All workers engaged in the inspection operation of the printed circuit board assemblies at Lockheed Martin, Ocean, Radar & Sensor Systems Division located in Utica, New York who become totally or partially separated from employment on or after October 5, 1994 are eligible to apply for adjustment assistance under Section 223 of the Trade Act of 1994.

Signed at Washington, D.C. this 1st day of November 1995.

Russell T. Kile,

Acting Program Manager, Policy and Reemployment Services, Office of Trade Adjustment Assistance.

[FR Doc. 95–30155 Filed 12–11–95; 8:45 am]

BILLING CODE 4510–30–M

[NAFTA–00358]

Sun Apparel, Inc., El Paso, Texas; Amended Certification Regarding Eligibility To Apply for NAFTA Transitional Adjustment Assistance

In accordance with Section 250(a), Subchapter D, Chapter 2, Title II, of the Trade Act of 1974, as amended (19 U.S.C. 2273), the Department of Labor issued a Certification for NAFTA Transitional Adjustment Assistance on March 10, 1955, applicable to all workers of Sun Apparel, Inc., Concepcion Plant located in El Paso, Texas. The notice was published in the Federal Register on March 22, 1995 (60 FR 15164).

At the request of the State Agency, the Department reviewed the subject certification. New findings show that worker separations have occurred at the Sun Apparel's Armour Plant in El Paso. The workers at the Armour Plant, like the Concepcion Plant, are engaged in employment related to the production of jeans. The intent of the Department's certification is to include all workers of the subject firm who were adversely affected by increased imports from Mexico or Canada. Therefore, the Department is amending the certification to expand coverage to all workers of Sun Apparel in El Paso, not just those workers at the Concepcion Plant.

The amended notice applicable to NAFTA–00358 is hereby issued as follows:

All workers of Sun Apparel, El Paso, Texas who became totally or partially separated from employment on or after February 2, 1994 are eligible to apply for NAFTA–TAA Section 250 of the Trade Act of 1974.

Signed at Washington, D.C., this 29th day of November 1995.

Russell T. Kile,

Acting Program Manager, Policy and Reemployment Services, Office of Trade Adjustment Assistance.

[FR Doc. 95–30148 Filed 12–11–95; 8:45 am]

BILLING CODE 4510–30–M

NATIONAL SCIENCE FOUNDATION

Collection of Information Submitted for OMB Review

In compliance with the requirement of Section 3506(c)(2)(A) of the Paperwork Reduction Act of 1995, on October 4, 19985, Federal Register No. 192, page 52024, the National Science Foundation (NSF) published, for public comment, a proposed collection of information, ''Survey of Industrial Research and Development, 1995–97.'' No public comments were received. The collection of information is now being forwarded to the Office of Management and Budget for consideration. To request more information on the proposed project or to obtain a copy of the data collection plans and instruments, call Herman Fleming, NSF Clearance Officer at (703) 306–1243, or send comments to: National Science Foundation, 4201 Wilson Boulevard, Suite 485, Arlington, VA 22230.

Written comments should be received by January 5, 1995.

Abstract: This survey measures the amount and indicates the direction of R&D expenditures by U.S. industry, Government agencies, corporations, academic researchers, trade associations, research organizations, and others use the survey statistics to analyze and forecast technological growth, investigate productivity determinants, formulate tax policies, and compare individual company performance with industry averages.

Companies with known R&D activity and samples of companies in selected industries that may conduct R&D are included.

Dated: December 7, 1995.

Herman G. Fleming,

NSF Clearance Officer.

[FR Doc. 95–30229 Filed 12–11–95; 8:45 am]

BILLING CODE 7555–01–M

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–309, 50–285, 50–317, 50–318, 50–336, and 50–335]

Maine Yankee Atomic Power Co., Omaha Public Power District, Baltimore Gas and Electric Co., Northeast Nuclear Energy Co., and Florida Power & Light Co.; Maine Yankee, Fort Calhoun Unit 1, Calvert Cliffs Units 1 and 2, Millstone Unit 2, and St. Lucie Unit 1; Issuance of Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has taken action with regard to a Petition dated May 2, 1995, by Mr. John F. Doherty, J.D. (Petition for action under 10 CFR 2.206). The Petition pertains to the following plants: Maine Yankee, Fort Calhoun Unit 1, Calvert Cliffs Units 1 and 2, Millstone Unit 2, and St. Lucie Unit 1.

