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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY AND NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY	)	Docket Nos. 50-400 OL 50-401 OL
	)	
(Shearon Harris Nuclear Power Plant, Units 1 and 2)	)	

APPLICANTS' RESPONSE TO "NEW CONTENTIONS RE SPENT FUEL CASK SAFETY" BY WELLS EDDLEMAN

By a pleading dated August 4, 1983, Intervenor Wells Eddleman seeks to have admitted in this proceeding proposed Contentions 164, 165 and 166, relating to public health and safety in transporting spent fuel from Applicant Carolina Power & Light Company's ("CP&L") Brunswick and Robinson Plants to the Harris Plant in CP&L's IF-300 spent fuel shipping cask. Applicants CP&L and North Carolina Eastern Municipal Power Agency oppose admission of Contentions 164, 165 and 166. Applicants submit that Mr. Eddleman has failed to meet the showing

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required by 10 C.F.R. § 2.714(a)(1) for late-filed contentions, that the Board has no jurisdiction over the issues raised in the three contentions involving transportation of spent fuel from Brunswick and Robinson in licensed shipping casks, and that, in any event, Mr. Eddleman has failed to support his proposed new contentions with adequate basis and specificity.

CONTENTIONS 164, 165 AND 166 ARE UNTIMELY AND MR. EDDLEMAN HAS FAILED TO ESTABLISH GOOD CAUSE FOR ADMISSION OF LATE CONTENTIONS

Nontimely-filed contentions are not to be entertained absent a determination by the Board which balances the five lateness factors set forth in 10 C.F.R. § 2.714(a)(1). See Duke Power Co. (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 N.R.C. ___ (June 30, 1983). Intervenors must establish "good cause" for failure to file on time to satisfy the first lateness factor. The Commission recently reiterated the obligation of intervenors "to diligently uncover and apply all publicly available information to the prompt formulation of contentions." Id., slip op. at 11. Even the institutional unavailability of a licensing-related document (not at issue here) does not in itself establish good cause for filing a contention late if information was available early enough to allow the timely filing of that contention. Id. at 11-12.

Mr. Eddleman cites to information obtained during discovery and during negotiations with Applicants in an attempt to resolve Mr. Eddleman's concerns regarding spent fuel casks pressure relief valves (Contention 64(f)) as "new" information that demonstrates good cause for the late-filed contention.1/ August 4 pleading at 4-5. He also relies on "PATRAM '80" and a letter from Battelle Columbus Laboratories dated May 19, 1980 (Applicants do not possess and have not located either document) and a book released on March 15, 1983 by Dr. Marvin Resnikoff, The Next Nuclear Gamble.

In fact the Cask Safety Analysis Report ("CSAR") on the IF-300 cask has been on file with the NRC and publicly available since 1973. The CSAR was amended by supplement dated March 15, 1982 to allow dry shipments with a rupture disc installed rather than a pressure relief valve. Shipments in the IF-300 cask have been limited to dry shipments since June 24, 1981. Letter to D. E. MacDonald, Chief - Transportation Branch, NRC Office of Nuclear Material Safety and Safeguards,

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1/ Mr. Eddleman states "I understood directly from Applicants that a condition of our negotiations on 64f was that I not bring certain facts before the Board; one such fact was the rupture disk being planned to be used." August 4 pleading at 4. This statement may be somewhat misleading. Applicants agreed to provide to Mr. Eddleman informally information regarding CP&L's IF-300 spent fuel shipping cask that Applicants believed was otherwise outside the scope of any admitted contention in an attempt to settle Contention 64(f). As a condition to providing such information, Mr. Eddleman agreed that such discussions would be "off the record" and the information so obtained would not be used in the licensing proceeding. Counsel for Applicants informed Mr. Eddleman that information found in responses to formal interrogatories or otherwise publicly available was not subject to the above agreement.

from E. E. Utley, CP&L, dated June 24, 1981. An NRC Staff analysis of the criticality issued raised in Contention 166 was released on March 22, 1983. Thus, the only information obtained by discovery by Mr. Eddleman, that was not publicly available and upon which he relies in proffering Contentions 164, 165 and 166, was the fact of CP&L's present intention to remove the pressure relief valve and replace it with a rupture disc prior to its next spent fuel shipment. This is not really of consequence since CP&L has had authority to ship spent fuel dry in the IF-300 cask with an installed rupture disc since April 1982.

The 1980 documents cited by Mr. Eddleman as basis for Contention 164 and 165 were clearly available much earlier. While the book, The Next Nuclear Gamble, was released in March 1983, the information in the book which Mr. Eddleman cites regarding the failure of rupture discs in a fire (at page 165) is taken from a Battelle, Pacific Northwest Laboratory Report, "An Assessment of the Risk of Transporting Spent Nuclear Fuel by Truck" (November 1978). Thus the underlying information upon which Mr. Eddleman relies in support of Contentions 164 and 165 has been available for a number of years. Applicants submit that the publication of a document which simply cites to previously available underlying data does not establish good cause for late filing. See Cleveland Electric Illuminating Co. (Perry Nuclear Power Plant, Units 1 & 2), LBP-82-104, 16 N.R.C.

