Docket No. 50-324

NOV 1 1 1975

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

The Commission has issued the enclosed Amendment No. 7 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit No. 2. This amendment includes Change No. 6 to the Technical Specifications and is in response to your request dated October 14, 1975. Our review of that portion of your request relating to the off-site review and audit function has not been completed and, therefore, will be considered separately.

The amendment incorporates into the Brunswick Steam Electric Plant, Unit No. 2, Technical Specifications changes to the reporting requirements. Changes to your proposal were necessary to meet our requirements. These have been discussed with your staff. The technical specifications are based on Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4. In addition, this amendment changes the composition of the Plant Nuclear Safety Committee by adding the Plant Superintendent as Vice Chairman.

We request that you use the formats presented in the Appendices to Regulatory Guide 1.16, Revision 4, for reporting operating information and that you report events of the type described under the section "Events of Potential Public Interest". Instructions for using these reporting formats are contained in Regulatory Guide 1.16 (a copy is enclosed for your use), and AEC report OOE-SS-001 titled "Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File" (a copy of which was provided you previously). This report is modified by updated instructions dated August 21, 1975 which are enclosed. Copy requirements are summarized in Regulatory Guide 10.1, "Compilation of Reporting Requirements for Persons Subject to NRC Regulations", a copy of which is also enclosed. This Guide will assist you in identifying reports that are required by the Commission's regulations set forth in Title 10 Code of Federal Regulations but are not contained in your technical specifications. Reports that are required by the regulations have not been repeated in your technical specifications.

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SURNAME ▶						
DATE ▶						

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1974 — 555-338

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Sincerely,

Original signed by
R. A. Purple
Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 7
2. Regulatory Guide 1.16
3. Updated Instructions
4. Regulatory Guide 10.1
5. Safety Evaluation
6. Federal Register Notice

cc w/enclosures:
See next page

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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1974 — 555-338

cc w/enclosures:
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Ms. Janet Godwin, President
Project Environment
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Southport - Brunswick County Library
109 W. Moore Street
Southport, North Carolina 28461

Mr. W. A. Kopp, Jr.
Chairman, Board of County
Commissioners of Brunswick County
Bolivia, North Carolina 28422

cc w/enclosures & incoming:
Office of Intergovernmental
Relations
116 West Jones Street
Raleigh, North Carolina 27603

Mr. Dave Hopkins
Environmental Protection Agency
1421 Peachtree Street, NE.
Atlanta, Georgia 30309

yellow

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-62

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated October 14, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-62 is hereby amended to read as follows:

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

"2.C.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 6 ."

3. This license amendment is effective 30 days after the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
R. A. Purple

Robert A. Purple, Chief,
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 6 to the
Technical Specifications

Date of Issuance: NOV 11 1975

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DATE						

ATTACHMENT TO LICENSE AMENDMENT NO. 7

CHANGE NO. 6 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Revise Appendix A as follows:

Remove pages listed below and insert identically numbered pages.

iii and iv

0-1 and 0-2

0-3 and 0-4

6.5-1 through 6.5-4

6.7-1

6.11-1 through 6.11-10

Figure 6.5-1 and Figure 6.5-2

6.13-1 and 6.13-2

6.6-1

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SURNAME ➤						
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CONTENTS (Continued)

TECHNICAL SPECIFICATIONS

NOV 11 1975

1.0 DEFINITIONS

The frequently used succeeding terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. 6
Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

B. Limiting Safety System Setting (LSSS) - The limiting safety system settings are nominal settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The limiting safety system setting plus the tolerance of the instrument as given in the system instrument list gives the limiting trip point for operation. This additional margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded. The inequality sign which may be given merely signifies the preferred direction of operational trip setting.

C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. The trip setting plus the tolerance of the instrument as given in the system instrument list gives the limiting trip point for operation. This additional margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded. The inequality sign which may be given merely signifies the preferred direction of operational trip setting.

NOV 1 1 1975

6 **Section 1.0.D is Deleted**

NOV 11 1975

6

E. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

F. Operating - Operating means that a system or component is performing its intended function in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

H. Reactor Power Operation - Reactor power operation is any operation with mode switch in the STARTUP or RUN position with the reactor critical and above one percent rated power.

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212 F, system pressure less than 1055 psig, the main steam isolation valves closed and the mode switch in STARTUP.