In the Petition, Petitioner requested that the following six pressurized-water reactors be immediately shut down: Maine Yankee, Fort Calhoun Unit 1, Calvert Cliffs Units 1 and 2, Millstone Unit 2, and St. Lucie Unit 1. In addition, the Petitioner requested that steam generator tubes be inspected immediately at those plants.

The Director of the Office of Nuclear Reactor Regulation has determined to deny the Petition. The reasons for this denial are explained in the ''Director's Decision Pursuant to 10 CFR 2.206'' (DD–95–22), the complete text of which follows this notice, and is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

A copy of the Decision will be filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206(c) of the Commission's regulations. As provided by this regulation, the Decision will constitute the final action of the Commission 25 days after the date of issuance unless the Commission, on its own motion, institutes a review of the Decision within that time.

Dated at Rockville, Maryland, this 6th day of December, 1995.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

Office of Nuclear Reactor Regulation, William T. Russell, Director

In the Matter of: Maine Yankee Atomic Power Co., Omaha Public Power District, Baltimore Gas and Electric Co., Northeast

Nuclear Energy Co., Florida Power & Light Co. (Maine Yankee, Fort Calhoun Unit 1, Calvert Cliffs Units 1 and 2, Millstone Unit 2, and St. Lucie Unit 1). Docket Nos. 50–309, 50–285, 50–317, 50–318, 50–336, and 50–335. License Nos. DPR–36, DPR–40, DPR–53, DPR–69, DPR–65, DPR–67.

I. Introduction

On May 2, 1995, Mr. John F. Doherty, J.D. (Petitioner), filed a Petition with the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 2.206. The Petitioner requested that the following six pressurized-water reactors be immediately shut down: Maine Yankee, Fort Calhoun Unit 1, Calvert Cliffs Units 1 and 2, Millstone Unit 2, and St. Lucie Unit 1. In addition, the Petitioner requested that steam generator tubes be inspected immediately at those plants. The Petitioner stated that an inspection by the license in April 1995 of the Maine Yankee plant using the newly developed Point Plus system revealed that the steam generator tubes are on the verge of rupture, threatening the release of radioactive liquid and gaseous material into the environment and consequent harm to human health and safety. Because the other plants the Petitioner identified were built by the same manufacturer (Combustion Engineering) and are of similar operating age, the Petitioner asked that they, along with the Maine Yankee, be immediately shut down and that all steam generator tubes be immediately inspected using the Point Plus Probe system.

On June 28, 1995, I informed the Petitioner that the Petition had been referred to my office for preparation of a Director's Decision. I further informed the Petitioner that his request for immediate shutdown and inspection was denied because continued operation of these units until their next scheduled outage posed no undue risk to public health and safety. I also informed the Petitioner that the NRC would take appropriate action within a reasonable time.

II. Discussion

The Petitioner requested that six CE-designed plants be shut down and their steam generator tubes inspected with the Plus Point inspection probe. The request appears to be based on concerns that without inspections using the Plus Point probe, the steam generators in these plants may be susceptible to one or more steam generator tube ruptures (SGTRs). However, the results of examinations of tubes removed from the Maine Yankee steam generators and in situ pressure tests of the most severely degraded tubes in the Maine Yankee steam generators have demonstrated that the tubes, although severely degraded, still had a significant margin before failure even under postulated accident conditions. Furthermore, the NRC has taken actions to ensure that other plants have performed appropriate steam generator tube inspections to assure tube integrity. These important actions are discussed below in greater detail.

The NRC applies a defense-in-depth approach toward protecting public health and safety from the potential consequences of events involving the rupture of steam generator tubes. Steam generator tube degradation is managed through a combination of several different elements, including inservice inspection, tube repair criteria, primary-to-secondary leak rate monitoring, water chemistry, and analyses to ensure safety objectives are met.

