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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)
NUCLEAR MANAGEMENT) Docket No. 50-255-LR
COMPANY, LLC)
(Palisades Nuclear Plant)) ASLBP No. 05-842-03-LR

CERTIFICATE OF SERVICE

I hereby certify that copies of the "PETITIONERS' Appendix of Evidence In Support of Contentions" in the above-captioned proceeding have been served with copies by U.S. First Class mail this 16th day of September, 2005:

Office of the Secretary
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Mail Stop: O-16C1
U.S. Nuclear Regulatory Commission
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Office of Commission Appellate
Adjudication
Mail Stop O-16C1
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dr. Anthony Baratta
Administrative Judge
Atomic Safety and Licensing Board Panel
Mail Stop: T-3F23
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dr. Nicholas G. Trikouros
Administrative Judge
Atomic Safety and Licensing Board Panel
Mail Stop: T-3F23
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Ann Marshall Young
Administrative Judge
Atomic Safety and Licensing Board Panel
Mail Stop: T-3F23
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Kary Love, Esq.
Executive Business Center
348 Waverly Road, Suite 2
Holland, MI 49423

Paul Gunter
Nuclear Information & Resource Service
1424 16th Street, NW
Suite 404
Washington, DC 20036

Chuck Jordan
Chairman
Green Party of Van Buren County
50521 34th Avenue
Bangor, MI 49013

Alice Hirt
Western Michigan Environmental Action Co.
1415 Wealthy Street, SE
Suite 280
Grand Rapids, MI 49506

Michael Keegan
Co-Chair
Don't Waste Michigan
2213 Riverside Drive, NE
Grand Rapids, MI 49505

Maynard Kaufman
Michigan Land Trustees
25485 County Road 681
Bangor, MI 49013

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SECY-02

Terry Lodge, Esq.
316 N. Michigan St. Suite 520
Toledo, OH 43624-1627

David R. Lewis, Esq.
Pillsbury Winthrop Shaw Pittman, LLP
2300 N Street, N.W.
Washington, DC 20037-1128

Jonathan Rogoff, Esq.
Vice President, Counsel, & Secretary
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

Susan Uttal, Esq.
U.S. Nuclear Regulatory Commission
Office of the General Counsel
Mail Stop: O-15D21
Washington, D.C. 20555

/s/ Terry J. Lodge / PG
Terry J. Lodge

EXHIBIT 1 A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 27, 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON PALISADES PLANT

Dear Dr. Seaborg:

At a Special Meeting, January 23-24, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by Consumers Power Company for authorization to operate the Palisades Plant at power levels up to 2200 MWt. This project was also considered at the 113th ACRS meeting, September 4-6, 1969, the 115th ACRS meeting, November 6-8, 1969, and the 116th ACRS meeting, December 11-13, 1969. Subcommittee meetings were held on July 31, 1969, at the site, and on October 29, 1969, December 3, 1969, and January 22, 1970, in Washington, D. C. During its review, the Committee had the benefit of discussions with representatives of Consumers Power Company, Combustion Engineering, Inc., Bechtel Corporation, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported to you on the construction of this plant in its letter dated January 18, 1967.

The site for the Palisades Plant consists of 487 acres on the eastern shore of Lake Michigan in Covert Township, approximately four and one-half miles south of South Haven, Michigan. The minimum exclusion radius for the site is 2300 feet and the nearest population center of more than 25,000 residents consists of the cities of Benton Harbor and St. Joseph, Michigan, which are approximately 16 miles south of the site.

The nuclear steam supply system for the Palisades Plant is the first of the Combustion Engineering line currently licensed for construction. A feature of the Palisades reactor is the omission of the thermal shield. Studies were made by the applicant to show that omission of the shield would not adversely affect the flow characteristics within the reactor vessel or alter the thermal stresses in the walls of the vessel in a manner detrimental to safe operation of the plant. Surveillance specimens in the vessel will be used to monitor the radiation damage during the life of the plant. If these specimens reveal changes that affect the safety of the plant, the reactor vessel will be annealed to reduce

radiation damage effects. The results of annealing will be confirmed by tests on additional surveillance specimens provided for this purpose. Prior to accumulation of a peak fluence of 10^{19} nvt (> 1 Mev) on the reactor vessel wall, the Regulatory Staff should reevaluate the continued suitability of the currently proposed startup, cooldown, and operating conditions.

The secondary containment is a reinforced concrete structure consisting of a cylindrical portion prestressed in both the vertical and circumferential directions, a dome roof prestressed in three directions, and a flat non-prestressed base. Before operation, it will be pressurized and extensive measurements will be made of gross deformations and of strains in the linear, reinforcement, and concrete, and the pattern and size of cracks in the concrete will be observed and measured. The applicant has proposed suitable acceptance criteria for the pressure test, and the ACRS recommends that the Regulatory Staff review and assess the results of this test prior to operation at significant power.

The prestressing tendons in the containment consist of ninety, one-quarter-inch diameter wires. They are not grouted or bonded, and are protected from corrosion by grease pumped into the tendon sheaths. The applicant has proposed that selected tendons be inspected periodically for broken wires, loss of prestress, and corrosion. If degradation is detected, the inspection can be extended to the remaining tendons, all of which are accessible. The applicant is performing studies to determine the appropriate number and interval for tendon inspection. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The core is calculated to have a slightly negative moderator coefficient at full power operation at beginning-of-life, but uncertainties in the calculations are such that the existence of a positive moderator coefficient cannot be precluded. The applicant has stated that the moderator coefficient will not exceed $+0.5 \times 10^{-4}$ Δ k/k/°F at beginning-of-life, computed from start-up test data on a conservative basis. The applicant also plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses required to establish the value of the moderator coefficient.

The meteorological observation program conducted at the site subsequent to the Committee's report to you on January 18, 1967, indicated the need for the addition of iodine removal equipment to the containment for use in the unlikely event of a loss-of-coolant accident. The applicant proposed to install means for adding sodium hydroxide to the water in the containment spray system. However, because of uncertainties regarding the generation of hydrogen and the effects of other materials resulting

from the reaction of this alkaline solution with the relatively large amounts of aluminum in the containment, this spray additive will not be used unless it can be shown by further studies that the use of sodium hydroxide is clearly acceptable. In addition, the applicant will carry out studies of iodine removal by borated water sprays without sodium hydroxide. If the results of these studies are not acceptable, a different iodine removal system satisfactory to the Regulatory Staff will be installed at the first refueling outage. A report on the applicant's plans will be submitted to the AEC within six months following issuance of a provisional operation license. The Committee believes that this procedure is satisfactory for operation at power levels not exceeding 2200 MWt.

The applicant has stated that if fewer than four primary coolant pumps are operating, the reactor overpower trip settings will be reduced such that the safety of the reactor is assured in the absence of automatic changes in the thermal margin trip settings.

The Committee believes that, for transients having a high probability of occurrence, and for which action of a protective system or other engineered safety feature is vital to the public health and safety, an exceedingly high probability of successful action is needed. Common failure modes must be considered in ascertaining an acceptable level of protection. Studies are to be made on further means of preventing common failure modes from negating scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant should consider the results of such studies and incorporate appropriate provisions in the Palisades Plant.

The Committee recommends that attention be given to the long-term ability of vital components, such as electrical equipment and cables, to withstand the environment of the containment in the unlikely event of a loss-of-coolant accident. This matter is applicable to all large, water-cooled power reactors.

Continuing research and engineering studies are expected to lead to enhancement of the safety of water-cooled reactors in other areas than those mentioned: for example, by determination of the extent of the generation of hydrogen by radiolysis and from other sources, and development of means to control the concentration of hydrogen in the containment, in the unlikely event of a loss-of-coolant accident; by development of instrumentation for inservice monitoring of the pressure vessel and other parts of the primary system for vibration and detection of loose parts in the system; and by evaluation of the consequences of water contamination by structural materials and coatings in a loss-of-coolant accident. As solutions to these problems develop and are evaluated

by the Regulatory Staff, appropriate action should be taken by the applicant on a reasonable time scale.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Palisades Plant can be operated at power levels up to 2200 MWt without undue risk to the health and safety of the public.

Sincerely yours,

/s/ Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References:

1. Final Safety Analysis Report for the Palisades Plant

2. Amendments No. 9-19 to license application

EXHIBIT 1 B

NRC STAFF PRESENTATION

TO THE

ACRS

SUBJECT: PALISADES PRESSURED THERMAL SHOCK

DATE: DECEMBER 9, 1994

PRESENTER: BARRY J. ELLIOT
SENIOR MATERIALS ENGINEER
MATERIALS AND CHEMICAL ENGINEERING BRANCH
DIVISION OF ENGINEERING
OFFICE OF NUCLEAR REACTOR REGULATION
(301) 504-2709

10 CRF 50.61 RT_{PTS} EVALUATION

RT_{PTS} SCREENING CRITERIA PER 10 CFR 50.61

-270 °F FOR AXIAL WELDS AND PLATES

-300 °F FOR CIRCUMFERENTIAL WELDS

$$RT_{PTS} \text{ VALUE} = I + M + \Delta RT_{PTS}$$

I = INITIAL REFERENCE TEMPERATURE (RT_{NDT}) OF THE UNIRRADIATED MATERIAL.

-MEASURED VALUES MUST BE USED IF AVAILABLE.

-IF GENERIC VALUE NOT AVAILABLE, GENERIC MEAN VALUES MUST BE USED

M = MARGIN TO COVER UNCERTAINTIES IN THE VALUES OF INITIAL RT_{NDT}, COPPER AND NICKEL CONTENTS, FLUENCE AND THE CALCULATION PROCEDURES.

ΔRT_{PTS} = MEAN VALUE OF THE ADJUSTMENT IN REFERENCE TEMPERATURE CAUSED BY IRRADIATION AND IS A FUNCTION OF NEUTRON FLUENCE, PERCENT COPPER AND PERCENT NICKEL

-CALCULATED USING SURVEILLANCE DATA

-IF SURVEILLANCE DATA IS UNAVAILABLE, THE ADJUSTMENT IN REFERENCE TEMPERATURE MAY BE CALCULATED FROM TABLES USING THE BEST-ESTIMATE PERCENT COPPER AND NICKEL

PALISADES PTS

SINCE THE SURVEILLANCE WELD MATERIAL IN PALISADES IS NOT THE SAME AS THE BELTLINE WELDS, THE LICENSEE MUST DETERMINE THE EFFECT OF RADIATION USING NUCLEAR INDUSTRY DATA

THE STAFF MET WITH THE LICENSEE ON MARCH 9, 1994 TO DISCUSS THE LICENSEES PROGRAM FOR FURTHER EVALUATION OF THE CRITICAL WELDS IN THEIR RPV

THE LICENSEE PLANNED TO:

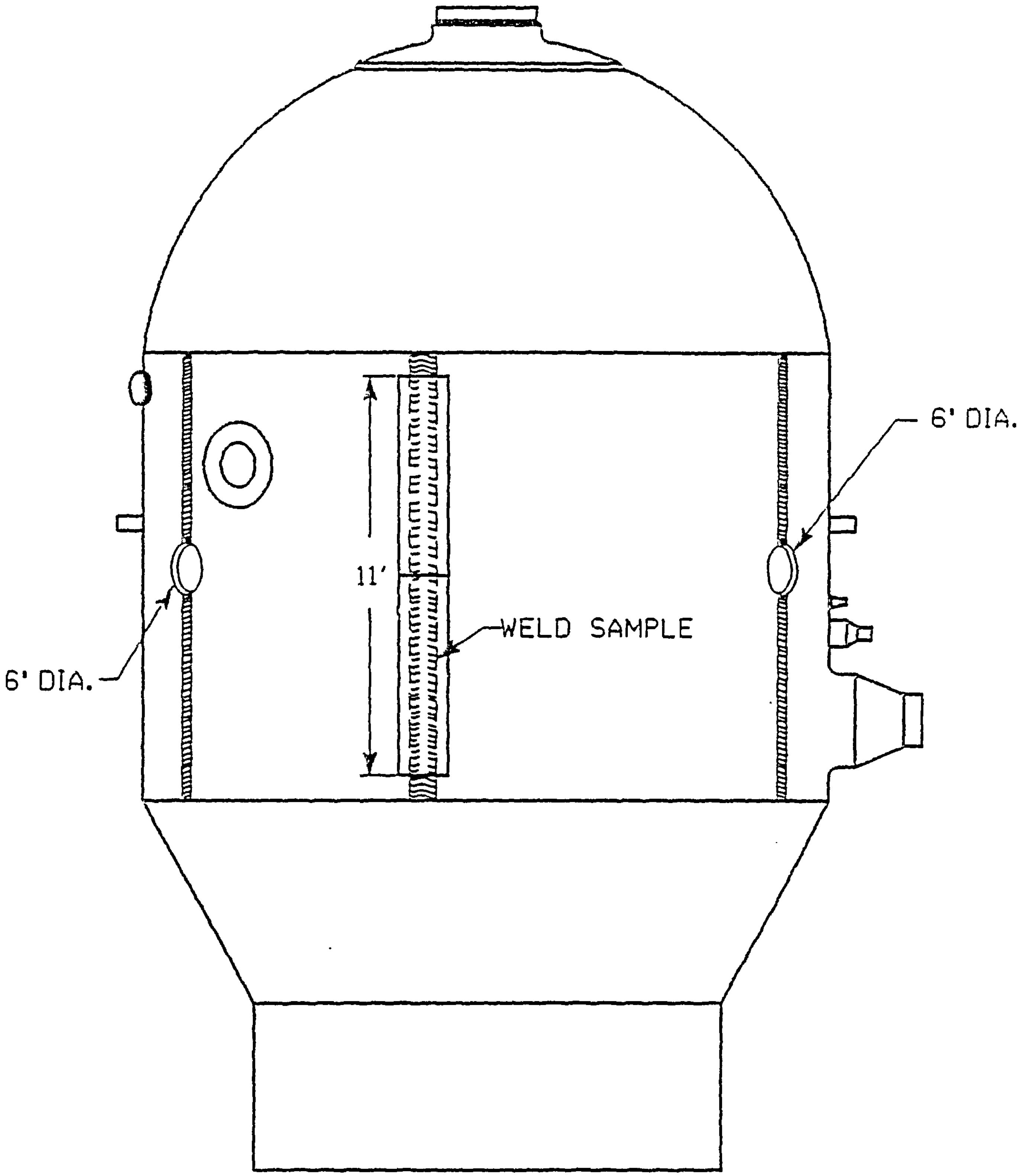
-GATHER ADDITIONAL MATERIALS PROPERTIES DATA FROM ITS RETIRED STEAM GENERATORS (WELDS FABRICATED USING W5214 AND 34B009 WELD WIRE)

-INSTITUTE AN AUGMENTED SURVEILLANCE PROGRAM THAT WOULD CONTAIN THE LIMITING WELD METAL

-EVALUATE ANNEALING OF THE REACTOR VESSEL

-CONSIDER INSTITUTING AN "ULTRA LOW" LEAKAGE FUEL STRATEGY

SGA W5214/SGB 34B009



PALISADES PTS cont.

STAFF ISSUED AN INTERIM SER ON JULY 12, 1994 AND ISSUED A COMMISSION PAPER AND NUREG REPORT ON RPVs ON OCTOBER 28, 1994. THESE DOCUMENTS STATED:

-BASED ON PREVIOUS NUCLEAR INDUSTRY DATA THE PALISADES REACTOR VESSEL WAS PROJECTED TO REACH THE PTS SCREENING CRITERIA IN 2004, PRIOR TO EOL, 2007

-STAFF SER NOTED THAT THE PTS EVALUATION COULD CHANGE BASED ON THE INFORMATION TO BE ACQUIRED FROM THE SG WELDS

ON NOVEMBER 1 THE LICENSEE INFORMED THE STAFF BY TELEPHONE THAT THE CHEMISTRY DATA FROM THE W5214 WELDS INDICATED HIGHER COPPER CONTENTS THAN PREVIOUSLY ASSUMED.

-EVALUATION OF THE STEAM GENERATOR WELD MATERIAL ALSO INDICATED A HIGHER INITIAL RT_{NDT} VALUE THAN THE MEAN GENERIC VALUE.

ON NOVEMBER 18 THE LICENSEE SUBMITTED THEIR ASSESSMENT OF THE IMPACT OF THESE NEW DATA ON THE RT_{PTS} VALUE. THIS ASSESSMENT INDICATES THAT PALISADES REACTOR VESSEL WOULD REACH THE PTS SCREENING CRITERIA IN 1999

STAFF MET WITH THE LICENSEE ON NOVEMBER 21, 1994 TO DISCUSS THE NEW INFORMATION.

STAFF REQUEST FOR ADDITIONAL INFORMATION SENT TO LICENSEE ON NOVEMBER 30, 1994.

STAFF EVALUATION IS SCHEDULED TO BE COMPLETED BY JANUARY 31, 1995.

PALISADES PTS cont.

THE STAFF IS CURRENTLY REVIEWING THE LICENSEE'S NOVEMBER 18 SUBMITTAL.

CRITICAL AREA BEING ASSESSED INCLUDE:

- -EFFECT OF THERMAL AGING, HEAT TREATMENT AND TEST METHOD ON UNIRRADIATED REFERENCE TEMPERATURE

- -BEST ESTIMATE CHEMICAL COMPOSITION FROM STEAM GENERATOR AND NUCLEAR INDUSTRY DATA

DEPENDING UPON HOW THE NEW DATA ARE USED IN THE ANALYSIS THE PTS SCREENING LIMIT COULD BE REACHED BEFORE 1999

STAFF WILL RECEIVE TECHNICAL ASSISTANCE FROM RES CONTRACTOR, ORNL

GENERIC IMPLICATIONS OF NEW DATA

REVIEW OF OTHER RPVs WITH PALISADES WELD MATERIAL (i.e. W5214 or 34B009 WELD METAL)

- -OTHER PLANTS STILL SATISFY PTS SCREENING CRITERIA AND UPPER SHELF ENERGY CRITERIA

- -LOWER FLUENCE OR USE OF ACTUAL SURVEILLANCE DATA

OTHER PLANTS THAT ARE PROJECTED TO BE NEAR THE PTS SCREENING CRITERIA BEFORE END-OF-LIFE ARE BEING ASSESSED

- -SENSITIVITIES BEING STUDIED

- -PROACTIVE MEASURES MAY BE APPROPRIATE

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FINANCIAL DESK

Cheap and Abundant Power May Shutter Some Reactors

By MATTHEW L. WALD (NYT) 1518 words
Published: April 14, 1992

Nuclear plants that provide 10 percent of the nation's nuclear power may be closed this decade because their operating costs are too high to compete against a rising tide of cheap surplus electricity, experts say.

More than 100 plants under construction were abandoned in the 1970's and 80's because of their cost. But the idea that an operating nuclear plant is not competitive with other sources of electricity violates the fundamental logic of nuclear power, which is that plants may be expensive to build but are cheap to run.

"It used to be that everyone said, once you built it, there wasn't any question that costs were lower," said Victor Gilinsky, an energy consultant and former member of the Nuclear Regulatory Commission. "Now they are more expensive to run than other plants." Aging Steam Generators

In the next few years, at least 10 utilities will need to replace steam generators, which are giant heat exchangers that have shown a tendency to rust and crack, said Gary R. Doughty, an expert on plant life extension with the Nielsen Wurster Group, a consulting firm in Hartford. The job generally runs about $150 million for each reactor.

Other utilities face questions about the condition of their reactor vessels, the great steel pots that hold the fuel. Years of bombardment by neutrons, the subatomic particles that sustain a chain reaction, are known to make metal more brittle, but the extent of the problem is not clear.

Some utilities that operate a single reactor may be tempted to pull the plug, he said, because that would allow elimination of an entire division.

In Rowe, Mass., the owners of the 32-year-old Yankee Rowe reactor decided in February that the plant was too small and too old to justify the investment needed to keep it in service, given the general power surplus in its region. Southern California Edison reached a similar judgment recently about its 24-year-old San Onofre 1 plant near San Clemente, although the plant has not yet been shut. And last year the Sacramento Municipal Utility District decided to shut the Rancho Seco plant as uneconomic at the age of 15. Others around the country were retired in earlier years, some at even younger ages.

With only a handful of additional plants likely to be finished and no new ones on order, the result could be an accelerated march to the extinction of nuclear power in the United States. Currently, 108 are

operating, producing about 20 percent of the nation's electricity. Some of those, however, are doing very well; in 1991, 25 plants set records for themselves in the number of kilowatt-hours produced.

John F. Ahearne, a former member of the Nuclear Regulatory Commission and now the director of Sigma Xi, the Scientific Research Society, said that plants that were not economic were more likely to be shut now than they would have been a few years ago. In the last 10 years, he said, the utilities have come to be dominated by business managers, replacing what he called "technologists," or "people who were committed to nuclear power because they thought it was just a good thing for this country." The Bottom Line

In the view of the business managers, he said, "the role of a utility is to make money." They are the people who canceled over-budget reactor construction projects in the 1980's, he said, and they are willing to shut plants now if there are cheaper alternatives.

The price of oil, which is currently low, plays a small role in keeping the electricity market highly competitive, especially in places like New York, which uses oil for about 20 percent of electricity generation. But nationally, electricity made from oil is less than 5 percent of total generation.

Natural gas plays a far larger role, because it represents about 10 percent of the utilities' fuel use nationally, and about half the generators recently completed or under construction use natural gas. On the basis of energy content, natural gas prices have been substantially below oil prices recently.

In addition, overall demand for power has been driven down by recession and by conservation measures, with utilities often subsidizing customers' installation of light bulbs, motors and other devices that will do the same work with less power. Price May Rise

Some experts believe that as the economy turns around, the demand for power will rise and hence its price. In addition, requirements of the new Clean Air Act will raise the cost of coal-fired power, and if the United States institutes a carbon tax in the next few years to stave off global warming, that would make nuclear power more competitive, too.

Experts are not sure how many nuclear plants will shut in this decade. The chairman of the Nuclear Regulatory Commission, Ivan Selin, said in a telephone interview that three or four were vulnerable soon. Mr. Ahearne said it could be 10 by the end of the decade.

Mr. Selin said it was unlikely that any utility would decide to close a plant that was running smoothly and was not in immediate need of any big investment. But if a plant required a large investment, he said, "that could push it over the brink." In that category he put the Consumers Power Company's Palisades plant, near South Haven, Mich., which opened in 1971, where the pressure vessel may now be brittle, the same weakness that was suspected at Yankee Rowe; Consumers Power's Big Rock Point plant, in Charlevoix, Mich., opened in 1965, which has no known significant flaws but is by far the smallest still operating, and Rochester Gas and Electric's Robert E. Ginna plant, near Rochester, which opened in 1970 and faces the expensive replacement of its steam generators.

All those plants are old and fairly small. Mr. Selin said it was far from clear whether the problem would extend into the large plants that entered service in the mid-1970's. But it might, he said in a telephone interview.

"There are two ways of looking at it," Mr. Selin said. "You can say each is different, and there is no trend, or you can say there's an underlying trend here. The financial people are beginning to worry

about an underlying trend."

In fact, Lehman Brothers organized a conference for utility investors last month on the question of whether old plants were still economic. It drew two dozen investment managers.

The Utility Data Institute, a firm in Washington that charts operating costs, reported recently that in 1990 fuel, operating and maintenance expenses at nuclear plants came to $21.89 for one thousand kilowatt-hours produced, about as much electricity as a typical household uses in two months. At a coal plant, the fuel, operating and maintenance cost for the same amount of energy was $20.24. The coal cost was up slightly in 1990 and the nuclear cost down compared with 1989, but nuclear has exceeded coal for the last several years.

Those figures are an average for all nuclear plants, meaning that some are significantly higher.
Relicensing a Question

The old reactors have a variety of factors working against them.

Mr. Doughty of Nielsen Wurster pointed out that a plant that was nearing the expiration of its 40-year operating license and needed major investments would have to face the economics of amortizing the expenses over the few remaining years of operation. The Nuclear Regulatory Commission has established a policy for granting license extensions, but no plant has yet applied and no one is sure how easy it will be to get one.

Carl A. Goldstein, a spokesman for the U.S. Council for Energy Awareness, the nuclear industry's public relations arm, said that more plants would probably be found to be uneconomic, but that the point at which a plant should be written off could not be defined until the Nuclear Regulatory Commission made clearer what would be required for a plant to be re-licensed. And nuclear economics could improve, he said, because plant operating and maintenance expenses could decline.