J. Cold Condition - Reactor coolant temperature equal to or less than 212 F.

K. Mode - The reactor mode is that which is established by the mode-selector switch. The modes are SHUTDOWN, REFUEL, STARTUP, and RUN which are defined as follows:

1. STARTUP - In this mode the reactor protection scram trips, initiated by main steamline isolation valve closure, are bypassed , the low

NOV 1 1 1975

pressure main steamline isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron monitoring system trips, the APRM 15% high flux trip and control rod withdrawal interlocks in service.

2. RUN - In this mode the reactor system pressure is at or above 850 psig and the reactor protection system is energized with APRM protection. The RBM interlocks are in service $\geq$30% power.

3. Shutdown - The reactor is in the shutdown mode when the reactor mode switch is in the SHUTDOWN position and no core alterations are being performed.

   a. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212 F.

   b. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212 F.

4. REFUEL - The reactor is in the refuel mode when the mode switch is in the REFUEL position. When the mode switch is in the REFUEL position, the refueling interlocks are in service and core alternations can be performed.

L. Rated Power - Rated power refers to operation at a reactor power of 2436 MWt; this is also termed 100 percent power. Rated steamflow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power (see also "Design Power" definition).

M. Design Power - Design power refers to the power level at which the reactor is producing 105 percent of reactor vessel rated steamflow. Because of differences in turbine-generator design and feedwater heating, design power does not necessarily correspond

6.5 Review and Audit

Organizational units for the review and audit of plant operations shall be constituted and have the responsibilities and authorities outlined below:

6.5.1 Plant Nuclear Safety Committee (PNSC)

6.5.1.1 Purpose

As an effective means for regular review, evaluation, and maintenance of plant operational safety, a Plant Nuclear Safety Committee (PNSC) will be established during the preoperational test period prior to fuel loading of the first unit. The committee will be chaired by the Plant Manager and composed of plant supervisory personnel. The organization of this committee is shown on Figure 6.5-1.

6.5.1.2 Composition

The Plant Nuclear Safety Committee shall be composed of the following:

(a)	Chairman:	Plant Manager
(b)	Vice Chairmen:	Operations Supervisor or Plant Superintendent 6
(c)	Secretary:	Administrative Supervisor
(d)		Engineering Supervisor
(e)		Maintenance Supervisor
(f)		Environmental and Radiation Control Supervisor
(g)		Quality Assurance Supervisor

Minimum qualifications of members of the Plant Nuclear Safety Committee with regard to educational background and experience shall meet or exceed the criteria included in ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel" for similar supervisory positions.

6.5.1.3 Alternates

Alternate members shall be appointed in writing by the PNSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in PNSC activities as voting members at any one time.

6.5.1.4 Consultants

Consultants shall be utilized as determined by the PNSC Chairman to provide expert advice to the PNSC.

6.5.1.5 Meeting Frequency

The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman.

6.5.1.6 Quorum

A quorum of the PNSC shall consist of the Chairman or Vice Chairman plus three members including alternates.

6.5.1.7 Responsibilities

a) Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

b) Review of all proposed test and experiments that affect nuclear safety.

c) Review of all proposed changes to the Technical Specifications.

d) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e) Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Manager of Nuclear Generation and to the Chairman of the Company Nuclear Safety Committee (CNSC).

f) Review of facility operations to detect potential safety hazards.

g) Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Company Nuclear Safety Committee (CNSC).

h) Review of the Plant Security Plan and implementing procedures,

i) Review of the Emergency Plan and implementing procedures.

j) Review of all events which are required by regulations or Technical Specifications to be reported to NRC within 24 hours. 6

6.5.1.8 Authority

a) The Plant Nuclear Safety Committee shall be advisory.

b) The Plant Nuclear Safety Committee shall recommend to the Plant Manager approval or disapproval of proposals under 6.5.1.7 a) through d) above.

In the event of disagreement between the recommendations of the Plant Nuclear Safety Committee and the actions contemplated by the Plant Manager, the course determined by the Plant Manager to be more conservative will be followed with immediate notification to the Manager of Nuclear Generation and to the Chairman of the Company Nuclear Safety Committee.

c) The Plant Nuclear Safety Committee shall make determinations as to whether or not proposals considered by the Committee involve unreviewed safety questions. This determination shall be subject to review by the Company Nuclear Safety Committee as specified under 6.5.2.9.(a).

