50-325/324

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 2, 1997

Mr. C. S. Hinnant, Vice President
Carolina Power & Light Company
Brunswick Steam Electric Plant
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: ISSUANCE OF AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 221 TO FACILITY OPERATING LICENSE NO. DPR-62 REGARDING CORRECTION OF A LICENSING BASIS ASSUMPTION FOR SPENT FUEL SHIPPING CASK HANDLING - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. M99490 AND M99491)

Dear Mr. Hinnant:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 190 to Facility Operating License No. DPR-71 and Amendment No. 221 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant Units 1 and 2 (BSEP). The amendments are in response to Carolina Power & Light Company's (CP&L's) application dated August 6, 1997, and correct an assumption used in the analysis of the credibility of a spent fuel shipping cask drop event.

Previously a load drop involving the spent fuel shipping cask was determined not to be a credible event based upon use of redundant primary and secondary lifting yokes. Carolina Power & Light Company (CP&L) recently recognized that, during a portion of the cask handling evolution, only the primary yoke is utilized. While this actual practice is consistent with the description of cask handling provided in the Updated Final Safety Analysis Report (UFSAR), it is not consistent with the redundant design described to the NRC in a CP&L letter dated November 16, 1982. Therefore CP&L determined that it constituted an unreviewed safety question (USQ).

CP&L reanalyzed the credibility of a fuel shipping cask drop event using the correct assumption for lifting yoke redundancy and concluded that BSEP did not operate outside its design basis because the design basis function of preventing a load drop was not compromised due to the inherent safety factors of the load handling system, previous load tests, and periodic inspections.

The conclusion that both prior and current analyses demonstrate that a cask drop event is not credible does not in itself justify the use of a different load handling design. The specific assumptions used in the analyses, such as the redundancy of the lifting yokes, are also important considerations. The fact that a portion of cask handling system operation had not been properly evaluated by CP&L and reviewed by the NRC is a matter of concern and is under review for possible enforcement action.

The NRC staff has reviewed the analysis and conclusions presented by CP&L in the August 6, 1997, submittal and, based on these findings and on the fact

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that lifting the cask using the non-redundant yoke is confined to a relatively short section of the load path, the staff concludes that failure of the yoke is not credible.

The staff noted that once the USQ was identified by your organization, CP&L took appropriate action, including the performance of a detailed evaluation, and submitted the subject amendment request within a relatively short period of time.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register Notice.

Sincerely,

(Original Signed By)

David C. Trimble, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-325
and 50-324

Enclosures:
1. Amendment No. **190** to License No. DPR-71
2. Amendment No. **221** to License No. DPR-62
3. Safety Evaluation

cc w/enclosures: See next

FILENAME - G:\BRUNSWIC\BR99490.AMD *See previous concurrence

PM:PDII-1	LA:PDII-1*	EMCB*	SPLB*	OGC RMW No legal objection	PD:II-1
DTrimble DT	EDunnington	ESullivan		RWeisman	JLyons
12/2/97	11/19 /97	11/14/97	12/1 /97	12/1 /97	12/2 /97
Yes/No	Yes/No	Yes/No	yes	Yes/No	Yes/No

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AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO. 221 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
License No. DPR-71

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated August 6, 1997, 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;

   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; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended to authorize changes to the Updated Final Safety Analysis Report to reflect the analysis of the credibility of a spent fuel shipping cask drop event as set forth in the application for amendment by Carolina Power & Light Company dated August 6, 1997.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

J. E. Lyons

James E. Lyons, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Date of Issuance: **December 2, 1997**

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 221
License No. DPR-62

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated August 6, 1997, 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;

   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; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended to authorize changes to the Updated Final Safety Analysis Report to reflect the analysis of the credibility of a spent fuel shipping cask drop event as set forth in the application for amendment by Carolina Power & Light Company dated August 6, 1997.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

J. E. Lyons

James E. Lyons, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Date of Issuance: **December 2, 1997**

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 221 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated August 6, 1997, Carolina Power and Light Company (CP&L)(the licensee), submitted a license amendment request pursuant to 10 CFR 50.90 for approval of changes to the Updated Final Safety Analysis Report (UFSAR). The proposed changes relate to the cask handling operations involving lifting Vectra IF-300 spent fuel shipping casks at the Brunswick Steam Electric Plant (BSEP), Units 1 and 2.