1626, 1627 (1982) (reliance on a Science News article cannot establish good cause for late filing where intervenor failed to demonstrate the extent to which information in the article differed from previously available information).

Regarding Contention 166, the NRC Staff's evaluation of the criticality issue was available in March 1983. Mr. Eddleman's failure to address the NRC Staff's analysis during the period of time between March 1983 and August 1983 certainly does not indicate a diligent investigation on his part. Furthermore, as will be discussed infra, his failure to address the Staff's analysis also establishes his failure to provide adequate basis and specificity for this contention.

With respect to the second lateness factor,2/ Mr. Eddleman has available to him an opportunity under 10 C.F.R. § 2.206 to challenge the cask license if he believes health and safety issues are raised. Since there are four IF-300 casks licensed, if Mr. Eddleman believes that he has discovered a generic safety concern, it would be more efficient for the Commission to review Mr. Eddleman's allegations in the context of the cask license applicable to all such casks.

Responding to the third lateness factor,3/ Mr. Eddleman

2/ The availability of other means whereby petitioner's interest will be protected. 10 C.F.R § 2.714(a)(1)(ii).

3/ The extent to which the petitioner's participation may reasonably be expected to assist in dveloping a sound record. 10 C.F.R. § 2.714(a)(1)(iii).

argues that his discovery on Contention 64(f) shows that he has the ability to help develop a sound record on these issues. Applicants suggest that in light of the contentions already admitted in this proceeding that were profferred by Mr. Eddleman or by Joint Intervenors it is highly unlikely that he will have the time and resources to litigate the issues already admitted much less any new ones.

Applicants submit that on balancing the five lateness factors, the Board should reject Contentions 164, 165 and 166 as failing to meet the requirements of 10 C.F.R. § 2.714(a)(1).

THIS BOARD LACKS JURISDICTION TO ADJUDICATE CONTENTIONS REGARDING ISSUES OF HEALTH AND SAFETY IN TRANSPORTING SPENT FUEL FROM CAROLINA POWER & LIGHT COMPANY'S LICENSED FACILITIES TO THE HARRIS PLANT IN SHIPPING CONTAINERS ALREADY LICENSED BY THE NRC

Contentions 164, 165 and 166 all seek to litigate health and safety issues involved in transporting spent nuclear fuel from CP&L's licensed nuclear facilities to the Harris Plant. This is clear from the wording of the contentions and the context of the "basis" for each contention.4/ Contention 164

4/ While Contention 164 does attempt to encompass "shipments to or from Harris", shipments from Harris are not discussed in the context of the "basis". Contention 164 focuses on the health and safety impacts of shipments with the IF-300 cask which CP&L plans to utilize in any shipments of spent fuel to Harris in the near term. Shipments of spent fuel from Harris will be performed by the Department of Energy pursuant to a contract with Applicants for waste disposal. See 10 C.F.R. Part 961 (48 Fed. Reg. 16,590 et seq. (April 18, 1983)).

alleges that "The health and safety of the public is insufficiently protected . . . ." August 4 pleading at 1 (emphasis supplied). Contention 165 alleges releases of radioactive materials will occur in the event of accidents and "persons equipped to go into a radioactive release to reseal the cask are not available in many towns along CP&L's rail routes to Harris." Id. at 4. Contention 166 asserts that use of the IF-300 cask "is unacceptable for the protection of public health and safety." Id. at 6 (emphasis supplied). The "basis" offered by Mr. Eddleman makes it clear that he is concerned with shipments of spent fuel from Brunswick and Robinson to Harris. For example, Mr. Eddleman states:

> CP&L has had substantial amounts of failed fuel at Brunswick which they may seek to ship to Harris . . . . The above also applies to failed fuel from the Robinson reactor.

Id. at 3.

This Board has already ruled that it has no jurisdiction over health and safety issues involved in the transportation of spent fuel from Brunswick and Robinson to Harris. Memorandum and Order (Reflecting Decisions Made Following Prehearing Conference) dated September 22, 1982, at 57 (Contention 64(k)). This ruling is supported by substantial authority which holds that health and safety issues involving the transportation of spent fuel are not subject to case by case resolution in operating license proceedings. See Wisconsin Electric Power

Co. (Point Beach Nuclear Plant, Unit No. 2), ALAB-31, 4 A.E.C. 689, 693, 697 (Contention 32) (1971); Trustees of Columbia University in the City of New York, ALAB-50, 4 A.E.C. 849, 863 (1972); Pennsylvania Power and Light Co. (Susquehanna Steam Electric Station, Units 1 and 2), LBP-79-6, 9 N.R.C. 291, 315 (1979); Philadelphia Electric Co. (Limerick Generating Station, Units 1 and 2), LBP-82-43A, 15 N.R.C. 1423, 1501 (1982); cf. Cincinnati Gas & Electric, (William H. Zimmer Nuclear Station), LBP-81-2, 13 N.R.C. 36, 42-3 (1981) (limitation on litigation of public health and safety aspects of transportation encompasses the adequacy of a transportation plan, including designation of routes and security required by 10 C.F.R. § 73.72).

