Functional Units:

    a. item 1, Reactor Vessel Water Level - Low, Level 2
    b. item 2, Reactor Vessel Water Level - High
    c. item 3, Condensate Storage Tank Level - Low (no change to Calibration interval)

42. Modify Bases Section 3/4.3.7 to reference generic LTR GENE-770-06-2P-A which provides the detailed basis for the above proposed changes.

III. BASIS FOR PROPOSED CHANGES

Carolina Power & Light Company has reviewed the referenced Licensing Topical Reports (LTRs) completed by the BWR Owners' Group and completed the plant-specific evaluations necessary to confirm that the generic analyses and conclusions of the LTRs apply to BSEP Units 1 and 2. As stated within the NRC Safety Evaluation Reports (SERs) for the LTRs, three issues must be addressed to justify application of the generic analyses to a specific facility's Technical Specifications. Two of the issues are applicable to each of the GE Licensing Topical Reports (LTRs). Issue 3 is applicable only to NEDC-30851P-A. The following discussion provides the information requested by the NRC staff in plant-specific submittals.

<u>SER Issue 1</u>

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

<u>Response to SER Issue 1</u>

1. Licensing Topical Report NEDC-30851P-A, Appendix L identifies CP&L (BSEP 1 & 2) as a participating utility in the development of the RPS Technical Specification Improvement Analysis. Section 7.4 of the LTR states:

> "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 to 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."

Furthermore, included in this submittal is GE plant specific report MDE-81-0485, Rev. 1 titled, "Technical Specification Improvement Analysis for the Reactor Protection System for Brunswick Steam Electric Plant, Units 1 and 2," dated August 1994, which concludes in Section 4 that "... the generic analysis in Reference 1 (NEDC-30851P-A) is applicable to BSEP." For a discussion of the differences between BSEP Units 1 and 2 and the generic plant analyzed in NEDC-30851P-A, see the response to Issue 3 below and the GE plant specific report MDE-81-0485, Rev. 1. CP&L has reviewed the LTRs and plant-specific reports and verified applicability to BSEP.

2. Licensing Topical Report NEDC-30851P-A, Supplement 2, Appendix A identifies CP&L as a participating utility in the development of the BWR Isolation Instrumentation Common to the RPS and ECCS Technical Specification Improvement Analysis. Section 3.3 specifically addresses BWR 3/4 plants. CP&L has reviewed this LTR and verified applicability to BSEP.

3. Licensing Topical Report NEDC-31677P-A, provides the Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation that is not common to the RPS or ECCS. Analysis and results for BWR 3/4 plants is presented in Section 5.1, and Appendix C1. In addition, GE's evaluation of the applicability of the LTR to the BSEP Units 1 and 2 is provided in the supplemental letter report (Enclosure 8) from R.P. Raftery/S. Visweswaran (GE) to A. Harris (CP&L), "Assessment of Brunswick Nuclear Plant Isolation Actuation Instrumentation Against NEDC-31677P-A Bounding Analyses," dated August 18, 1994. CP&L has reviewed the topical and supplemental report and verified applicability to BSEP.

4. Licensing Topical Report NEDC-30936P-A, Part 1, Appendix N and Part 2, Appendix B identifies CP&L as a participating utility in the development of the BWR Owners' Group Technical Specification Improvement Methodology for ECCS Actuation Instrumentation. Section 5.4 (Part 2) describes the generic analyses performed for BWR 3/4 plants. The requisite plant-specific evaluation for BSEP is provided in the enclosed GE Report RE-011, Rev. 1, titled, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Brunswick Steam Electric Plant, Units 1 & 2," dated August 1994. CP&L has reviewed the LTRs and plant specific report and verified applicability to BSEP.

5. Licensing Topical Report GENE-770-06-1-A, identifies application of changes to surveillance test intervals and allowable out-of-service times for selected Instrumentation Technical Specifications for all BWR plants. CP&L has reviewed this LTR and verified applicability to BSEP.

6. Licensing Topical Report GENE-770-06-2P-A, Section 3.1 concludes that the proposed changes to the RCIC Instrumentation Technical Specifications for BWR 3/4 plants are consistent with the changes justified by LTR NEDC-30851P-A. CP&L has reviewed these LTRs and verified applicability to BSEP.