The requested UFSAR changes are due to a discrepancy between the licensing basis and site procedures regarding the handling and control of casks. The site procedures require casks to be lifted over a short section of the load path without the single failure proof lifting system and with the cask valve box covers removed. Past correspondence to the NRC indicates that the same lift is made with single failure proof capability and with the valve box covers installed.

Through Licensee Event Report Number (LER) 97-004, dated June 5, 1997, CP&L informed the NRC of the discrepancy between the site procedures and the UFSAR regarding the movement of a spent fuel shipping cask. In the LER, CP&L indicated that the cask handling operations were performed under conditions that are outside the design basis and involve an unreviewed safety question (USQ). In the August 6, 1997, submittal, CP&L provided a detailed analysis pursuant to 10 CFR 50.59 regarding a postulated cask drop accident to support its conclusion that the cask handling operation does involve a USQ but is not outside the design basis of the plant.

2.0 BACKGROUND

In a letter dated November 16, 1982, CP&L informed the NRC that BSEP's reactor building crane design satisfied the requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." In that letter and other correspondence provided to the NRC, dated June 18, 1976, and June 22, 1981, CP&L indicated that BSEP uses a lifting device consisting of a yoke (primary yoke) and a lifting basket (a secondary yoke) which are of "redundant" design to lift spent fuel shipping casks. CP&L also indicated that these components are used in conjunction with a single failure proof crane during transfer of the casks.

Based on these capabilities, CP&L concluded that a load drop of a cask is not credible.

NRC Technical Evaluation Report/Safety Evaluation, dated May 18, 1984, approved CP&L's load handling operations, and acknowledged that BSEP's special lifting devices satisfy the single failure proof guidelines as provided by NUREG-0612 and the requirements of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds or More for Nuclear Materials." Therefore, the staff accepted the licensee's position that a load drop of the cask is not credible.

NRC Bulletin (NRCB) 96-02, dated April 11, 1996, requests that all licensees review their plans and capabilities for handling heavy loads while the reactor is at power in accordance with existing regulatory guidelines. The bulletin also requests that licensees determine whether their current heavy loads handling activities are within their current licensing basis and, if not, submit a license amendment request. In addition, a Request for Additional Information (RAI), dated December 6, 1996, regarding further evaluation of load handling activities was issued to BSEP. This evaluation was to determine if a cask tipping-over hazard exists while dry casks are being moved by plant cranes.

BSEP responded by letters dated May 10, 1996, and February 7, 1997, to NRCB 96-02 and the associated RAI, respectively. In both responses, BSEP indicated that the cask drop and cask tipping-over hazards are not credible at the Brunswick plant due to the redundant design of the special lifting device, the single failure proof upgrade of the crane, and BSEP's compliance with NUREG-0612. Again, the staff accepted BSEP's conclusion that a cask drop is not credible.

BSEP's recent discovery that there is a discrepancy between its procedures and existing analysis is contrary to information previously provided to the NRC that indicated that single failure proof capability is used throughout the entire cask handling operation.

3.0 EVALUATION

3.1 Analysis of Postulated Cask Drop Accident

The cask lift occurs at the 20' elevation (ground level) of the Reactor Building at the lower level of the equipment hatch with a lift height of approximately 7' above the rail car. As the cask is raised to the vertical position from a horizontal position and lifted into the cask lifting basket, redundancy in the design of the yoke does not exist. Therefore, the yoke is not single failure proof in accordance with Section 5.1.6 of NUREG-0612. Once the cask is in the lifting basket, the "redundant" yoke is used to lift the assembly to the 117' elevation. All the components of the yoke including the cross members, arms, and J-hooks are made of ASTM A514 material except for the yoke pin which is made of ASTM 4340 steel. The rated load of the yoke is 140,000 lbs. and the Vectra IF-300 cask loaded with spent fuel weighs approximately 140,000 lbs.

NUREG-0612 Section 5.1.6, "Single Failure Proof Handling System," provides the alternative of upgrading an existing crane in lieu of complying with certain recommendations of NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," to achieve improved reliability in a load handling system. Accordingly, upgrades to the crane and associated lifting devices can be achieved through an increase in the factors of safety, and through redundancy or duality in certain active components.