Mr. Doughty said that investing new money still made good sense for most plants, but that he feared that reactors with 6,000 megawatts of capacity, or about 6 percent of the nation's total nuclear capacity, would shut in the next few years. Reason to Stay Open

How much is ultimately closed may depend on how state rate regulators handle the costs, said Peter Bradford, the chairman of the Public Service Commission in New York and also a former member of the Nuclear Regulatory Commission. Mr. Bradford, a speaker at the Lehman Brothers session, said a utility with a large investment in a reactor might seek to keep it running so it could continue to collect depreciation, even if cheaper power were available elsewhere.

That, he said, would create a conflict between the interest of customers, who would want the plant closed, and the interest of the utility, which would want to let it run. The solution, he said, would be to allow utilities to write off plants that had become economically obsolete, and collect the investment from customers.

"Otherwise, the utility doesn't have the incentive to make the right decision," he said.

Photo: Utilities may be tempted to pull the plug on existing nuclear plants as they become too expensive to operate. Owners of the 32-year-old Yankee Rowe reactor in Rowe, Mass., closed it in February because the plant was too small and too old to justify the investment needed to keep it in service. (Associated Press) (pg. D25) Table: "Nuclear Plant Retirements" Plant, location Years in operation

Size, in megawatts San Onofre 1, San Clemente, Calif. 1968-1992 or 1993 436 Yankee Rowe, Rowe, Mass. 1961-1991 175 Rancho Seco, Sacramento, Calif. 1975-1989 918 Fort St. Vrain, Platteville, Colo. 1979-1989 330 La Crosse, Genoa, Wis. 1969-1988 50 Dresden 1, Morris, Ill. 1959-1978 207 Humboldt Bay, Eureka, Calif. 1962-1976 65 Shippingport, Shippingport, Pa. 1957-1982 60 Indian Point 1, Buchanan, N.Y. 1962-1980 265 Peach Bottom 1, Peach Bottom, Pa. 1966-1974 40 Fermi 1, Newport, Mich. 1963-1972 61 Elk River, Elk River, Minn. 1962-1968 22 CVTR, Puerto Rico 1962-1967 17 Pathfinder, Sioux Falls, S.D. 1964-1967 59 Piqua, Piqua, Ohio 1962-1967 59 Hallam, Hallam, Neb. 1962-1964 75 Graph: "At What Cost" shows average cost, in cents per kilowatt hour, for fuel, operation and maintenance of nuclear power plants, 1982-1990. The cost at the most economical nuclear plant was 1.21 cents per kilowatt-hour in 1990. The highest cost was more than 5 cents. (Source: Utility Data Institute) (pg. D25)

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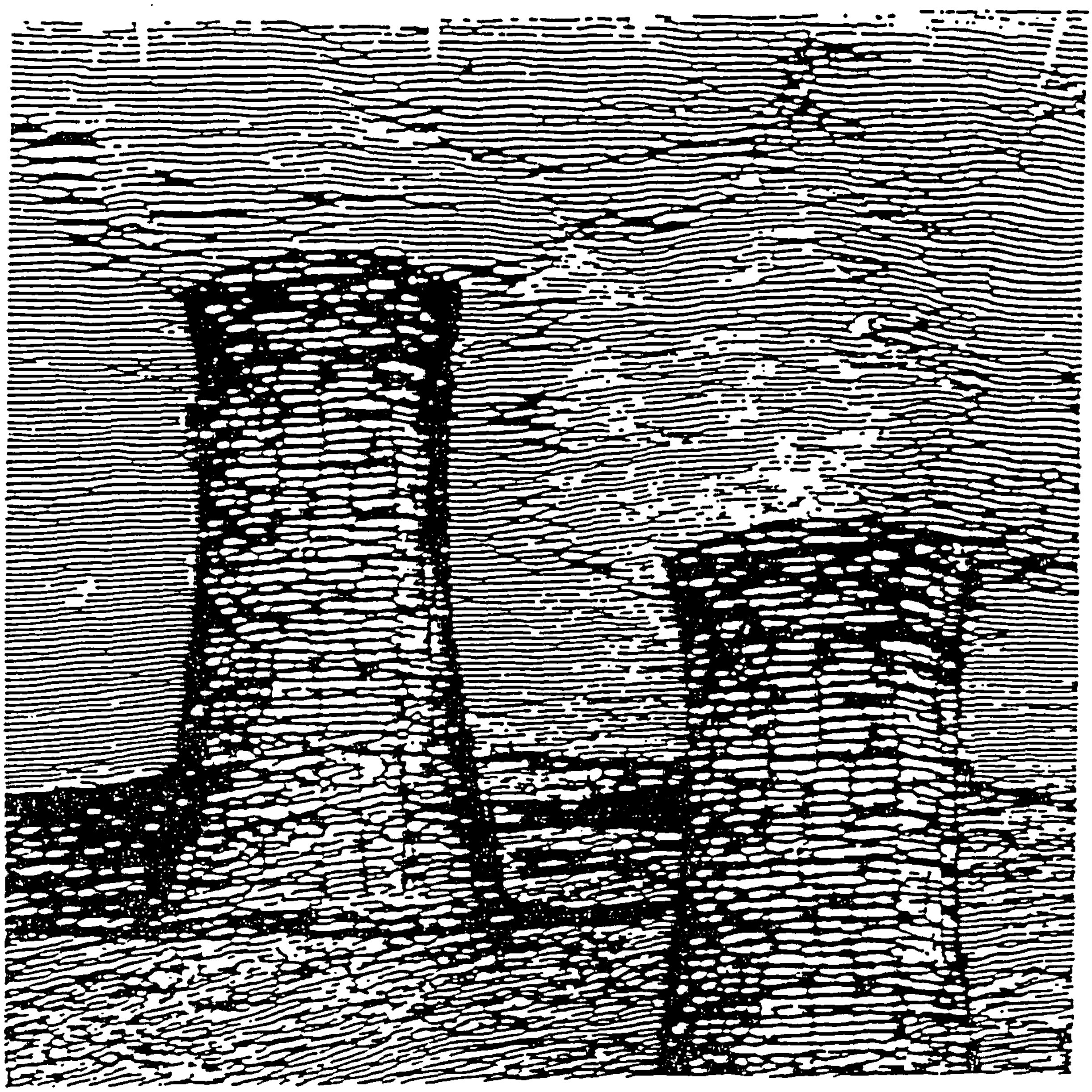
EXHIBIT 1 D

The Aging of Nuclear Power Plants: A Citizen's Guide to Causes and Effects



Nuclear Information and Resource Service

The Aging of Nuclear Power Plants: A Citizen's Guide to Causes and Effects

by James Riccio

and

Stephanie Murphy

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Nuclear Information and Resource Service

1424 16th Street NW, Suite 601, Washington, DC 20036

(202) 328-0002

IV. Embrittlement of Reactor Pressure Vessels and Reactor Pressure Vessel Supports in Pressurized Water Reactors

Irradiation embrittlement of the reactor pressure vessels (RPVs) may be the single most important factor in determining the operating life of a PWR. The design of pressure vessels is generally the same for all PWRs. Combustion Engineering (CE) and Babcock and Wilcox (B&W) manufacture their own vessels while Westinghouse either purchases its vessels from CE, B&W, Chicago Bridge and Iron or Rotterdam Dockyard Company. Regardless of the manufacturer, PWR vessels are generally constructed from eight inch thick steel plates, formed and welded to create the vessel structure.

The major age-related mechanism associated with this component is embrittlement. Embrittlement is the loss of ductility, i.e., the ability of the pressure vessel metals to withstand stress without cracking. It is caused by neutron bombardment of the vessel metals and is contingent upon the amount of copper and nickel in the metal and the extent of neutron exposure or fluence. As the metal in the reactor pressure vessel is bombarded with radiation, high-energy atomic particles pass through the steel wall. In doing so, these atoms collide with atoms in the metal and knock them out of position. Over time this results in a loss of ductility.

In an unirradiated vessel the metal loses its ductility at about 40 degrees Fahrenheit. As the vessel becomes embrittled, the temperature at which it loses its ductility rises. This change in the mechanical properties of the metal from ductile to brittle is characterized as the "reference temperature for nil ductility transition" or RTndt. Thus as the reactor ages and the pressure vessel is exposed to more radiation, the RTndt can shift from its original 40 degrees F to as much as 280-290 degrees F or more in extreme cases.[48]

Embrittlement is of even greater concern to those plants constructed prior to 1972. According to thermal shock experts from the Electric Power Research Institute (EPRI), records show that there is copper in the walls of older vessels. Theodore Marston, who works on thermal shock for EPRI, stated that,"(w)e used a lot of auto stock (for the vessel metal), when you melt it you can't get all the wire out." The use of copper was also extensive in the welds of the vessel walls in older reactors. Copper coated wire was routinely used to weld together the large plates which make up the RPV. The NRC's director of safety technology stated that "the copper was used to prevent rust, someone probably got a $10 prize for the suggestion."[49]

The significance of reactor pressure vessel embrittlement and the

concomitant shift in RTndt is the increased susceptibility to pressurized thermal shock (PTS). Pressurized thermal shock occurs when the reactor pressure vessel is severely overcooled. RPV technical specifications generally limit cool down to a rate of 100 degrees F. per hour. However, during an overcooling event the vessel may experience a drop in temperature of several hundred degrees per hour. This extreme drop in temperature of the vessel creates thermal stresses through the RPV wall. As the RPV is overcooled, there is a drop in the pressure of the primary coolant loop. This rapid decrease in the pressure of the primary coolant causes the high pressure injection pumps in the emergency core cooling system to automatically inject coolant into the primary loop. As the injection of coolant repressurizes the RPV, the vessel is subjected to pressure stresses. The stresses placed on the reactor pressure vessel by overcooling and repressurization cause pressurized thermal shock.[50]

Pressurized Thermal Shock (PTS) can be initiated by a host of mishaps including: instrumentation and control system malfunctions; small-break loss-of-coolant accidents; main steam line breaks; feed water pipe breaks; and steam generator tube ruptures. Any of these incidents can initiate a PTS event, but as long as the fracture resistance of the reactor pressure vessel remains high, i.e. the RTndt remains low, such transients are not likely to cause the RPV to fail. After the fracture resistance of the RPV is reduced through neutron bombardment, however, severe overcooling accompanied by repressurization could cause flaws in the inner surface of the RPV to propagate into a crack which breaches the vessel wall.[51]

For failure of the reactor pressure vessel to occur several factors must be present: (1) the vessel must have a flaw of sufficient size to propagate; (2) the vessel material must be susceptible to irradiation embrittlement due to copper and nickel content; (3) the vessel must be sufficiently irradiated to cause a decrease in ductility, represented by an increase in the RTndt value; (4) an event must initiate a severe overcooling transient with repressurization; and (5) the resulting crack must be of such a size and location that the RPV's ability to maintain core cooling is affected. This type of failure is beyond the design basis of PWRs: the safety systems, including the emergency core cooling system and the containment, are not designed to withstand cracks in the pressure vessel. Without the reactor pressure vessel surrounding the radioactive fuel, it would be impossible to sufficiently cool the reactor core and a meltdown would ensue.[52]

Pressurized thermal shock is a safety issue for every pressurized water reactor. PTS is of lesser concern for boiling water reactors because radiation embrittlement is not as severe a problem with BWR vessels. This is due to the greater amount of water between the reactor core and the vessel walls in BWRs. The

additional water absorbs a greater amount of neutrons so that fewer bombard the walls of the RPV. The walls of a BWR vessel are also thinner than that of a PWR. Therefore, there is less of a temperature differential between the inner and outer walls of the vessel during a cooldown and thus less stress.[53]

While every PWR vessel is susceptible to pressurized thermal shock, those designed by Babcock & Wilcox (B&W) are inherently more susceptible to accidents that can initiate PTS. This is primarily due to the unique design of the B&W steam generators. B&W reactors use once through steam generators, or OTSGs (see Appendix D). OTSGs differ from other PWR steam generators in that the generator tubes are only partially covered with water and contain a smaller volume. This makes the B&W reactor much more sensitive to changes in feed water flow-changes in the flow can cause large rapid changes in the temperature of the reactor. As a consequence, incidents which interrupt feed water flow present more severe challenges to the safety systems than would be experienced in other PWRs. The result is an increased incidence of overcooling events in B&W reactors and an increased probability of pressurized thermal shock.[54]

On December 26, 1985, a severe overcooling event occurred at a B&W facility near Sacramento, California. A loss of power to the "non-safety" integrated control system at the Rancho Seco facility caused a reduction in the main feed water flow to the steam generators. Coolant level in the steam generators decreased, reactor temperature and pressure increased and the reactor scrammed. Feed water valves controlled by the integrated control system could not be operated and remained open. A rapid and severe overcooling event ensued and was exacerbated by the start up of the auxiliary feed water system which sprayed even colder water directly onto the steam generator tubes. The reactor temperature dropped 180 degrees F in 24 minutes, easily violating the technical specification limits of 100 degrees/hour. Additionally, the recommended pressure/temperature limits for pressurized thermal shock were exceeded, although the RTndt limit was not.[55]

If the overcooling event had been more severe or the reactor pressure vessel more embrittled, the RTndt limit may have been reached and the vessel could have ruptured precipitating a meltdown. Equally as disturbing as the accident itself is the fact that the failures and consequences of the event were essentially the same as those previously experienced at Rancho Seco and other plants designed by B&W. In fact, many of the safety problems experienced during the transient were identical to those supposedly resolved by the "short-term" modifications imposed on B&W plants by the NRC in the wake of the Three Mile Island accident.[56]

In May 1979, after the TMI accident, the NRC shut down every B&W

facility, including Rancho Seco. The Commission ordered that procedures and training be implemented to assure that steam generator levels could be maintained if the integrated control system failed. Approximately a month later, the NRC staff concluded that "the licensee has developed adequate procedures and operator training to control AFW (auxiliary feedwater) flow to the steam generators to specific values independent of the ICS, should a failure of the ICS occur, and therefore, is in compliance with this part of the order."[57] However, on December 26, 1985, the staff's conclusions were proven incorrect when operators at Rancho Seco were unable to control the feedwater flow to the steam generators. The NRC's reaction was to conduct a year-long review of problems that were supposedly resolved six and a half years earlier.

The Nuclear Regulatory Commission has vacillated on the issue of pressurized thermal shock for over ten years now. As early as 1977, test samples placed in B&W reactors were indicating that embrittlement was progressing at a faster rate than had been expected. RTndt limits had been originally set at 200 degrees Fahrenheit. However, as these limits were reached in the early to mid 1980s, the NRC began developing new limits within the framework of the PTS rule.

In a briefing to its Advisory Committee on Reactor Safeguards in 1982, the NRC staff considered RTndt limits of 230 and 250 degrees F for longitudinal and circumferential welds respectively. However, by 1985, the NRC sought to amend its regulations on pressurized thermal shock. The proposed amendments would establish an RTndt below which the risk from a PTS event is considered acceptable. These new reference temperatures established limits of 270 degrees F. for plate materials and axial welds and 300 degrees F. for circumferential welds.[58]

The Commission attempted to gloss over the fact that an increase in the RTndt translated into a decreased margin of safety. The NRC press release said the rule constituted "further protection from pressurized thermal shock." At least one expert was not buying the NRC's line. Demetrios Basdekas, an NRC safety engineer and long time critic of the Commission's handling of the PTS issue, opposed the new rule on the grounds that the reference temperatures were unrealistically high.

Dissatisfied with the NRC's handling of the PTS issue, Basdekas made his opinion known in a letter to the New York Times. The letter stated that while,"(t)he Nuclear Regulatory Commission is charged with ensuring that nuclear plants are operated 'with adequate protection' of the public health and safety. . . bureaucratic foot dragging and preoccupation with public relations and financial problems of the industry are contributing to a shortsighted view - that technical problems can wait or do not exist."[59]

Basdekas contended that the new PTS rule was flawed in that it failed to recognize control system failures as a possible initiator of accidents that could challenge the pressure vessel. The NRC was not only failing to acknowledge Basdekas' contentions but plant operating experience as well. On March 20, 1978, the B&W designed Rancho Seco nuclear power plant experienced a PTS event precipitated by a control system failure. While replacing a light bulb in the integrated control system, an operator dropped the bulb into the control panel shorting out the control room instrumentation which eventually led to an overcooling of the reactor accompanied by repressurization of the vessel. The event is believed to represent the most severe and prolonged overcooling event to date with a change in temperature of 300 degrees F. per hour.[60] Basdekas was able to convince the NRC that control system failures were an unresolved safety issue, but the Commission continued to ignore these failures in their calculations on pressurized thermal shock.

In response to the NRC's ambivalence, Basdekas wrote the Commissioners suggesting an independent panel review the PTS issue. The nuclear safety engineer stated that,

> . . .our understanding and treatment of both the systems/process and materials/mechanics aspects of this issue remain wanting. I also believe that the agency and the public would benefit from the opportunity of an independent panel of experts to contribute to your decision making. . . . I might not have accomplished a great deal beyond receiving punishment and intimidation, but I am satisfied that I have stayed away from what appears to be increasingly in vogue within the agency to literally give the store away.[61]

Basdekas further explained the prevailing attitude within the NRC when asked by the Chairman of the House Subcommittee on Energy and the Environment, Rep. Morris K. Udall (D-Ariz.), to comment on NRC responses to the Committee on the topic of pressurized thermal shock. Basdekas stated that:

> A satisfactory resolution, however, cannot be achieved under currently prevailing attitudes within the NRC. On one hand the NRC left it up to the utilities operating the plants chosen for evaluation to provide design and operational information on a voluntary basis, and on a schedule of their convenience, while internally establishing an arbitrary schedule for producing a "resolution" document and withdrawing previously allocated resources while engaging in a variety of prohibited personnel actions and abuse of authority to intimidate and impede if not silence those voicing concern or disagreement.[62]

The NRC adopted the PTS rule in July 1985. In less than six months from the date of its adoption, control system failure had precipitated a severe overcooling event at the Rancho Seco facility (discussed above). Yet the NRC still failed to acknowledge control system failures in their analysis of embrittlement and pressurized thermal shock.

The NRC has continued its research on the PTS issue, focusing on methods to calculate and mitigate embrittlement of reactor vessels. To cope with the most severely embrittled reactors, the NRC has allowed some plants to redesign the configuration of the fuel rods so that fewer neutrons bombard the pressure vessel wall. The NRC has also released for comment a second revision of a regulatory guide (1.99) which specifies how utilities are to calculate the extent of embrittlement and the limits for operating with embrittled pressure vessels. The revision is an improvement in that it takes into consideration the copper and nickel content of the RPV materials. However, a major source of uncertainty still exists due to the limited accuracy and the variable range of the data base (a comparison of embrittlement limits under each revision of regulatory guide 1.99 is provided for each plant in appendix E). [63]

The NRC has attempted to put the PTS issue behind it, but the problem of embrittlement has been recurring like a bad dream. New questions involve the reactor pressure vessel supports. These hold the pressure vessel in place and, depending upon the design, can be exposed to substantial amounts of radiation. There are five major types of RPV supports, four of which are used in PWRs. The major factor in determining embrittlement of the supports is their exposure to reactor core beltline neutron flux. Two types of supports are directly exposed to irradiation from this area of the reactor, the neutron shield tank supports and the column supports. These two types of supports are used in 90% of the operating PWRs in the United States.[64]

The danger of embrittlement of the structural steel supports is the possibility that the neutron bombardment has so irradiated the metal that it cracks under the stress of the combined loads it was designed to bear. In the NRC jargon this is known as catastrophic brittle failure. This type of accident is beyond the design basis for safety systems and could result in total loss of reactor cooling capability. For catastrophic brittle failure to occur three conditions must be present: (1) there must be a flaw of critical size; (2) there must be a sufficient load on the support to create critical stress at the crack tip of the flaw; and (3) the temperature must be low enough to promote a cleavage fracture at the crack.[65]

It appears that the NRC is once again attempting to finesse the issue of embrittlement. In May of 1975 it was discovered that the

asymmetric loads placed on reactor pressure vessel supports because of postulated loss-of-coolant-accidents were not taken into consideration in the design of the supports for the reactors at North Anna units 1 and 2. This underestimation of the potential burden on the RPV supports, coined the "North Anna syndrome," prompted the NRC to require all PWRs to reevaluate the loads placed on the structures. It was discovered that the additional load resulting from a double ended rupture of the reactor coolant piping, also known as a guillotine break, was equal to the combined loads the structures were thought to support.[66]

The Commission responded to this issue in April of 1986 by exempting guillotine breaks from consideration. The NRC has stated that:

> . . .the dynamic effects associated with postulated pipe ruptures of primary coolant loop piping in pressurized water reactors may be excluded from design basis when analyses demonstrate the probability of rupturing such piping is extremely low under design basis conditions.[67]

The NRC bases this exemption on the "leak-before-break" theory. In essence the NRC is saying that the additional load placed on the RPV support in the event of a guillotine break need not be taken into consideration because the pipes will leak before they break. However, as previously noted, leak-before-break is neither an "established law", nor should it be the, "sole basis for continued safe operation."

Another factor contributing to the issue of embrittlement of RPV supports is the accelerated shift in RTndt of the support materials. Data from the test reactor at the Department of Energy's Oak Ridge National Laboratory (ORNL) has shown a greater than expected rate of embrittlement for steel that has been exposed to low temperature irradiation. A letter from the Advisory Committee on Reactor Safeguards (ACRS) to Victor Stello, NRC Executive Director for Operations, stated that the RTndt of steel, "irradiated slowly at 120 degrees can rise much more rapidly with exposure to fast neutrons than would be expected from the available experimental work obtained in test reactors."[68]

The ACRS requested that Stello look into the implications of the ORNL findings on embrittlement and the impact on the NRC's plans to extend reactor life past the 40 year license. Stello's response stated that,"(t)he ORNL summary coincides with our evaluation that the neutron shield tanks and support structures do not appear to pose any safety problems." However, close examination of the report reveals that the ORNL did not conclusively state that embrittlement was not a problem. In fact

the report found that, "plant specific data are required for an accurate evaluation of the potential for LWR vessel support failure."[69]

The ACRS was understandably "concerned and perplexed" by Mr. Stello's response. Interpretation of the data revealed that structural steel supports are experiencing 2 to 3 times the embrittlement as might have been predicted. However, Mr. Stello failed to draw any inferences from this information. The ACRS stated that they could, "see no reason to be sanguine about the safety of operating nuclear power plants with the largest, heaviest component in the primary system supported on a structure, parts of which are fully brittle. This is unsafe by any type of analysis."[70]

The NRC's final word on embrittlement of RPV supports is still out as the staff seeks further documentation. In the meantime, support reliability has been judged to be adequate. It escapes comprehension how the supports could be found adequate without inspecting them or determining the extent of actual embrittlement.

EXHIBIT 1 E

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards Joint Subcommittees: Materials and Metallurgy Thermal Hydraulic Phenomena Reliability and Probabilistic Risk Assessment
Docket Number:	(not applicable)
Location:	Rockville, Maryland
Date:	Wednesday, December 1, 2004
Work Order No.:	NRC-114

Pages 1-137

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rationale.

MR. ERICKSONKIRK: Yes. What we do find on the graph on the lower right-hand side is that the flaws that are driving the through-wall cracking frequency fully 90 percent of them are fairly small flaws and that's the observation.

DR. WALLIS: Because there aren't very many big ones? Is that what it is? It's more probable that you would have a small flaw under the surface?

MR. ERICKSONKIRK: Absolutely. There's a very low probability of having big flaws and even if you increase the big flaw probability by credible, or even incredible factors, it wouldn't matter much. I apologize for that. You are absolutely correct. The first rational was erroneous.

DR. WALLIS: This flaw distribution is based on rather skimpy evidence. This is one of the areas where -- I mean, heat transfer Dittus-Boelter if you believe that. It's based on data points. But the floor distribution in these walls is based on a few examinations. Isn't it?

MR. ERICKSONKIRK: A few examinations but infinitely more than we had the first time.

DR. WALLIS: It's much better than you had

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the first time.

MR. ERICKSONKIRK: Much better than we had the first time. I think as a laboratory geek at heart I have to admit I would really like to have more data on this and I don't think there's anybody in the technical community that would disagree with this.

But I think it's also important to recognize that the flaw distribution doesn't rest on experimental evidence alone. Certainly we started with -- excuse me. We start with experimental evidence both from destructive and nondestructive evaluations but that's then also bolstered by --

DR. WALLIS: But those were of individual reactor vessels.

MR. ERICKSONKIRK: That's right.

DR. WALLIS: But there are a hundred reactor vessels. I don't know how convincing it is that the flaw distribution that you measured in a couple of vessels which were taken apart is typical of all other vessels.

MR. ERICKSONKIRK: No. I think it would be unfair to say that a single experimental distribution derived from two vessels could be just looked at and thought to be representative of the other vessels.