7. Licensing Topical Report NEDC-30851P-A, Supplement 1, Appendix B identifies CP&L as a participating utility in the development of the Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation. Section 4.0 specifically includes BWR 4 plants. CP&L has reviewed this LTR and verified applicability to BSEP.

SER 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 the (RPS, Isolation, ECCS, Control Rod Block, Control Room Emergency Ventilation, ATWS-RPT, EOC-RPT and RCIC) channels in the plant are bounded by the assumptions used in the Topical Report when the functional test interval is extended from monthly (or weekly) to quarterly.

Response to SER Issue 2

This requirement, as stated, was difficult to address because the LTRs do not contain quantitative instrument drift assumptions. In order to resolve this issue, the BWR Owner's Group and the NRC staff reviewed BWR setpoint calculation methodology and decided that additional clarification was in order. As a result, the NRC staff provided the following additional guidance in a letter from C.E. Rossi (NRC) to R.F. Janecek (BWROG), dated April 27, 1988:

> "...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."

For BSEP instrumentation addressed by the Topical Reports, the sensor (Transmitter) Calibration intervals are quarterly or longer and are therefore unaffected by the changes proposed in this amendment. Also, CP&L has evaluated field data from monthly calibration checks performed over a nine year period (over 2900 datapoints) for the affected analog trip units (Trip Logic). Results of that evaluation showed that the quarterly drift will remain within the present tolerances. As a result, CP&L has concluded that increasing the Channel Functional Test intervals and, where applicable, the analog trip unit Calibration intervals from weekly or monthly to quarterly will not result in excessive drift relative to the current setpoints.

SER 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 specific analysis done using the procedures of Appendix K to NEDC-30851P-A. (This issue applies to LTR NEDC-30851P-A only.)

Response to SER Issue 3

Included in this submittal (Enclosure 6) is GE Report MDE-81-0485, Rev. 1, titled "Technical Specification Improvement Analysis for the Reactor Protection System For Brunswick Steam Electric Plant, Units 1 and 2." The analysis documented in this report utilized the procedures of GE Licensing Topical Report NEDC-30851P-A, Appendix K to identify and evaluate the differences between the parts of the RPS that perform the trip functions at BSEP Units 1 and 2 and those analyzed in the generic study. Results of the analysis indicate that, although the RPS configuration for BSEP Units 1 and 2 differs from the configuration of the base case plant, the differences do not have a significant impact. Therefore, the conclusions reached in NEDC-30851P-A apply to BSEP and the plant-specific changes contained in this license amendment request are bounded by both the generic analysis and the NRC's SER.

IV. CONCLUSION

As discussed in Section III above, Carolina Power & Light Company has satisfactorily addressed the three issues which the NRC staff has indicated are necessary to implement the generic Technical Specification changes on a plant-specific basis. Two of the issues are applicable to each of the GE Licensing Topical Reports (LTRs). Issue 3 is applicable only to NEDC-30851P-A. The first issue, which applies to each LTR, required confirmation of the applicability of the generic analyses to BSEP Units 1 and 2. Two required plant-specific analyses, one for RPS and one for ECCS Instrumentation, demonstrate that the generic analyses are applicable to BSEP Units 1 and 2. The information provided in the plant-specific reports contained in this submittal addresses the differences between BSEP Units 1 and 2 and the generic analyses and, when applied with the conclusions contained in NEDC-30851P-A, and NEDC-30936P-A, justifies the proposed changes.

The second issue required confirmation 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. This issue is only applicable to those instrument channels that have sensor or trip unit calibration intervals that are shorter than the proposed quarterly functional test interval. For those cases, an extension of the Channel Functional Test interval would require a corresponding extension of the Channel Calibration interval and therefore consideration of the effects on setpoint drift. Sensor calibration intervals for the BSEP instrumentation associated with the proposed changes were verified to be equal to or longer than once per quarter and therefore unaffected by the proposed changes. Also, setpoint drift was confirmed to be within current tolerances for those trip units that required their calibration intervals increased to quarterly.

