UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	:	Docket Nos. 50-352
	:	50-353
Philadelphia Electric Company	:	
(Limerick Generating Station,	:	
Units 1 and 2)		

AFFIDAVIT OF LUBOMIR B. PYRIH REGARDING STORAGE OF UNIRRADIATED FUEL AT THE LIMERICK GENERATING STATION

Lubomir B. Pyrih being first duly sworn according to law, deposes and states:

My name is Lubomir B. Pyrih. A statement of my professional qualifications is attached and incorporated herein by reference. This affidavit addresses whether there is a credible potential for violating the Nuclear Regulatory Commission regulations applicable to onsite and offsite radiological consequences in the event of an accident to the new fuel assemblies to be delivered to and stored at the Limerick Generating Station. The new fuel assemblies contain unirradiated, low enriched uranium oxide fuel pellets in a metal cladding.

As discussed below, there is no potential for violating the provisions of 10 CFR Part 20 as a result of credible accidents affecting the new fuel assemblies to be stored at Limerick. This is demonstrated by comparing the dose effects of a postulated accident, which is not deemed credible, wherein all barriers between the unirradiated, low enriched uranium fuel pellets and the environment are non-mechanistically removed, with the requirements of 10 CFR Part 20. This comparison, as described below, shows that the regulatory

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limits are not exceeded. Thereafter, the conservatisms inherent in the simplifying non-mechanistic assumptions will be reviewed to further demonstrate the incredibility of exceeding the limits established by regulation.

1. Description of Incredible Accident

A. Fuel Storage Configuration. The fuel is stored in shipping containers in 3 piles containing 64 boxes and in 4 piles containing 48 boxes as is described in the amended Special Nuclear Material application dated January 23, 1984. The boxes are oriented on the site as shown on Figure 1 attached. For the purposes of the incredible accident, all protective material surrounding the fuel pellets in a given pile or piles was assumed to disappear non-mechanisticly. The pellets were then assumed to be located at a point at the geometric center of the given pile.

B. The incredible accident evaluated consists of three scenarios:

1. One pile of 48 boxes was assumed to non-mechanisticly disappear. The pellets contained in the fuel assemblies in the 48 boxes were considered to be concentrated at the geometric center of the pile in the form of an unshielded point source of 17.5 metric tons of uranium enriched to an average enrichment 1.85% in isotope U-235.

2. One pile of 64 boxes was assumed to non-mechanisticly disappear. The pellets contained in the fuel assemblies in the 64 boxes were considered to be concentrated at the geometric center of the pile in the form of an unshielded point source of 23.3 metric tons of uranium enriched to 1.85% in isotope U-235.

3. All of the 7 piles of boxes were assumed to non-mechanisticly disappear. The pellets contained in the fuel assemblies in each of the 7 piles were considered to be concentrated at the center of the pile in which the pellets were originally resident. Each of the 7 piles then was considered as a point source of either 17.5 or 23.3 metric tons of uranium enriched to 1.85% in isotope U-235.

C. Dose Consequences of Incredible Accident Scenarios.

1. Scenario B.1. above produced exposures below the limits of 10 CFR Part 20 for unrestricted areas, 10 CFR Section 20.105 (b)(1)(2 millirem per hour), inside the fence of the new fuel storage area shown on Figure 1.

2. Scenario B.2. above produced exposures below the limits of 10 CFR Part 20, as stated above, inside the fence of the new fuel storage area shown on Figure 1.

3. Scenario B.3. above produced exposures below the limits of 10 CFR Part 20, as stated above, at a distance of no more than 25 feet outside the fence of the new fuel storage area shown on Figure 1. The area which is no more than 25 feet outside of the fence of the new fuel storage area is all within the construction security boundary of the plant and further, is within the area which is under the control of Philadelphia Electric Company.

For scenario B.3, the exposure rate at a distance of 530 feet from the centroid of the fuel stored in the new fuel storage area is only 2% of the limit of 10 CFR Part 20 for unrestricted areas discussed above.