The primary means for assessing steam generator tube degradation is through inservice inspections. Plant technical specifications require a periodic inspection of the steam generator tubes. Any tubes with identified degradation in excess of the repair criteria are repaired or removed from service. In order to assess the condition of steam generator tubing, the industry primarily relies on eddy current inspection techniques, which includes the motorized rotating pancake coil (MRPC) test. Circumferential cracking in steam generator tubing has been identified at expansion transitions, small radius U-bends, dented tube support plate intersections, and sleeved joints. Based on the utilities' responses to GL 95–03, the inservice CE steam generators (i.e., not including retired CE steam generators) have been inspected in these areas with techniques capable of detecting circumferential cracking and, to date, such cracking was found only at the expansion transitions.

Experience to date, including experience at the Maine Yankee plant, shows that the standard MRPC probe is a reliable means for detecting structurally significant cracking in steam generator tubes. The use of an MRPC probe in conjunction with adequate inspection procedures is a reliable means for detecting circumferential cracking in steam generator tubes. As discussed above, metallographic examinations of removed tubing and in situ pressure testing of degraded tubes continue to support the staff's conclusion that properly conducted MRPC inspections can identify circumferential cracking before the cracking exceeds the structural limits.

In addition to requiring periodic steam generator tube inspections, the NRC requires an operational leak rate limit to provide reasonable assurance that should a primary-to-secondary leak be experienced during service, it will be detected and the plant will be shut down in a timely manner before rupture occurs and with no undue risk to public health or safety. Requiring operation within these limits decreases the possibility that steam generators may be vulnerable to tube ruptures during postulated accidents such as a main steamline break or a loss-of-coolant accident.

Inspection findings at Maine Yankee in 1994 revealed indications of large circumferential cracks that had been missed in previous inspections because of inadequacies in MRPC test and analysis procedures. The test and analysis procedures were upgraded accordingly. However, subsequent inspections at Maine Yankee performed with the MRPC in early 1995 revealed circumferential indications that were more numerous and larger than expected based on the short operating interval since the previous inspection. The 100-percent MRPC inspection of the expansion transitions were supplemented by inspections with the recently developed Plus Point probe and a specially wound high-frequency MRPC coil. These latter probes offer improved sensitivity to inner-diameter-initiated circumferential cracks of the type present at the Maine Yankee expansion transitions and identified substantial numbers of relatively small circumferential cracks not detected with the conventional MRPC.

Three tubes were removed from these steam generators in early 1995. Before the tubes were removed, they were tested by ultrasonic, visual (fluorescent penetrant dye), and eddy current techniques to confirm the nature of the indications. Eddy current methods included examination with a standard rotating pancake coil, a Plus Point coil, and a high-frequency pancake coil. The indications were sized with various techniques and the tubes were then destructively examined so that the actual size of the indications could be determined. The results of the destructive examinations are provided in NRC Information Notice 95–40, ''Supplemental Information Pertaining to Generic Letter 95–03, 'Circumferential Cracking of Steam Generator Tubes.' '' The destructive examination results and data obtained with a high-frequency pancake coil suggest that many of the indications may not have been as structurally significant as the standard pancake coil appeared to indicate.

In situ pressure tests were conducted on the tubes with the largest MRPC indications and the results indicate acceptable margins against burst under normal operating and postulated accident conditions. The NRC had a review conducted by an independent contractor of the in situ test method used at Maine Yankee and determined that it provides a reasonable simulation of the hydraulic pressure loads induced during a postulated main steamline break.

Thus, it has been demonstrated that the tubes with the largest indications at Maine Yankee continued to exhibit adequate structural integrity at the time they were found. This finding is attributable to the morphology of the cracks as determined from metallographic examinations of pulled tube specimens from Maine Yankee. This morphology consists of cracks that were not coplanar but rather of short circumferential length and staggered around the circumference over a short axial region with ligaments of material between the cracks. These ligaments add considerably to the strength of the tube, but these ligaments are generally not detectable by the MRPC.

The findings at Maine Yankee nevertheless raised concern that large undetected circumferential cracks could possibly exist at other plants. Therefore, the NRC issued Generic Letter (GL) 95–03, ''Circumferential Cracking of Steam Generator Tubes,'' on April 28, 1995, notifying licensees of the Maine Yankee experience and requesting that they evaluate recent operating experience concerning the detection and sizing of circumferential cracks and the potential applicability of this experience to their plants. On the basis of the results of this evaluation, past inspections and the results thereof, and other relevant factors, licensees were requested to develop a safety assessment justifying continued operation until the next scheduled steam generator tube inspections were to be performed. The generic letter also requested that licensees develop and submit their plans for the next steam generator tube inspection as they pertain to the detection of circumferential cracks. The utilities were required to respond to GL 95–03 within 60 days. By now, the utilities that own the six plants listed in the Petition have responded to GL 95–03 and the responses have been evaluated by the staff.