EXHIBIT 1 F

PALISADES COULD REACH ITS PTS SCREENING LIMIT EARLIER THAN EXPECTED

Consumers Powers' Palisades may reach its PTS screening limit, a key indicator of reactor vessel brittleness, next year—eight years earlier than NRC staff reported to the commission as recently as late October—according to testimony at a December 8 commission meeting.

Consumers Power advised NRC on November 18 that new information on reactor vessel integrity showed that the critical PTS limit likely will be reached by 1999, five years earlier than an NRC staff analysis in late October showed. Jack Strosnider of the Office of Nuclear Reactor Regulation's (NRR) Division of Engineering told Commissioners Kenneth Rogers and Gail de Planque that information developed even more recently by testing of material from the plant's decommissioned steam generators may have pushed up the date—perhaps to as early as next year. NRC staff said they hope to produce a new evaluation of Palisades data by the end of January.

According to NRC regulations, a plant that has reached its PTS (pressurized thermal shock) screening limits may not operate unless the licensee presents additional information to justify the safety of the decision.

A Consumers Power executive who observed the meeting expressed disappointment that the new information showed the Palisades reactor vessel might reach the limits earlier than previously thought. The purpose of the testing was to justify operation to the end of the plant's original operating license in 2007.

The inability of Yankee Atomic Electric Co. to provide sufficient information about the integrity of Yankee's reactor vessel, together with economic issues, prompted the Yankee to shut that unit permanently in 1992.

As recently as October 28, when NRC staff issued Secy 94-267, "Status of Reactor Pressure Vessel Issues," the agency projected that Palisades would reach its PTS screening criteria in 2004. On November 18, Consumers Power submitted a revised evaluation of the PTS issue that indicated the vessel would reach the critical level in 1999.

Analysis of the critical beltline welds, along the axis of the reactor vessel, depends a lot on exactly what proportion of copper is present in the weld wires used to join the metal plates. Analysis of the metallurgy of the welds continues apace at Palisades, a nuclear engineer for the company told Inside N.R.C.

In a separate development at the meeting, NRC staff told the commissioners that they would, if necessary, compel ABB Combustion Engineering to divulge data on reactor vessel weld integrity that the vendor seeks to keep confidential.

The clash over the data from the Combustion Engineering Reactor Vessel Owners Group—data owned by ABB C-E—attracted attention from Commissioner Rogers, who told staff, including Office of Nuclear Reactor Regulation (NRR) chief William Russell, that "I would like to be kept informed of the discussions with the industry group (led by ABB C-E).

"I think the commissioners would be interested (in updates on the matter)," Rogers continued. "What are the proprietary aspects here that they are concerned about? I would hope that we could get over that hurdle."

Russell told Rogers that the basis for ABB C-E's request that the data be kept confidential is that it contains information on how C-E reactor vessel welds were carried out—"using the methods of twenty to thirty years ago that are no longer used." Russell said he was hopeful that the agency and the company would come to a voluntary agreement on the issue.

At the end of the meeting, Rogers repeated his call for the data to be made public. "I hope we'll be successful, ultimately, in filling out that data set."

NRC wants to include the data in a database called the Reactor Vessel Integrity Data Base, or RVID.

RVID summarizes the properties of reactor pressure vessel materials for all plants; it is based on docketed information and is scheduled for public availability in the first quarter of 1995.

David Jaffe - **Palisades** Dhone call Paae 1 H

From: Stephanie Coffin
To: Hoffman, Stephen
Date: 11/24/04 3:05PM
Subject: **Palisades** phone call

We had a phonecall with them Monday.

They no longer plan on submitting an exemption to apply "Master Curve" at their facility. Instead, they will be managing it in accordance with the May 27, 2004 guidance from Reyes to the Commissioners. They are following Point Beach and Beaver Valley closely.

I gave them feedback especially about the flux reduction requirements of the current rule and suggested they review the Point Beach submittal and our associated SER with Open Items, and to check for applicability to their plant.

FYI for Matt and Barry and Neil:

If they see that the new **PTS** rule will not be published in time for them (they currently exceed the screening criteria in 2014 - I don't know if we agree with that), they will submit the Master Curve exemption in 2007.

Stephanie

CC: Duvigneaud, Dylanne; Elliot, Barry; Mitchell, Matthew; Ray, Nihar; Stang, John

Outlook On Life Extension

EXHIBIT 1 H

SPECIAL REPORT TO THE READERS OF NUCLEONICS WEEK, INSIDE N.R.C., AND NUCLEARFUEL

As the nuclear industry tries to hang on in an increasingly competitive marketplace, considerations of extending plant lifetimes beyond their allotted 40 years sometimes seem an academic exercise. Yet whether the U.S. will have nuclear power as a future energy option—regardless of the cost or availability of other options—depends in large degree on license renewal and life cycle management decisions being made today.

The licenses of 49 of the 110 nuclear units in the U.S. are set to expire in the coming two decades, by 2014 (though some could be further delayed by recapturing their construction period and adding it to the license period). Will utilities refurbish and recertify them as safe to produce power beyond their current 40-year licenses? Or will they be shuttered one by one as their licenses run out, with some forced into premature shutdown?

NRC Chairman Ivan Selin counts license renewal as the *"number one topic" before the commission.* He is optimistic that at least some utilities will apply to extend the operating lives of their plants. "Interest is higher now than it has been for two years. I'm certain that many reactors will come in for license renewal," he asserted.

Selin says the NRC is on the verge of coming out with a rule that will make the process of applying for plant life extension simple, cheap, and predictable: "I feel good about the license renewal process we're developing—though my satisfaction is tainted by the fact that we should have got things right the first time."

Among government and industry experts, the NRC chief's may be a lone voice of optimism. A recent study by the Edison Electric Institute concluded that—far from life extension being the question—cost-related, premature shutdowns are likely to be an issue for utilities as early as this year (Nucleonics Week, 17 March, 1).

Even traditional boosters of nuclear energy express serious doubts about how many—if any—plants will see life beyond their 40th birthdays. "I'm relatively sure some plants will go in for license renewal, although it all depends on load growth at the time they go in," said Scott Peters, a spokesman for the Nuclear Energy Institute (NEI). The organization—born March 16 when four U.S. nuclear trade groups consolidated—is emblematic of the squeeze nuclear utilities face: paring down and cutting costs as they navigate the uncertainties economic competitiveness has thrust on the industry. Ultimately, the decision to seek license renewal "will be an economic decision utilities will make, taking into account many factors," said Peters.

"There may be people who, out of sheer stubbornness, continue to pursue license renewal—and NRC may let them—but whether it will be viable for these plants to continue after 30 years is questionable," said Jim Riccio, an attorney with Washington, D.C.-based Public Citizen, a Ralph Nader lobby. Riccio doubts that any of the U.S. nuclear units will even get to the ends of their licenses. "It won't be the antinuclear forces that force them to close, but the people who do their ledger books," he said, adding, "Economic and safety considerations will shut these plants down."

Only 46% of electric utility executives expect to see operating licenses extended for most plants, according to a survey released by the Washington International Energy Group in January. Odds remain in favor of most units continuing to operate through their first 40 years, the executives said. But only 37% of respondents believed there would be a resurgence of nuclear power in the U.S. "Privately, CEOs talk about someday turning over title of [nuclear] plants—even the best run ones—to the government," said the report, "1994 Electric Utility Outlook."

Those utilities that set their sights on plant life extension will have to brave uncharted territory. A few years ago, the path appeared relatively straightforward. The industry would develop pilot license renewal submittals for a lead BWR and a lead PWR, which would pave the way for other utilities to follow with renewal applications. But the process has taken unexpected turns. Yankee Rowe's experience as a lead PWR led ultimately to its shutdown, and the other utilities have hung back.

Competition has changed the landscape of the electric utility industry entirely in two years. Economic decisions to seek license renewal will be made on a

plant-by-plant basis, industry experts say. Few generalizations can be drawn to predict likely candidates: some utilities view their nuclear units as albatrosses, others embrace them as assets. The price of other energy sources in a given region, the philosophy of state regulators, public perceptions about nuclear power in a county or state, the number of units a utility is operating, and the age, condition, and operating history of each plant—not to mention the cost and degree of hassle involved in meeting NRC's forthcoming rule—will all influence the life extension decision.

Key players and issues to watch include:

- Virginia Power, the only utility to announce plans to apply for life extension. It surprised industry peers and NRC by unveiling plans to initially seek five-year license renewals—instead of 20 years—for Surry and North Anna.
- The Babcock & Wilcox Owners Group, which plans to submit a license renewal application on one of its PWRs by 1997. Duke Power's Oconee-1 is one of the top candidates for their submission.
- Baltimore Gas & Electric Co., which has spent $15-million on a combined life cycle management and license renewal program. The company may decide to apply to extend the Calvert Cliffs licenses for 20 years.
- NRC's rule rewrite, ordered by the commissioners in February. Will the new rule provide the cheap, simple, predictable application basis Selin has promised? And how will utilities implement it?
- Other nations' experience. Electricite de France (EDF) has taken the lead in grappling with many of the aging issues that U.S. utilities must evaluate in the technical assessments of their plants. EDF has developed a list of 18 essential components. In Sweden, the clash between politics and performance once again is coming to a head, where nuclear power opponents want to see lifetimes limited on plants considered by U.S. analysts to be among the best performers in the world.

VIRGINIA POWER TAKES THE BULL BY THE HORNS

Virginia Power Co. could be the first utility to test NRC's promise of a simpler and cheaper approach to license renewal. The company unveiled plans in February to file an application early in 1995 to renew the operating licenses for the Surry and North Anna nuclear stations for five years (Inside N.R.C., 21 Feb., 1).

Virginia Power's nuclear units are "very competitive," said William Stewart, vice president-nuclear. "Our production costs are excellent. Our units were built in the 1970s, so the [capital] cost was low." Stewart characterizes the decision as one that could make the utility even more competitive. "We're not in a scramble here," he said. "We're just looking ahead.

Stewart said the five-year life extension initiative has received the support of Wall Street utility analysts, whom he briefed on the plan in February. "Their response was favorable. The feeling was that a five-year renewal is feasible."

Reducing Busbar Costs

Company officials concede the decision to pursue five-year license renewals is "an economic rather than an operational" one, primarily driven by the current climate of economic competitiveness. Martin Bowling, manager of the utility's nuclear licensing programs, said that renewing the licenses at its two dual-unit nuclear stations by five years would lower "busbar" costs—the amount it costs to produce electricity at the point it leaves the plant, including capital costs, taxes, fuel, and operations and maintenance costs.

In 1992, Virginia Power's busbar costs were 2.84 cents per kilowatt-hour (KWH) at Surry and 3.42 cents/KWH at North Anna. Overall busbar costs for nuclear generation are 3.1 cents/KWH, Stewart said.

Costs would be lowered by changing depreciation rates and reducing near-term decommissioning trust collections. Lower costs would help hold down future rate increases and keep the utility competitive, Bowling explained. In their economic analysis, utility officials calculated that, as a near-term effect, "depreciation expense is reduced each year, $21-million the first year and up to $49-million by 2011."

Virginia Power has replaced steam generators at three of the four units and plans to replace the steam generators at North Anna-2 in 1996 at a 1996 dollar equivalent to the $120-million spent last year to replace the steam generators in unit 1.

Stewart said the company has spent more than $5-million to date on license renewal work, most of it on technical analyses under the old NRC rule. Surry-1 began operating in December 1972 and its license expires in May 2012; Surry-2 started up in May 1973 and its license expires in January 2013. North Anna-1 started operating in June 1978 and its license expires in April 2018; unit 2 began operations in December 1980 and its license expires in August 2020.

Company officials recently proposed a streamlined integrated plant assessment (IPA) process to NRC staffers. They say the technical analyses under their IPA would be done no differently whether they were applying for a five-year extension or a 20-year one. Utility plans call for submitting a license renewal application during the first quarter of 1995, but that could change if NRC's revised license renewal rule is not out yet, Stewart noted.

Assuming Virginia Power's license renewal application proceeds on track, the utility should provide a case study on how NRC will have utilities show compliance

reliance on the maintenance rule, completing the reviews needed for a license renewal application is not going to be a piece of cake. There will be considerable documentation required.

Thermal fatigue monitoring of the primary system is an example of how life extension work provides short-term and long-term benefits, Doroshuk said. Engineers analyzed all the Class 1 piping, for instance, and created a computerized data bank that not only is used to track aging mechanisms like fatigue, but can be used in operations. "We've been able to provide an analysis of the plant and respond to transients that [operators] see during a startup and allow them to continue on without doing holds and analysis," an engineer on the team said.

A priority of the life cycle management program has been evaluating the reactor pressure vessel. Calvert Cliffs-1 is currently going to exceed NRC's pressurized thermal shock (PTS) screening criteria between 2004 and 2006. Unit 2 was built using lower-copper weld material, so it doesn't have the same problem.

"There are plant-specific differences that make us believe unit 1's embrittlement is much less than NRC's correlation would predict," said Marvin Bowman, an engineer on the life cycle management team. "These include fabrication process differences, heat treatment differences, materials differences, weld materials, weld fluxes that were used—they're all part of that embrittlement process."

BG&E officials submitted the technical evaluation to NRC last November and are expecting to receive NRC's review by this summer. "We think we have a very sound technical basis for continued operation of unit 1," Doroshuk said.

"We've done substantial flux reduction and we can do much more aggressive flux reduction if we have to, involving radical fuel management," Bowman added.

The conclusion of BG&E's plant assessment activities to date: "We've found that, materially, the plant should last for 60 years," Doroshuk said. "We haven't found the show-stopper—even in the reactor vessel."

REACTOR VESSEL INTEGRITY, COSTS, CRUCIAL TO LIFE EXTENSION

Coping with reactor vessel embrittlement is a priority for managing plant life extension, as several U.S. reactors likely will face the problem after about 40 years of operation.

PWR vessels are more susceptible than BWRs' because the PWR vessels are narrower and contain less coolant, so more neutrons reach the vessel walls. Most PWR vessels in the U.S. are made of steel known as A533-B alloy, an alloy of iron, carbon, and manganese with some nickel. In a number of older vessels, the welds joining the vessel's curved plates contain some copper. As neutrons from the core strike the steel, they change the crystalline structure of the A533-B alloy. The weld material is especially affected, with the neutrons disrupting the crystal lattice, creating clumps of copper atoms and vacancies in the matrix. This process makes the steel more brittle.

The brittleness is measured in two ways: the upper shelf energy loss and the reference temperature at nil ductility temperature (RT-NDT). The loss of ductility can leave the metal vulnerable to ductile fractures, a tearing that can take place in seconds or minutes, and to pressurized thermal shock (PTS). The latter could occur if, during an accident, coolant is suddenly restored to an overheated vessel, and it could cause an abrupt fracture of the metal. Reactor operators must show NRC that their vessels are proof against both types of failure.

Because ductility loss is cumulative, operators plot the time it will take their vessel to reach the RT-NDT and upper shelf criteria that NRC selected as a conservative danger signals of growing embrittlement, set in 10 CFR 50 and its Appendix G. Utilities have a number of ways to cope with embrittlement as operating lifetime progresses, including flux reduction through fuel management, such as installing neutron absorbers on the core periphery. Utilities can also analyze their vessels' metal to prove they won't approach the NRC criteria within their operating lives, and monitor the accuracy of those analyses by regularly removing and testing specimens of vessel material, which are kept inside the vessels in capsules.

As vessel lifetime is lengthened, a rate of embrittlement which was no concern for 40 years of operation may become a barrier to reaching 60 years, and utilities that did not previously have to be concerned about PTS are having to look again.

NRC has required utilities to submit information on the status of their vessels with regard to PTS and upper shelf energy. From this information, the agency has deduced a list of especially vulnerable reactors that has fluctuated from around one to six reactors, as utilities take measures to alter the rate of embrittlement. Calvert Cliffs, Duquesne Light Co.'s Beaver Valley, and Consumer Power's Palisades have been mentioned as particularly vulnerable. While opinions within NRC and the industry appear to vary, the utilities generally hold that further analysis of the metallurgy of their reactors will prove that the vessels will last to the ends of their license periods.

According to the latest NRC list, seven units' vessels likely will encounter PTS concerns before the ends of

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REACTOR VESSELS WITH PRESSURIZED THERMAL SHOCK (PTS) CONCERNS

Plant Name	Estimated Year PTS Screening Criteria will be Reached	License Expiration Date per NUREG 1350
Palisades	1997-2005	03/14/2007
Fort Calhoun	2013	06/07/2008
Calvert Cliffs-1	>2005	07/31/2014
Point Beach-2	>2013	03/08/2013
Point Beach-1	>2010	10/05/2010
Beaver Valley-1	2014	01/29/2016
PTS CONCERNS BEYOND CURRENT END-OF-LICENSE LIFE		
Zion-1	2011	12/26/2008
Oconee-2	2019	10/06/2013
Surry-1	>2012	05/25/2012
Salem-1	2020	09/25/2008
Zion-2	2023	12/26/2008
Ginna	2026	04/25/2006
Diablo Canyon-1	2034	04/23/2008
Cook-1	2037	03/25/2009
Farley-1	>2050	06/25/2007
St. Lucie-1	>2050	03/01/2016

Source: NRC

their licenses, and nine other units' vessels will fall in the category after their current licenses expire (see table).

Annealing: The Last Ditch

Utilities that look soon enough can alter the rate of embrittlement early in a vessel's life, but for older reactors the standard methods may not be able to change the rate enough. That can leave utilities facing an expensive problem for life extension. YAEC chose to close Yankee because the process of proving the vessel's stage of embrittlement was too expensive—and NRC's requirements for that proof too open-ended—to be justified economically for a 185-MW reactor.

One option is to replace the vessel, which has never been done and is estimated to cost as much as $100-million. Another choice being viewed with increasing favor by reactor owners is annealing the vessel, or heating it to the point where the crystalline structure of the steel is partly or fully restored to its original fracture resistance. Estimates for annealing range around $10-million.

Annealing is a routine process in metallurgy and has been extensively modeled, but it is complicated by vessel radioactivity. For U.S. vessels, it would involve heating the beltline weld, and in some cases the axial welds or some vessel plates, to about 850 degrees F for about a week. The longer the heat is applied, the more complete the restoration of the metal's crystalline structure. Theoretically, heating the vessel for as much as two weeks could restore the metal 100%.

Different annealing specialists offer different estimates of how long the repair done by an annealing job would last. Some estimate annealing could restore a vessel to service for five or six years, while others say field experience indicates 60 to 70 years. Researchers say more study is needed.

Alan Hiser, of the materials engineering branch of NRC's Office of Nuclear Regulatory Research (RES), said that the level of restoration of embrittlement due to annealing, and the rate of re-embrittlement, is dependent on a number of factors. If the material is a weld, rather than a plate, the annealing repair will be less effective and the re-embrittlement rate faster. The chemistry of the material is crucial, as well—steels or welds containing nickel or copper are more subject to both embrittlement and re-embrittlement.

Hiser emphasized that the difference between the reactor operating temperature—around 550 degrees F—and the temperature of annealing (850 degrees) has an important effect. The greater the difference between the two temperatures, the more successful the annealing will be and the longer its effects will last. NRC has funded research on annealing that was carried out by

DOE at Oak Ridge, and Hiser's views also take into account research performed by Westinghouse on behalf of EPRI, the U.S. Navy, and Russian annealing specialists.

Hiser said that most studies have considered annealing for 168 hours, or one week. On the basis of 168 hours and 850 degrees, Hiser said that an annealing job can restore as much as 100% of the damage due to neutron flux, depending on the factors mentioned. "From the data we've seen so far, it appears that (after annealing) you get the same embrittlement rate you got initially," he said.

The American Society of Testing & Materials' Standard E509-86 model posits that re-embrittlement occurs

YANKEE EFFORT FOUNDERED ON VESSEL

Zeus does not bring all men's plans to fulfillment.
—Homer

At the time, it seemed logical. Yankee Atomic Electric Co.'s (YAEC) Yankee was by far the oldest PWR operating in the U.S. Though small and unique in design, it had a good operating record and YAEC wanted to extend the operating license that was to expire in 2000.

When DOE and the Electric Power Research Institute (EPRI) were looking for candidates to test the license extension waters, the 185-MW Yankee got the nod for PWRs, despite EPRI's expressed reservations about using older, smaller plants as license extension guinea pigs and despite the fact that Virginia Power had already done extensive initial screening of possible license extension roadblocks for Surry-1, an 824-MW PWR of a more common design.

The issues were timing and money. Surry's license wouldn't expire until 2012. The Yankee plant was YAEC's only asset. Therefore, YAEC had no choice but to go forward with license extension, even if it wasn't chosen as a lead plant. "We were going to file a license extension application," said YAEC's Bill Szymczak, a member of the Yankee license extension team. "There was sentiment at the time that (the lead plant) should have been Surry, but we were going to be in the queue anyway."

"As a single-asset company, regardless of what anyone else was doing, we were going forward," Szymczak added. Having YAEC before the NRC for a license extension at the same time the pilot plants were going through would have been a "complicating factor," he said.

DOE and EPRI were, in fact, very concerned about maintaining a uniform front on license renewal before the NRC. When YAEC signed on as lead PWR, a provision in its contract with DOE and EPRI required that its licensing submittals be "sufficiently consistent" with those filed by Northern States Power Co. for the lead BWR, Monticello, that NRC wouldn't be able to "leverage one plant against the other."

If YAEC had filed a license extension application as a third party, it would not have been constrained to maintain that uniformity.

Ultimately, Yankee's unique design proved to be its undoing. On February 26, 1992—41 months after YAEC received DOE's proposed contract for the five-year lead plant effort—the YAEC board voted to permanently close the unit. Yankee had been "voluntarily" shut down the previous October after the NRC staff recommended that it be laid up until questions regarding the degree of embrittlement of its reactor pressure vessel (RPV) could be resolved. They never were.

Yankee's RPV design made it virtually impossible—or at least extremely expensive—to answer the questions NRC and intervenors had posed. The RPV's configuration severely limited inspection of the beltline weld region. Yankee's vessel consists of a rolled 0.109-inch-thick stainless steel sheet welded in several locations—but not completely bonded—to the shell. That incomplete bond made ultrasonic inspection of the welds very difficult, because the "gap" between the cladding and the vessel shell hindered the ultrasonic instruments' ability to accurately identify and size weld flaws.

Yankee's RPV also has a thermal shield just inside the vessel wall. The proximity of the shield to the vessel shell restricts the accessibility to the beltline region and restricts inspections of critical welds. To position instruments for inspection, the thermal shield would have to be destroyed and removed, unless new tools were manufactured to make the welds accessible. YAEC started investing in new inspection equipment, but that cost a lot and the economics of saving Yankee—which already wasn't selling electricity at a competitive price in recession-ravaged New England—just didn't add up.

YAEC even briefly considered replacing the RPV. Though DOE's Sandia National Laboratories concluded that the replacement was "technically achievable," the decision on replacement was ultimately an economic one. It was estimated that RPV replacement could be paid off over 20 years with a rate increase of 1/2 to 1 cent per KWH for the first year, which would drop to 0.1 cent the last year. Of course, that payback scenario assumed license extension.

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at the same rate as before annealing, but that embrittlement restarts from a point of greater ductility—a "lateral shift" in the embrittlement curve. Russian experience in annealing 13 reactor vessels has verified the lateral shift approach to analyzing re-embrittlement, Hiser said. Even if an annealing job does not repair all flux damage, he said. "You will always end up better (after an annealing), but there's no significant impact on the rate of embrittlement."

However, each reactor vessel responds to fluence according to the particular circumstances of its construction, including the type of material used in the steel and the welds. The rate of fluence accumulation before and after annealing will also have an effect, Hiser said.

"Assuming the same rate of fluence accumulation of the vessel and 100% recovery (from embrittlement), the vessel should be good for the same period as before you annealed," Hiser said.

Hiser's views on annealing and re-embrittlement are based on more than a decade of research on the topic in the U.S., where sample coupons have been tested many times for embrittlement rates and response to annealing, and on full-vessel annealing in Russia.

Westinghouse has moved strongly into the nascent field. David Howell, manager of engineering services for the vendor, said, "The nuclear industry needs this program for life extension and to meet license requirements, plus we see annealing as a sales opportunity."

Westinghouse has raised more than $2-million of $4-million it is seeking from the industry to conduct an annealing demonstration at the never-completed Marble Hill reactor in Indiana. Westinghouse plans to use indirect gas-fired heating to raise the vessel's temperature. Extensive monitoring will be designed to answer critical engineering questions, such as how the process affects nozzles and pipes attached to the vessel.