Because ANSI N14.6, Section 6 allows increased stress factors to be used to demonstrate reliability where redundancy is not an inherent part of the design of the yoke, the licensee analyzed the failure potential of the components of the yoke in accordance with the standard. ANSI N14.6 requires that special lifting devices be designed to a safety factor of three with respect to yield strength, and five with respect to the ultimate strength of the material. NUREG-0612, Section 5.1.6(1)(a) requires that these factors be doubled for special lifting devices that are used to handle heavy loads where an upgraded crane is involved. With respect to the yield and ultimate strengths of the components of the yoke, the following compares the licensee's findings with regard to the nonredundant yoke to the requirements of NUREG-0612.

	Yield strength	NUREG /ANSI	Ultimate strength	NUREG /ANSI
Cross members	7.92	6	9.24	10
Arms	20.52	"	23.94	"
J-hooks	3.63	"	4.24	"
Yoke pin	6.21	"	**	"
Worst case welds	6.18.	"	**	"

** The ultimate strength cannot be demonstrated or is not available.

Based on the above, the safety factors of the cross members and arms approximate or exceed that required by the standard. With regard to the J-hooks, the licensee used stress analysis to demonstrate that the J-hooks would not fail under design basis earthquake conditions.

The licensee found that the cross members, arms, and J-hooks of the non-redundant yoke have yield strengths in excess of 80% of the ultimate strength of the material. As a result, the licensee based its acceptability of these components on the material fracture toughness as is allowed by ANSI N14.6, Section 4.2.1.1. Based on the fracture toughness analysis of these components, the licensee found that unstable cracking propagation throughout the material due to maximum loading of the components will not occur. Thus, the licensee concluded that a cask drop due to a failure of the components of the yoke is not credible.

Since the safety factors regarding the ultimate strength of the yoke pin and worst case welds could not be demonstrated, the licensee evaluated these components based on safety margins and inspections as is permitted by NUREG-0612, Section 5.1.1(4). Based on this evaluation, the licensee found that prior load tests (original and post modification tests) had been performed on the yoke at 200% of its rated load of 140,000 lbs (i.e., at 280,000 lbs.). Since ANSI N14.6, Section 6.2 requires that after fabrication and prior to initial use, special lifting devices will be load-tested at 150% of the rated load, BSEP's prior load tests on the yoke exceeded the requirements of the standard.

In addition to the above analyses, the licensee performed a walk-down of the systems located within the load path and verified that, due to the availability of redundant safe shutdown systems, shutdown of the plant can be achieved following a dropped cask. Accordingly, the licensee concluded that the potential damage that could result from a cask drop would not preclude the operation of sufficient equipment needed to achieve safe shutdown.

Based on the analyses discussed above, the licensee reviewed its crane design, the strength of materials for the components of the lifting yoke, prior load tests, the load lifting history, previous inspection records, and cask loading and unloading processes and concluded that a cask drop is not credible. These findings, as supported by the licensee's analysis, are acceptable.

3.2 Changes to UFSAR Sections 9.1.4.2.2 and 9.1.4.2.3.2

The licensee proposed to change UFSAR Sections 9.1.4.2.2, "Spent Fuel Shipping Cask Handling," and 9.1.4.2.3.2, "Spent Fuel Shipping Cask Hoisting." The changes are intended to make the licensing basis consistent with the site procedures governing the load handling operations. The changes clearly indicate when the non-redundant and redundant yokes are used to lift the cask. Changing the UFSAR to more accurately reflect actual plant conditions during the transfer of the cask will help to assure compliance with and operation within the plant-specific design basis. Therefore, the proposed changes to the UFSAR are acceptable to the staff.

4.0 Results Of Staff Review

Based on the above discussions, the staff finds that the licensee has appropriately proposed changes to the UFSAR to make its load handling operation consistent with BSEP's design basis. In support of the proposed changes, the licensee has evaluated the potential for a cask drop accident using the non-redundant yoke. Through analysis, the licensee has demonstrated that the components of the non-redundant yoke should not fail.

In addition, the licensee has reviewed the results of prior nondestructive examinations and inspections of the worst case welds and finds that no prior material defects had been observed. Also, the lift history of the system, and the use of qualified operators and adequate load handling and inspection procedures further assure the licensee that the crane and lifting device will not experience failure without prior detection. Based on these findings, and on the fact that lifting the cask using the non-redundant yoke is confined to a relatively short section of the load path, the staff concludes that failure of the yoke is not credible.

In addition, the staff concludes that BSEP has been operating outside the licensing bases for the facility because its load handling operations do not conform with information provided to the NRC in CP&L's November 16, 1982, letter.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 48897). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Brian E. Thomas
Date: **December 2, 1997**