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PROD. & UTIL. FAC. 50-329,330

McDERMOTT, WILL & EMERY

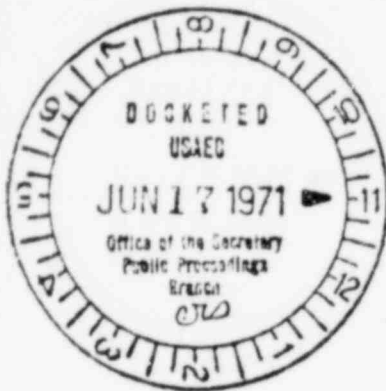
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CHICAGO, ILLINOIS 60603

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June 15, 1971

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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

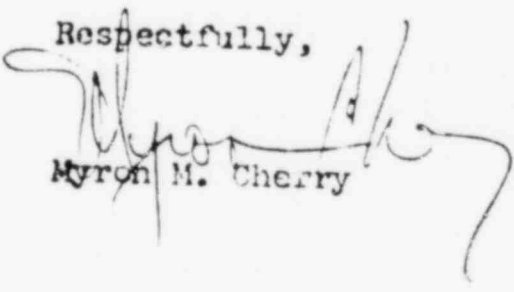
Re: AEC Docket Nos. 50-329 and 50-330

Dear Mr. Chairman:

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

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

Respectfully,

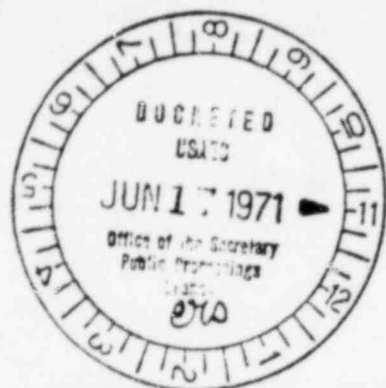

Myron M. Cherry

MMC/cam
Enclosure

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

8007170

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CKET NUMBER
PROD. & UTIL. FAC. 50-329,330

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-72002 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 PKTR 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
- KAPL-2221 Effect of Hydrogen on the Strain Fatigue Properties of Zircaloy-2 Weld Metal
- NMI-1233 Interdiffusion in Zircaloy-2 Clad U₂ w/o Zr Fuel Materials and its Effect Upon Corrosion Behavior
- NMI-1243 Deformation Modes of Zirconium at 77°K, 300°K, and 1075°K
- ORNL-3039 Review and Correlation of In-Pile Zircaloy-2 Corrosion Data and a Model for the Effect of Irradiation
- ORNL-3398 The Effect of Stress State on High-Temperature Low-Cycle Fatigue
- ORNL-3514 Mechanical Cladding-Fuel Interactions During Thermal Cycling of Metal-Clad Fuel Elements
- ORNL-TM-2347 Failure Modes of Zircaloy-Clad Fuel Rods

STRUCTURAL STEELS

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ANL-6389 Design Criteria for Steel in Nuclear Reactors

ANL-6781 Analytical and Experimental Investigation of a
Nuclear Reactor Support Structure

ANL-6868 Annual Report for 1963 Metallurgy Division

ANL-7162 Stress Analysis of a Reactor Core Support Structure
Consisting of Two Interconnected Multiregion Plates

ANL-7266 Hydrogen Embrittlement in Irradiated Steels

BMI-1813 Investigation of the Initiation and Extent of Ductile
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BMI-1828 October-December, 1967

BMI-1834 Effects of Large Fast Fluences on Mechanical Properties
of Type 347 Stainless Steel and Aluminum

BMI-1836 Investigation of...Pipe Rupture January-March, 1968

BMI-1873 Ditto

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BMI-1876 Ditto, Phase II

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- BNL-1093 Calculated Damage Functions for Determining
Irradiation Effectiveness
- DP-579 Mechanical Properties of Irradiated Stainless Steels
- DP-531 Mechanical Properties of Irradiated Plain Carbon and
Alloy Steels
- DP-1089 A Mechanism for Stress Corrosion Cracking of Stainless
Steel in Reactor Systems
- DP-1199 Repair of a Nuclear Reactor Vessel
- GA-3585 A Program of Basic Research on Mechanical Properties
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- HW-66425 Some Effects of Neutron Radiation on the Mechanical
Properties and Structural Characteristics of High-
Purity Iron
- IN-1112 Characterization of the Fracture Surface of the PM-2A
Pressure Vessel
- IN-1398 Incipient Failure Detection by Acoustic Emission
A Development and Status Report
- ORNL-2834 An Investigation of the Corrosion Resistance of
Brazing Alloys for Austenitic Stainless Steel Fuel
Elements for Service in 565°F Pressurized Water
- ORNL-2972 Effect of Environment on the Creep Properties of Type
304 Stainless Steel at Elevated Temperatures
- ORNL-4512 Heavy-Section Steel Technology Program Semiannual
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- ORNL-4590 February 28, 1970
- ORNL-4315 February 29, 1968
- ORNL-NSIC-15 The Integrity of Reactor Pressure Vessels
- TID-7625 Technical Papers of the Thirteenth Metallographic
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- 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

IITRI-578-P-21-39 Water Decompression Experiments and Analysis
for Blowdown of Nuclear Reactors

ANL-7609 Laboratory Simulations of Cladding-Steam Reactions
Following Loss-of-Coolant Accidents in Water-Cooled
Power Reactors