As noted above, all three proposed contentions involve health and safety issues in the transportation of spent fuel from Robinson and Brunswick to Harris and are clearly outside this Board's jurisdiction.

Furthermore, Contentions 164, 165 and 166 seek to litigate issues that challenge the findings of the NRC in its issuance of a license to General Electric Company for shipments with the IF-300 shipping cask.5/ The jurisdiction of this Board does not extend to General Electric Company's cask license or to CP&L's license to transport spent fuel.6/

5/ See Certificate of Compliance 9001, attached hereto as Appendix A.

6/ The transportation of spent fuel from Brunswick and Robinson involves licenses other than Applicants' operating

(Continued Next Page)

Accordingly, Contentions 164, 165 and 166 should be rejected as outside of this Board's jurisdiction.

ASSUMING, ARGUENDO, THAT THE BOARD FINDS PROPOSED CONTENTIONS 164, 165 AND 166 ARE UNTIMELY BUT FOR GOOD CAUSE AND THAT IT HAS JURISDICTION TO ADJUDICATE THE ISSUES RAISED THEREIN, THE BOARD SHOULD NEVERTHELESS REJECT THE CASK SAFETY CONTENTIONS BECAUSE MR. EDDLEMAN HAS FAILED TO PROVIDE ADEQUATE BASIS AND SPECIFICITY TO SUPPORT THE ALLEGATIONS MADE

Pursuant to 10 C.F.R. Parts 71 and 73, the NRC has issued to General Electric Company Certificate of Compliance 9001 regarding shipments of spent fuel in the IF-300 shipping casks. Applicants submit, supra, that Mr. Eddleman should not be permitted to attack that license and the findings of the NRC Staff which permitted issuance of the cask license in the Harris operating license proceeding. In the alternative, Applicants contend that the basis and specificity required to support a contention that the IF-300 cask is not safe for shipment of spent fuel, in light of the NRC's finding to the contrary, must overcome the presumption that attaches to the NRC license and

(Continued)

license for the Harris Plant. Such transportation would be accomplished by one of the Applicants in this proceeding, CP&L, pursuant to authority granted to CP&L alone in conjunction with its licenses to operate the Brunswick and Robinson units. Certificate of Compliance No. 9001, governing the use of the IF-300 spent fuel shipping cask, was issued to General Electric Company and covers four IF-300 casks -- three owned by General Electric and one owned by CP&L.

must address the existing NRC Staff analysis on cask safety. Mr. Eddleman, at a minimum, has an obligation to state with basis and specificity what he faults in the CSAR. This he has not done.

As will be discussed below, Mr. Eddleman has utterly failed to provide adequate basis and specificity for proposed contentions 164, 165 and 166.

Contentions 164 and 165 (Radioactive Releases During a Spent Fuel Transport Accident)

It is not clear to Applicants precisely what distinctions Mr. Eddleman means to draw between Contention 164 and Contention 165. Contention 164 alleges Applicants will ship failed spent fuel, the uranium aerosols of which will be released through a breached rupture disc during an accident. Contention 165 alleges uranium aerosols and other radioactive materials will be released during a fire or a crash (both presumably accidents). Contention 165 adds the thought that "persons equipped to go into a radioactive release to reseal the cask are not available in many towns along CP&L's rail routes to Harris." The same basis is offered for both contentions.

Mr. Eddleman has been afforded the opportunity to discover the two volume CSAR for the IF-300 cask. He has requested and obtained copies of most of the pages in the CSAR. Section IX of the CSAR discusses in considerable detail why the radiological releases described in the above contentions will not occur during accident conditions. The CSAR has been reviewed and approved by the NRC. Mr. Eddleman has an obligation at this stage of the proceeding to state with specificity what he finds wanting in the accident analysis in the CSAR. This he has not done.