6.5.1.9 Records

Minutes shall be kept at the plant of all meetings of the Plant Nuclear Safety Committee and copies shall be sent to the Manager of Nuclear Generation and to the Chairman of the Company Nuclear Safety Committee.

6.5.1.10 Procedures

Written administrative procedures for committee operation shall be prepared and maintained.

6.5.2 Company Nuclear Safety Committee (CNSC)

The purpose of the Carolina Power & Light Company Nuclear Safety Committee is to function as an independent technical advisory group to the senior management of Carolina Power & Light Company on all matters concerning the safe performance and operation of the Company's nuclear power plants.

6.5.2.1 Technical Review Areas

The CNSC shall function to provide independent review and audit of designated activities in the areas of:

a) Nuclear power plant operations
b) Nuclear engineering
c) Chemistry and Radiochemistry
d) Metallurgy
e) Instrumentation and control
f) Radiological safety
g) Mechanical and electrical engineering
h) Quality assurance practices

6.5.2.2 Composition

The Committee shall consist of at least nine persons including:

a) Chairman
b) Vice Chairman
c) Secretary
d) Four technically qualified persons who are not members of a plant staff.
e) One member from the supervisory staff of each nuclear plant.
f) At least one qualified noncompany affiliated technical consultant and others as required. Duly appointed consultants shall have equal vote with permanent members of the committee.

Members in a) through f) above shall be designated by the Executive Vice President - Engineering, Construction and Operation Group.

6.6 Reportable Occurrence Action 6

The following actions shall be taken in the event of a reportable occurrence:

a. The NRC shall be notified and/or a report submitted by the Plant Manager pursuant to the requirements of Specification 6.11.

b. Each reportable occurrence report shall be submitted to the Manager - Nuclear Generation and the Chairman - CSNC. 6

NOV 11 1975

6.7 Safety Limit Violation

If a safety limit is exceeded, the affected reactor shall be shut down and reactor operation shall only be resumed in accordance with the authorization within 10CFR50.36(c)(1)(i). An immediate report shall be made to the Manager - Nuclear Generation, to the Chairman - CNSC, and to the Office of Inspection and Enforcement, NRC. 6

A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations to prevent a recurrence shall be prepared by the Plant Nuclear Safety Committee. This report shall be submitted to the Manager - Nuclear Generation and the Chairman, CSNC for independent review. Appropriate analyses or reports shall be submitted to the NRC by the Bulk Power Supply Department within 14 days of the violation. 6

6.11 Reporting Requirements

Information to be reported to the Commission, in addition to the reports required by Title 10, Code of Federal Regulations, shall be as indicated in the following sections. Reports shall be addressed to the Director of the appropriate Regulatory Operation Region Office unless otherwise noted.

6.11.1 Routine Reports

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. Annual Operating Report.[2,3] Routine operating reports covering the operation of the unit during the previous calendar year shall be

[2] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

[3] Much of the information in the Annual Report was previously submitted in a Semiannual Report.

submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

The primary purpose of annual operating reports is to permit annual evaluation by the NRC staff of operating and maintenance experience throughout the nuclear power industry. The annual operating reports made by licensees shall provide a comprehensive summary of the operating experience gained during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear. |6

Each annual operating report shall include, for example:

(1) A narrative summary of operating experience during the report period relating to safe operation of the facility, including safety-related maintenance not covered in 6.11.1b.(2)(e) below.

(2) For each outage or forced reduction in power[4/] of over twenty percent of design power level where the reduction extends for greater than four hours:

 (a) the proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction);

 (b) a brief discussion of (or reference to reports of) any abnormal occurrences pertaining to the outage or power reduction;

 (c) corrective action taken to reduce the probability of recurrence, if appropriate;

 (d) operating time lost as a result of the outage or power reduction (for scheduled or forced outages,[5/] use the generator off-line hours; for forced reductions in power, use the approximate duration of operation at reduced power);

4/ The term "forced reduction in power" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance and calibration activities requiring power reductions are not covered by this section.

5/ The term "forced outage" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

(e) a description of major safety-related corrective maintenance performed during the outage or power reduction, including the system and component involved and identification of the critical path activity dictating the length of the outage or power reduction; and

(f) a report of any single release of radioactivity or single 6
radiation exposure specifically associated with the outage which accounts for more than 10% of the allowable annual values.