Westinghouse faces competition from a team formed of Framatome subsidiary B&W Nuclear Technologies, MPR Associates Inc. of Alexandria, Va., and a consortium called Russian Annealing Moht. The moht, or consortium, is formed of Moscow's Kurchatov Institute; vendor Gidropress; Cnittmash; and other Russian organizations, which has annealed 13 vessels in Russia. MPR Associates principals Bill Schmidt and Noman Cole praise the Russian technology, which relies on electric resistance heating, as simple and reliable. Schmidt said that Electricite de France (EDF) specialists who met with his company said that they were concerned that the Westinghouse process could not be licensed in France because of the dangers of working with natural gas.

U.S. annealing specialists visited Novovoronezh in 1992 to witness the annealing of a VVER-440 vessel. The Russian approach is different, the specialists say, because Russian vessels are made of ring forgings, so the circumferential weld area to be annealed is only about three feet wide. U.S. reactors, which are longer and have axial welds, would have to be annealed in a band about 12 feet wide. Also, some U.S. reactors would require annealing on plate sections.

MPR's Cole said that the recovery in RT-NDT at Novovoronezh was greater than 80%. While Russian annealing specialists have achieved recovery of 100% in that index of PTS vulnerability from some annealing projects, they guarantee recovery of 80% from their annealing process.

Keith Wichman, annealing specialist with NRC's research arm, agreed Russian annealing projects have achieved recovery rates over 90%. That level of recovery is equivalent to full recovery for his purposes, Wichman said, adding that above 90% recovery, distinctions are meaningless.

No full-scale annealing project is now on the drawing boards at an operating U.S. reactor, despite the optimism of the annealing vendors. Utilities have reason to be cautious, since the first utility to anneal almost certainly will have to pay extra to defray the costs of licensing the process and the extra costs associated with the learning curve on the technology.

In any event, the first U.S. annealing likely will not take place before the turn of the century, since other methods of lowering embrittlement rates will work until then. Jack Hanson, an annealing specialist at Palisades, said his utility's decision on annealing would be governed by the economics of the process when license renewal is evaluated years from now. Hanson added that the most important factor in that calculation will be the price of natural gas—the strongest competitor to nuclear power plants.

NRC DRAFT ENVIRONMENTAL REVIEW RULE ANGERS STATES

In addition to the safety reviews, NRC will require an environmental review as part of the license renewal process. NRC's proposed amendments to tailor the existing environmental review rule (10 CFR Part 51) to the license renewal process have come under attack by states for what they view as an attempt by NRC to preempt their traditional review of need for generating capacity and alternate energy sources.

The amendments include a draft Generic Environmental Impact Statement (GEIS), published in September 1991, in which NRC decided to treat the issues of need for power and alternative energy sources in the same way they are handled in an operating license review. While the agency performs a detailed analysis

EXHIBIT 1 I

Consumers Power

POWERING MICHIGAN'S PROGRESS

General Offices: 1945 West Parnall Road, Jackson, MI 49201 • (517) 788-0550

May 17, 1990

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT -
COMPLIANCE WITH PRESSURIZED THERMAL SHOCK REGULATION 10CFR50.61 AND REGULATORY GUIDE 1.99 REVISION 2 (TAC NO. 59970)

Consumers Power Company (CPC) submittal on April 3, 1989 provided a revised report on reactor vessel fluence for Cycles 1 - 8. Attached is the vessel fluence reduction report describing the effect of incorporating low-leakage fuel management for the Cycle 9 core loading pattern. In this proposed Cycle 9 design, 16 thrice-burned fuel assemblies with zircaloy-clad hafnium absorber rods will be used at the selected core peripheral locations to protect the vessel axial welds from neutron fast flux E>1.0 MeV. Remaining core peripheral locations will be loaded with twice-burned fuel assemblies. All once-burned and fresh fuel assemblies will be inside the core away from the peripheral locations.

This report reflects results based upon the development of in-house methodology utilizing the DOT 4.3 discrete ordinates transport code and Reactor Engineering Analyses performed during the period of 1987-1990. It concludes that the PTS screening criteria will be exceeded at the axial welds in September, 2001, as opposed to the previously reported exceed date of March, 2002. The difference reflects an improvement in vessel flux reduction in Cycle 9 relative to Cycle 8 and slightly higher vessel flux levels calculated by the refined in-house transport methodology relative to the Westinghouse methodology previously utilized. Thus, the previously

derived conclusion that the flux reductions achieved in the Cycle 8 and 9 core loading patterns are, by themselves, insufficient to allow plant operation to the current expected end of life in 2011 remains valid. Further measures, eg, greater flux reduction, Regulatory Guide 1.154 analysis, vessel shielding etc, are necessary to allow plant operation to the nominal end of plant life and beyond.

Richard W Smedley
Staff Licensing Engineer

CC Administrator, Region III, USNRC
NRC Resident Inspector - Palisades

ANALYSIS OF THE REACTOR PRESSURE VESSEL FAST NEUTRON FLUENCE AND PRESSURIZED THERMAL SHOCK REFERENCE TEMPERATURES FOR THE PALISADES NUCLEAR PLANT

May 1990

Performed by the
Reactor Engineering Department
Palisades Nuclear Plant
Consumers Power Company

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1.0 INTRODUCTION

Consumers Power Company previously submitted to the NRC a report describing Cycle 8 fluence reduction measures for the Palisades Nuclear Plant reactor pressure vessel [1]. It was committed that an additional fluence report reflecting Cycle 9 fuel management and extrapolated to nominal plant end-of-life would be submitted to the NRC. The information contained herein is intended to address the fluence re-evaluation and reduction program as previously committed to and to describe the methodology utilized for determining vessel incident fast fluxes and fluence levels.

In order to accurately calculate pressure vessel fluence levels, in-house methodology was developed utilizing the DOT 4.3 discrete ordinates transport code as the base model. Training in the use of DOT 4.3 and associated cross section libraries and support codes was obtained from Combustion Engineering. The scope of the training included code usage and model development as well as results evaluations. In-house methodology of flux calculations was further refined via consultation with Westinghouse Electric Company, Radiation and Systems Analysis-Nuclear Technology Division. Westinghouse determined that Consumers Power's neutron transport methodology represented state-of-the-art practice consistent with Westinghouse methodology [2,8].

The modeling of the vessel and fluence analysis was performed using DOT 4.3 and the SAILOR cross-section library. Cycles 1 through 7 core loading patterns were typical of out-in fuel management in that the fresh fuel was placed on the core periphery. This approach results in the maximum overall core neutron leakage and flux to the reactor pressure vessel. The Cycle 8 core was loaded with thrice burned fuel assemblies with stainless steel shielding rods located near the axial weld locations. In the previously submitted report [1], flux reductions of a factor of two were achieved at the axial weld locations from the Cycle 8 loading pattern. The design goal for Cycle 9 was to meet or exceed the flux reductions achieved in Cycle 8. The proposed Cycle 9 loading pattern consists of thrice burned fuel assemblies with hafnium absorbers located at the same core peripheral locations that utilized stainless steel shielding rods in

Cycle 8. The remaining core peripheral locations will be loaded with twice burned fuel assemblies. All of the new fuel assemblies will be located within the core interior.

In this report, cycle-specific calculations have been performed for Cycles 1 through 9. Results presented address the accumulated vessel fluence through the end of Cycle 7 as well as the flux reductions obtained for the Cycle 8 (currently in operation) and Cycle 9 (under design) low leakage loading patterns. Vessel fluence limits based on the 10CFR50.61 PTS screening criteria and both the 10CFR 50.61 and Regulatory Guide 1.99, Revision 2, reference temperature correlations are calculated based on the vessel material chemistries. Vessel lifetimes are calculated relative to the fluence limits assuming the flux-reduction fuel management for Cycle 9 and beyond utilizing the Regulatory Guide 1.99, Revision 2 reference temperature correlations. In addition, details are provided about the in-house methodology and data [3] and the status of Consumers' in-house flux reduction and measurement program.

2.0 SUMMARY

Neutron transport calculations were performed using the DOT 4.3 computer code and SAILOR cross section library. The 2D R-θ neutron fluxes (E>1.0 MeV) were computed using DOT 4.3 with consideration of axial flux peaking. For each of Cycles 1 through 9, cycle specific DOT runs have been made. For Cycles 1 through 7 on-line core monitoring energy generation data and actual cycle operational history data were utilized for vessel flux and fluence calculations; calculations for Cycles 8 and 9 utilized predictive core simulator data. A comparison between calculated and measured fluxes at the W-290 wall capsule location, analyzed at the end of Cycle 5, was made. It was found that the calculated fluxes were about 4% higher than the measured values, thus assuring reasonable flux predictions for the models.

Flux levels for Cycle 8 were compared with that of Westinghouse methodology [1]. The in-house model indicates a positive bias in the flux calculations relative to the Westinghouse methodology and this bias varied with the azimuthal locations. Maximum variation was on the order of about 12% at 45° location [3].

Pressure vessel fluence limits based on the PTS screening criteria of 10CFR50.61 were calculated using the reference temperature (RT) correlations of both 10CFR50.61 and Regulatory Guide 1.99, Rev 2 using the vessel chemistries provided in Reference 4. The results are summarized in Table 2.1 and show the dramatic reduction in the vessel weld fluence limits with the use of the Regulatory Guide 1.99 RT correlation. With the pending issuance of a revised 10CFR50.61 incorporating the Regulatory Guide 1.99, Rev 2 RT correlation, the more restrictive Regulatory Guide fluence limits were utilized in this study.

Core loading patterns for Cycles 8 and 9 are designed to provided substantial flux reduction at the axial weld locations in comparison to previous cycles. The associated flux reductions for the primary vessel materials are shown in Table 2.2. Fast flux (E>1.0 MeV) reductions of more than 50% were obtained at the axial weld locations for Cycles 8 and 9 in comparison with Cycle 7. For Cycle 8, at the circumferential weld and base metal (peak) locations, about 20%

flux reduction was obtained. However, for Cycle 9 the flux reductions are on the order of 48% at these locations.

Vessel lifetimes based on when the PTS screening criteria are met were determined for fuel management schemes with flux reduction for Cycles 8, 9, and beyond. Operation beyond end of Cycle 8 (September 1990) was assumed to occur at 75% capacity. With no flux reduction utilized, the PTS screening criteria would be exceeded at the axial welds in 1995; utilizing Cycle 8 flux reductions, this would be extended to 2000. With flux reduction incorporated in Cycle 9 and beyond, the PTS limit would be exceeded at the axial welds again, but not until about September, 2001. These predicted dates are far short of the assumed nominal plant operating license expiration date of March, 2011.

While the flux reduction obtained in Cycles 8 and 9 substantially reduced the axial weld flux levels, the reduction is insufficient to remain within the PTS screening criteria through the minimum plant life (nominal end of operating license). Some additional flux reduction will be possible through more aggressive low-leakage fuel management in Cycle 10 and beyond. However, in order to allow plant operation at least until the nominal license expiration date, additional PTS-addressing measures will have to be implemented (eg, Regulatory Guide 1.154 analysis, vessel shielding, etc). Activities are currently underway with the Combustion Engineering Owners' Group in the areas of additional vessel surveillance data and model development for a Regulatory Guide 1.154 analysis; initial conceptual discussions are underway with other vendors for incorporation of weld specific vessel shielding in Cycle 10.

An ex-vessel dosimetry program was developed by Westinghouse and hardware installation was completed during the end of Cycle 7 refueling outage. This program would supplement the existing surveillance program. In addition to the ex-vessel program, Combustion Engineering will install an in-vessel dosimetry capsule at the W-290 capsule holder vacated following Cycle 5. These in-vessel and ex-vessel dosimetry programs will provide measured data for use in vessel wall and vessel support fluence evaluations.

Updates on vessel fluence levels and adjusted reference temperatures will be provided to the NRC as actual operational data including vessel dosimetry information is obtained. In addition, developments in fuel management, vessel materials information, vessel shielding and other PTS-related areas that substantially impact the vessel lifetime will be reported as required in 10CFR50.61.

3.0 METHODOLOGY

3.1 Overview

The pressure vessel fast neutron fluence levels (E>1.0 MeV) were calculated utilizing available historical and predictive fuel cycle information.

The primary analytical model was based on a two dimensional (R,θ) discrete ordinates code DOT 4.3 representation [5] of the Palisades reactor vessel configuration. The representation includes a model of the core/vessel geometry, the neutron source distribution, and nuclear interactions as represented by cross section data. Measurement data was available for comparison from an analysis of radiometric dosimeters irradiated in the W-290 vessel wall surveillance capsule [6], which was removed at the end of Cycle 5. The measured fast neutron flux as calculated from the measured activities using reactor power history, dosimetry cross sections and basic nuclear data was used to compare the DOT calculated neutron fluxes for Cycles 1 through 5. Individual DOT calculations for remaining Cycles 6 through 9 were also made. To-date fluence levels were calculated and end-of-life fluence levels were extrapolated based upon anticipated capacity factors for the remaining life of the Palisades Plant.

3.2. Fuel Management

Palisades followed a standard out-in fueling scheme through Cycle 7 (Figure 3.1). In this scheme, only fresh fuel was placed around the core periphery. This approach results in the maximum overall core neutron leakage and fast flux to the reactor vessel, but minimizes power peaking and generally provides the greatest thermal margin.

Utilization of the Regulatory Guide 1.99, Rev 2, reference temperature correlations for comparison to the 10CFR50.61 PTS screening criteria determined that the axial welds would be responsible for limiting the life of the Palisades reactor vessel. It was decided to alter the fuel management strategy to

distribute the power away from these critical weld locations for Cycle 8 operation. A low leakage loading pattern was adopted to improve the neutron economy and to reduce the fluence levels at the axial welds.

A total of 16 thrice-burned stainless steel shielded assemblies were installed at the core periphery. In addition, eight twice burned assemblies were placed on the core periphery. The remaining 24 peripheral locations were filled with fresh fuel assemblies (Figure 3.2). With this arrangement, it was anticipated that the reduced power in the peripheral assemblies would reduce the primary source of fast neutrons reaching the reactor vessel axial welds.

Design of the Cycle 9 core is based upon 52 fresh, 60 once-, 76 twice-, and 16 thrice-burned fuel assemblies. All thrice burned assemblies will have zircaloy-clad hafnium rods placed in eight guide tube locations. These assemblies will be placed on the edge of the core near critical weld locations (Figure 3.3). Hafnium is an effective absorber primarily for neutrons in the thermal through epithermal energy ranges. It is anticipated that the power in these thrice-burned fuel assemblies will be greatly reduced along with the neutron source. Therefore, there will be fewer neutrons reaching the vessel at the critical weld locations.

3.3 Geometry

The Palisades reactor exhibits one-eighth (1/8) core symmetry, thus only a zero to 45 degree sector has been included in the DOT model (Figure 3.4). In this figure two surveillance capsules attached to the inner vessel wall are shown. A plan view of the Palisades capsule arrangement is shown in Figure 3.5, with specific surveillance capsules dimensions shown in Figure 3.6. Figure 3.5 shows that four of the 45 degree sectors do not have any capsules. Two other sectors have one accelerated (attached to core support barrel) and one wall capsule. The remaining two sectors have two vessel wall capsules at the 10° and 20° locations. The utilized DOT model contains two wall capsules at the 10° and 20° locations. This model utilizes 99 radial and 98 azimuthal intervals for a total of 9702 meshes in polar (R, θ) geometry. Fine mesh detail has been utilized as necessary in setting up the geometry model to accurately represent the reactor core, shroud, bypass flow, core support barrel, inlet

flow, surveillance capsules, vessel clad and the vessel wall regions. A total of 15 outer assemblies have been modeled to represent the detailed core; the total model mesh extends to just outside the vessel in the reactor cavity area. Various regions of the DOT model are represented in such a way that their volumes are close to that of the physical volumes of the reactor internals.

3.4 Material Cross Sections

The DOT model analysis employed a P_3 expansion of the scattering cross sections. The microscopic cross sections used in the analysis were obtained from the SAILOR cross section library. Macroscopic cross sections were calculated for each region in the model using the computer code GIP. Plant specific material compositions and the corresponding atomic densities were used for this analysis.

3.5 Neutron Source

Assembly-wise radial power distributions were obtained from the Palisades incore monitoring system (INCA) for Cycles 1 through 7; fuel vendor-generated discrete PDQ bundle power data were used for Cycles 8 and 9. Average energy generated by fuel assemblies was obtained from the exposure data and the heavy metal weight of the assemblies to calculate cycle average assembly powers. Figures 3.7 through 3.15 exhibit the fifteen (15) outer peripheral normalized bundle powers for Cycles 1 through 9. Cycle 8 actual assembly power data to date is adequately modeled by utilizing the predictive core simulator information. Local pin power distributions were derived from discrete PDQ model calculations. The local pin power distributions and the average assembly powers were combined to determine core normalized pin power distributions.

Axial peaking was accounted for by applying the bundle specific axial peaking factors to the normalized pin powers of the fifteen modeled fuel assemblies. This approach conservatively defines the axial variation of the vessel incident neutron/source. Axial power information was obtained from INCA core monitoring data for Cycles 1 through 7 and 3D XTG core simulator models for Cycles 8 and 9. The core power distributions were initially calculated in Cartesian (x,y) geometry from the original data sources. The Cartesian geometry was converted to a polar (R,θ) geometry using an algorithm that maintained equivalent average source strength over the affected surface area between coordinate systems.

3.6 Burnup Corrections

As the fuel starts to deplete in the core during plant operation, exposure of the individual fuel assemblies increases and a build up of plutonium isotopes occurs. Plutonium isotopes have higher υ (neutrons/fission) and κ (energy/fission) values and exhibit fission spectra shifted towards the higher energies (harder spectra) than uranium isotopes. The contributions of the individual isotopes U235, U238, Pu239 and Pu241 to the core neutron source have been accounted for in the present set of flux calculations. Since the fission spectra and effective neutron yield differs for the above isotopes, the core neutron source and the vessel wall flux will generally increase with the fuel depletion for given peripheral assembly power levels. This is especially important for the twice and thrice burned fuel at the core periphery for Cycles 8 and 9. Composite fission spectra for each of Cycles 1 through 9 have therefore been developed. Individual isotopic fission spectra were obtained from ENDF-B/V for the uranium and plutonium isotopes. The spectra were collapsed to 47 energy groups similar to the SAILOR Library [7]. The exposure dependent neutron source for each cycle was then determined by weighting the individual group-wise neutron yields with the corresponding exposure dependent isotopic fission fractions based on the cycle average exposure of five peripheral assemblies. Only 19 groups above 1 MeV have been employed in the DOT model for the fast flux calculations. Cycle specific fission spectra are shown in Table 3.1 in comparison with the SAILOR Library fission spectra. Fission spectra are normalized to one (1) neutron in the 47 groups, similar to the SAILOR Library. From Table 3.1, it can be noted that high energy neutron groups have higher yields for Cycles 8 and 9, compared to the previous seven cycles.

In the DOT model, cycle-specific υ/κ ratios for fifteen (15) fuel assemblies were obtained from CASMO lattice depletion code data for a standard Palisades fuel type, utilizing middle-of-cycle exposure values. The effect of neutron yield and energy generated in these assemblies were incorporated in the neutron source. These effects are more important on the fast flux at the reactor vessel for Cycles 8 and 9 as compared to previous Cycles 1 through 7.

3.7 Neutron Transport Analysis

The spatial distribution of neutron flux in the reactor was calculated using the DOT 4.3 computer code. The DOT program solves the Boltzman transport equation in two-dimensional geometry using the method of discrete ordinates. Third order scattering (P_3) and S_8 angular quadratures were used. The cycle-by-cycle neutron flux distributions were calculated using the cycle-dependent neutron sources and material compositions.

3.8 Vessel Fluence Calculations

Fluence levels of a given cycle were obtained by multiplying the flux at the clad-base metal interface by the effective full power seconds at 2530 MWTH for that cycle. Accumulated fluence values at the EOC 8 were calculated by adding the fluence for all the Cycles 1 through 8. Further extrapolation to end-of-life fluence is based upon the estimate that the plant will operate at 75% capacity factor after EOC 8 at the calculated fluence rate for the Cycle 9 proposed core loading scheme.

3.9 Fluence Limits/Reference Temperature Calculations

Target fluence limits for pressure vessel welds and base metals are calculated using the 10CFR50.61 correlation for RT_{PTS} and the vessel material PTS screening criteria. The reference temperature correlation is given as:

$$RT_{PTS} = I + M + (-10 + 470Cu + 350CuNi)f^{0.270}$$

where:

RT_{PTS} is the adjusted reference temperature for pressurized thermal shock considerations (°F)

I is the initial reference temperature (°F)
M is the margin term (°F)
Cu, Ni are the copper and nickel content (in weight percent), respectively
f is the accumulated fluence (E>1.0 MeV)) in units of $10^{19} n/cm^2$

The corresponding fluence limits are determined by solving the RT correlation for the fluence value. Initial reference temperature and chemistry information and corresponding fluence limits are shown in Table 3.2.

Target fluence limits for pressure vessel welds and base metals are also calculated using Regulatory Guide 1.99, Rev 2 reference temperature correlation and the 10CFR50.61 PTS screening criteria. The adjusted reference temperature for each material in the beltline is given as:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

or

$$ART = I + M + \Delta RT_{NDT}$$

where: $\Delta RT_{NDT} = (CF{*}f^{(0.28-0.10 \log f)})$

I, M and f have the same meaning as above. The chemistry factor, CF (°F) depends on the content of copper and nickel in the belt line materials. This factor is provided in Regulatory Guide 1.99, Rev 2. The corresponding fluence limits are determined by solving the RT correlation for the fluence value and are shown in Table 3.3.

For each Cycles 1 through 9, fluence values were obtained for the base metal, and axial and circumferential weld materials. Using the parameters of Table 3.3 and the accumulated fluence at the end of each cycle, the corresponding adjusted reference temperatures were calculated.

4.0 RESULTS

4.1 Comparison to Measured Data

The W-290 surveillance capsule was removed at the end-of-Cycle 5 and was analyzed by Westinghouse [6]. The measured average flux at the W-290 capsule was corrected for a discrepancy in the power irradiation history data (Figure 4.1) versus that utilized in Reference 6. The corrected measured flux at the W-290 capsule was $6{\cdot}73\times10^{10}$ n/cm^2-sec. DOT calculations for Cycles 1 through 5 provide a cycle-energy averaged flux at the W-290 locations of $7{\cdot}02\times10^{10}$ n/cm^2-sec [3], 4% higher that the measured value. It was also noted that the lead factors obtained for Cycles 1 through 5 from DOT calculations were fairly constant (between 1·24-1·27). These facts indicate that calculated flux values from the in-house DOT results can be directly used for reasonable end-of-life fluence calculations. It should be noted that these results are a slight improvement over the DOT calculations utilized in Reference 1, which exhibited a +11% bias relative to the measured W-290 fluxes.

4.2 Flux/Fluence Distribution

For Cycles 1 through 9, the maximum fast flux occurs at the azimuthal interval between 16·44°-17° at the clad-base metal interface (Table 4.1). Flux distributions for Cycles 3 through 7 are very similar. Comparison of flux distribution between different cycles is presented in Figures 4.2 and 4.3. These figures confirm that the maximum flux occurs around 17°. Wall capsules at 10° and 20° exhibit an attenuating effect in their immediate vicinities, but do not affect the peak fluxes. For Cycles 1 though 7, a second peak occurs around the 32° azimuthal location. For Cycles 8 and 9, this peak is eliminated as a result of the implementation of low leakage fuel management schemes. Substantial flux reduction for the low leakage fuel management schemes relative to the high neutron leakage loading patterns is apparent. Radial flux distributions at the 0, 17 and 30 degree azimuthal locations for Cycles 7 (representative of previous cycles), 8 and 9 are presented in Appendix 7.1.

Accumulated fast fluence distributions at the end of Cycle 9 and EOL at the clad-base metal interface is shown in Figure 4.4. Based upon Reg. Guide 1.99, Revision 2, fluence limits corresponding to base metal, axial, and circumferential welds are also presented in Figure 4.4. From this figure it can be noted that the fluence values at the axial welds at 0° and 30° are limiting the life of the Palisades reactor pressure vessel.

Table 4.2 summarizes the cycle specific fluence ($\Delta\Phi$) and accumulated fast fluence ($\Sigma\Phi$) at the clad-base metal interface for each of Cycles 1 through 9. For the selected azimuthal locations: 0° (axial weld location), 17° (maximium of peak at base metal), 30° (axial weld location) and 45°, effective full power years (ΔEFPY) for each cycle and the accumulated EFPY's are also presented. Table 4.3 provides the fluence limit violation dates with Cycle 9 fluence rates for plant operations beyond the end of Cycle 8 date of September, 1990.