Mr. Eddleman's sole basis for his postulated release of radioactive material due to failure of a rupture disc is a reference to page 165 of Dr. Marvin Resnikoff's book, The Next Nuclear Gamble. Dr. Resnikoff cites to a November 1978 study by Pacific Northwest Laboratory which inter alia determined cask failure thresholds. In a water cooled truck cask exposed to a 1010°C (1850°F) fire for 15 minutes a cask rupture disc would fail from overpressurization in 2.5 hours, according to the study. This study is not relevant to dry shipments in an IF-300 rail cask and Mr. Eddleman does not address the findings of this study.7/ It does support the requirement for a pressure relief valve for water cooled shipments. The accident analysis in the CSAR demonstrates that under design accident

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7/ Mr. Eddleman is certainly aware of the 1978 study as he asked Applicants about the study in an interrogatory and a copy of those sections of the study in the possession of Applicants was made available to Mr. Eddleman for inspection. See "Wells Eddleman's General Interrogatories and Interrogatories on Contentions 29, 37B, 64f and 67 to Applicants Carolina Power & Light et al (Second Set)," dated April 22, 1983, at Interrogatory 64-6(a).

conditions for a dry shipment, the maximum cask cavity pressure is 248 psia, significantly less than the minimum relief pressure of the rupture disc of 350 psig. CSAR, §§ 6.3.20(4); 6.5.B.1; 9.7.3.2.

The rupture disc will not rupture during a dry shipment even under accident conditions. Mr. Eddleman has not addressed the accident analysis in the CSAR nor has he pointed to any data that address dry shipments. The citation to The Next Nuclear Gamble, which in turn cites to the 1978 Pacific Northwest Laboratories' study, is simply inapposite.

Contentions 164 and 165 depend on the following assumption: "CP&L has had substantial amounts of failed fuel at Brunswick which they may seek to ship to Harris." August 4 pleading at 3. This is a bald assertion on the part of Mr. Eddleman without any support; furthermore it is incorrect. Out of a total 476 spent fuel assemblies stored at the Brunswick Unit 1 spent fuel pool as of June, 1982, only 14 were detected to be in any way defective -- including mechanical defects (no breach of the cladding) and as little as a "pin hole" breach of the cladding. Out of 424 spent fuel assemblies stored in Brunswick Unit 2's spent fuel pool as of that same date, only 19 were defective. For Robinson Unit 2, six out of 157 assemblies were determined to be defective.8/ Nor has Mr.

8/ This information on defective spent fuel assemblies for all three units reflects data submitted to The U.S. Department

(Continued Next Page)

Eddleman offered any basis for the speculation that CP&L would be likely to elect to ship failed fuel to Harris.

Contentions 164 and 165 rely on the assertion that CP&L has "substantial" quantities of spent fuel to ship to Harris and that a rupture disc will rupture during a dry shipment, releasing the uranium aerosols from failed fuel to the environment. As publicly available information substantiates, CP&L does not have "substantial" quantities of spent fuel at its Brunswick and Robinson Plants. The reference to a study on over-pressurization of a water cooled truck cask under certain extreme accident conditions does not support an allegation that a rupture disc will rupture during dry shipments. Mr. Eddleman has failed to address the analysis to the contrary in the CSAR. He provides no basis with specificity for Contentions 164 and 165.

Contention 166 (Criticality in Spent Fuel Cask)

Contention 166 alleges that the IF-300 spent fuel cask "has a basket that cannot assure that spent fuel contained in it will not experience nuclear criticality ... under all

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(Continued)

of Energy and publicly available. The last such submission was based on data as of June 1982 by letter dated November 23, 1982 from L. H. Martin, CP&L, to B.M. Cole, Battelle Northwest Laboratories (under contract to DOE).

conditions." The only basis offered for this allegation is a footnote in a book entitled The Next Nuclear Gamble and the fact that Applicants utilized the IF-300 spent fuel shipping cask. The footnote referred to by Mr. Eddleman reads as follows:

> In September, 1982, the IF-300 was entirely removed from service for BWR fuel because the lack of criticality under all conditions could not be assured with the fuel basket employed.

Indeed, in September 1982, General Electric and CP&L reported to the NRC an error in the structural analysis for the BWR fuel basket for the IF-300 cask. The companies voluntarily limited use of the IF-300 cask to shipments of PWR fuel, pending reanalysis. A new structural analysis of the BWR fuel basket was submitted to the NRC in October 1982. After obtaining additional information from the companies, the NRC Staff agreed with the revised structural analysis and concluded that the BWR fuel basket "can withstand a 30-foot free drop of the cask without any significant effect on safety or the performance of the package."9/ A copy of the NRC Staff's analysis

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9/ The nuclear criticality safety of the cask is based in part on the spacing between the fuel assemblies. Buckling of the fuel basket tie rods as a result of a 30-foot drop (or crash yielding the same force) could affect the spacing. (Although, shipping dry there is no possiblity of criticality in any event unless the cask was involved in an accident exceeding the hypothetical accident described in 10 C.F.R. Part 71 and a large quantity of water leaked into the cavity.) Reanalysis of structural integrity demonstrated a sufficient safety margin against rod buckling. U.S. NRC Transprotation Certification Branch, "Approval Record," Model No. IF-300 Package, Docket No. 71-9001 (March 22, 1983) (Appendix B hereto).

and "Approval Record" is attached hereto as Appendix B.