4. The exposure limits discussed above are those for persons in unrestricted areas, as defined in 10 CFR Part 20, under conditions of normal

operation. The consequences listed in 1., 2., and 3. above result from non-mechanistic accident scenarios. The exposure consequences of these three non-mechanicistic accident scenarios have been found to be acceptable using the regulatory acceptance criteria of 10 CFR Part 20 discussed above.

D. The conclusions in C. above are based on the affidavit of Dr. Paul S. Stansbury of General Electric Company, dated March 12, 1984, which is attached hereto.

1. The locations of the point sources shown on Attachment A to Dr. Stansbury's affidavit were selected as described in A. above.

2. The consequences in C.1. above are taken from Attachment D to Dr. Stansbury's affidavit.

3. The consequences in C.2. above are taken from Attachment E to Dr. Stansbury's affidavit.

4. The consequences in C.3. above are taken from Attachments B. and C. to Dr. Stansbury's affidavit.

5. The statements of the consequences made in C. above do not take credit for the reduction in dose which would result from the conservatisms detailed in Dr. Stansbury's affidavit.

2. There are many other conservatisms inherent in the postulated incredible accident compared with the conditions of fuel storage and handling as described in Philadelphia Electric Company's amended Special Nuclear Material application dated January 23, 1984. These are discussed, starting with the unirradiated, low enriched uranium fuel pellets and moving outward to the shipping container:

A. The exposure consequences discussed in C.1. above are assumed to be to persons in unrestricted areas who are not subject to a radiological monitoring program. In the event of an accident which would in any way affect the stored new fuel, health physics personnel, who are part of the Limerick Station staff and whose qualifications are discussed in the application, will perform complete radiation and contamination surveys and take appropriate actions to limit exposure.

B. If all protective and encapsulating material surrounding the fuel pellets were to be removed non-mechanisticly, and the resulting concentrated pile of fuel assemblies (not a point source as conservatively assumed by Stansbury) were to be impacted by a force due to an accident, the pellets could be dispursed over an area that, by definition, would be of greater areal extent than that of a point source. The radiological consequences of such a dispursed pile of fuel pellets would be less than that resulting from a point source, and hence even further below the limit of 10 CFR Part 20 discussed above.

C. It is likely that the impact of any accident would not be the same on each of the seven piles of boxes. Therefore, seven point sources as assumed in B.3 would not be created. More likely, some degree of encapsulation would be retained for some of the above piles which are less damaged. Illustrative of this, Attachment E to Stansbury's affidavit shows the exposure resulting from the non-mechanistic establishment of a point source equal to a 64-box pile would be less than 1% of Part 20 limits at a distance of less than 266 feet.

D. The fuel pellets are encased in metal rods (cladding) which are

sealed by welding. This cladding constitutes a radiation barrier and a barrier against release of the fuel pellets. Rather than non-mechanisticly disappearing in an accident, these tubes would remain. Although the degree of damage to these rods (if any) has not been quantified, the continued presence of the cladding would serve to reduce the radiological consequences of any accident which would disturb the pile of fuel assembly boxes.

E. The fuel rods are joined together into an 8 x 8 array. The rods are held in their 8 x 8 geometric arrangement by spacer hardware. Thus assembled, with associated upper and lower tie plates, they constitute a fuel bundle. This bundle is a flexible structural assembly. When subjected to an accident loading, it is more likely that the fuel bundle would deflect and deform, absorbing energy, rather than releasing fuel pellets. This absorption of energy would serve to reduce the radiological consequences of any accident which would disturb the pile of boxes.

F. Shipping Container Information

1. Background Information

The fuel bundles are shipped to Limerick and stored at Limerick in a package that is licensed by the U.S. Nuclear Regulatory Commission in accordance with 10 CFR Part 71, "Packaging & Transportation of Radioactive Material".

The package identification number assigned to this package by the Nuclear Regulatory Commission is USA/4986/AF. Design details and evaluations, and test results, submitted by licensee to the U.S. Nuclear Regulatory Commission and determinations by the U.S. Nuclear Regulatory Commission

relating to this package are matters of public record in Docket File 71-4986. An authenticated copy of the current Certificate of Compliance for Radioactive Materials Packages (USA/4986/AF, Revision 15, dated September 6, 1983) is attached as Exhibit 1.