4.3 Calculational Uncertainty

A number of factors contribute to the uncertainty in the projected peak fast fluence at the reactor vessel wall. These factors are due to the conversion of measured activity data to fluxes, uncertainties in material composition, neutron cross sections, power distributions, as-built core/vessel dimensions and cycle-by-cycle variation in the fast flux lead factors. An uncertainty of ±25% is estimated in the calculated vessel wall fluence, typical of current neutron transport methodology uncertainties. The calculated +4% flux bias relative to actual W-290 measured fluxes indicates that vessel wall flux predictions are reasonable given the inherent uncertainty in the methodology.

4.4 Adjusted Reference Temperatures and Screening Limits

Adjusted reference temperatures (ARTs) as a function of effective full power years (EFPYs) corresponding to the fluence values at the end of Cycle 1 through 9 and projected to plant EOL, have been plotted in Figure 4.5. PTS screening limits for each of the beltline materials are provided. This figure

suggests that the axial welds are the limiting material for the Palisades reactor pressure vessel relative to PTS limits. Table 4.4 provides the summary of PTS adjusted reference temperatures for base metal, axial and circumferential weld materials. Note that for the licensed end-of-life date of March, 2011, ARTs for the axial welds at 30 degrees exceed the PTS screening limit of 270°F.

5.0 DISCUSSION

5.1 Impact of Results

Modifications to the Cycles 8 and 9 loading patterns substantially reduce the flux at the critical weld locations and delays exceeding the PTS screening criteria to about September 2001, as opposed to in 1995 if no flux reduction measures are taken. The flux reduction is insufficient, however, to allow operation of the plant within the PTS screening criteria until the minimum expected plant life, corresponding to the expiration of the pending full term operating license in March, 2011.

In-house flux calculations have a positive bias with respect to Westinghouse model [1], mainly due to the slightly larger core size in the in-house model. The bias ranges from +0.5% at 0° and increases to about +11.7% at the 45° location. In addition, more realistic plant-specific design and operational data have been utilized in the in-house model. This approach therefore does not depend very heavily on assumptions used for the flux calculations, but relies on the plant specific parameters.

In order to maximize vessel lifetime, further measures must be taken in the areas of greater flux reduction, Reg Guide 1.154 analysis to properly define the real Palisades PTS risk, and possible vessel annealing/shielding actions to reduce the accumulated vessel embrittlement rate.

5.2 Additional Flux Reduction

The most straightforward method of reducing the vessel fast flux level is reduction of the source itself, which has been initially addressed with the incorporation of low-leakage fuel management and stainless steel shield rods in Cycle 8 and thrice burned fuel with hafnium absorbers for Cycle 9. While flux reduction gains are predicted for Cycle 9, some further reductions are believed to be obtainable via fuel management alone. Cycle 9 will be the first cycle with the new steam generators installed. The new generators are expected to provide substantially higher primary coolant flow than the current generators.

The increased flow, which can be quantified accurately during Cycle 9 operation, will provide additional core operating thermal margin and thus allow higher power peaking limits to be utilized in developing the Cycle 10 loading pattern. The higher peaking will provide additional fuel management flexibility and support more aggressive low-leakage fuel management for further reductions in vessel wall fluxes.

Additionally, Cycle 9 will be the first cycle to incorporate a new high thermal performance (HTP) spacer grid design in the fresh reload fuel. Insertion of a second reload of fuel with the HTP spacers in Cycle 10, along with development of a Palisades-specific DNB correlation for the HTP fuel, will provide additional allowable peaking factor increases to be utilized in Cycle 10.

A third area design to allow greater fuel management flexibility in the Cycle 10 core design will be the installation, utilization, and optimization of a new full core power monitoring system beginning in Cycle 9. This monitoring system will allow the Cycle 10 loading pattern design to utilize 1/4 core symmetry, as opposed to current 1/8 core symmetry utilized in Cycles 1-9, and will provide more options for reducing power and flux levels in peripheral fuel assemblies.

Discussions have been held with NSSS vendors on the possibility of installing critical material area neutron shields. A shield between the fuel and the vessel wall would act to reflect, slow down, or absorb high-energy neutrons before they could reach the vessel wall. Stainless steel shielding pads could be designed to mount near the core support barrel to maximize the attenuation of the high energy neutrons of concern. The possibility exists to use other hybrid materials which are better neutron shielding then stainless steel and therefore provide further neutron flux reduction beyond that attainable with low leakage fuel management alone. It is estimated that internal vessel shielding could reduce the flux at the critical axial weld locations a minimum of 25%.

5.3 Refined Flux Measurement

In order to benchmark vessel fluence calculations, an upgraded vessel dosimetry program has been initiated to supplement the existing surveillance capsule program. An ex-vessel dosimetry program was developed by Westinghouse and hardware installation occurred during the end of Cycle 7 refueling outage. The

dosimetry installed will provide detailed azimuthal and axial mapping of the 270-360 degree vessel quadrant, with gradient chains installed in the other three quadrants to provide accurate axial and cross-quadrant mapping. It is intended to exchange this dosimetry at the end of Cycle 8 with similar sets of dosimeters for the Cycle 9 irradiation period. The dosimetry will provide measured data for use in vessel wall and supports fluence evaluations. In addition to the ex-vessel program, Combustion Engineering has been contracted to fabricate and install a replacement in-vessel dosimetry capsule to be inserted into the W-290 capsule holder vacated following Cycle 5. Installation will occur during the next refueling outage (Fall 1990). When installed, this capsule will provide an excellent through-wall correlation with the ex-vessel dosimetry installed in the same quadrant.

In addition to implementing the supplemental dosimetry program, efforts will be made to extend the DOT model up to the reactor cavity area to analyze the ex-vessel dosimeters. A further enhancement planned to the DOT model will be to synthesize a 3-D model for flux calculations to remove some of the inherent conservatism in the calculations due to utilization of the bundle-specific peak axial power over the entire core axial height.

5.4 Other PTS Activities

Planned flux reduction measures do not appear to fully solve the vessel fluence issue relative to PTS. Consumers Power Company is pursuing a methodology through the Combustion Engineering Owners Group (CEOG) to augment plant data by correlating surveillance material and data from other plants to Palisades vessel materials. Such data could allow Palisades to reduce operating restrictions caused by Regulatory Guide 1.99, Rev 2/10CFR50.61 default margin terms and initial reference temperatures for generic weld material in absence of actual Charpy weld test specimen data.

A detailed risk evaluation based on Regulatory Guide 1.154 analysis is also being pursued through CEOG. Such analysis will identify and summarize the potential risk of a PTS event occurring. This risk would be based on the known

operating activities or transients which could lead to a PTS event. The program is being undertaken in a phased approach with the currently in-progress Phase I dealing with generic model development only. The analysis, if actually needed, would be completed at least three years prior to the predicted exceed date of the PTS screening criteria.

6.0 REFERENCES

1. Letter from R W Smedley (CPCo) to NRC, "Docket 50-255 - License DPR-20 - Palisades Plant - Compliance with Pressurized Thermal Shock Rule 10CFR50.61 and Regulatory Cuide 1.99 Revision 2 - Fluence Reduction Status (TAC No. 59970)," April 3, 1989.

2. Letter from J C Hoebel (Westinghouse) to R A Klavon (CPCo) "Interim Report of Westinghouse Review of Consumers Power PTS Calculations," August 29, 1989.

3. Engineering "Analysis Package for PTS study, Reactor Engineering Department, Palisades Plant (1987-90).

4. Letter from K W Berry to NRC, "Response to Request for Additional Information - Pressurized Thermal Shock (PTS) Rule 10CFR50.61," August 7, 1986.

5. RSIC Computer Code Collection DOT IV Version 4.3 (Report No. CC-429).

6. WCAP - 10637, Analysis of Capsules T-330 and W-290 from the Consumers Power Company Palisades Reactor Vessel Radiation Surveillance Program, M K Kunka and C A Cheney, September, 1984.

7. RSIC Library Collection SAILOR DLC-76.

8. Telecopy of E.P. Lipincott (Westinghouse) to O.P. Jolly (CPCo), "Final Report on Westinghouse Review of Consumers Power PTS Calculations," April 20, 1990.

EXHIBIT 1 J

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

May 19, 1995

NRC GENERIC LETTER 92-01, REVISION 1, SUPPLEMENT 1: REACTOR VESSEL STRUCTURAL INTEGRITY

Addressees

All holders of operating licenses (except those licenses that have been amended to possession-only status) or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this supplement to Generic Letter (GL) 92-01, Revision 1, to require that all addressees identify, collect and report any new data pertinent to analysis of structural integrity of their reactor pressure vessels (RPVs) and to assess the impact of that data on their RPV integrity analyses relative to the requirements of Section 50.60 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.60), 10 CFR 50.61, Appendices G and H to 10 CFR Part 50, (which encompass pressurized thermal shock (PTS) and upper shelf energy (USE) evaluations) and any potential impact on low temperature overpressure (LTOP) limits or pressure-temperature (P-T) limits.

Background

The staff issued GL 92-01, Revision 1, "Reactor Vessel Structural Integrity," on March 6, 1992, to obtain information necessary to assess compliance with requirements regarding RPV integrity in view of certain concerns raised in its review of RPV integrity for the Yankee Nuclear Power Station. All licensees submitted the information requested by July 2, 1992. Following receipt and review of licensee supplements responding to requests for additional information, the staff completed its review of licensee responses to GL 92-01, Revision 1, in the fall of 1994. The staff issued NUREG 1511, "Reactor Vessel Status Report," summarizing key aspects of the work in December 1994 [Ref. 1].

The staff has recently reviewed data relevant to the PTS evaluations of several plants. These reviews showed that licensees may not have considered all pertinent data in their responses to GL 92-01, Revision 1, or in their RPV integrity evaluations. It has now become apparent to the staff that no single organization has all the data relevant to RPV integrity evaluations. A major complicating element in this regard is that proprietary considerations have inhibited effective sharing of information.

It has been demonstrated that some RPV integrity evaluations are very sensitive to consideration of new data. For example, under certain conditions, changing the mean copper content for the limiting vessel beltline material by a few hundredths weight percent can change the predicted date for reaching the PTS screening criteria of 10 CFR 50.61 by several years. In addition, changes in estimates of mean copper content can affect the validity of PTS evaluations based on surveillance data. The staff will be considering the impact of these findings in plant-specific evaluations and in its longer-term reassessment of 10 CFR 50.61. PTS is a concern only for pressurized water reactors (PWRs) because boiling water reactors (BWRs) operate with a large inventory of water at saturated steam conditions and, therefore, are not subject to PTS.

However, in addition to concerns regarding PTS evaluations, consideration of additional, unreviewed RPV data can also affect evaluations for USE, P-T limits, and LTOP limits. These evaluations pertain to both PWRs and BWRs, except for LTOP limits, which apply only to PWRs. The staff recognizes that addressees have previously submitted data pertinent to these evaluations as required by the regulations and in responses to GL 92-01, Revision 1, and GL 88-11.

Based on currently available information, the staff believes that the near- term focus for RPV integrity will be the Palisades RPV which is predicted to reach the PTS screening criteria by late 1999, before any other plant. However, because of the importance of RPV integrity and the potential impact of additional, unreviewed data on existing RPV evaluations, the staff believes that this issue needs to be resolved on an expedited basis. Although the issues raised in this GL supplement were highlighted by concerns pertaining to PTS analyses, licensees should consider the effect of the reexamination of RPV data on all aspects of RPV structural integrity.

Regulatory Requirements

As required by 10 CFR 50.60(a), licensees for all light water nuclear power reactors must meet fracture toughness requirements and maintain a material surveillance program for the reactor coolant pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. 10 CFR 50.60(b) provides that proposed alternatives to the requirements of Appendices G and H to 10 CFR Part 50 may be used when an exemption is granted under 10 CFR 50.12. 10 CFR 50.61 provides fracture toughness requirements for protecting PWRs against PTS events. Licensees and permit holders have also made commitments in response to GL 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," to use the methodology in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to predict the effects of irradiation as required by Paragraph V.A of Appendix G to 10 CFR Part 50.

Discussion

The staff focused its examination of the GL 92-01, Revision 1, data and other docketed

below or the appropriate NRR project manager.
/s/'d by RPZimmerman

Roy P. Zimmerman
Associate Director for Projects
Office of Nuclear Reactor Regulation
Technical contacts: Edwin M. Hackett
(301) 415-2751

Keith R. Wichman
(301) 415-2757

Lead project manager: Daniel G. McDonald
(301) 415-1408
Attachments: 1. References
2. List of Recently Issued NRC Generic Letters

(NUDOCS Accession Number 9505090312)

ATTACHMENT 1

GL 92-01, Rev. 1, Supp.
May 19, 1995

References

[1] NUREG-1511, "Reactor Pressure Vessel Status Report," U.S. Nuclear Regulatory Commission, Washington, DC, December, 1994.
[2] Letter from Elinor Adensam, USNRC, to Kurt Haas, Consumers Power Company forwarding, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the Evaluation of the Pressurized Thermal Shock Screening Criteria, Consumers Power Company, Palisades Plant, Docket No. 50-255", April 12, 1995.

EXHIBIT 1 K

POLICY ISSUE
(Information)

October 28, 1994

SECY-94-267

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: STATUS OF REACTOR PRESSURE VESSEL ISSUES

PURPOSE:

To provide an update of the status of plants with regard to Appendix G, "Fracture Toughness Requirements," to Part 50 of the Code of Federal Regulations (10 CFR) and 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

BACKGROUND:

In SECY-93-048, the staff of the Nuclear Regulatory Commission (NRC) stated that it was performing detailed reviews of licensee responses to Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)." As part of this review, the staff has assessed the upper-shelf energies (USEs), transition temperatures, and reference temperature for pressurized thermal shock $RT_{(PTS)}$ or adjusted reference temperatures (ARTs) for all domestic commercial nuclear power plants. Appendix G to 10 CFR Part 50 requires licensees (1) to operate their reactor vessels with pressure-temperature limits that are dependent on the amount of increase in the transition temperature resulting from neutron radiation, and (2) to maintain the Charpy USE throughout the life of the vessel of no less than 41 Joules (50 ft-lb), unless it is demonstrated that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the American Society of Mechanical Engineers Boiler and Pressure Vessel Codes (ASME) Code. The analyses submitted by licensees to demonstrate margins of safety equivalent to those required by Appendix G of the ASME Code are called equivalent margins analyses. The increase in the transition temperatures affects the RT_{PTS} values for pressurized water reactors (PWRs) that are calculated in accordance with 10 CFR 50.61 and the ART that is calculated in determining the pressure-temperature limits for both PWRs and boiling water reactors (BWRs).

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 10 WORKING DAYS FROM THE DATE OF THIS PAPER

Contact:
B. Elliot, NRR/DE/EMCB
504-2709

Plant Name: Palisades

Docket Number: 50-255

NSSS Vendor: Combustion Engineering

Vessel Manufacturer: Combustion Engineering

Edition of ASME Code for Design: Winter 1965 Addenda to 1965 ASME Code

Date of Commercial Operation: December 31, 1971

Date of License Expiration: March 14, 2007

RT_{pts} for the Limiting Beltline Material:

Limiting Beltline Material: Axial welds, heat W5214
ID Fluence at EOL: 1.91E19 n/cm^2
Initial RT_{NDT}: -56°F
Method of Determining Chemistry Factor: Chemistry data per Paragraph C.1.1 of RG 1.99, Rev. 2
Increase in RT_{NDT} at EOL: 265°F
Margin: 66°F
RT_{pts} at EOL: 275°F
Date at which PTS Screening Limit will be exceeded: 2004

USE for the Limiting Beltline Material:

Limiting Beltline Material: Plate D-3804-1, heat C-1308
1/4T Fluence at EOL: 1.615E19 n/cm^2
Initial USE: 72 ft-lb
Percent Drop at EOL: 31%
USE at EOL: 50 ft-lb
Date USE Screening Limit will be Exceeded: After EOL
Bases for Accepting the USE at EOL: Chemistry data per Paragraph C.1.2 of RG 1.99, Rev. 2

REFERENCES:

July 3, 1992, letter from G. B. Slade (CPCo) to USNRC Document Control Desk, Subject: Palisades Plant--Reactor Vessel Structural Integrity--Response to Generic Letter 92-01, Revision 1

February 23, 1994, letter from D.W. Rogers (CPCo) to USNRC

August 31, 1990, letter from G.B. Slade (CPCo) to USNRC

July 12, 1994, letter from A. Hsia (NRC) to CPCo

EXHIBIT 1 L

Office of Nuclear Reactor Regulation
Items of Interest
Week Ending November 4, 1994

SECY-94-267, "Status of Reactor Pressure Vessel Issues"

In Commission Paper SECY-94-267, "Status of Reactor Pressure Vessel Issues," the staff indicated that the Palisades reactor pressure vessel would reach the pressurized thermal shock (PTS) screening criteria in the year 2004. We also indicated that the licensee was gathering additional materials properties data from its retired steam generators (the welds in the retired steam generators were fabricated using the same materials as used in the fabrication of the limiting Palisades reactor vessel beltline welds) and the results of these tests could change the date when the plant will reach the PTS screening criteria.

During telephone conversations with the licensee on November 1 and 2, 1994, the staff was informed of preliminary data from the retired steam generators that indicates the Palisades reactor pressure vessel could reach the PTS screening criteria earlier than 2004. The licensee is continuing to evaluate the new data and to gather additional materials properties from its retired steam generators. If the preliminary data are confirmed, the plant would reach the PTS screening criteria at the next outage in May 1995. A meeting between the staff and the licensee has been tentatively scheduled for November 18, 1994, to discuss the test results.

EXHIBIT 1 M

MATERIALS ISSUES IN PALISADES PTS EVALUATION

PRESENTED TO NSRRC SUBCOMMITTEE ON MATERIALS AND ENGINEERING

January 24, 1995

Michael E. Mayfield, Chief
Electrical, Materials and Mechanical
Engineering Branch
Division of Engineering Technology, RES
U.S. Nuclear Regulatory Commission
Washington, D.C.

Ed Hackett NRR
Core Shroud
&
PTS Evaluation

INTRODUCTION

- Materials Research Program Providing Direct Support to NRR on Current Regulatory Issues, Including
  - o Palisades PTS Evaluation
  - o BWR Core Shroud Cracking
- Both Issues Have Generic Implications
  - o Research Program Adjusted to Address Those Implications
- Presentations Will Summarize the Issues and the Research Programs

BACKGROUND -- PALISADES PTS EVALUATION

- 10 CFR 50.61 Fracture toughness requirements for protection against pressureized thermal shock events

  - o Includes embrittlement screening criteria -- RT_{PTS}
    - 270°F for axial welds and plates
    - 300°F for circumferential welds

  - o If criteria are to be exceeded, flux reduction and plant specific analyses may be required

  - o $RT_{PTS} = I + M + \Delta RT_{PTS}$

    I = Initial reference temperature (RT_{NDT}) of the unirradiated material
    - Measured values must be used if available
    - If plant specific values not available, generic mean must be used

    M = Margin to cover uncertainties

    ΔRT_{PTS} = Mean value of shift in reference temperature due to neutron irradiation
    - A function of fluence, and chemical composition (copper and nickel)

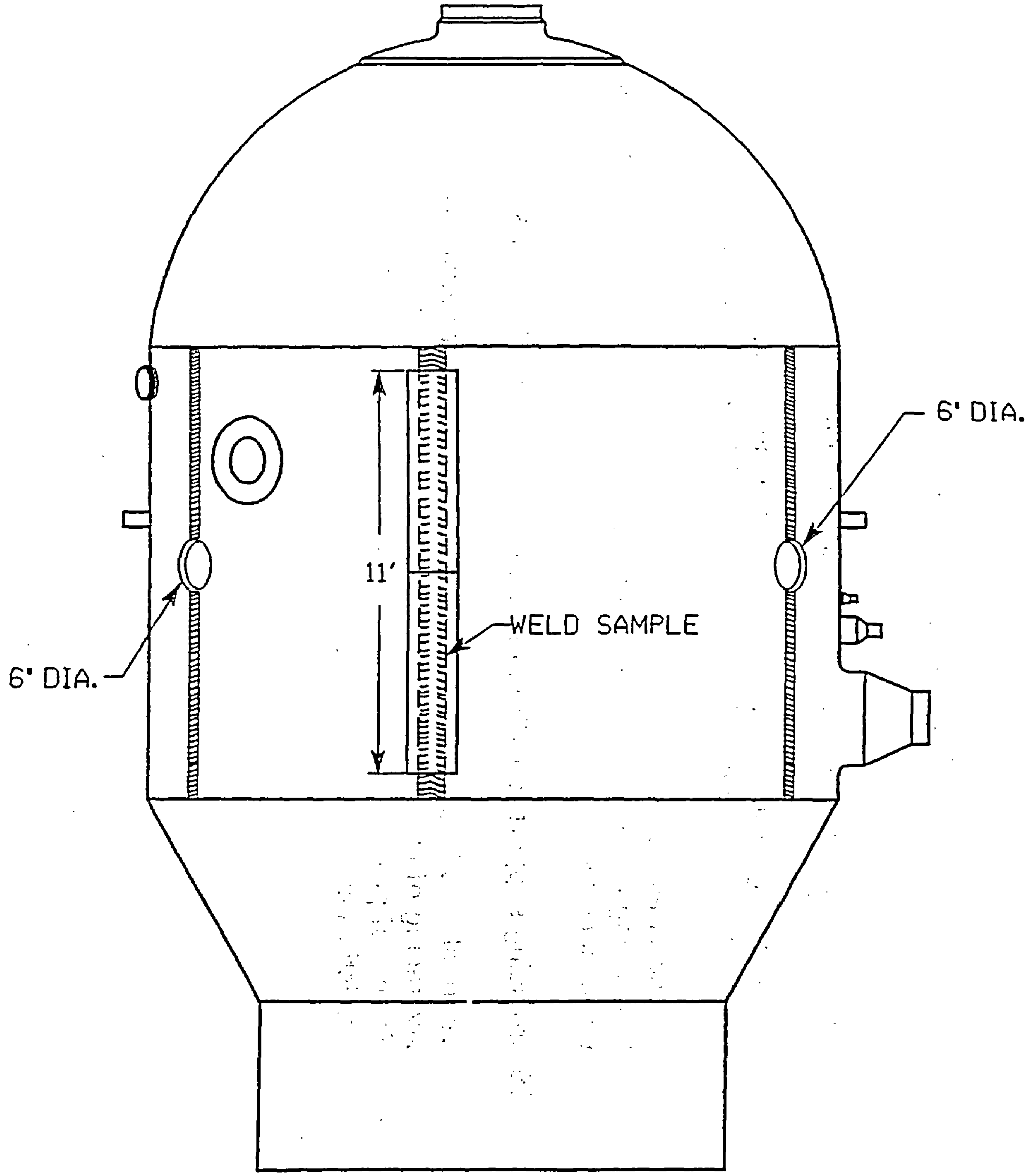
PALISADES PTS

- Palisades surveillance data not same as beltline weld material
    - Requires use of industry generic data

- Licensee took actions to provide representative data

    - Gathered additional material properties from welds in retired steam generators
    - Instituted an augmented surveillance program
    - Evaluated annealing of the reactor vessel
    - Considered instituting an "ultra low" leakage fuel strategy "how low can you go" reactor head cup

- Staff SER dated July 12, 1994 and NUREG-1511
    - Palisades vessel projected to reach PTS screening criteria in 2004 -- prior to EOL in 2007
    - Noted that evaluation could change based on information from SG welds


6' DIA.
11'
WELD SAMPLE
6' DIA.