Mr. Eddleman offers no mention or refutation of the NRC Staff's analysis. He provides no basis with specificity for the allegation made in Contention 166. For this reason alone Contention 166 must be rejected.

CONCLUSION

For all of the reasons discussed above, Mr. Eddleman's proposed Contentions 164, 165 and 166 must be rejected.

Respectfully submitted,

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(919) 836-6839

Counsel for Applicants

Dated: August 24, 1983

August 24, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CAROLINA POWER & LIGHT COMPANY AND NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY (Shearon Harris Nuclear Power Plant, Units 1 and 2)	))))))))	 Docket Nos. 50-400 OL 50-401 OL

CERTIFICATE OF SERVICE

I hereby certify that copies of "Applicants' Response To 'New Contentions Re Spent Fuel Cask Safety' By Wells Eddleman" were served this 24th day of August, 1983 by deposit in the United States mail, first class, postage prepaid, to the parties on the attached Service List.

John H. O'Neill, Jr.

John H. O'Neill, Jr.

Dated: August 24, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
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CAROLINA POWER & LIGHT COMPANY and NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY	)	Docket Nos. 50-400 OL 50-401 OL
	)	
(Shearon Harris Nuclear Power Plant, Units 1 and 2)	)	

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APPENDIX A

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

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MAR 22 1983

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MAR 28 1983

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

RADIOLOGICAL & CHEMICAL
SUPPORT SECTION

FCTC:RHO
71-9001

General Electric Company
ATTN: Mr. D. M. Dawson, MC-861
175 Curtner Avenue
San Jose, CA 95125

Gentlemen:

As requested by your application dated October 21, 1982, as amended, enclosed is Certificate of Compliance No. 9001, Revision No. 15, for the Model No. IF-300 shipping container. This certificate supersedes, in its entirety, Certificate of Compliance No. 9001, Revision No. 14, dated September 30, 1982.

Changes made to the enclosed certificate are indicated by vertical lines in the margin.

Those on the attached list have been registered as users of this package under the general license provisions of 10 CFR §71.12(b) or 49 CFR §173.393a.

This approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR §173.393a.

Sincerely,

Charles E. MacDonald

Charles E. MacDonald, Chief
Transportation Certification Branch
Division of Fuel Cycle and
Material Safety

Enclosures:
1. Certificate of Compliance
No. 9001, Rev. 15
2. Approval Record

cc: w/encls
Mr. Richard R. Rawl
Department of Transportation

Model No. IF-300 Packaging
USA/9001/B( )F

Addressees: w/encls

Ltr dtd: MAR 22 1983

Commonwealth Edison Company
ATTN: Director of Nuclear Licensing
P. O. Box 767
Chicago, IL 60690

Carolina Power & Light Company
ATTN: Mr. B. H. Webster
Route 1, Box 327
New Hill, NC 27562

General Electric Company
ATTN: Mr. D. M. Dawson, MC-861
175 Curtner Avenue
San Jose, CA 95125

Nebraska Public Power District
ATTN: Mr. Jerry V. Sayer
P.O. Box 98
Brownville, NE 68321

CERTIFICATE OF COMPLIANCE

For Radioactive Materials Packages

Certificate Number	1.(b) Revision No.	1.(c) Package Identification No.	1.(d) Pages No.	1.(e) Total No. Pages
9001	15	USA/9001/B( )F	1	6

2. PREAMBLE

2.(a) This certificate is issued to satisfy Sections 173.393a, 173.394, 173.395, and 173.396 of the Department of Transportation Hazardous Materials Regulations (49 CFR 170-189 and 14 CFR 103) and Sections 146-19-10a and 146-19-100 of the Department of Transportation Dangerous Cargoes Regulations (46 CFR 146-149), as amended.

2.(b) The packaging and contents described in item 5 below, meets the safety standards set forth in Subpart C of Title 10, Code of Federal Regulations, Part 71, "Packaging of Radioactive Materials for Transport and Transportation of Radioactive Material Under Certain Conditions."

2.(c) This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. This certificate is issued on the basis of a safety analysis report of the package design or application—

3.(a) Prepared by (Name and address):
General Electric Company
175 Curtner Avenue
San Jose, CA 95125

3.(b) Title and identification of report or application:
General Electric Company application dated October 8, 1979, as supplemented.

3.(c) Docket No. 71-9001

4. CONDITIONS
This certificate is conditional upon the fulfilling of the requirements of Subpart D of 10 CFR 71, as applicable, and the conditions specified in item 5 below.