(3) A tabulation of man rem for (Supplementing the requirements of § 20.407 of 10 CFR Part 20) the tabulated number of personnel receiving exposures greater than 100 mrem in the reporting period according to duty function, e.g., routine plant surveillance and inspection (regular duty), routine plant maintenance, special plant maintenance (describe maintenance), routine fueling operation, special refueling operation (describe operation), and other job-related exposures. Estimates of the dose assignment to various duty functions shall be based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the total dose need not be individually accounted for; however, in the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific work functions. See Appendix A to Regulatory Guide 1.16* for the 6
required format for providing this information.

(4) Findings from irradiated fuel examinations, including results of eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

c. <u>Monthly Operating Report</u>. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis. The report formats set forth in Appendices B, C, and D to Regulatory 6
Guide 1.16* shall be completed in accordance with the instructions provided. The completed forms should be submitted by the tenth of the month following the calendar month covered by the report to the Office of Inspection & Enforcement, U. S. Nuclear Regulatory 6
Commission, Washington, D. C. 20555, with a copy to the appropriate NRC Regional Office.

* Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A to Technical Specifications," Rev. 4, August 1975. 6

NOV 11 1975

6.11.2 Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of the occurrence. In cases of corrected or supplemental reports, a licensee event report should be completed and reference should be made to the 6
original report date.

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate Regulatory Operations Regional Office, or his designate no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of the licensee event report form (see Appendix E to Regulatory Guide 1.16*) 6
used for entering data into the NRC's computer-based file of information concerning licensee events. (Instructions for completing these licensee event report forms[6/] are issued individually to each licensee.) Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Failure of the reactor protection system, or other systems subject to limiting safety systems settings, to initiate the required protective function by the time a monitored parameter reaches the value specified as the limiting safety system setting in the technical specifications, or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item (but see 6.11.2.a(5), 6.11.2.a(6), and 6.11.2.b(1) below).

6/ Instruction Manual, Licensee Event Report File, Office of Management Information and Program Control, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555. 6

* Regulatory Guide 1.16, "Reporting of Operating Information Appendix A Technical Specifications," Rev. 4, August 1975.

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NOV 11 1975

(2) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the the limiting condition for operation is not considered to have been violated and no report need be submitted under this section (but see 6.11.2.b(2) below).

(3) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary or primary containment.

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this section.

(4) Reactivity anomalies involving disagreement with predicted value of reactivity balance under steady state conditions greater than or
equal to 1% Δk/k; a calculated reactivity balance indicating shutdown 6
margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds, or if subcritical, an unplanned reactivity
insertion of more than 0.5% Δk/k; or any unplanned criticality. 6

(5) Failure or malfunction of one or more components, which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

(6) Presonnel error or precedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

Note: For 6.11.2.a(5) and 6.11.2.a(6) reduced redundancy that does not result in loss of system function need not be reported under this section (but see 6.11.2.b(2) and 6.11.2.b(3) below).

(7) Conditions arising from natural or man-made events that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

(8) Errors discovered in the transient or accident analyses or in the

NOV 11 1975

methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(9) Performance of structures, systems, or components that require remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

b. Thirty-Day Written Reports. The reportable occurrences discussed below shall be the subject of written 6
reports to the Director of the appropriate Regulatory Operations Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of the licensee event report form (see Appendix E to Regulatory Guide 1.16* used for entering data into the NRC's computer-based file of information concerning licensee events. (Instructions for completing these licensee event report forms[6/] are issued individually to each licensee.) Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems (but see 6.11.2.a(1) and 6.11.2.a(2) above).

* Regulatory Guide 1.16, "Reporting of Operating Information Appendix A Technical Specifications," Rev. 4, August 1975.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation (but see 6.11.2.a(2) above).

Note: Routine surveillance testing, instrument calibration, or preventive maintenance which require system configurations as described in 6.11.2.b(1) and 6.11.2.b(2) above need not be reported except where test results are not satisfactory.

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems (but see 6.11.2.a(6) above).

(4) Abnormal degradation of systems other than those specified in 6.11.2.a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in 6
technical specifications need not be reported under this item.