PALISADES PTS

- November 1, 1994, licensee informed staff that data from SG welds
  - Indicated higher copper contents than previously assumed
  - Indicated higher initial RT_{NDT} than mean generic value
  - Licensee assessment indicated reaching PTS screening criteria in 1999

- Staff's assessment on-going
  - Depending on how the new data are evaluated, PTS screening criteria could be reached before 1999 SER being prepared

- RES providing support to NRR in resolving issues raised by licensee

  - Thermal embrittlement of SG welds cause of higher initial RT_{NDT}
  - Averaging of chemistry values
  - Effects of post-weld heat treatments on physical properties
  - Statistical analyses of chemistry and initial property data

RESEARCH EFFORT ON PALISADES ISSUES

- RES staff and contractors involved in providing independent assessment of data and analysis methods

- Thermal Embrittlement
  - Literature survey
  - Evaluation of data from Power Reactor Embrittlement Data Base
  - Evaluation of mechanisms of thermal embrittlement
  - Combined effects of thermal aging and neutron flux
  - Examination of mechanical properties data related to Palisades
  - Drop-Weight Specimen fabrication techniques

- Chemical Composition
  - Analysis of generic data
  - Fabrication techniques -- single arc versus tandem arc
  - Data weighting
  - Coil-weighted averaging
  - Recommendations on data treatment

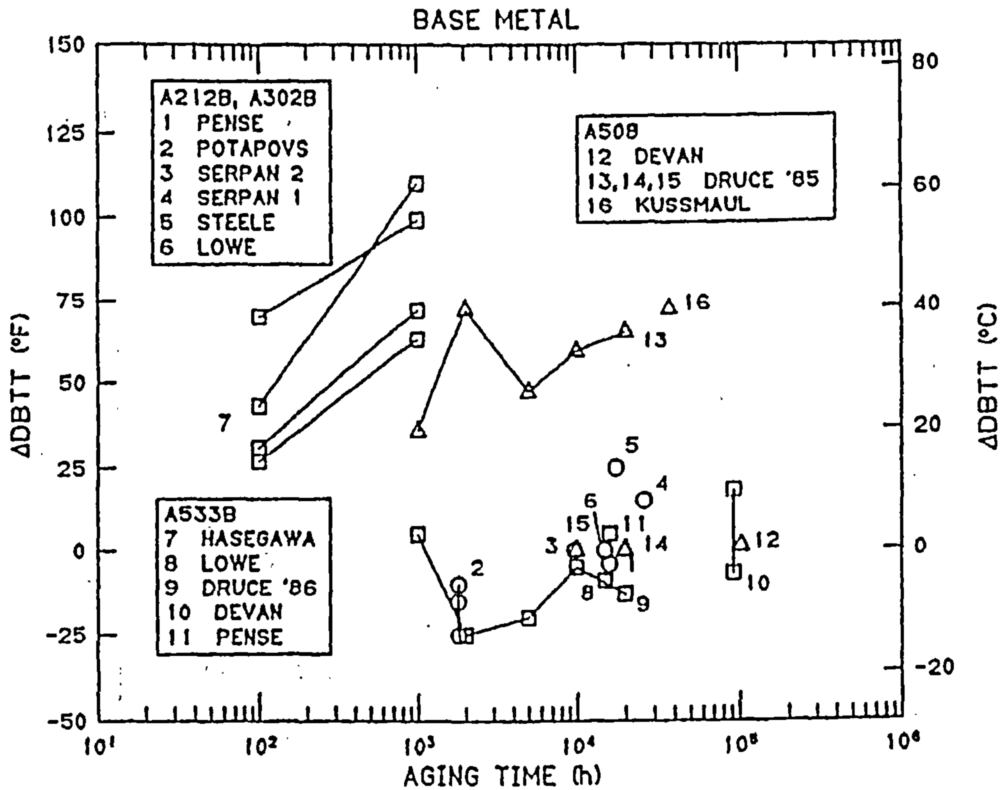
THERMAL EMBRITTLEMENT

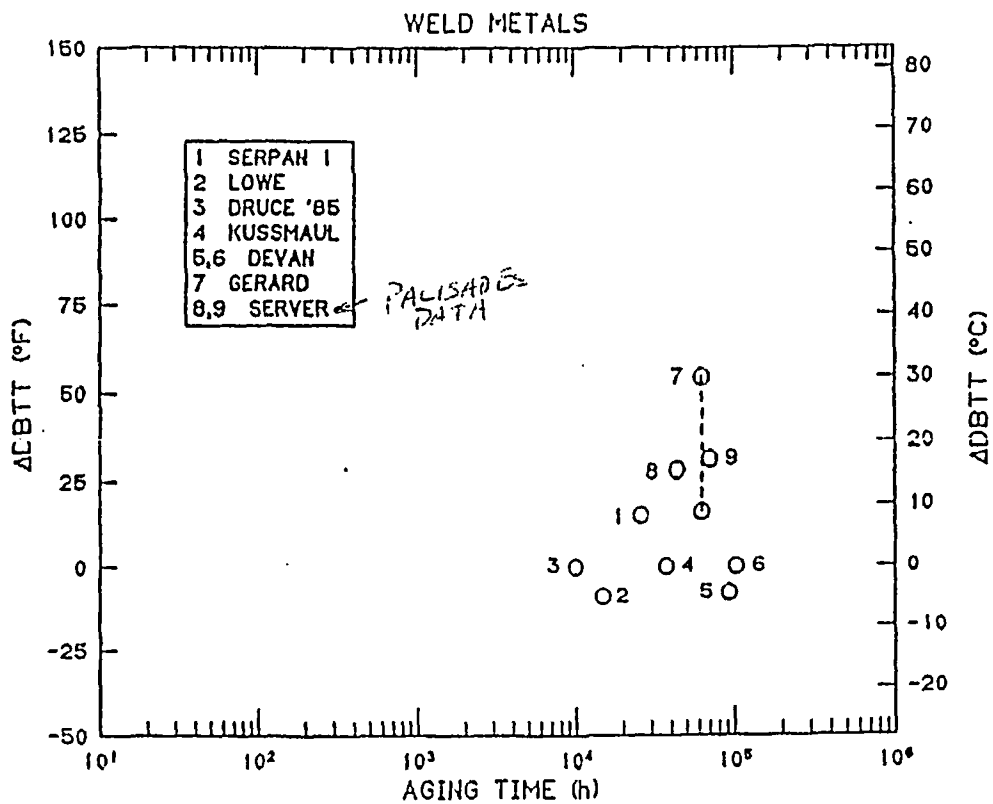
- Palisades SG RT_{NDT} tests indicate an initial value of -20°F compared to generic mean of -56°F
    - Cited thermal embrittlement as likely cause
    - 10 EFPY at approximately 500°F
    - Presented results from Belgian paper -- suggests 70°F shifts possible
    - Inconsistent with "common wisdon"
    - RES effort to look at available data and known mechanisms

- Literature survey

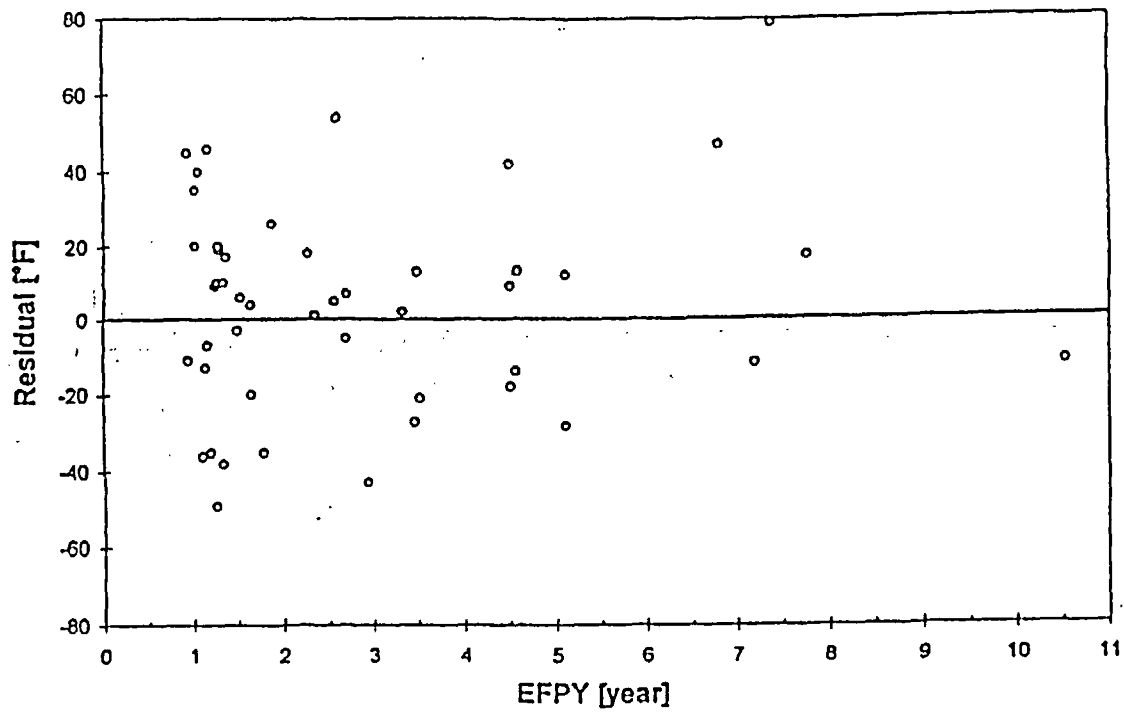
- Review of PR-EDB

- Evaluation of Belgian data

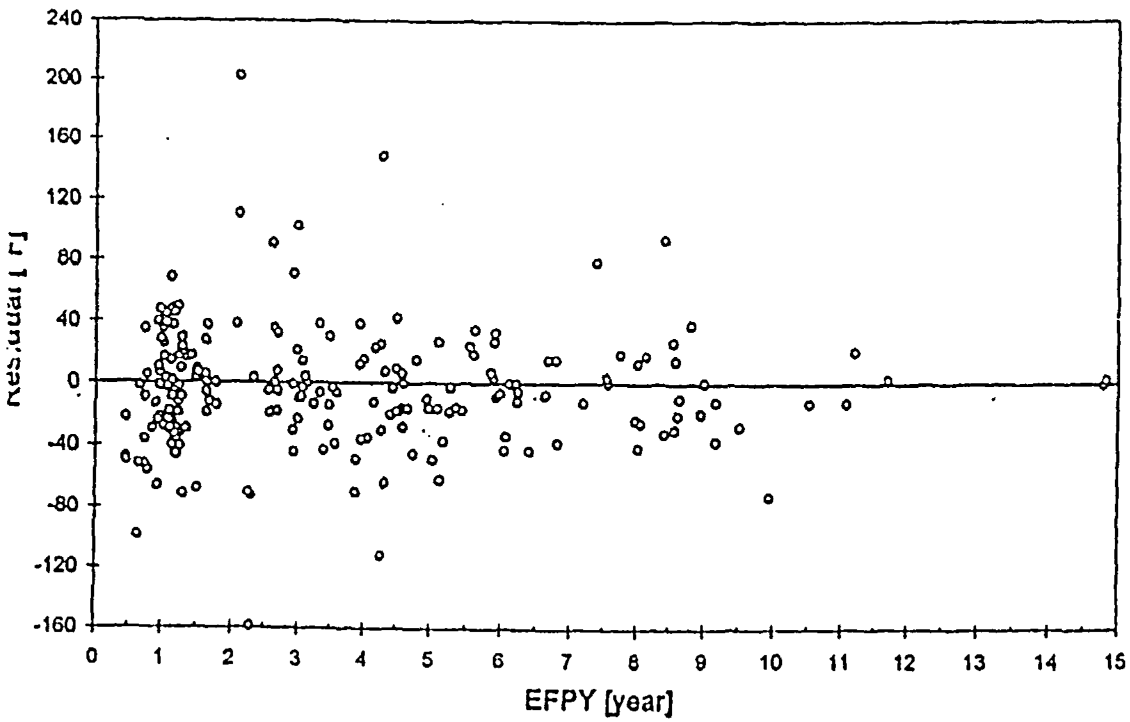

BASE METAL
A212B, A302B
1 PENSE
2 POTAPOVS
3 SERPAN 2
4 SERPAN 1
5 STEELE
6 LOWE
A508
12 DEVAN
13,14,15 DRUCE '85
16 KUSSMAUL
A533B
7 HASEGAWA
8 LOWE
9 DRUCE '86
10 DEVAN
11 PENSE
ΔDBTT (°F)
ΔDBTT (°C)
AGING TIME (h)

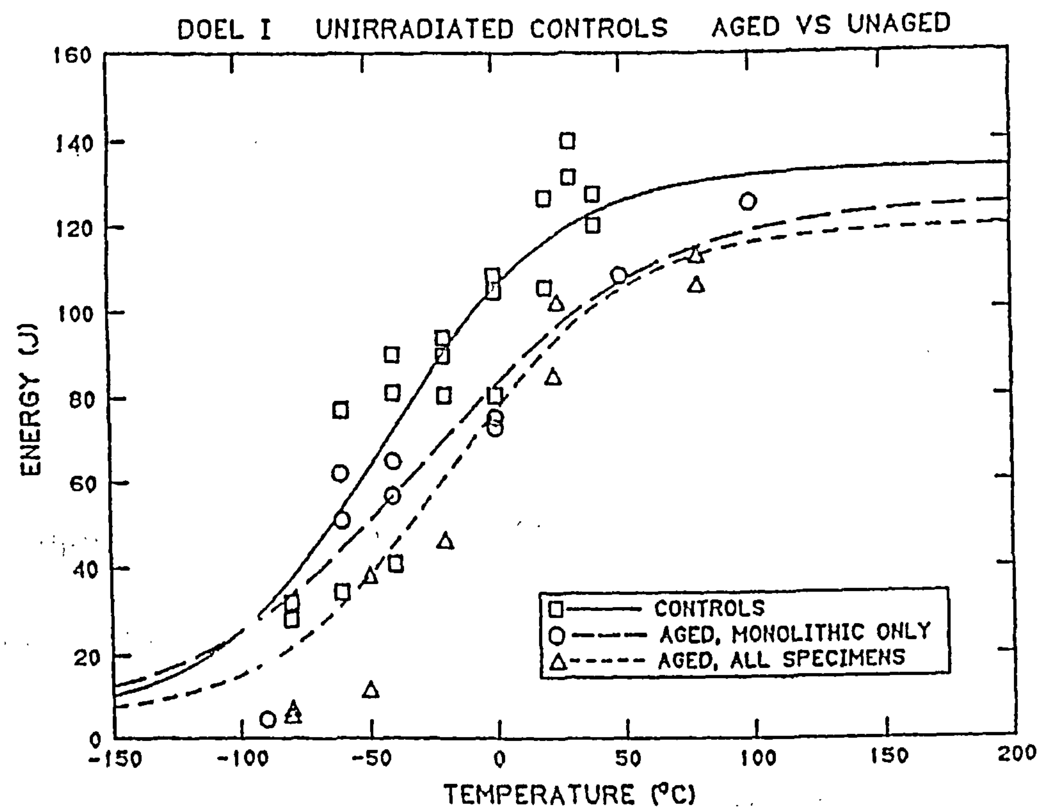

WELD METALS
1 SERPAN I
2 LOWE
3 DRUCE '85
4 KUSSMAUL
5,6 DEVAN
7 GERARD
8,9 SERVER
PALISADES DATA
ΔDBTT (°F)
ΔDBTT (°C)
AGING TIME (h)

Plot of Residual vs. Effective Full Power Year for Weld Materials per 177 Data Points



PLot of Residual vs. Effective Full Power Year for Weld Materials per PR-EDB





DOEL I UNIRRADIATED CONTROLS AGED VS UNAGED
ENERGY (J)
TEMPERATURE (°C)
CONTROLS
AGED, MONOLITHIC ONLY
AGED, ALL SPECIMENS

CONCLUSIONS ON THERMAL EMBRITTLEMENT

- Thermal embrittlement not likely source of higher RT_{NDT}

- Statistical variablity more likely

  - -20°F value is within $+2\sigma$ of generic mean

PALISADES CHEMICAL COMPOSITION ISSUE

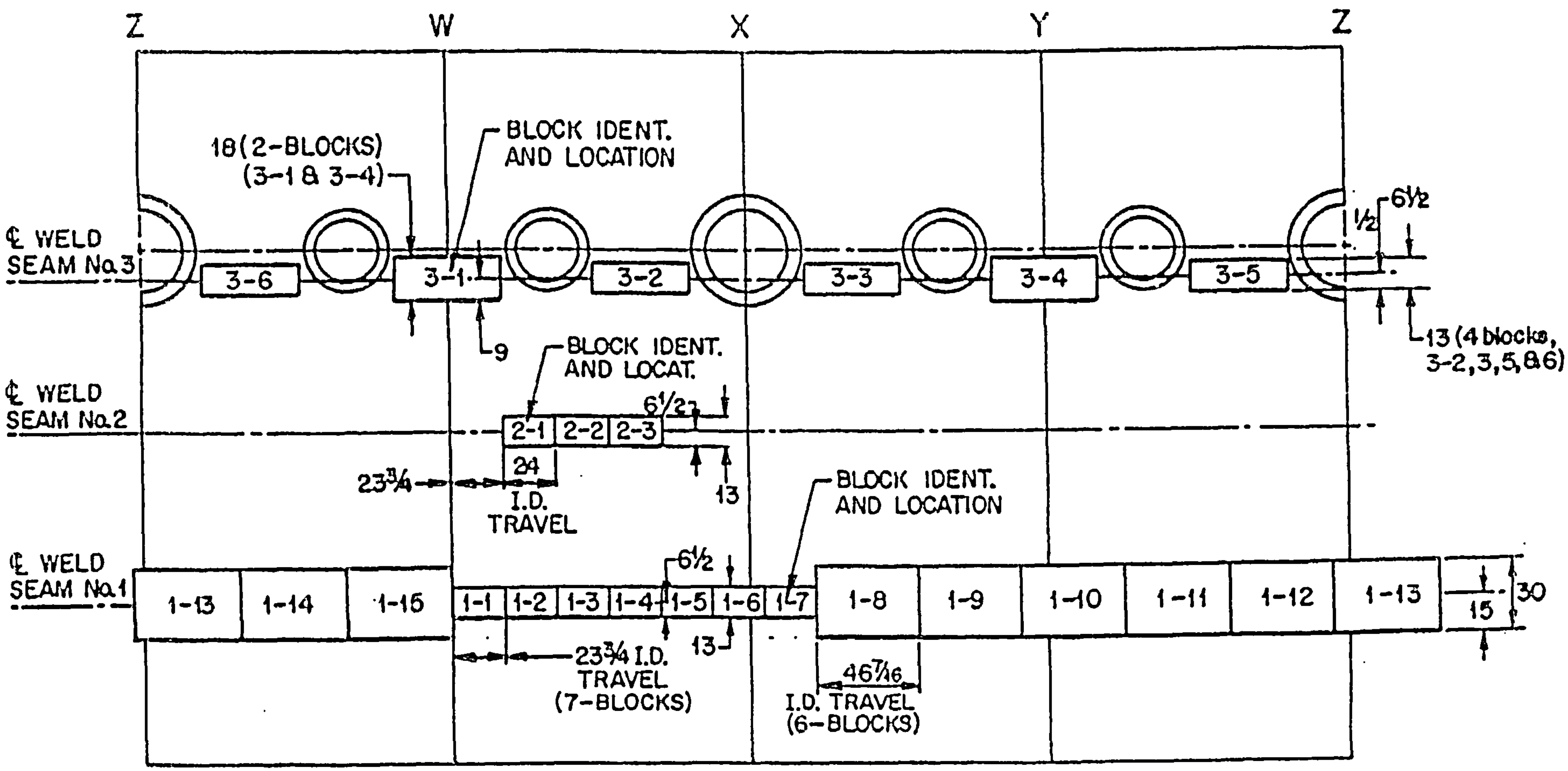
- Key issue is how to weight new data from SG welds
- Several proposals are being considered
- Cannot provide more details until staff discussions are completed and SER issued

CHEMISTRY CONTENT ISSUES

- Palisades SG data
- Weld fabrication procedure copper content
- Six different coils used in fabricating welds
- How does chemical variability relate to wire coil composition
  - Statistical treatment of data

RESULTS FROM MIDLAND UNIT 1 RPV

ORNL-DWG 92-11823A



OVERALL COPPER CONTENT IN MIDLAND WELDS SHOWS WIDE VARIATION

REGION	SECTION	COPPER (WT %)
Beltline	9	0.22 to 0.34
	11	0.16 to 0.34
	13	0.21 to 0.32
	15	0.22 to 0.33
Nozzle	31	0.37 to 0.46
	34	0.38 to 0.42

.16 to .46 very wide margin / problematic
greatly sensitive to copper content

RESEARCH PLAN TO ADDRESS THE GENERIC ISSUES

- Evaluating the effects material chemistry and radiation environment on irradiation embrittlement of RPV steels

  - o Examine samples from two representative RPV welds at ORNL
  - o Examine samples of thermally aged samples from retired SG shells
  - o Evaluate variability in chemistry and mechanical properties both along the weld and through the thickness
  - o Develop generic guidance on estimating chemistry and properties variability

- Evaluate the combined effects of thermal aging and neutron irradiation

  - o Thorough assessment of technical literature
  - o Review and assessment of foreign positions on thermal aging
  - o Detailed metallurgical assessment of thermally aged materials
  - o Evaluate thermal aging, heat treatments, and flux effects on neutron embrittlement

SCHEDULE

- FY 1995

  - Determine the variability of chemistry in representative RPV welds
  - Assessment of technical literature and foreign positions on thermal aging
  - Initiate metallurgical assessment of thermally aged materials
  - Continue irradiations of RPV materials

- FY 1996

  - Complete report on irradiation effects on old-fabrication practice plate material
  - Establish generic guidance for estimating material chemistry and properties variability

- FY 1997

  - Complete report on irradiation effects in the Midland weld
  - Complete determination of effects of thermal aging on RPV materials from SGs

SUMMARY

- PTS continues to be a significant issue for PWRs
- Palisades PTS evaluation highlights uncertainty in initial properties and in embrittlement estimates
- Research program addressing the key factors

EXHIBIT 2 A

NUREG/CR-2907
BNL-NUREG-51581
Vol. 14

Radioactive Materials Released from Nuclear Power Plants

Annual Report 1993

Prepared by
Tichler, K. Doty, K. Lucadamo

Brookhaven National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

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CR-2907 R PDR

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1.0 Introduction

1.1 Purpose

This report, prepared annually for the staff of the U.S. Nuclear Regulatory Commission, presents measured data on radioactive materials in effluents released from licensed commercial reactor power plants. These data were reported by licensees for plant operations during 1993. This information supplements earlier annual reports issued by the former Atomic Energy Commission and Nuclear Regulatory Commission. [1]

1.2 Scope

Releases of radioactive materials are governed by 10 CFR Part 20 and 50 and by limits established in the Technical Specifications for each facility. The requirement for reporting effluent releases by nuclear power plant operators is described in 10 CFR 50.36a. Through its Office of Nuclear Reactor Regulation, the Nuclear Regulatory Commission maintains a knowledge of radioactive releases from licensed nuclear reactors to ensure that they are within regulatory requirements. This report summarizes data from the licensed nuclear power plants that were declared by the utilities to be in commercial operation as of December 31, 1993. Data are included for several licensed facilities which are permanently or indefinitely shut down (Browns Ferry 1 & 3, Brunswick 1, Dresden 1, Fort St. Vrain, Humboldt Bay, Indian Point 1, LaCrosse, Rancho Seco 1, San Onofre 1, Three Mile Island 2, Trojan 1, Yankee Rowe 1) and Shoreham which was never in commercial operation.

1.3 Source of Data

The information included in this report was obtained from data reported by the licensees. Individual licensee reports are available in the NRC Public Document Room, Gelman Building, 2120 L Street, Washington, D.C. 20555 and in local Public Document Rooms located near each licensed facility. Licensee reports varied in the format and extent of information provided.

Data from prior years used in the comparison tables were obtained from the previous annual summaries.

2.0 Tabulated Data

2.1 Airborne and Liquid Effluents

Tables 1 through 4 list for each reactor, the measured quantities of total noble gases and of I-131 and particulates (with half lives greater than 8 days) released in effluents to the atmosphere during each of the years 1974 through 1993. Tables 5 and 6 list the total measured quantities of tritium released in liquid effluents in each of the years. Tables 7 and 8 list the mixed fission and activation products not including noble gases, tritium and alpha released in liquid effluents in each of the years.

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[1] Previous reports in this series are listed on page ii and iii.