The USA/4986/AF package has been identified by the Nuclear Regulatory Commission in the Certificate of Compliance as being Fissile Class I in accordance with the definition of that term in 49 CFR 173.455. This means that the USA/4986/AF package may be transported in unlimited numbers and in any arrangement, and that no criticality safety controls are required during transport.

2. Applicability of License Regulations

In August, 1983, the U.S. Nuclear Regulatory Commission published revisions in 10 CFR 71 to achieve compatibility with the transport regulations of the International Atomic Energy Agency. These revisions modified in part the container testing requirements of 10 CFR Part 71, Appendix A (currently 10 CFR 71.71) and 10 CFR 71, Appendix B (currently 10 CFR 71.73). The final rule became effective September 6, 1983, but has not been retroactively applied to this package.

Therefore, notwithstanding the changes to 10 CFR Part 71, the General Electric USA/4986/AF package meets the proper demonstration of safety and is properly licensed as provided in 10 CFR 71.31(b), which specifies information to demonstrate that the design satisfies the package standards in effect at the time the application is filed, in this case, March, 1982.

G. The shipping containers described in the Certificate of Compliance

(Exhibit 1) have characteristics which would reduce the damage to a fuel bundle below that which it would experience if it were to be subjected to an accident unprotected by such containers. The protection of the bundles would serve to reduce the radiological consequences of any accident which would disturb the pile of boxes to the extent that the shipping containers would be damaged.

The ability of the shipping containers to withstand damage has been demonstrated. The containers have been qualified under normal and hypothetical accident conditions as specified in the Certificate of Compliance. The shipping containers satisfactorily passed the following NRC required tests of hypothetical accident conditions as part of this qualification:

(a)Free Drop - Four individual drop tests through a distance of 30 feet were conducted on the package. One test was conducted using only the loaded steel inner container, excluding the customary added protection of the wooden overpack. Two dummy fuel bundles were loaded into the inner container to simulate the normal gross weight. In all tests, the cover and end caps remained intact. The inside angle spacers maintained the annulus required so that criticality safety considerations were not affected.

(b)Puncture - A puncture test was likewise performed on the loaded steel inner container without the added protection of the customary wooden overpack. The test on the inner metal container produced an indentation, but no puncture. There were no ruptured fuel rods, and even though the container was bowed approximately 2 inches, the angle spacers maintained the spacing required so that criticality safety considerations were

not affected.

(c)Thermal - A thermal test was also performed on the loaded steel inner container without the added protection of the customary wooden overpack. A gasoline fire test was selected to be most representative of the accident considered. The thermal test produced a maximum flame temperature of 893C (1640F). The gasket and other combustible materials inside the container, including foamed polyethylene cushioning and plastic rod vibration surpression spacers, completely burned away during the thirty minute test. Five hundred gallons of gasoline were consumed during the test and no abnormal thermal distortion was observed. The pressure relief valve and the burnt gasket permitted the pressure inside to be vented away and prevented rupture of the container. There was no release of the contents of the container and no physical damage to the fuel assemblies.

(d)Water Immersion - Even though the criticality analysis assumed water inleakage and demonstrated safety under these conditions, the prescribed water immersion tests were performed after the fire test to evaluate any effects. In this test, water leaked into the container since the gasket was consumed during the fire. Residue and debris remaining did not restrict the free flow of water into and out of the container. The presence of water for 8 hours caused no damage to the fuel rods.

These conditions were considered in the criticality calculations which showed the reactivity of such an array to be subcritical when flooded.

H. The piles of fuel element shipping containers are covered by a 5 sided, open bottom prefabricated metal building once the piles are created in

the new fuel storage area. This metal building would have the effect of abosrbing energy and spreading any external loading on the pile uniformly over the face of the pile. The existence of this prefabricated metal building was not considered in performing the calculations of the radiological consequences and hence its contribution to the lessening of exposures has been neglected.

3. While the consequences of the incredible non-mechanistic accidents reported were, as has been discussed, extremely small and localize , to further quantify the conservatism inherent in the storage of the new fuel on site, an evaluation was performed to evaluate the potential effects of the design railroad TNT blast on the piles of fuel storage boxes.