6 This page has been deleted.

NOV 1 1 1975

6.11.4 Special Reports

a. Reports on the following areas shall be submitted as noted:

	Area	Submittal Date
a)	Initial containment structural test[1]	Within three months following completion of test 6
b)	In-service inspection evaluation	Within three months of completion of five years of operation

b. An analysis and report shall be submitted to the AEC on all surveillance specimens removed form the reactor vessel as described in Appendix I. These reports shall include: 1) the information specified in ASTM E-185-66, "Recommended Practices for Surveillance Tests on Structural Materials in Nuclear Reactors", 2) information obtained on the level of integrated fast neutron irradiation received by the specimens and the actual vessel material; and 3) revised limitations on startup, cooldown and operating conditions as designed by Technical Specifications.

c. Other Reports

(1) Annually: Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination. 6

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(1) The initial containment structural test shall be the subject of a report which includes summary of measurements of deflections, strains, crack width, crack patterns observed, and detailed observations around tendon anchorage zones, as well as comparisons with predicted values of acceptance criteria.

NOV 11 1975

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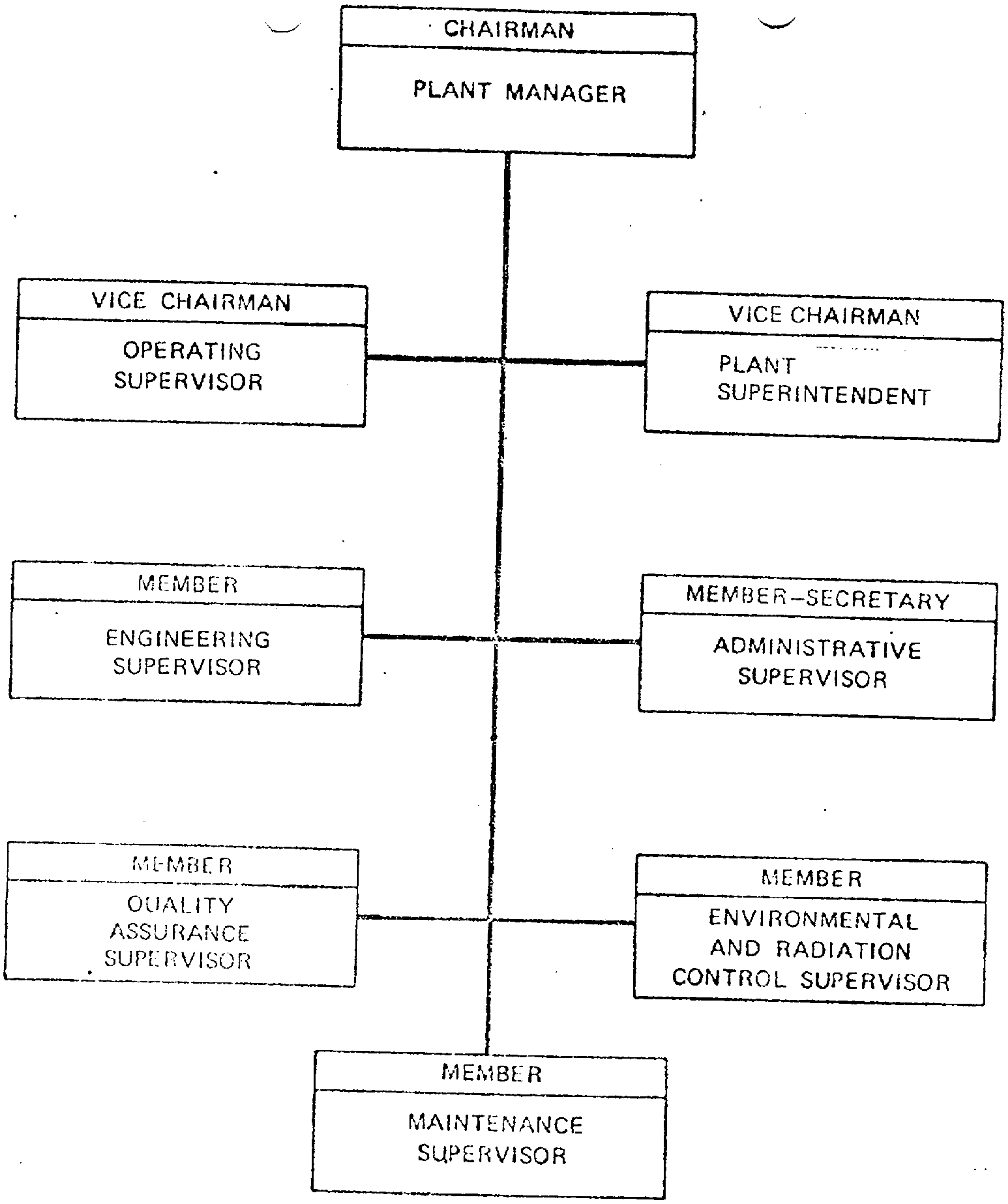

NOTES:

(1) ANY THREE MEMBERS OR DESIGNATED ALTERNATES PLUS CHAIRMAN OR VICE CHAIRMAN CONSTITUTES A QUORUM.

