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Arthur W. Murphy, Esq., Chairman
Atomic Safety and Licensing Board
Columbia University School of Law
Box 38, 435 West 116th Street
New York, New York 10027

Re: AEC Docket Nos. 50-329 and 50-330

Dear Mr. Chairman:

Enclosed is a copy of the document list referred to in Intervenors' letter of June 10, 1971.

This list is not intended to be exhaustive of the documentary material which Intervenors may offer into evidence.

Respectfully,

Myron M. Cherry

MMC/cam
Enclosure

cc: Dr. David B. Hall
Dr. Clark Goodman
Mr. Stanley T. Robinson, Jr.
All Counsel of Record

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ZIRCONIUM

WAPD-TRANS-119 Zirconium as Material for the Reactor Core

AECL-3375 On the Oxidation of Zirconium Alloys in Air and the Dimensional Changes Associated with Oxidation

GEMP-669 Thermal Conductivity and Electrical Resistivity of Zircaloy-4

ANL-6160 Examination of an Irradiated Prototype Fuel Element for the Elk River Reactor

ANL-6232 Studies of the Hydrogen Damage Mechanism in the Corrosion of Zirconium

ANL-6370 Corrosion Studies of Ternary Zirconium Alloys in High-Temperature Water and Steam

ANL-7252 Studies in Zirconium Oxidation

BMI-1689 Studies in the UO_2-ZrO_2 System

BMI-1803 Specific Heats and Heats of Transformation of Zircaloy-2 and Low-Nickel Zircaloy-2

DP-859 Irradiation of Tandem-Extruded Joints Between Zircaloy and Stainless Steel

GA-2229 Irradiation Effects on the Surface Reactions of Metals

GA-2235 A Program of Research on Mechanical Metallurgy as Related to Fuel-Element Fabrication

GEAP-3739 Plastic Strain in Thin Fuel Element Cladding Due to UO_2 Thermal Expansion

GEAP-3983 Slow Cycle Strain Fatigue in Thin Wall Tubing

GEAP-3999 Corrosion Mechanism of Zirconium and its Alloys - Diffusion of Oxygen in Zirconium Dioxide

HW-62347 Extrusion Characteristics of Uranium-Zirconium and Uranium-Carbon Alloys

HW-65465 Hydriding in Purposely Defected, Zircaloy-Clad Fuel Rods

HW-67677 The Physical Integrity and Corrosion Resistance of the Zircaloy-2 Pressure Tubes for the PRTR

HW-67949 REV Evaluation of Properties of Irradiated Zircaloy-2 Pressure Tube from KER Loop 1

HW-68195 A Study of the Wear and Galling of Autoclaved Zircaloy-2 by Various Materials

HW-69679 Recovery and Recrystallization of Zirconium and its Alloys, Part 2 Annealing of Cold-Worked Zirconium

HW-69680 Part 3 Annealing Effects in Zircaloy-2 and Zircaloy-3

HW-70151 The Activation Energies for Creep of Zircaloy-2

HW-72602 Ultrasonic Testing of Zircaloy Sheath Tubing for Fuel Elements

HW-73398 Strength and Metallurgical Properties of the Zircaloy-2 Pressure Tubes for the PRTR

HW-73511 High Temperature Oxidation of Zirconium and Zircaloy-2 in Oxygen and Water Vapor

HW-73698 REV Postirradiation Evaluation of Zircaloy-2 PRTR Pressure Tubes

HW-74339 Impact Testing and Slow Notch-Bend Testing of Zircaloy-2

HW-74955 Effects of Cold Work and Neutron Irradiation on the Tensile Properties of Zircaloy-2

HW-75052 Postirradiation Evaluation of Zircaloy-2 Pressure Tube from KER-3, Preliminary Report

HW-75267 Creep Properties of Zircaloy-2 for Design Application

HW-76562 REV Role of the Oxidation Rate on the Hydriding of Zirconium Alloys in Gas Atmospheres Containing Hydrogen

HW-76636 Neutron Irradiation and Cold Work Effects on Zircaloy-2 Corrosion and Hydrogen Pickup

HW-80309 The Effects of Hot-Water Thermal Treatments on the Cold Work Recovery of Zircaloy-2

HW-80567 Crack Propagation Tests on Normal and Hydrided Zircaloy-2 Reactor Pressure Tubing

HW-82102 Effects of Neutron Irradiation on the Flow and Fracture Behavior of Zircaloy-2

HW-82631 Fracture Studies of Zircaloy-2

HW-83164 Postirradiation Evaluation of Zircaloy-2 PRTR Pressure Tubes

IN-1389 Brittle Behavior of Zircaloy in an Emergency Core Cooling Environment

KAPL-2111 Mechanical Properties of Zircaloy-2 Weld Metal

KAPL-2110 Mechanical Properties of Zircaloy-2

KAPL-2149 Hydrogen Absorption by Zirconium-2 at. % Tin-2 at. % Niobium Alloy During Corrosion

KAPL-2203 Corrosion of Zircaloy in Crevices Under Nucleate Boiling Conditions

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NMI-1233 Interdiffusion in Zircaloy-2 Clad U_2 w/o Zr Fuel Materials and its Effect Upon Corrosion Behavior

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ORNL-3514 Mechanical Cladding-Fuel Interactions During Thermal Cycling of Metal-Clad Fuel Elements

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ANL-7266 Hydrogen Embrittlement in Irradiated Steels

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DP-1199 Repair of a Nuclear Reactor Vessel

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ORNL-2834 An Investigation of the Corrosion Resistance of Brazing Alloys for Austenitic Stainless Steel Fuel Elements for Service in 565°F Pressurized Water

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ORNL-NSIC-15 The Integrity of Reactor Pressure Vessels

TID-7625 Technical Papers of the Thirteenth Metallographic Group Meeting

TID-17682 A Survey of 21 Reactor Vessels in Light of the Anticipated Effects of Neutron Irradiation on the Brittle-Rupture Properties of Materials of Construction

TID-17887 The True Stress-Strain Properties of Brittle Materials to Very High Temperatures

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WAPD-T-2222 Assessment of Safety Injection Strategies During Loss-of-Coolant Accidents

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IDO-17219 Loft Core Design Report

IDO-17258 Loft Engineered Safety Systems Investigations

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IN-1355 SECHT III An Experimental Investigation of Top and Bottom Flooding of a Nuclear Bundle Simulator

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IN-1390 Experimental Results of the Fuel Heatup Simulation Tests (FHUST) - Emergency Core Cooling Test Series

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ORNL NSIC 22	Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants
ORNL-NSIC-25	Air Cleaning as an Engineered Safety Feature in Light-Water-Cooled Power Reactors
ORNL NSIC 26	Testing of Containment Systems Used with Light-Water-Cooled Power Reactors
ORNL-NSIC-27	Review of Methods of Mitigating Spread of Radioactivity from a Failed Containment System
ORNL-NSIC-15	The Integrity of Reactor Pressure Vessels
ORNL-NSIC-28	Earthquakes and Nuclear Power Plant Design
ORNL-NSIC-29	Protection Instrumentation Systems in Light-Water-Cooled Power Reactor Plants

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ORNL-3776	for period ending December 31, 1964
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