A. It was first assumed that the pile (conservatively assumed to be a 48 box pile, 4 boxes high x 12 boxes long) of shipping containers was placed near the fence of the new fuel storage area where the fence is the closest to the railroad. This is conservative since the piles are placed a minimum of 25 feet inside the new fuel storage area fence. The pile was assumed to be oriented with the long side of the boxes parallel to the railroad, and therefore parallel to the blast front. The effect of the blast on the top layer of shipping containers was calculated. The calculations showed that the railroad blast would accelerate the top layer of shipping boxes to a velocity such that if the boxes were postulated to strike an unyielding surface, the energy absorbed by the shipping container would be approximately 35% of that which would be absorbed in the 10 CFR Part 71 accident 30 foot free drop test. It is therefore concluded that the effect of such a blast on the shipping container and the fuel being stored inside would be less than that

experienced by the 10 CFR Part 71 hypothetical accident free drop test. This conclusion provides additional assurance that the consequerces of an accident would be further below the 10 CFR Part 20 limits than those calculated by Stansbury. B.

It was then assumed that the pile of shipping containers was placed as close to the fence as is described in A. above. For this calculation, the pile was assumed to be oriented such that the face of the pile made up of the ends of the boxes was parallel to the railroad and therefore parallel to the blast front. For this case, the blast force was applied to the end of the boxes, accelerating one box as opposed to accelerating a layer of 12 boxes as was done in A. above. This calculation showed that the railroad blast would accelerate one shipping box to a velocity such that if the boxes were postulated to strike an unyielding surface, the energy absorbed by the shipping container would be approximately 53% of that which would be absorbed in the 10 CFR Part 71 accident 30 foot free drop test. It is therefore concluded that the effect of such a blast on the shipping container and the fuel being stored inside would be less than that experienced by the 10 CFR Part 71 hypothetical accident free drop test. This conclusion provides additional assurance that the consequences of an accident would be further below the 10 CFR Part 20 limits than those calculated by Stansbury.

4. The normal handling of the fuel within its shipping containers is described in the amended Special Nuclear Materials application referenced in 2 above. The amended application details how the containers are monitored and carefully handled from the time they are delivered through inspection and unloading at the new fuel storage area. The amendment also describes the

handling of the shipping container and the fuel bundles from transfer of the shipping containers from the new fuel storage area through the placement of individual bundles in the spent fuel storage pool in the reactor building. The consequences of any accident to which the shipping containers and the fuel bundles would be subjected during this onsite handling would be bounded by the radiological consequences of the point source piles evaluated by Stansbury.

5. It is therefore concluded that there is no credible potential for violating the regulations applicable to onsite and offsite releases in the event of an accident to the unirradiated fuel bundles to be delivered to and stored at Limerick Generating Station.

Lubomir B. Pyrih

Lubomir B. Pyrih

Subscribed and sworn to
before me this 13th day
of March, 1984.

Patricia D. Scholl

Notary Public

PATRICIA D. SCHOLL
Notary Public, Philadelphia, Philadelphia Co.
My Commission Expires February 10, 1986

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	:	Docket Nos. 50-352
	:	
PHILADELPHIA ELECTRIC COMPANY	:	50-353
	:	
(Limerick Generating Station	:	
Units 1 and 2)	:	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Transmittal of Affidavits Related to Storage of Unirradiated Fuel at the Limerick Generating Station, including the affidavits referred to therein were served on the following by deposit in the United States mail, first-class postage prepaid on this 13th day of March, 1984.

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PROFESSIONAL QUALIFICATIONS
LUBOMIR B. PYRIH
ENGINEER-IN-CHARGE
NUCLEAR AND ENVIRONMENTAL SECTION
PHILADELPHIA ELECTRIC COMPANY

My name is Lubomir B. Pyrih. My business address is 2301 Market Street, Philadelphia, PA 19101. I am the Engineer-in-Charge of the Nuclear and Environmental Section. In this position, I supervise a group of engineers engaged in nuclear plant licensing, reactor core analysis, nuclear fuel management, probabilistic risk assessment, radiological environmental monitoring and aquatic environmental monitoring.