(2) MEETINGS HELD AT LEAST ONCE A MONTH

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT
UNITS 1 & 2
Final Safety Analysis Report

ORGANIZATION OF PLANT NUCLEAR SAFETY COMMITTEE

FIG. NO. 6.5–1

NOV 11 1975

CHAIRMAN
DIRECTOR OF TECHNICAL SERVICES

VICE CHAIRMAN
DIRECTOR REASEARCH SECTION

MEMBER
VICE PRES. POWER PLANT ENGINEERING DEPARTMENT

SECRETARY
SENIOR ENGINEER TECHNICAL SERVICES UNIT

MEMBER
VICE PRESIDENT BULK POWER SUPPLY DEPARTMENT

MEMBER
PRINCIPAL RADIATION CONTROL ENGINEER NUCLEAR GENERATION SECTION

MEMBER
NUCLEAR SAFETY CONSULTANT

MEMBER
MANAGER NUCLEAR GENERATION SECTION

MEMBERS
PLANT SUPERVISORY STAFF MEMBERS

MEMBER
MANAGER LICENSING & TECHNOLOGICAL SERVICES SECTION

MEMBER
MANAGER NUCLEAR ENGINEERING SECTION

MEMBER
MANAGER FUEL SECTION

AMENDMENT 26
SEPTEMBER 1974

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT
UNITS 1 & 2
Final Safety Analysis Report

COMPANY NUCLEAR SAFETY COMMITTEE

FIG. NO. 6.5–2

NOV 11 1975

6.13 Unit Operating Records Retention

6.13.1 Facility records shall be retained in accordance with ANSI-N45.2.9-1974.

6.13.2 The following records shall be retained for at least five years:

a. Records of sealed source leak tests and results.

b. Records of annual physical inventory of all source material of record.

6.13.3 The following records shall be retained for the duration of the Facility Operating License:

a. Records of review, performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

b. Transient or operational cycling for particular plant components. Records of transient or operational cycling for those plant components that have been designed to operate safely for a limited number of transients or operational cycles.

(1) Design fatigue usage evaluation - Monitoring, recording, evaluating, and reporting requirements will be met for various portions of the reactor coolant pressure boundary for which detailed fatigue usage evaluation per the ASME Boiler and Pressure Vessel Code Section III was performed for the conditions defined in the specification. The locations to be monitored shall be:

(a) The feedwater nozzles.

(b) The shell at or near the water line.

(c) The flange studs.

(2) Monitoring, recording, evaluating and reporting

(a) Operational transients that occur during plant operations will, at least semi-annully, be reviewed and compared to the number of transients defined in the component stress report for the locations listed in (1) above, and used as a basis for fatigue evaluation.

(b) The number of transients which are comparable to or more severe than the transients evaluated in the stress report will be re-
6 corded in the annual operating report. For those transients which are more severe, available data, such as the metal and fluid temperatures, pressures, flow rates, and other conditions
6 will be recorded in the annual operating report.

(c) The number of transient events that exceed the design specification quantity and the number of transient events with a severity greater than that included in the existing Code fatigue usage calculations shall be added. When this sum exceeds the predicted number of design condition events by twenty-five[1], a fatigue usage evaluation of such events will be performed for the affected portion of the RCPB.

[1] The Code rules permit exclusion of twenty-five (25) stress cycles from secondary stress and fatigue usage evaluation. (See paragraphs N-412(t)(3) and N-417.10(f) of the Summer 1968 Addenda to ASME Section III, 1968 Edition).

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 7 TO FACILITY LICENSE NO. DPR-62

CHANGE NO. 6 TO TECHNICAL SPECIFICATIONS

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-324

Introduction

By letter dated October 14, 1975, Carolina Power & Light Company (the licensee) proposed changes to the Technical Specifications appended to Facility Operating License No. DPR-62, for the Brunswick Steam Electric Plant, Unit No. 2. The proposed changes involve changes to the administrative controls and changes to the reporting requirements.