5. Description of Packaging and Authorized Contents, Model Number, Fissile Class, Other Conditions, and References:

(a) Packaging

(1) Model No.: IF-300

(2) Description

A stainless steel encased, depleted uranium shielded cask. The cask is cylindrical in shape, 64 inches in diameter and a maximum of 210 inches long with maximum cavity dimensions of 37-1/2 inches in diameter by 180-1/4 inches long. Shielding is provided by 4 inches of depleted uranium, 2-1/8 inches of stainless steel and a minimum of 4-1/2 inches of water.

Two closure heads are provided for the shipment of BWR and PWR fuel assemblies. The heads are 304 stainless steel forgings and end plates which encase the 3-inch thick depleted uranium shielding.

The closure heads are secured to the cask body by means of 32, 1-3/4 inch studs and nuts. The cask is sealed with a metallic ring gasket.

The cavity is penetrated by a vent line at the top and a drain line at the bottom. These lines are sealed by bellows seal stainless steel globe valves and valved quick-disconnect couplings. The vent line is also equipped with a 375 psig relief valve or a 350-400 psig rated rupture disk. All valves are housed in protected boxes on the cask exterior.

5. (a) Packaging (continued)

(2) Description (continued)

Neutron shielding is provided by a liquid-filled, thin-walled, corrugated containment on the cask exterior. This cylindrical structure is separated into two longitudinal compartments, each equipped with two expansion tanks, fill and relief valves. The fill line from each compartment is terminated by a stainless steel globe valve in a protected box (separate from cavity boxes) on the cask exterior. The vent line from each compartment goes to an expansion tank which is provided with a pressure relief valve set at 200 psig.

The cask has two types of fuel baskets which can be interchanged to accommodate various fuels. The PWR basket holds 7 assemblies, the BWR basket hold 18 assemblies. The BWR fuel basket may be provided with supplementary shielding (depleted uranium) near the cask closure.

The cask is shipped horizontally with the bottom supported in a tipping cradle between two pedestals and the upper end resting in a semi-circular saddle; the upper end is pinned to the saddle. The cask supports are welded to the framing of a 37-1/2-foot long by 8-foot wide structural steel skid. The skid also holds the cask cooling system which consists of two diesel engines driving two blowers which discharge into common ducting. Four ducts run the length of the cask and direct cooling air to the corrugated surface. Operation of the auxiliary cooling system is not a requirement of this package approval.

The entire cask and cooling system is covered by a retractable aluminum enclosure. Access to the enclosure is via locked panels in the side and a locked door in one end. Although the Model No. IF-300 cask can be transported for short distances on the highway, its principal mode of transportation is by railroad.

The gross weight of the cask is approximately 140,000 pounds. The skid and other external components weigh approximately 35,000 pounds.

(3) Drawing

The Model No. IF-300 shipping cask is described by the following General Electric Company Drawing No.: 159C5238 - Sheets 1 thru 2, Rev. 3; Sheet 3, Rev. 5; Sheet 4, Rev. 6; Sheet 5, Rev. 5; Sheet 6, Rev. 5; Sheet 7, Rev. 4; Sheet 8, Rev. 5; Sheet 9, Rev. 4; Sheet 10, Rev. 5; and Sheet 11, Rev. 2.

5. (a) (4) Basic Components

The Basic Components of the Model No. IF-300 shipping cask that are important to nuclear safety are listed in Subsection 10.5.

(b) Contents - air as primary coolant

(1) Type and form of material

Irradiated PWR and BWR uranium oxide fuel assemblies. PWR assemblies may be shipped with or without control rods. Partial fuel assemblies that is, assemblies from which fuel pins are missing, must not be shipped unless dummy fuel pins are used to displace an amount of wat equal to that displaced by the original pins. The specific power of each fuel assembly must not exceed 40 Kw/KgU and the burnup of each fuel assembly must not exceed 35,000 MWD/MTU. The minimum cooling time of each assembly must be no less than 120 days. Prior to irradi the BWR and PWR fuel assemblies must have the following dimensions an specifications:

(i) Group I fuel assemblies

	PWR	BWR
Fuel form	Clad UO_2 pellets	Clad UO_2 pellets
Cladding material	Zr or SS	Zr or SS
Maximum initial U content/assembly, kg	465	198
Maximum initial U-235 enrichment, w/o	4.0	4.0
Maximum bundle cross section, inches	8.75	5.75
Fuel pin array	14x14/15x15	7x7
Fuel diameter, inch	0.380-0.460	0.500-0.600
Fuel pin pitch range, inch	0.502-0.582	0.647-0.809
Maximum active fuel length, inches	145	146

5. (b) Contents - air as primary coolant (continued)

(ii) Group II fuel assemblies

	PWR	BWR
Fuel form	Clad UO_2 pellets	Clad UO_2 pellets
Cladding material	Zr or SS	Zr or SS
Maximum initial U content/assembly, kg	475	198
Maximum initial U-235 enrichment, w/o	4.0	4.0
Maximum bundle cross section, inches	8.75	5.75
Fuel pin array	16x16/17x17	8x8
Fuel diameter, inch	0.376-0.400	0.475-0.505
Fuel pin pitch range, inch	0.496-0.507	0.630-0.645
Maximum active fuel length, inches	150	150

(2) Maximum quantity of material per package

Maximum decay heat per package not to exceed 40,000 Btu/hr. Maximum 5,725 Btu/hr/PWR assembly. Maximum 2,225 Btu/hr/BWR assembly.