I received my Bachelor of Science in Mechanical Engineering from Drexel Institute of Technology (now Drexel University) in 1960 and my Master of Science in Mechanical Engineering from the same institution in 1965. I attended LaSalle College and received my Master of Business Administration degree in 1983.

I am a registered Professional Engineer in the Commonwealth of Pennsylvania, a member of the American Society of Mechanical Engineers and a member of the Philadelphia Section of the American Nuclear Society.

I have been employed by Philadelphia Electric Company since June of 1960. My initial employment with the Company was spent in the Electric Production Department primarily at Richmond Station, a fossil-fueled electric generating station. Since March of 1966 to the present I have been employed in various positions in the Mechanical Engineering

Division of the Engineering and Research Department. These assignments included the position of project engineer of the Muddy Run Pumped Storage Generating Plant (1967-1968), engineering review responsibilities for the design of the Peach Bottom and Limerick circulating water systems (1969-1974), engineering review responsibilites for the design of the Peach Bottom and Limerick Nuclear Steam Supply Systems (1974-1975) and the position of mechanical project engineer for the Peach Bottom Atomic Power Station (1975-1977).

In April of 1977, I was appointed Branch Head of the Nuclear Steam Supply Branch. In this position, I was responsible for the direction of all mechanical engineering activities pertaining to the Nuclear Steam Supply System and the supervision of the Peach Bottom Atomic Power Station mechanical project staff.

In January of 1981, I was appointed Branch Head of the Nuclear Branch. In this position I was responsible for the direction of nuclear plant licensing, out-of-core nuclear fuel management, and reactor core analysis.

I was appointed to my present position in January of 1984.

LBP/gra/031384930

EXHIBIT 1

NRC FORM 618
(6-83)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
4986	15	USA/4986/AF	1	4

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging of Radioactive Materials for Transport and Transportation of Radioactive Material Under Certain Conditions."

b. 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

a. PREPARED BY (Name and Address):

General Electric Company
P.O. Box 780
Wilmington, NC 28401

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

General Electric Company application dated March 1, 1982, as supplemented.

c. DOCKET NUMBER 71-4986

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model Nos.: RA-2 and RA-3

(2) Description

A fuel assembly and fuel rod shipping container. Packagings are right rectangular boxes consisting of an outer container of wooden construction and a metal inner container separated by cushioning material.

The metal inner container is 11-1/2 inches by 18 inches by 179 inches long and is positioned within a Model No. RA-3 wooden outer container approximately 30 inches by 31 inches by 207 inches long. Cushioning is provided between the inner and outer containers by phenolic impregnated honeycomb and ethafoam, or equivalent. Closure is accomplished by bolts, latches, or equivalent. A pressure relief (breather) valve is provided on the inner container, and is set for 0.5 psi differential. The total weight of the packaging and contents is 2,800 pounds.

AUTHENTICATED RECORDS
NAME Charles M. Efaughn
DATE 12 March 1984

Page 2 - Certificate No. 4986 - Revision No. 15 - Docket No. 71-4986

5. (a) Packaging (continued)

(3) Drawings

(i) The packagings are constructed in accordance with the following GE Drawing Nos.:

769E229, Revision 1, Model RA-3 Wooden Outer Container
769E231, Revision 3, Model RA-3 Inner Container
769E232, Revision 3, Model RA-2 Inner Container

The RA-3 inner container may be constructed in accordance with Exxon Nuclear Company, Inc. Drawing No. XN-NF-304,416, Rev. 2.

(4) Product Container

Five-inch, Schedule 40, stainless steel, pipe fitted with screw type or flange closure. Container shall be vented in the event it contains materials which decompose at less than 1475°F.

(b) Contents

(1) Type and form of material

(i) UO_2 fuel assemblies with a maximum average U-235 enrichment of [illegible]% by weight. Assembly rods are clad with a minimum 0.029-inch thickness of zircaloy and a maximum fuel pellet outside diameter of 0.416 inch. Each assembly, made up of a maximum 8 x 8 square array of fuel rods, must have a maximum fuel length of 174 inches with a maximum fuel cross-sectional area of 2[illegible] square inches.