Discussion

The proposed changes to the administrative controls involve modifications to both the on-site and off-site review committees. The licensee proposes to add the Plant Superintendent to the Plant Nuclear Safety Committee (PNSC) as Vice Chairman. The present specification designates the Operating Supervisor as Vice Chairman; the Plant Superintendent is not a PNSC member. With the requested change then, the PNSC would have an additional member and two assigned Vice Chairmen.

Our evaluation of the changes requested to the structure of the off-site review and audit function has not been included here since our review of this aspect of the proposal of October 14, 1975, has not been completed.

The balance of the requested changes involve changes in reporting requirements and a definition change for "abnormal occurrence".

In Section 208 of the Energy Reorganization Act of 1974 "abnormal occurrences" is defined as an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety. The term "abnormal occurrence" is reserved for usage by NRC. Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4, enumerates required

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reports consistent with Section 208. The proposed change to required reports identifies the reports required of all licensees not already identified by the regulations and those unique to this facility. The proposal would formalize present reporting and would delete any reports no longer needed for assessment of safety related activities.

Evaluation

The new guidance for reporting operating information does not identify any event as an "abnormal occurrence". The proposed reporting requirements also delete reporting of information no longer required and duplication of reported information. The standardization of required reports and desired format for the information will permit more rapid recognition of potential problems.

During our review of the proposed changes, we found that certain modifications to the proposal were necessary to have conformance with the desired regulatory position. These changes were discussed with the licensee's staff and have been incorporated into the proposal.

Provisions for review of facility operations should be in accord with Regulatory Guide 1.33, "Quality Assurance Program Requirements", which endorses proposed standard ANS 3.2, which was subsequently issued as ANSI N18.7-1972. The licensee's proposal to add the Plant Superintendent (who reports to the Plant Manager) to the Plant Nuclear Safety Committee as Vice Chairman is consistent with ANSI N18.7 and is acceptable.

We have concluded that the proposal as modified improves the licensee's program for evaluating plant performance and the reporting of the operating information needed by the Commission to assess safety related activities and is acceptable. The modified reporting program is consistent with the guidance provided by Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4.

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Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: NOV 11 1975

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-324

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 7 to Facility Operating License No. DPR-62 issued to Carolina Power & Light Company for operation of the Brunswick Steam Electric Plant, Unit 2, located in Brunswick County, North Carolina. The amendment is effective 30 days after the date of issuance.

This amendment revises the provisions in the Technical Specifications relating to Reporting Requirements and changes the composition of the Plant Nuclear Safety Committee by adding the Plant Superintendent as Vice Chairman.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated October 14, 1975, (2) Amendment No. 7 to License No. DPR-62, with Change No. 8 , and (3) the Commission's related

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Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Southport-Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this NOV 11 1975

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

10/31/75

OFFICE➤	RL:ORB#1	OELD	RL:ORB#1			
SURNAME➤	CTrammell:1b		RAPurple			
DATE➤	10/31/75	10/ /75	10/ /75			

→ S. Ley ← attach a copy of this to the yellows of all our packages on this subject.

ROUTING AND TRANSMITTAL SLIP

TO (Name, office symbol or location)	INITIALS / DATE	ACTION
1 OELD - f/concurrences	INITIALS / DATE	CIRCULATE / COORDINATION
2 DLZiemann - f/signatures	INITIALS / DATE	FILE / INFORMATION
3 Reba - for final checks	INITIALS / DATE	NOTE AND RETURN / PER CONVERSATION
4	INITIALS / DATE	SEE ME / SIGNATURE

REMARKS

Attached for your concurrence are five packages (Dresden Station, Quad Cities Station, Cooper, Pilgrim and Calvert Cliffs) of nine from ORB 2 which incorporate standard reporting requirement sections into the Appendix A Technical Specifications. One package, Pilgrim also revises the entire administrative controls section.

It is requested that, in the interest of review consistency, these packages (and the 4 future reporting requirements packages) be assigned to one OELD reviewer.

Questions may be directed to the PM for the particular case or to Mike Fletcher, coordinator for reporting. (Exts. 7403, 7450)

11/3/75 No need for OELD concurrence this time on this subject

Do NOT use this form as a RECORD of approvals, concurrences, disapprovals, clearances, and similar actions

FROM (Name, office symbol or location)	DATE
DLZiemann DLZ	11-3-75
	PHONE 7380

11/3