IDO-17219 Loft Core Design Report

IDO-17258 Loft Engineered Safety Systems Investigations

IDO-17258C Semiscale Blowdown and ECC

IDO-17258J ORNL-Loft Fission Product Support Program

IDO-17258K Loft Integral Test Program

IDO-17258F Fuel Heatup Simulation Tests - FHUST

IN-1321 Relap3 - A Computer Program for Reactor Blowdown Analysis

IN-1383 Technical Assistance in Reactor Safety Analysis

IN-1384 Semiscale Blowdown and Emergency Core Cooling Project

IN-1388 Review of Heat Transfer Coefficients for Condensing
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Coolant Accident

UC-80 Semiscale Blowdown and Emergency Core Cooling Project

IN-1393 Test Report - Tests 822 and 823

UC-80 IN-1404 Tests 803-820

IN-1428 Particle Size Distributions from Fuel Rods Fragmented
During Power Burst Tests in the Capsule Driver Core

- IN-1348 Experimental Investigations of Reactor System Blowdown
- IN-1354 Subcooled-Blowdown Forces on Reactor-System Components:
Calculational Method and Experimental Confirmation
- IN-1355 SECHT III An Experimental Investigation of Top and
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- IN-1362 STRAP A Computer Code for Static and Dynamic Structural
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- IN-1390 Experimental Results of the Fuel Heatup Simulation
Tests (FRUST) - Emergency Core Cooling Test Series
- IN-1391 Loft Core Length Study
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- ANL-6702 Heat Capacity Studies of Uranium and Uranium-Fissium Alloys
- ANL-6883 Examination of Irradiated Ag-In-Cd Alloys
- ANL-7070 Mechanical Properties of Uranium Compounds
- HW-67072 Final Report: Irradiation of Zircaloy-2 Clad Uranium Rods in NaK-Filled Capsules,
- HW-69234 Irradiation Effects on Uranium Dioxide Melting
- HW-69393 Swelling in Uranium, a Comparison of the Effects of Irradiation and Postirradiation Annealing
- NLCO-988 Structural Changes Associated with High-Temperature Deformation of Uranium
- NLCO-990 Rolling Studies of Uranium Rods and Tubes
- K-1778 Formation of Corrosion-Resistant Oxide Film on Uranium
- AEC-tr-4467 Thermal Cycling Equipment and Experimental Data on Uranium. Studies on Uranium Fuel Element 1.
- GA-2691 A Program of Research on Mechanical Metallurgy as Related to Fuel-Element Fabrication
- TID-11295 Nuclear Fuels and Materials Development Program Summary for 1965
- ORNL-4440 Fuels and Materials Development Program Quarterly Progress Report for period ending June 30, 1969
- ORNL-4480 for period ending September 30, 1969
- ORNL-4520 for period ending December 31, 1969
- ORNL-4560 for period ending March 31, 1970

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- ORNL-NSIC-23 Potential Metal-Water Reactions in Light-Water-Cooled Power Reactors
- 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

SAFETY RESEARCH REPORTS

BNWL-1315-1 Nuclear Safety Quarterly Report, Nov, Dec, 1969, Jan, 70

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MISCELLANEOUS GENERAL REFERENCES

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- LA-4316 The Application of Risk Allocation to Reactor Siting and Design
- LA-4449 A Risk Analysis of the Omega West Reactor
- ORNL-HUE-11 The Siting of Nuclear Reactors as Related to Urban Heat Supply
- ORNL-NSIC-64 Abnormal Reactor Operating Experiences 1966-68
- ORNL-NSIC-69 Safety-Related Occurrences in Nuclear Facilities as Reported in 1967 and 1968
- IDO-17252 A Digital Computer Code for Predicting the Pressure-Temperature History within a Pressure-Suppression Containment Vessel in Response to a Loss-of-Coolant Accident
- IN1330 Analysis of Fault Trees by Kinetic Tree Theory
- TID-25537 Operating History - U.S. Nuclear Power Reactors
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TWO PHASE FLOW AND OSCILLATION

- CNEC 156 Water Injection Test Simulating Moderator Flooding of
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- NASA TN D-3553 Stability of Intermixing of High-Velocity
Vapor with its Subcooled Liquid in Cocurrent Streams
- NASA Experimental Study of Subcooled Nucleate Boiling of
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Pressures
- WAPD-T-1824 Flow Patterns in High Pressure Two-Phase (Steam-
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FUEL ROD FAILURE EXPERIMENTS PERFORMED AT O.R.N.L.

ORNL-TM-2919	ORNL Nuclear Safety Research and Development Program Bimonthly Report for January-February 1970
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ORNL-TM-3061	ORNL ...Bimonthly Report for May-June 1970
ORNL-TM-2538	ORNL ...Bimonthly Report for March-April 1969
ORNL-TM-2548	Failure Modes of Zircaloy-Clad Fuel Rods
ORNL-4635	Final Report on the first Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in Treat
UC-80	A Metallurgical Evaluation of Simulated BWR Emergency Core Cooling Tests
ORNL-TM-2829	ORNL Nuclear Safety Research and Development Program Bimonthly Report for November-December 1969
GEMP-1008	Physico-Chemical Studies of Clad UO_2 Under Reactor Accident Conditions
ORNL-TM-2850	Design, Equipment, and Program for Fuel Rod Burst Experiments