Table 6

Liquid Effluents Comparison By Year

Tritium (Curies)

Pressurized Water Reactors

Facility	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
Arkansas One 1	2.56E+01	4.60E+02	2.12E+02	2.45E+02	2.94E+02	1.63E+02	2.12E+02	4.42E+02	2.06E+02	1.09E+02
Arkansas One 2						5.27E+01	2.89E+02	2.44E+02	1.29E+02	2.28E+02
Beaver Valley 1&2			8.60E+00	1.08E+02	3.49E+02	9.59E+01	3.98E+01	1.40E+02	1.84E+02	4.60E+02
Braidwood 1										
Braidwood 2										
Byron 1&2										
Callaway 1										
Calvert Cliffs 1&2		2.63E+02	2.74E+02	5.75E+02	4.56E+02	5.14E+02	4.91E+02	1.00E+03	4.35E+02	7.56E+02
Catawba 1										
Catawba 2										
Comanche Peak 1										
Donald C. Cook 1&2		5.64E+01	1.92E+02	2.86E+02	6.24E+02	1.22E+03	7.82E+02	9.15E+02	1.23E+03	8.85E+02
Crystal River 3				1.66E+02	1.54E+02	1.65E+02	1.95E+02	2.71E+02	1.82E+02	1.99E+02
Davis-Besse 1				9.01E+00	2.15E+02	2.45E+02	1.08E+02	1.57E+02	5.68E+01	1.14E+02
Diablo Canyon 1&2										
Joseph M. Farley 1					5.91E+01	9.40E+01	5.70E+02	1.65E+02	3.37E+02	4.12E+02
Joseph M. Farley 2								6.34E+02	3.59E+02	3.17E+02
Fort Calhoun 1.	1.24E+02	1.11E+02	1.22E+02	1.57E+02	1.50E+02	2.58E+02	5.44E+01	2.42E+02	3.08E+02	1.53E+02
R. E. Ginna	1.95E+02	2.60E+02	2.42E+02	1.19E+02	2.42E+02	2.40E+02	1.60E+02	2.40E+02	3.08E+02	3.50E+02
Haddam Neck	2.24E+03	5.67E+03	4.85E+03	6.67E+03	3.94E+03	3.55E+03	3.29E+03	5.29E+03	4.05E+03	3.90E+03
Harris 1										
Indian Point 1&2	4.79E+01	7.94E+01	3.32E+02	3.71E+02	5.12E+02	3.75E+02	2.76E+02	2.41E+02	1.72E+02	3.43E+02
Indian Point 3			Shown With	Other Unit	2.56E+02	1.15E+02	4.27E+02	6.42E+02	1.94E+02	3.19E+01
Kewaunee	9.24E+01	2.77E+02	1.80E+02	2.95E+02	2.96E+02	2.49E+02	2.33E+02	2.51E+02	3.18E+02	2.92E+02
Maine Yankee	2.19E+02	1.77E+02	3.67E+02	1.53E+02	3.15E+02	2.02E+02	2.18E+02	2.16E+02	1.85E+02	2.87E+02
McGuire 1								6.25E+00	1.60E+02	1.49E+02
McGuire 2										1.49E+02
Millstone 2		7.60E+00	2.77E+02	2.11E+02	2.01E+02	2.54E+02	2.68E+02	3.71E+02	2.91E+02	1.21E+02
Millstone 3										
North Anna 1&2					2.82E+02	3.13E+02	4.03E+02	1.28E+03	5.71E+02	1.61E+03
Oconee 1,2 & 3	3.50E+02	3.55E+03	2.19E+03	1.92E+03	1.17E+03	8.94E+02	7.12E+02	5.07E+02	3.54E+02	1.28E+03
Palisades	8.10E+00	4.16E+01	9.63E+00	5.58E+01	1.01E+02	1.26E+02	7.47E+01	2.78E+02	1.79E+02	2.25E+02
Palo Verde 1										
Palo Verde 2										
Palo Verde 3										
Point Beach 1&2	8.33E+02	8.85E+02	6.94E+02	9.99E+02	1.29E+03	8.92E+02	7.61E+02	6.52E+02	5.03E+02	5.39E+02
Prairie Island 1&2	1.42E+02	4.54E-01	1.00E-01	1.35E+03	5.51E+02	6.25E+02	5.43E+02	5.62E+02	6.00E+02	5.20E+02
Rancho Seco 1		1.32E+02	N/D	8.55E-02	N/D	N/D	1.47E-02	8.35E+01	6.46E+01	7.43E+01
H. B. Robinson 2	4.49E+02	6.24E+02	9.80E+02	6.85E+02	4.73E+02	4.29E+02	1.89E+02	1.86E+02	9.51E+01	2.40E+02
Salem 1			4.00E-02	2.96E+02	4.46E+02	7.26E+02	N/D	4.93E+02	7.22E+02	2.08E+02
Salem 2							N/R	8.42E+02	5.25E+02	2.23E+02
San Onofre 1	3.81E+03	4.00E+03	3.39E+03	1.79E+03	2.50E+03	2.32E+03	1.03E+03	2.97E+02	5.45E+02	1.57E+01
San Onofre 2-3									8.92E+00	2.38E+02
Seabrook 1										
Sequoyah 1&2							3.23E-01	7.65E+01	9.34E+02	7.35E+02
South Texas 1										
South Texas 2										
St. Lucie 1			1.33E+01	2.42E+02	1.23E+02	1.28E+02	2.72E+02	3.25E+02	3.21E+02	3.46E+02
St. Lucie 2										3.77E+01
Summer 1									3.19E-01	2.27E+02
Surry 1&2	2.45E+02	4.42E+02	7.82E+02	4.08E+02	7.47E+02	3.57E+02	3.85E+02	5.31E+02	9.10E+02	7.17E+02
Three Mile Island 1	1.30E+02	4.63E+02	1.59E+02	1.92E+02	1.55E+02	5.59E+01	3.26E+01	7.11E+00	3.91E+00	3.09E+00
Three Mile Island 2					3.83E+01	7.81E+01	6.10E-04	5.06E-02	7.20E-02	3.75E-04
TMI 2/Epicor							N/D	N/D	N/D	N/D
Trojan			3.60E+01	3.11E+02	1.59E+02	6.80E+01	1.24E+02	1.03E+02	2.00E+02	2.34E+02

N/R = Not Reported

N/D = Not Detectable

Table 8

Liquid Effluents Comparison By Year

Tritium (Curies)

Pressurized Water Reactors

Facility	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993
Arkansas One 1	3.05E+02	3.27E+02	2.12E+02	1.50E+02	2.50E+02	3.81E+02	2.67E+02	5.13E+02	5.06E+02	4.51E+02
Arkansas One 2	3.09E+02	2.41E+02	2.30E+02	3.52E+02	2.44E+02	4.40E+02	5.22E+02	9.40E+02	2.98E+02	3.08E+02
Beaver Valley 1&2	4.12E+02	1.50E+02	2.06E+02	5.72E+02	4.09E+02	6.21E+02	4.91E+02	4.85E+02	4.65E+02	5.53E+02
Braidwood 1				4.12E+01	2.74E+02	5.58E+02	6.50E+02	3.43E+02	9.58E+02	8.05E+02
Braidwood 2					2.44E+02	5.58E+02	6.50E+02	3.42E+02	9.58E+02	8.05E+02
Byron 1&2		2.61E+02	6.70E+01	4.10E+02	1.01E+03	1.29E+03	9.98E+02	1.43E+03	1.58E+03	2.06E+03
Callaway 1	2.90E+01	5.88E+02	4.25E+02	4.48E+02	8.93E+02	6.09E+02	1.02E+03	1.22E+03	5.92E+02	1.41E+03
Calvert Cliffs 1&2	7.87E+02	4.83E+02	7.25E+02	7.38E+02	8.24E+02	2.36E+02	7.29E+01	1.02E+03	1.77E+03	6.36E+02
Catawba 1		1.75E+02	1.18E+02	3.64E+02	3.53E+02	4.45E+02	2.97E+02	3.23E+02	3.86E+02	4.13E+02
Catawba 2			1.18E+02	3.64E+02	3.53E+02	4.45E+02	2.97E+02	3.23E+02	3.86E+02	4.13E+02
Comanche Peak 1							1.87E+02	4.60E+02	8.11E+02	5.04E+02
Donald C. Cook 1&2	1.37E+03	1.14E+03	6.95E+02	1.97E+03	1.10E+03	8.74E+02	1.56E+03	1.55E+03	4.23E+02	6.01E+02
Crystal River 3	4.20E+02	1.76E+02	1.73E+02	3.56E+02	5.11E+02	3.44E+02	5.10E+02	4.49E+02	3.64E+02	5.89E+02
Davis-Besse 1	1.22E+02	6.74E+01	2.09E+01	2.46E+02	3.50E+01	2.39E+02	1.27E+02	3.26E+02	3.80E+02	1.81E+02
Diablo Canyon 1&2	1.07E+00	4.28E+02	6.98E+02	6.91E+02	4.29E+02	9.35E+02	9.68E+02	1.05E+03	1.22E+03	1.03E+03
Joseph M. Farley 1	4.23E+02	6.03E+02	7.14E+02	6.37E+02	5.16E+02	6.99E+02	7.25E+02	4.71E+02	8.18E+02	9.35E+02
Joseph M. Farley 2	3.56E+02	5.02E+02	6.22E+02	5.05E+02	7.53E+02	6.08E+02	6.72E+02	3.53E+02	7.90E+02	8.85E+02
Fort Calhoun 1	2.35E+02	1.67E+02	1.84E+02	2.23E+02	2.32E+02	2.28E+02	1.74E+02	1.77E+02	1.06E+02	2.39E+02
R. E. Ginna	4.59E+02	5.01E+02	3.57E+02	5.64E+02	3.47E+02	5.92E+02	3.21E+02	3.76E+02	2.13E+02	1.77E+02
Haddam Neck	3.66E+03	5.76E+03	2.58E+03	3.17E+03	1.18E+03	4.81E+03	9.89E+02	4.63E+03	8.63E+02	4.00E+03
Harris 1				2.48E+02	4.01E+02	4.58E+02	7.26E+02	2.92E+02	9.02E+02	5.55E+02
Indian Point 1&2	2.22E+02	3.51E+02	3.36E+02	5.63E+02	4.39E+02	5.60E+02	6.44E+02	5.45E+02	6.95E+02	2.89E+02
Indian Point 3	5.87E+02	3.40E+02	5.67E+02	3.40E+02	5.73E+02	3.51E+02	3.23E+02	5.38E+02	4.50E+02	2.95E+02
Kewaunee	4.40E+02	3.79E+02	2.94E+02	3.51E+02	3.32E+02	3.41E+02	3.79E+02	4.34E+02	2.90E+02	2.25E+02
Maine Yankee	1.72E+02	1.84E+02	3.50E+02	1.18E+02	2.91E+02	4.22E+02	2.43E+02	3.89E+02	2.17E+02	2.72E+02
McGuire 1	3.23E+02	4.02E+02	4.58E+02	4.92E+02	5.29E+02	4.23E+02	4.58E+02	4.39E+02	4.33E+02	3.88E+02
McGuire 2	3.23E+02	4.02E+02	4.58E+02	4.92E+02	5.29E+02	4.23E+02	4.58E+02	4.39E+02	4.33E+02	3.88E+02
Millstone 2	3.97E+02	1.66E+02	2.80E+02	2.86E+02	2.59E+02	3.66E+02	5.28E+02	2.56E+02	1.06E+02	3.29E+02
Millstone 3			5.41E+02	5.90E+02	5.47E+02	6.97E+02	7.74E+02	3.04E+02	5.96E+02	5.16E+02
North Anna 1&2	6.20E+02	1.48E+03	1.56E+03	8.36E+02	1.94E+03	1.40E+03	1.67E+03	1.16E+03	9.29E+02	6.93E+02
Oconee 1,2,& 3	1.28E+03	1.24E+03	1.34E+03	9.49E+02	7.10E+02	1.02E+03	9.92E+02	1.12E+03	9.98E+02	1.10E+03
Palisades	6.95E+01	4.29E+02	8.22E+01	1.19E+02	2.83E+02	8.06E+01	1.49E+02	5.52E+01	8.09E+01	2.10E+02
Palo Verde 1		N/D	N/D	N/D	N/D	N/D	N/D	N/D	N/D	N/D
Palo Verde 2			N/D	N/D	N/D	N/D	N/D	N/D	N/D	N/D
Palo Verde 3				N/D	N/D	N/D	N/D	N/D	N/D	N/D
Point Beach 1&2	2.10E+03	8.05E+02	8.11E+02	7.09E+02	3.57E+02	5.59E+02	8.72E+02	7.87E+02	4.16E+02	4.64E+02
Prairie Island 1&2	6.41E+02	6.96E+02	6.70E+02	4.49E+02	4.05E+02	4.64E+02	3.98E+02	5.58E+02	4.72E+02	4.80E+02
Rancho Seco 1	2.97E+02	9.00E+01	6.50E+01	1.83E+01	1.01E+02	7.29E+01	1.37E+01	9.84E-01	2.42E+01	7.44E+00
H. B. Robinson 2	1.34E+01	3.09E+02	3.42E+02	2.74E+02	5.36E+02	1.64E+02	3.53E+02	1.88E+02	3.94E+02	8.45E+02
Salem 1	3.30E+02	9.22E+02	4.10E+02	3.79E+02	6.35E+02	6.09E+02	3.53E+02	6.06E+02	2.45E+02	3.93E+02
Salem 2	3.08E+02	5.77E+02	4.38E+02	6.81E+02	3.68E+02	5.11E+02	3.03E+02	4.42E+02	2.25E+02	5.08E+02
San Onofre 1	3.39E+01	2.38E+03	4.53E+02	2.27E+03	1.53E+03	9.62E+02	1.42E+03	1.25E+03	3.00E+03	4.45E+03
San Onofre 2-3	4.55E+02	4.75E+02	7.41E+02	8.20E+02	6.43E+02	1.30E+03	9.27E+02	1.08E+03	9.69E+02	9.78E+02
Seabrook 1						1.33E-03	1.13E+02	3.86E+02	5.01E+02	5.83E+02
Sequoyah 1&2	1.82E+03	6.23E+02	2.46E+02	1.19E+03	2.01E+02	1.15E+03	8.53E+02	1.65E+03	1.44E+03	5.60E+02
South Texas 1					1.99E+02	3.17E+02	3.45E+02	6.21E+02	6.19E+02	1.13E+02
South Texas 2						2.72E+02	4.70E+02	4.69E+02	7.42E+02	1.13E+02
St. Lucie 1	2.21E+02	2.36E+02	2.78E+02	3.38E+02	2.75E+02	4.05E+02	2.84E+02	4.06E+02	4.00E+02	2.58E+02
St. Lucie 2	2.21E+02	3.64E+02	2.78E+02	3.38E+02	2.75E+02	4.05E+02	2.84E+02	4.06E+02	4.00E+02	2.51E+02
Summer 1	2.25E+02	3.11E+02	3.75E+02	7.36E+02	7.55E+02	6.85E+02	4.22E+02	8.13E+02	6.08E+02	4.79E+02
Surry 1&2	8.12E+02	7.50E+02	8.73E+02	8.15E+02	4.94E+02	4.29E+02	1.11E+03	9.13E+02	9.74E+02	1.22E+03
Three Mile Island 1	1.72E+00	9.08E+00	1.69E+02	1.97E+02	3.02E+02	3.73E+02	2.10E+02	3.59E+02	5.61E+02	3.76E+02
Three Mile Island 2	1.35E-04	2.22E-03	1.60E-03	1.48E-03	5.49E-03	9.76E-04	8.30E-04	6.19E-03	3.53E-03	1.59E-02
TMI 2/Epicor	N/D	--	--	--	--	--	--	--	--	--
Trojan	1.87E+02	2.65E+02	2.43E+02	1.75E+02	3.75E+02	3.18E+02	2.19E+02	1.69E+02	1.96E+02	1.22E+03

-- Included with Three Mile Island 2 total

N/D = Not Detectable

Table 8

Liquid Effluents Comparison By Year

Mixed Fission and Activation Products (Curies)

Pressurized Water Reactors

Facility	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
Arkansas One 1	6.50E+00	3.11E+00	1.31E+01	4.50E+00	6.05E+00	3.09E+00	3.42E+00	7.50E+00	5.80E+00	4.30E+00
Arkansas One 2						1.30E+00	4.13E+00	2.95E+00	5.90E+00	3.70E+00
Beaver Valley 1&2			1.70E-01	6.52E-01	2.63E-01	1.21E-01	1.04E-01	1.44E-01	1.47E-01	6.09E-02
Braidwood 1										
Braidwood 2										
Byron 1&2										
Callaway 1										
Calvert Cliffs 1&2		1.44E+00	1.18E+00	3.48E+00	6.13E+00	7.80E+00	4.53E+00	2.68E+00	5.26E+00	2.24E+00
Catawba 1										
Catawba 2										
Comanche Peak 1										
Donald C. Cook 1&2		2.60E-01	1.87E+00	1.52E+00	1.48E+00	2.58E+00	1.37E+00	1.86E+00	1.90E+00	6.83E-01
Crystal River 3				1.54E-02	2.96E-02	4.16E-01	1.46E-01	1.29E-01	1.07E-01	1.50E-01
Davis-Besse 1				2.60E-02	9.01E-02	4.28E-02	2.07E-01	7.92E-01	2.19E-01	5.39E-01
Diablo Canyon 1&2										
Joseph M. Farley 1					1.03E-01	5.86E-02	6.18E-02	1.31E-01	5.94E-02	5.75E-02
Joseph M. Farley 2								2.69E-02	2.90E-02	2.04E-02
Fort Calhoun 1	2.30E+00	3.60E-01	5.50E-01	3.63E-01	5.95E-01	2.45E-01	5.33E-01	1.75E-01	2.03E-01	1.44E-01
R. E. Ginna	1.00E-01	4.20E-01	6.90E-01	6.47E-02	6.07E-02	8.63E-02	1.96E-02	3.85E-02	6.17E-01	1.93E-01
Haddam Neck	2.20E+00	1.20E+00	1.30E-01	1.71E+00	9.50E-01	8.67E-01	2.76E-01	7.12E-01	6.93E-02	4.80E-01
Harris 1										
Indian Point 1&2	4.20E+00	4.93E+00	< 4.98E+00	3.02E+00	1.99E+00	1.94E+00	1.26E+00	5.67E+00	2.41E+00	4.02E+00
Indian Point 3			Shown With	Other Unit	1.03E+00	4.02E-01	2.90E+00	2.62E+00	5.46E-01	5.44E-01
Kewaunee	4.00E-01	7.20E-01	2.83E+00	1.26E+00	6.99E-01	8.94E-01	6.17E-01	8.15E-01	1.52E+00	5.43E-01
Maine Yankee	4.00E+00	3.21E+00	< 2.84E+00	4.42E-01	1.04E-01	4.63E-01	2.97E-01	4.36E-01	7.03E-01	1.99E-01
McGuire 1								3.94E-01	1.75E+00	1.87E+00
McGuire 2										1.87E+00
Millstone 2		2.00E-02	2.60E-01	1.56E+00	2.79E+00	4.87E+00	2.81E+00	4.18E+00	1.39E+01	7.81E+00
Millstone 3										
North Anna 1&2					2.68E-01	5.89E-01	1.05E+00	6.76E-01	1.32E+00	5.88E+00
Oconee 1,2,& 3	1.90E+00	5.05E+00	7.93E+00	3.62E+01	6.51E+00	9.24E-01	1.54E+00	1.75E+00	1.04E+00	1.43E+00
Palisades	5.90E+00	3.45E+00	4.40E-01	9.29E-02	9.65E-02	1.28E-01	8.73E-03	3.31E-02	1.27E-01	7.48E-02
Palo Verde 1										
Palo Verde 2										
Palo Verde 3										
Point Beach 1&2	2.00E-01	2.34E+00	3.24E+00	1.50E+00	6.86E-01	7.25E-01	6.29E-01	1.01E+00	2.95E+00	1.27E+00
Prairie Island 1&2	< 1.00E-01	4.50E-01	1.00E-01	1.33E-02	4.94E-03	9.00E-03	1.32E-02	9.12E-03	2.23E-03	3.16E-02
Rancho Seco 1		< 1.00E-02	N/D	N/D	N/D	N/D	3.78E-03	5.92E-01	2.16E-01	2.81E-01
H. B. Robinson 2	2.50E+00	4.50E-01	3.80E-01	3.29E-01	1.78E-01	2.99E-01	3.58E-01	1.84E+00	1.20E+00	8.23E-01
Salem 1			< 1.00E-02	2.88E+00	4.02E+00	3.98E+00	2.65E+00	2.80E+00	3.22E+00	2.97E+00
Salem 2							3.89E-01	1.51E+00	3.21E+00	2.85E+00
San Onofre 1	5.00E+00	1.22E+00	7.43E+00	9.84E+00	1.18E+01	1.10E+01	1.12E+01	3.64E+00	2.15E+00	1.22E+00
San Onofre 2-3									6.32E-01	2.79E+00
Seabrook 1										
Sequoyah 1&2							N/R	2.76E+00	9.82E+00	4.61E+00
South Texas 1										
South Texas 2										
St. Lucie 1			8.00E-02	5.80E+00	2.80E+00	2.67E+00	2.36E+00	2.46E+00	3.07E+00	2.99E+00
St. Lucie 2										4.37E-01
Summer 1									1.24E-04	1.47E+00
Surry 1&2	3.80E+00	9.27E+00	3.37E+01	6.55E+01	2.41E+00	2.53E+00	3.85E+00	6.11E+00	6.68E+00	1.45E+01
Three Mile Island 1	1.30E+00	7.00E-02	1.00E-01	1.94E-01	6.14E-01	4.91E-01	1.83E-01	8.69E-02	5.29E-02	8.12E-02
Three Mile Island 2					3.92E-01	3.31E-01	1.45E-05	2.22E-05	4.25E-05	9.03E-05
TMI 2/Epicor							N/D	N/D	N/D	N/D
Trojan			2.77E+00	4.19E+00	7.07E-01	5.55E-01	7.87E-01	9.94E-01	8.56E-01	3.10E-01

N/R = Not Reported
N/D = Not Detectable

Table 8

Liquid Effluents Comparison By Year

Mixed Fission and Activation Products (Curies)