The third issue, which applies only to the plant-specific application of LTR NEDC-30851P-A, required confirmation that differences between the parts of the RPS that perform the trip functions in BSEP and those of the base case plant were included in the plant-specific analysis done using the procedures of NEDC-30851P-A, Appendix K. The

enclosed proprietary GE Report, MDE-81-0485, Rev. 1, titled, "Technical Specification Improvement Analysis for the Reactor Protection System for Brunswick Steam Electric Plant, Units 1 and 2," was performed using the Appendix K procedures and the results confirm that the differences between BSEP and the base case plant do not significantly impact the results and conclusions of the generic LTR.

Finally, approved BWR Owners' Group clarifications, used in the development of NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," were implemented in the proposed changes to the allowable out-of-service times (AOTs). The revised AOT wording precludes plant operation for an extended period of time with any combination of inoperable instrument channels that could prevent a trip function from completing its required action.

ENCLOSURE 2

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NO SIGNIFICANT HAZARDS EVALUATION

In accordance with 10 CFR 50.91(a), Carolina Power & Light Company is providing an analysis of no significant hazards consideration for this issue, using the standards in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed this proposed license amendment request and believes that its adoption would not involve a significant hazards consideration. The basis for this determination follows.

1) **Operation of Brunswick Steam Electric Plant, Units 1 and 2, in accordance with the proposed amendment, would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The generic Licensing Topical Report, NEDC-30851P-A, assessed the impact of changing RPS surveillance test intervals (STIs) and allowable out-of-service times (AOTs) on the RPS failure frequency, the scram frequency and equipment cycling. Specifically, Section 5.7.4, "Significant Hazards Assessment" of NEDC-30851P-A states:

"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 time is negligible."

From this generic analysis, the BWR Owners' Group concluded and CP&L concurs 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 the RPS instrumentation logic operates undisturbed is increased, resulting in fewer inadvertent scrams during testing and repair. The proprietary plant-specific analysis contained in this submittal (Enclosure 6) demonstrates that, although BSEP Units 1 and 2 differ from the generic plant analyzed in LTR NEDC-30851P-A, the net effect of the plant-specific differences does not alter the generic conclusions.

The generic Licensing Topical Reports, NEDC-30851P-A, Supplement 2 and NEDC 31677P-A, assessed the impact of changing STIs and AOTs for BWR Isolation Instrumentation. Section 4.0, "Summary of Results," of NEDC-30851P-A, Supplement 2 states:

> "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 allowable 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:

> "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".

From this generic analysis, the BWR Owners' Group concluded and CP&L concurs 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. For those parameters common to RPS, the overall probability of an accident is actually decreased as the time the RPS instrumentation logic operates undisturbed is increased, resulting in less inadvertent scrams during testing and repair. The plant-specific evaluation provided with this submittal (Enclosure 8) demonstrates that the conclusions of the generic analyses are applicable to BSEP Units 1 and 2.

The generic Licensing Topical Report, NEDC-30936P-A (Parts 1 and 2), assessed the impact of changing STIs and AOTs for all BWR ECCS Actuation Instrumentation. Section 4.0, "Technical Assessment of Changes," of NEDC-30936P-A (Part 2) states:

> "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 and CP&L concurs 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 analysis contained in this submittal (Enclosure 7) demonstrates that, although BSEP Units 1 and 2 differ from the generic model analyzed in LTR NEDC-30936P-A, the net effect of the plant specific differences does not alter the generic conclusions.

The generic Licensing Topical Reports, NEDC-30851P-A, Supplement 1, and GENE-770-06-1-A assessed the impact of changing Control Rod Block STIs and AOTs on Rod Block failure frequency. GENE-770-06-1-A also assessed the impact of changing STIs and AOTs on ATWS-RPT and EOC-RPT failure frequency. Section 5 (Brookhaven National Laboratory's Technical Evaluation Report - Attachment 2 to the NRC SER) of NEDC-30851P-A, Supplement 1 states:

> "The BWR Owners' Group 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."

and Section 2.0, "Summary" of GENE-770-06-1-A states:

> "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".