Seven (7) PWR fuel assemblies, or eighteen (18) BWR fuel assemblies.

Above assemblies to be contained in their respective fuel baskets as shown in GE Drawing No. 159C5238 - Sheet 6, Rev. 5.

(c) Unloaded package - contents and maximum quantity of material

Greater than a Type A quantity of residual radioactive material consisting of mixed-fission and activation products adhering to interior cavity and fuel basket surfaces.

(d) Fissile Class I

The end of life total calculated residual gas that could become available from the fuel pins must not exceed 0.23 lb moles for content 5(b) and individual calculated fuel pin pressure must not exceed 2,500 psia, at 900°F.

7. The maximum gross weight of the cavity contents must not exceed 21,000 pounds.

8. For the contents described in 5(b), the cavity fill specifications must include the following: An air void must be established such that not more than 1.0 cu ft of water (corresponding to a bulk water temperature of 70°F) remains in the cavity. The licensee must take sufficient time-temperature-pressure data to ensure that the cavity pressure will not exceed 45 psig, and that the average cavity wall temperature will not exceed 210°F during the 130°F day with no auxilary cooling.

   For the contents described in 5(b) and (c) the air coolant is considered part of the package contents. The radioactivity limits specified in 10 CFR §71.35(a)(4) do not apply.

9. Prior to each shipment, the licensee must confirm that the cask is properly sealed by testing as Subsection 11.3.3.1.

10. The cask contents shall be so limited that under normal conditions prior to transport, 111 times the neutron dose rate plus 11.3 times the gamma dose rate will not exceed 1000 mrem/hr at a distance of (6) feet from the side of the cask (10 feet from the cask center-line).

11. The neutron shielding tanks must be filled with approximately a 50/50 volume percent mixture of ethylene glycol and water during the months of October through May.

12. In addition the requirements of Subpart D of 10 CFR Part 71, each package prior to first use must meet all of the acceptance tests and criteria specified in Subsections 6.7.6.2, 11.3.1.1, and 11.3.1.7.

13. Each cavity relief valve, typical glove valves, and typical shielding tank (barrel expansion tank) relief valves must be tested as stated in Subsections 6.5.3.3, 6.6.1.1, and 6.6.1.2.

    In lieu of the requirements of 10 CFR §71.54(h), valve testing and maintenance frequency must be as stated in Subsections 6.5.3.4, 6.6.2.1, and 6.6.2.2 except during periods of cask inactivity. If a rupture disk device is utilized for dry shipments, the rupture disk device must be maintained and replaced as stated in Subsection 6.5.B.2 in lieu of the requirements of 10 CFR §71.54(h). During inactive periods the maintenance and testing frequency may be disregarded provided that the package is brought into full compliance with these requirements prior to the next use of the package.

14. The cask cavity must be equipped with either (1) a Target Rock 73J pressure relief valve set at a pressure of 375 psig (450°F). The valve is shown in Target Rock Corporation Drawing No. 73J-001, Rev. H, J, K, or L; or (2) a rupture disk device with a bust pressure within the range of 350-400 psig (443°F) including all tolerances.

15. The uranium shielding material must be separated from all steel surfaces with a minimum copper thickness of 4-mils, except that the stud bolts attaching the shield assemblies to top of the BWR basket must be coated with a minimum of 1/2-mil of copper.

16. No shutoff valve shall be installed between each neutron shield tank and its respective thermal expansion tank.

17. The package authorized by the certificate is hereby approved for use under the general license provisions of 10 CFR §71.12(b).

18. Expiration date: October 31, 1984.

REFERENCES

General Electric Company consolidated application dated October 3, 1979.

Supplements dated: May 12, July 21, and November 26, 1980; February 6 and December 29, 1981; March 15 and September 20, 1982; and March 18, 1983.

Section XI, Quality Assurance and Testing, is deleted from the application.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles E. MacDonald, Chief
Transportation Certification branch
Division of Fuel Cycle and
Material Safety

Date: MAR 22 1983

U.S. Nuclear Regulatory Commission
Transportation Certification Branch
Approval Record
Model No. IF-300 Package
Docket No. 71-9001

BACKGROUND

By letters dated September 7 and 13, 1982, the General Electric Company and Carolina Power and Light Company, respectively, notified NRC that they were voluntarily limiting the use of the Model No. IF-300 casks to PWR fuel shipments.