Based on the utilities' responses to GL 95–03, with the exception of Millstone Unit 2, the CE plants listed in the Petition have been inspected in those areas susceptible to circumferential cracking with improved eddy current inspection probes equally capable as the Point Plus system of detecting circumferential cracking. All tubes with detected cracks have been removed from service. The licensee for Millstone Unit 2 replaced the original CE steam generators during an outage that ended in January 1993. The new steam generators incorporated many new design features that are expected to eliminate or greatly reduce the potential for circumferential tube cracking. These include the use of Inconel 690, a material that has significantly greater resistance to cracking and hydraulic expansion of tubes, which reduces the potential for cracking in the expansion transitions. The limited operational time, improvements in design, and favorable plant operating conditions minimize the potential for the development of circumferential cracking in the Millstone Unit 2 steam generators. Millstone Unit 2 steam generators will continue to be inspected during refueling outages.

The NRC has studied the risk and potential consequences of a range of SGTR events in NUREG–0844, ''NRC Integrated Program for the Resolution of Unresolved Safety Issues A–3, A–4, and A–5 Regarding Steam Generator Tube Integrity.'' The staff estimated the risk contribution due to the potential for single and multiple SGTRs. The study also examined the expected consequences of SGTR scenarios, including beyond-design-basis situations, such as the potential for release as a result of containment bypass because of failed tubes concurrent with a breach of secondary system integrity. A combination of circumstances and conditions is required to produce such simultaneous failures: (1) Main steamline break or other less severe loss of secondary system integrity, (2) the potential that a population of tubes susceptible to rupture exists in a particular steam generator, (3) the potential that operators would not take actions to avoid high differential pressures, and (4) the probability that a large number of tubes would actually fail simultaneously. In the NUREG –0844 assessment, the staff concluded that the probability of simultaneous multiple tube failure was small (approximately 10^{-5}), and that the risk resulting from releases during SGTRs with loss of secondary system integrity was small (about 10^{-7} latent fatalities per reactor year).

III. Conclusion

Based on the fact that (1) adequate steam generator tube inspections have been performed, (2) primary-to-secondary leakage is being monitored on a continuing basis, and (3) the risk of multiple SGTR events is low, I have concluded that an immediate shutdown and Plus Point probe inspection of Maine Yankee, Fort Calhoun Unit 1, Calvert Cliffs Units 1 and 2, St. Lucie Unit 1, and Millstone Unit 2 is not warranted.

The Petitioner's request for action pursuant to 10 CFR 2.206 is denied. As provided in 10 CFR 2.206(c), a copy of the Decision will be filed with the Secretary of the Commission for the Commission's review. This Decision will constitute the final action of the Commission 25 days after issuance unless the Commission, on its own motion, institutes a review of the Decision within that time.

Dated at Rockville, Maryland, this 6th day of December , 1995.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 95–30176 Filed 12–11–95; 8:45 am]

BILLING CODE 7590–01–P–M

[Docket Nos. 50–325 AND 50–324]

Carolina Power & Light Company; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The U.S. Nuclear Regulatory Commission (the Commission or NRC) is considering issuance of an amendment to Facility Operating License Nos. DPR–71 and DPR–62 issued to the Carolina Power & Light Company (the licensee) for operation of the Brunswick Steam Electric Plant, Units 1 and 2 (BSEP) located in Southport, North Carolina.

Effective October 26, 1995, the Commission amended its regulations (10 CFR Part 50, Appendix J) to provide a performance-based option for leakage-rate testing of containments of light-water-cooled nuclear plants. The proposed amendment would permit the licensee to implement this performance-based option, which allows leakage testing intervals to be based on system and component testing performance.

The proposed amendment requires the establishment of a ''Primary Containment Leakage Rate Testing Program'' (program) and makes general reference to the NRC guidance utilized by the licensee for development of this program, i.e. Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program''. Regulatory Guide 1.163 addresses the acceptability of industry-