Based on the generic analysis in NEDC-30851P-A, Supplement 1, the BWR Owners' Group concluded and CP&L concurs that the proposed changes to Control Rod Withdrawal

Block instrumentation do not significantly increase the probability or consequences of an accident previously evaluated. Also, based on the generic assessment in GENE-770-06-1-A, the BWR Owners' Group concluded and CP&L concurs that the proposed changes to the ATWS-RPT and EOC-RPT instrumentation do not significantly increase the probability or consequences of an accident previously evaluated.

Bases contained in GE Topical Report GENE-770-06-2P-A, assessed the impact of changing STIs and AOTs on BWR RCIC failure frequency. Section 2.0, "Summary" of GENE-770-06-2P-A states:

> "The STI and AOT changes to the RCIC actuation instrumentation are justified based on their small effect on the water injection function unavailability and consistency with comparable changes to actuation instrumentation for the other ECCS subsystems".

On this basis, the BWR Owners' Group concluded and CP&L concurs that the proposed changes to RCIC instrumentation do not significantly increase the probability or consequences of an accident previously evaluated.

2) **Operation of Brunswick Steam Electric Plant, Units 1 and 2, in accordance with the proposed amendment, would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed changes do not alter the physical characteristics or function of any plant systems or components and they do not introduce any new mode of operation. Therefore, system and component performance would not be challenged in a manner that could create the possibility of a new or different kind of accident from any accident previously evaluated.

3) **Operation of Brunswick Steam Electric Plant, Units 1 and 2, in accordance with the proposed amendment, would not involve a significant reduction in a margin of safety.**

The NRC staff has reviewed and approved the generic studies contained in the LTRs and has concurred with the BWR Owners' Group that the proposed changes do not significantly affect the probability of failure or availability of the affected Instrument Systems. The proposed changes to AOTs provide realistic times to complete the required actions without increasing the overall instrument failure frequency. Likewise, the extended STIs do not result in significant changes in the probability of instrument failure. Furthermore, the proposed changes will reduce the probability of test-induced plant transients and equipment failures. Finally, instrument setpoint drift will remain within present tolerances, thereby assuring that the margin of safety, as demonstrated by applicable safety analyses, remains unchanged. Therefore, it is concluded that the proposed changes would not result in a reduction in the margin of safety.

ENCLOSURE 3

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ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides a criterion for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) result in an increase in individual or cumulative occupational radiation exposure. Carolina Power & Light Company has reviewed this request and believes that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows.

1. This amendment does not involve a significant hazards consideration, as shown in Enclosure 2.

2. This amendment does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The amendment only changes the allowable out-of-service times for instrument repair and testing and the frequency at which instrument testing is performed. It does not alter the physical characteristics or function of any plant structure, system, or component; and it does not introduce any new mode of plant operation. Therefore, no negative impact on any systems which could affect the types or amounts of effluents released would occur.

3. This amendment does not result in an increase in individual or cumulative occupational radiation exposure. The amendment only changes the allowable out-of-service times for instrument repair and testing and the frequency at which instrument testing is performed. The proposed changes will not affect radiation levels and exposure times will decrease as a result of less frequent surveillance testing. Therefore, no increase in individual or cumulative occupational radiation exposure would result from this amendment.

ENCLOSURE 4

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MARKED-UP TECHNICAL SPECIFICATIONS PAGES - UNIT 1

The following pages have been revised and an 'X' has been placed in the margin to indicate where changes occur.

Technical Specifications

3/4 3-1, 2, 3, 5, 7, 8, 9	(RPS Instrumentation)
3/4 3-10, 11, 17a, 27 through 32	(Isolation Actuation Instrumentation)
3/4 3-33 through 35, 37, 38, 43 through 46	(ECCS Actuation Instrumentation)
3/4 3-49, 51	(Control Rod Withdrawal Block Instrumentation)
3/4 3-64c	(Control Room Emergency Ventilation Instrumentation)
3/4 3-88, 89, 91	(ATWS-RPT Instrumentation)
3/4 3-92 through 94, 96	(RCIC Actuation Instrumentation)

Bases

B 3/4 3-1
B 3/4 3-2
B 3/4 3-3d
B 3/4 3-3e
B 3/4 3-6