The licensees reported that an error was discovered in the structural analysis for the BWR fuel basket. Specifically, a value of 3,840 pounds was used for the basket weight when analyzing the effect on the basket of the 30-foot drop onto the cask's bottom end (Section 5.6.3.4, p 5-95 of the application). The correct maximum of the basket is 5,675 pounds. Substitution of the correct weight into the analysis of Section 5.6.3.4 results in a compressive stress which exceeds the critical buckling stress of the 2-1/4-inch diameter fuel basket tie-rods.

The nuclear criticality safety of the cask is primarily based on the spacing between the fuel assemblies and the presence of boron carbide rods (neutron absorbers). During transport, the cask is shipped dry with no possibility of a criticality event occuring unless the cask was involved in an accident exceeding the hypothetical accident defined in 10 CFR Part 71 and a large quantity of water leaked into the cavity.

Until the nuclear criticality safety of the BWR fuel shipments is firmly established, shipment of BWR fuel assemblies were deleted from the certificate by Revision No. 14, dated September 30, 1982.

APPLICATIONS

By application dated October 21, 1982, General Electric Company requested a revision to the certificate of compliance for the Model No. IF-300 irradiated fuel cask to reinstate its capability to transport BWR fuel assemblies. The submittal contained a new structural evaluation of the BWR basket if it were subjected to 30-foot horizontal and vertical drop tests.

By letter dated October 29, 1982, the NRC staff raised several questions regarding the technical bases for the safety conclusions and a number of specific questions regarding the modeling and boundary conditions used in the analysis.

On November 22, 1982 and January 21, 1983, the General Electric Company responded to our October 29 letter by revising and performing new calculations for the BWR fuel basket. By letter dated February 22, 1983, the NRC staff raised several questions pertaining to the tie rod buckling analysis and the spacer disk horizontal drop analysis.

Additional information was furnished by letters dated February 28 and March 4, 1983 in response to our February 22 letter. By letter dated March 9, 1983, an additional question was raised concerning the 30-foot vertical drop of the BWR fuel basket. The applicant responded on March 16, 1983. On March 18, 1983, the General Electric Company consolidated the previously submitted analysis for the BWR fuel basket.

SUMMARY OF BWR FUEL BASKET ANALYSIS AND CONCLUSIONS

The fuel basket was analyzed by the applicant for the vertical 30-foot bottom drop by the ADINA finite element computer program. The analysis was a transient dynamic analysis using the cask deceleration and revised fuel basket mass as input force to the fuel basket tie rods. Emphasis has been placed on the possibility of tie rod buckling. By gradually increasing the load until the tie rod buckling occurs, the applicant has shown that a load factor (e.g., safety margin) of 1.2 exists for the tie rods against buckling. The analysis assumes bi-linear stress-strain curve which may be non-conservative under large strain. Also, the inelastic buckling of the tie rods is sudden and instantaneous even with decreasing load. Thus, the 1.2 load factor does not provide a sufficient safety margin to ensure no buckling. However, it is noted that the applicant has conservatively neglected all nine, 1/2"x6.0" bar cell spacers (AISI 200, Type 216 SS, welded to each spacer disk (telecon between Al Fanning, GE, and R.H. Odegaarden, NRC, 03/18/83)) and all 112 poison rods (304 SS Tubes 0.5" ODx0.02" wall) between the spacer disks. The actual load factor or safety margin against tie rod buckling is considerably higher than 1.2. For added protection, shipments should be limited to dry shipments until the applicant derives a more accurate load factor for buckling. The applicant also reduced the time steps by one half (2.5×10^{-6} vs. 5×10^{-6} seconds) for the case that load factor equals to unity. The results have shown very small changes (approximately 2 percent) indicating that time steps used are sufficiently small to provide a stable solution.

The fuel basket horizontal 30-foot drop was analyzed by the ANSYS computer code. Based on a separate static analysis, the highest loaded spacer disk was selected for the detailed dynamic analysis by ANSYS. The dynamic analysis was carried out using the highest cask deceleration time-history (horizontal drop orientation) and the inertial forces of all basket components and fuel loads. The analysis also considered material yielding and the effects of the radial gap between the outer rim of the spacer disk and the inner surface of the cask. The results of the analysis showed some yielding in small parts of the disk, but the deformations were insignificant (maximum 0.06 inches) and the stresses were below those required for buckling.

The staff agrees with the applicant's conclusion that the fuel basket can withstand a 30-foot free drop of the cask without any significant effect on safety or the performance of the package.

Charles E. MacDonald

Charles E. MacDonald, Chief
Transportation Certification Branch
Division of Fuel Cycle and
Material Safety, NMSS

Date: MAR 22 1983