(ii) UO_2 fuel assemblies with a maximum average U-235 enrichment of 2.7% by weight. Assembly rods are clad with a minimum 0.032-inch thickness of zircaloy and a maximum fuel pellet outside diameter of 0.490 inch. Each assembly, made up of a maximum 7 x 7 square array of fuel rods, must have a maximum fuel length of 174 inches with a maximum cross-sectional area of 25 square inches.

AUTHENTICATED RECORDS
NAME Charles M [illegible]
DATE 13 March 1984

5. (b) Contents (continued)

(1) Type and form of material (continued)

(iii) UO_2 fuel rods with a maximum U-235 enrich... of 5.0% by weight. Rods are clad with zircaloy, incaloy, incone... stainless steel such that the ratio of clad to fuel cross sec... l area be at least 0.26, and a maximum fuel pellet outside di. eter of 0.508 inch. Each rod must have a maximum length of 174 inches. The clad rods must be bundled (contained) in the product container described in 5(a)(4).

(iv) UO_2 fuel rods with a maximum U-235 enrichment of 6.5% by weight. Rods are clad with zircaloy, incaloy, inconel or stainless steel such that the ratio of clad to fuel cross sectional area be at least 0.26, and a maximum fuel pellet outside diameter of 0.508 inch. Each rod must have a maximum length of 174 inches. The clad rods must be bundled (contained) in the product container described in 5(a)(4).

(v) UO_2 fuel assemblies with a maximum average U-235 enrichment of 3.3% by weight for assembly rods clad with a minimum 0.030-inch thickness of zircaloy and a maximum fuel pellet outside diameter of 0.356 inch. Each assembly to be made up of a 9x9 square array of fuel rods with a maximum fuel length of 174 inches and a maximum fuel cross-sectional area of 25 square inches.

(2) Maximum quantity of material per package

(i) For the contents described in 5(b)(1)(i), 5(b)(1)(ii), and 5(b)(1)(v):

Two (2) fuel assemblies.*

(ii) For the contents described in 5(b)(1)(iii) and 5(b)(1)(iv):

Two (2) fuel bundles.*

(A bundle is defined as an arrangement of rods which are either contained within a product container or strapped together.)

*Two short fuel assemblies or bundles may be substituted for a full length fuel assembly or bundle provided the individual assemblies (or bundles) are shipped end-to-end and the total length does not exceed 174 inches.

(c) Fissile Class I

AUTHENTICATED RECORDS
NAME Charles M. Zaughn
DATE 12 March 1984

Page 4 - Certificate No. 4986 - Revision No. 15 - Docket No. 71-4986

6. Each fuel assembly must be unsheathed or must be enclosed in an unsealed, polyethylene sheath which may not extend beyond the ends of the fuel assembly. The ends of the sheath may not be folded or taped in any manner that would prevent the flow of liquids into or out of the sheathed fuel assembly.

   Polyethylene shipping shims may be inserted between rods within the fuel assemblies up to a maximum of 0.20 g H_2O equivalent per cubic centimeter averaged over the assembly.

7. In lieu of the product container specified in 5(a)(4), except for UO_2 fuel rods with U-235 enrichment greater than 3.2% and for rods described in 5(b)(1)(v), the clad rods must be bundled (bound with steel strappings at two or more locations) with a maximum cross sectional area of 20.0 square inches. The total breaking strength of the strapping must be 30 times the weight of the bound rods.

8. The maximum spacing between adjacent rods within the bundle must be 0.012 inch. The spacing must be maintained by the product container wall, metal strappings or peripheral metallic dunnage with a melting point greater than 1475°F within the bundle.

9. The package authorized by this certificate is hereby authorized for use under the general license provisions of 10 CFR §71.12.

10. Expiration date: March 31, 1987.

REFERENCES

General Electric Company application dated March 1, 1982.

Supplement dated: March 15, 1982.

Exxon Nuclear Company supplements dated: June 24, 1981; and January 11 and February 9, 1982.

Westinghouse Electric Company supplement dated: February 4, 1983.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles E. MacDonald

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

Date: SEP 06 1983

AUTHENTICATED RECORDS
NAME Charles M. Vaughan
DATE 12 March 1984