Pressurized Water Reactors

Facility	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993
Arkansas One 1	4.10E+00	3.53E+00	5.09E+00	2.45E+00	3.73E+00	2.04E+00	2.36E+00	1.12E+00	3.59E+00	1.75E+00
Arkansas One 2	2.48E+00	4.36E+00	3.43E+00	1.85E+00	4.46E+00	2.65E+00	2.52E-01	2.73E+00	1.85E+00	4.77E-01
Beaver Valley 1&2	2.03E-01	1.13E-01	1.19E-01	6.69E-01	1.02E-01	5.45E-01	2.55E+00	3.14E-01	3.41E-01	3.96E-01
Braidwood 1				5.00E-02	8.57E+00	2.50E+00	2.13E+00	1.01E+01	5.23E-01	4.77E-01
Braidwood 2					3.04E+00	2.52E+00	2.13E+00	1.01E+01	5.23E-01	4.77E-01
Byron 1&2		1.63E+01	4.05E+00	2.48E+00	1.40E+00	6.35E-01	1.18E+00	6.70E-01	4.10E+00	1.26E+00
Callaway 1	1.07E-03	4.97E-03	3.83E-02	4.92E-01	7.74E-02	1.01E-02	3.86E-02	1.59E-02	4.54E-03	4.01E-02
Calvert Cliffs 1&2	1.64E+00	2.38E+00	1.79E+00	5.19E+00	2.64E+00	2.07E+00	1.42E+00	1.59E+00	1.44E+00	1.55E+00
Catawba 1		1.26E+00	3.82E-01	6.53E-01	5.42E-01	3.42E-01	9.78E-01	3.81E-01	4.65E-01	4.47E-01
Catawba 2			3.82E-01	6.53E-01	5.42E-01	3.42E-01	9.78E-01	3.81E-01	4.65E-01	4.47E-01
Comanche Peak 1							1.19E-02	1.57E-01	3.99E-01	4.18E-01
Donald C. Cook 1&2	1.19E+00	2.26E+00	3.34E-01	2.00E+00	4.44E-01	8.06E-01	1.61E+00	1.03E+00	1.12E+00	5.37E-01
Crystal River 3	2.34E-01	1.51E+00	8.12E-01	9.55E-01	2.31E-01	2.36E-01	6.19E-01	1.80E-01	1.63E+00	5.30E-01
Davis-Besse 1	1.89E-01	1.85E-01	6.15E-02	6.51E-02	1.68E-01	1.84E-01	1.41E-01	1.84E-01	1.10E-01	5.21E-02
Diablo Canyon 1&2	1.16E-02	3.20E+00	1.11E+01	2.86E+00	2.00E+00	1.61E+00	2.80E+00	8.47E-01	7.44E-01	9.85E-01
Joseph M. Farley 1	6.34E-02	6.72E-02	1.02E-01	5.09E-02	7.97E-02	7.31E-02	7.47E-02	2.14E-01	1.77E-01	7.60E-02
Joseph M. Farley 2	8.63E-02	3.77E-02	8.28E-02	4.63E-02	8.53E-02	7.34E-02	8.29E-02	1.90E-01	1.77E-01	1.12E-01
Fort Calhoun 1	2.91E+00	2.88E-01	8.37E-02	2.03E-01	3.08E-01	5.62E-01	8.05E-01 *	2.08E+00	5.90E-01	5.19E-01
R. E. Ginna	1.69E-01	5.22E-01	6.47E-02	5.88E-02	3.43E-02	8.12E-02	1.50E-01	1.52E-01	3.42E-01	1.37E-01
Haddam Neck	2.63E-01	8.44E-02	3.10E-01	4.26E-01	6.87E-01	3.90E-01	2.69E+00	7.43E-01	1.73E-01	8.36E-01
Harris 1				9.08E-01	8.04E-02	2.42E-01	7.31E-01	6.62E-01	3.14E-01	7.79E-02
Indian Point 1&2	2.67E+00	1.85E+00	3.61E+00	6.02E+00	2.84E+00	6.38E-01	1.06E+00	1.30E+00	1.53E+00	7.24E-01
Indian Point 3	1.26E+00	4.18E-01	1.95E-01	3.47E-01	3.22E-01	5.92E-01	3.09E-01	2.86E-01	2.13E-01	1.07E-01
Kewaunee	1.01E+00	1.35E+00	5.33E-01	1.29E+00	5.01E-01	1.22E+00	2.06E-01	2.35E-01	6.42E-02	1.20E-01
Maine Yankee	8.62E-02	3.11E-02	2.99E-01	8.81E-01	3.49E-01	1.83E-01	1.87E-01	4.13E-01	2.51E-01	1.62E-01
McGuire 1	1.51E+00	6.21E-01	7.73E-01	1.57E+00	2.57E+00	1.54E+00	2.00E+00	1.04E+00	3.27E-01	2.85E-01
McGuire 2	1.51E+00	6.21E-01	7.73E-01	1.57E+00	2.57E+00	1.54E+00	2.00E+00	1.04E+00	3.27E-01	2.85E-01
Millstone 2	3.55E+00	4.60E+00	4.49E+00	4.07E+00	8.89E+00	1.06E+01	8.76E+00	2.06E+00	2.14E+00	1.18E+00
Millstone 3			3.01E+00	5.40E+00	3.15E+00	5.94E+00	2.47E+00	2.99E+00	2.42E+00	2.24E+00
North Anna 1&2	4.51E+00	5.07E+00	9.41E-01	1.33E+00	4.32E-01	1.16E+00	6.75E-01	3.20E-01	4.98E-01	4.83E-01
Oconee 1,2,& 3	1.58E+00	4.16E+00	3.02E+00	2.90E+00	3.10E+00	3.82E+00	3.11E+00	1.40E+00	2.58E+00	4.70E-01
Palisades	3.68E-02	5.83E-02	1.40E-01	9.23E-02	3.43E-02	3.75E-03	7.75E-03	1.14E-02	3.88E-03	1.40E-02
Palo Verde 1		N/D	N/D	N/D	N/D	N/D	N/D	N/D	N/D	N/D
Palo Verde 2			N/D	N/D	N/D	N/D	N/D	N/D	N/D	N/D
Palo Verde 3				N/D	N/D	N/D	N/D	N/D	N/D	N/D
Point Beach 1&2	1.22E+01	1.90E+00	1.60E+01	7.55E-01	9.58E-02	5.58E-02	1.16E-02	5.89E-02	4.29E-01	2.32E-01
Prairie Island 1&2	1.91E-02	2.75E-02	6.01E-01	6.04E-02	2.55E-01	1.73E-01	1.30E-01	1.85E-01	6.66E-01	1.95E-01
Rancho Seco 1	6.33E-01	7.39E-03	1.45E-03	5.78E-04	5.79E-03	2.15E-03	2.08E-04	2.04E-04	4.83E-04	3.92E-04
H. B. Robinson 2	3.90E-01	9.41E-02	2.61E-01	7.36E-01	9.64E-01	2.82E-01	3.60E-01	2.36E-01	2.20E-01	5.47E-02
Salem 1	3.31E+00	2.88E+00	4.35E+00	3.33E+00	3.21E+00	3.11E+00	3.00E+00	3.35E+00	3.27E+00	3.21E+00
Salem 2	2.75E+00	2.80E+00	6.11E+00	4.07E+00	3.23E+00	3.56E+00	3.14E+00	2.31E+00	3.63E+00	3.65E+00
San Onofre 1	2.74E+00	7.79E+00	8.51E-01	8.42E-01	7.11E-01	6.87E-01	4.03E-01	4.22E-01	3.79E-01	1.14E+00
San Onofre 2-3	1.30E+01	1.12E+01	8.20E-01	5.37E-01	1.16E+00	9.19E-01	2.02E-01	9.94E-02	1.03E-01	2.94E-01
Seabrook 1						1.09E-04	2.21E-03	1.22E-01	1.19E-01	9.18E-02
Sequoyah 1&2	3.23E+00	1.45E+00	1.65E-01	4.66E-01	4.48E-01	3.54E-01	1.22E+00	1.48E+00	1.45E+00	1.52E+00
South Texas 1					2.24E-01	3.02E+00	7.09E+00	5.08E+00	2.12E+00	5.73E-01
South Texas 2						1.17E-02	5.72E+00	3.61E+00	1.74E+00	2.94E-01
St. Lucie 1	1.93E+00	2.72E+00	2.53E+00	5.95E-01	2.64E-01	2.56E-01	8.27E-01	3.98E-01	5.12E-01	7.55E-01
St. Lucie 2	1.93E+00	2.75E+00	2.43E+00	5.42E-01	2.59E-01	2.53E-01	7.68E-01	3.09E-01	5.12E-01	6.79E-01
Summer 1	4.54E+00	7.09E-01	3.26E-01	4.88E-01	7.55E-01	1.37E+00	3.56E-01	6.08E-01	2.23E-01	1.93E-01
Surry 1&2	9.73E+00	8.55E+00	8.77E+00	5.17E+00	2.41E+00	3.87E+00	4.60E+00	2.84E+00	8.27E-02	2.08E-02
Three Mile Island 1	3.41E-02	6.30E-03	1.41E-02	4.41E-02	4.68E-02	1.61E-02	2.36E-02	3.50E-02	2.60E-02	8.82E-02
Three Mile Island 2	6.46E-04	1.77E-04	1.87E-04	1.16E-04	1.12E-03	3.15E-04	1.77E-04	8.82E-05	1.22E-04	7.68E-04
TMI 2/Epicor	N/D	**	**	**	**	**	**	**	**	**
Trojan	3.49E-01	4.65E-01	2.64E-01	2.09E-01	2.01E-01	1.61E-01	1.44E-01	5.80E-02	8.95E-02	1.06E-01

* This number is a correction to that reported in the 1990 report

** Included with Three Mile Island 2 total

N/D = Not Detectable

Table 2

Airborne Effluents Comparison By Year

Fission and Activation Gases (Total Curies)

Pressurized Water Reactors

Facility	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983
Arkansas One 1	1.96E+02	1.03E+03	5.69E+03	1.39E+04	7.50E+03	8.51E+03	3.80E+04	3.73E+03	2.10E+03	9.83E+02
Arkansas One 2						4.53E+03	9.37E+03	4.35E+03	9.78E+03	1.34E+03
Beaver Valley 1&2			1.07E+00	4.73E+01	3.90E+02	1.75E+03	8.64E+01	8.06E+02	1.31E+02	1.98E+02
Braidwood 1										
Braidwood 2										
Byron 1&2										
Callaway 1										
Calvert Cliffs 1&2		7.72E+03	9.40E+03	2.23E+04	2.76E+04	1.02E+04	2.96E+03	2.18E+03	8.00E+03	9.75E+03
Catawba 1										
Catawba 2										
Comanche Peak 1										
Donald C. Cook 1&2		2.64E+00	9.75E+02	3.80E+03	4.85E+04	1.09E+04	3.76E+03	5.42E+03	3.88E+03	3.28E+02
Crystal River 3				3.35E+03	6.86E+03	7.26E+04	3.65E+04	3.96E+04	6.85E+03	3.38E+03
Davis-Besse 1				1.27E+03	2.10E+03	< 1.68E+03	< 3.35E+03	1.01E+03	5.35E+02	9.15E+02
Diablo Canyon 1&2										
Joseph M. Farley 1					3.53E+03	3.18E+03	1.92E+04	2.21E+02	3.81E+04	2.20E+04
Joseph M. Farley 2								2.60E+00	3.54E+03	8.47E+02
Fort Calhoun 1	3.03E+02	4.29E+02	1.94E+03	3.81E+03	1.36E+03	7.06E+02	2.97E+02	1.22E+03	3.46E+02	8.79E+02
R. E. Ginna	7.57E+02	1.04E+04	5.52E+03	3.20E+03	9.72E+02	7.62E+02	8.61E+02	5.46E+02	1.95E+03	7.12E+02
Haddam Neck	7.00E+00	4.80E+02	4.52E+02	3.12E+03	2.14E+03	5.53E+03	2.68E+03	1.83E+03	7.54E+02	2.76E+03
Harris 1										
Indian Point 1&2	5.58E+03	8.20E+03	1.16E+04	1.60E+04	1.41E+04	9.03E+03	9.38E+03	9.13E+03	7.27E+03	9.58E+03
Indian Point 3			Shown with	Other Unit	8.09E+02	2.47E+02	1.11E+03	6.57E+03	2.58E+03	5.60E+02
Kewaunee	3.35E+03	2.45E+03	1.40E+03	2.43E+03	4.44E+02	1.52E+02	1.22E+02	1.18E+02	1.66E+02	< 2.25E+02
** Maine Yankee	6.36E+03	4.09E+03	1.30E+03	3.57E+03	1.55E+03	2.09E+03	4.07E+03	3.28E+02	1.53E+03	5.07E+01
McGuire 1								1.58E-01	1.65E-03	1.60E+03
McGuire 2										1.60E+03
Millstone 2			1.57E+03	2.28E+03	7.64E+02	3.59E+02	1.33E+03	2.24E+03	9.09E+03	9.06E+03
Millstone 3										
North Anna 1&2					1.51E+04	6.28E+03	3.50E+03	5.30E+03	4.34E+03	2.22E+04
Oconee 1,2,& 3	1.94E+04	1.51E+04	4.39E+04	3.56E+04	4.33E+04	4.79E+04	1.92E+04	1.63E+04	2.41E+04	2.40E+04
Palisades	< 1.00E+00	2.61E+03	2.99E+01	5.99E+01	3.23E+02	6.84E+01	1.40E+02	3.00E+03	7.38E+03	3.00E+03
Palo Verde 1										
Palo Verde 2										
Palo Verde 3										
Point Beach 1&2	9.74E+03	4.45E+04	1.91E+03	1.13E+03	5.16E+02	9.68E+02	6.41E+02	6.11E+02	9.93E+02	7.68E+02
Prairie Island 1&2	3.62E+02	2.17E+03	1.74E+03	6.73E+02	1.26E+03	6.97E+02	2.60E+02	4.65E+01	5.47E+02	2.76E+02
Rancho Seco 1		1.18E+02	1.27E+02	2.00E+03	7.10E+03	8.81E+03	1.58E+03	1.37E+03	1.48E+03	6.89E+02
H. B. Robinson 2	2.31E+03	1.17E+03	6.40E+02	4.76E+02	8.84E+02	1.52E+03	5.82E+02	5.13E+02	1.75E+02	2.93E+02
Salem 1			< 1.00E-02	1.96E+01	1.02E+01	2.49E+02	7.82E+01	1.06E+03	2.34E+02	1.25E+02
Salem 2							7.74E+00	6.09E+02	1.11E+03	7.44E+02
San Onofre 1	1.78E+03	1.11E+03	4.16E+02	1.54E+02	1.81E+03	6.37E+02	1.05E+03	4.17E+02	8.61E+01	1.06E+01
San Onofre 2-3									6.40E+00	7.43E+03
Seabrook 1										
Sequoyah 1&2							3.01E+03	9.03E+03	5.74E+03	3.92E+03
South Texas 1										
South Texas 2										
St. Lucie 1			1.72E+03	2.54E+04	2.93E+04	1.54E+04	8.97E+03	2.30E+04	2.33E+04	2.16E+04
St. Lucie 2										1.25E+03
Summer 1									1.40E+02	3.88E+02
Surry 1&2	6.86E+03	8.04E+03	1.91E+04	1.90E+04	4.36E+03	1.78E+03	6.17E+03	1.41E+04	2.11E+04	5.49E+03
Three Mile Island 1	9.16E+02	3.63E+03	2.76E+03	1.66E+04	1.57E+04	2.24E+03	4.64E-03	5.81E-02	7.56E-03	2.01E+01
Three Mile Island 2					8.73E+00	9.97E+06	4.72E+04	2.88E+02	4.89E+02	1.73E+02
TMI 2/Epicor						-	2.16E+00	1.84E+02	4.26E+02	3.61E+01
* Trojan			7.66E+02	4.45E+03	3.26E+02	9.47E+02	4.10E+02	1.24E+03	9.02E+02	2.29E+02

* Changes to the entries for Trojan for 1976 - 1987 represent corrections which were reported and explained in the Trojan July-December 1990 Effluent and Waste Disposal Report.

** Changes to the entries for Maine Yankee for 1977 - 1988 represent corrections which were reported and explained in the Maine Yankee report "Revised Semiannual Effluent Release Report for 770131 - 901231" Docket Date 92/01/08.

Table 2

Airborne Effluents Comparison By Year

Fission and Activation Gases (Total Curies)

Pressurized Water Reactors Facility	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993
Arkansas One 1	2.90E+03	8.10E+03	1.71E+03	3.26E+02	1.24E+03	2.33E+03	7.00E+02	4.95E+02	8.93E+02	1.79E+01
Arkansas One 2	3.26E+03	8.91E+03	3.46E+03	2.06E+02	2.16E+03	2.76E+03	1.89E+02	1.59E+03	1.70E+03	5.21E+01
Beaver Valley 1&2	1.16E+03	3.92E+01	7.57E+01	2.25E+02	9.41E+01	1.57E+02	8.17E+01	1.49E+02	1.55E+02	5.56E+02
Braidwood 1				2.81E-01	4.19E+01	1.17E+03	1.42E+03	5.24E+03	7.71E+01	3.49E+02
Braidwood 2					3.82E+01	5.07E+02	1.02E+03	5.28E+03	1.56E+02	2.40E+03
Byron 1&2		2.79E+02	6.36E+02	1.30E+03	1.78E+03	8.16E+02	1.24E+03	1.04E+02	3.77E+02	1.22E+02
Callaway 1	2.00E+02	1.67E+03	5.19E+03	2.90E+03	6.89E+02	7.22E+02	9.02E+02	1.36E+02	4.01E+02	8.08E+02
Calvert Cliffs 1&2	3.83E+03	3.98E+03	7.65E+03	4.55E+03	5.70E+03	3.28E+03	6.72E+02	2.57E+03	5.87E+03	2.14E+02
Catawba 1		2.77E+02	1.36E+03	2.41E+03	1.56E+03	3.15E+02	5.33E+02	4.01E+02	4.28E+02	6.48E+02
Catawba 2			1.36E+03	2.41E+03	1.56E+03	3.15E+02	5.33E+02	4.01E+02	4.28E+02	6.48E+02
Comanche Peak 1							9.06E+02	5.89E+03	1.76E+03	1.92E+02
Donald C. Cook 1&2	3.50E+03	4.94E+03	3.29E+02	8.75E+02	2.58E+02	1.15E+02	1.88E+02	8.10E+01	2.04E+02	2.06E+03
Crystal River 3	1.96E+03	1.05E+03	2.76E+03	1.10E+03	3.41E+03	4.54E+03	7.31E+03	1.41E+03	7.86E+02	3.82E+01
Davis-Besse 1	5.02E+02	1.18E+02	5.09E-04	3.80E+02	1.09E+02	3.78E+02	1.09E+03	1.16E+03	3.62E+01	3.48E+02
Diablo Canyon 1&2	5.86E-02	5.72E+02	2.32E+03	7.14E+02	3.27E+02	3.35E+02	5.63E+01	4.62E+01	2.46E+00	2.14E+00
Joseph M. Farley 1	3.73E+03	1.70E+03	1.28E+03	1.30E+03	9.60E+02	9.92E+01	8.72E+01	1.09E+02	6.82E+02	1.94E+02
Joseph M. Farley 2	3.99E+03	6.63E+02	1.84E+03	7.22E+02	5.92E+02	1.60E+02	3.38E+01	3.56E+02	2.68E+01	2.61E+01
Fort Calhoun 1	1.52E+03	1.48E+03	5.68E+02	4.23E+02	7.85E+02	1.64E+02	4.59E+02	3.58E+02	1.51E+02	9.26E+00
R. E. Ginna	2.96E+02	4.06E+02	2.09E+02	1.77E+02	5.17E+01	5.11E+02	5.95E+02	5.14E+02	5.41E+02	1.40E+02
Haddam Neck	7.52E+03	2.76E+03	2.33E+03	3.58E+03	2.55E+03	1.71E+04	1.46E+03	6.11E+03	2.79E+00	2.08E+03
Harris 1				1.71E+03	2.25E+03	1.15E+03	5.96E+02	8.62E+02	1.36E+03	3.49E+02
Indian Point 1&2	3.78E+03	1.88E+03	2.05E+03	4.68E+03	2.27E+02	8.77E+01	2.23E+03	1.41E+03	5.25E+03	1.68E+03
Indian Point 3	1.88E+03	1.54E+03	1.93E+03	1.82E+03	3.10E+02	3.14E+02	6.26E+02	6.05E+01	2.15E+01	4.17E+01
Kewaunee	< 4.04E+01	< 4.97E+01	< 6.55E+01	< 3.19E+01	< 2.91E+01	6.52E+01	2.31E+00	1.81E+00	1.60E+00	3.67E-01
** Maine Yankee	1.54E+02	4.41E+02	1.07E+03	8.34E+02	9.19E+01	2.02E+01	9.46E+02	1.13E+03	4.01E+02	4.50E+01
McGuire 1	2.28E+03	1.93E+03	1.05E+03	2.04E+03	1.95E+03	7.19E+02	5.18E+02	4.49E+02	4.05E+02	4.84E+02
McGuire 2	2.28E+03	1.93E+03	1.05E+03	2.04E+03	1.95E+03	7.19E+02	5.18E+02	4.49E+02	4.05E+02	4.84E+02
Millstone 2	4.19E+03	4.00E+02	1.02E+02	3.97E+02	6.34E+02	2.46E+02	2.89E+03	3.89E+02	6.36E+02	1.32E+01
Millstone 3			2.39E+01	1.05E+02	8.44E+01	2.96E+02	2.11E+02	1.25E+02	1.13E+00	3.00E+01
North Anna 1&2	1.76E+04	8.05E+03	5.71E+03	1.05E+03	4.83E+02	1.44E+03	9.52E+02	2.24E+03	1.23E+03	2.51E+02
Oconee 1,2,& 3	2.28E+04	2.35E+04	2.43E+04	1.05E+04	2.59E+04	8.97E+03	8.84E+03	3.45E+03	3.29E+03	6.58E+02
Palisades	2.84E+01	3.68E+03	1.73E+02	1.75E+03	2.43E+03	1.52E+02	1.21E+02	6.26E+01	7.46E+01	9.29E+01
Palo Verde 1		2.53E+02	2.67E+03	1.27E+03	1.84E+03	6.41E+02	7.08E+02	2.91E+03	2.22E+03	5.79E+02
Palo Verde 2			1.97E+03	5.47E+03	2.97E+03	4.29E+02	6.76E+02	5.29E+02	2.01E+02	2.62E+02
Palo Verde 3				2.52E-02	1.36E+02	8.34E+02	1.20E+03	4.38E+02	4.35E+01	1.97E+02
Point Beach 1&2	9.30E+01	1.16E+02	2.78E+01	4.82E+01	8.08E+01	1.50E+01	8.03E+00	2.00E+01	5.06E+01	1.01E+01
Prairie Island 1&2	7.58E+01	4.59E+01	3.03E+01	8.77E-01	1.42E-01	1.73E+02	8.28E+01	5.60E+01	2.54E+01	3.68E+01
Rancho Seco 1	3.83E+03	4.67E+03	9.30E+01	2.16E-02	1.52E+03	2.00E+03	2.20E-01	N/D	6.93E-02	N/D
H. B. Robinson 2	4.90E+01	2.14E+03	6.59E+02	7.70E+02	1.04E+03	2.79E+01	7.20E+00	2.26E+00	7.59E+00	3.99E+02
Salem 1	1.95E+02	1.68E+03	1.39E+03	3.64E+03	5.29E+02	1.39E+03	3.13E+02	3.66E+02	6.75E+02	1.12E+03
Salem 2	1.81E+03	1.15E+03	8.56E+02	1.06E+03	1.18E+03	7.30E+01	1.49E+02	1.92E+02	2.68E+02	3.42E+02
San Onofre 1	8.62E+01	3.83E+03	4.11E+02	9.81E+02	2.99E+03	9.05E+02	1.80E+03	2.49E+03	4.12E+03	4.20E+02
San Onofre 2-3	4.00E+04	2.53E+04	8.25E+03	2.18E+04	5.12E+03	2.46E+03	1.16E+03	1.30E+03	1.41E+03	1.54E+03
Seabrook 1						N/D	1.07E+02	2.92E+01	9.13E-01	1.09E-01
Sequoyah 1&2	6.68E+03	4.57E+03	1.21E+00	N/D	2.25E+02	3.85E+03	6.07E+03	1.42E+03	2.07E+02	7.71E+01
South Texas 1					8.64E+02	4.45E+02	1.72E+02	8.55E+01	2.89E+02	2.42E+01
South Texas 2						1.16E+02	1.09E+02	4.67E+01	6.23E+02	1.79E+01
St. Lucie 1	3.53E+04	5.08E+04	3.33E+04	6.21E+03	1.42E+03	4.53E+03	6.19E+02	2.05E+03	3.30E+02	2.61E+02
St. Lucie 2	7.68E+03	9.55E+03	9.98E+03	8.60E+03	9.16E+03	2.22E+03	5.34E+02	4.90E+02	6.59E+02	8.62E+01
Summer 1	1.64E+01	1.40E+02	1.39E+01	6.34E+02	3.32E+02	1.82E+03	7.51E+02	4.34E+02	3.38E+02	2.43E+02
Surry 1&2	6.95E+03	2.07E+03	1.99E+03	3.08E+02	3.66E+02	1.37E+02	4.51E+02	3.54E+01	1.61E+01	4.15E+01
Three Mile Island 1	3.62E-01	1.08E+02	3.80E+03	7.89E+02	1.87E+03	2.10E+03	6.66E+02	1.22E+02	5.73E+02	2.40E+03
Three Mile Island 2	2.07E+02	N/D	2.80E-01	N/D	4.40E-01	N/D	N/D	4.18E-05	5.81E-05	4.41E-02
TMI 2/Epicor	3.99E+01	*+	*+	*+	*+	*+	*+	*+	*+	*+
* Trojan	8.98E+02	1.10E+03	9.42E+02	2.55E+02	4.25E+02	5.94E+02	2.06E+02	1.66E+02	2.07E+02	5.34E+01

* Changes to the entries for Trojan for 1976-1987 are corrections which were reported and explained in the Trojan July-December 1990 Effluent and Waste Disposal Report.

** Changes to the entries for Maine Yankee for 1977 - 1988 are corrections which were reported and explained in the Main Yankee report "Revised Semiannual Effluent and Release Reports for 770131 - 901231" Docket Date 92/01/03.

*+ Included with Three Mile Island 2 total

N/D = Not Detectable

EXHIBIT 2 B

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
PALISADES NUCLEAR PLANT

RADIOACTIVE EFFLUENT RELEASE REPORT
GASEOUS EFFLUENTS - SUMMATION OF RELEASES

January - December 2001

3 pages follow

TABLE HP 10.5-2

PALISADES PLANT RADIOACTIVE EFFLUENT REPORT

GASEOUS EFFLUENTS - SUMMATION OF RELEASES

January 1, 2001 to December 31, 2001

A. FISSION AND ACTIVATION GASES	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr	Est Total Error %
1. Total release	Ci	3.01E+00	2.92E+00	2.21E-02	0.00	5.6
2. Average release rate for period	μCi/sec	3.88E-01	3.72E-01	2.78E-03	0.00	
3. Percent of annual avg EC	%	1.67E-04	1.84E-04	1.16E-06	0.00	
B. IODINES						
1. Total Iodine *	Ci	5.52E-04	3.85E-04	5.69E-05	0.00	10.3
2. Average release rate for period	μCi/sec	7.09E-05	4.90E-05	7.16E-06	0.00	
3. Percent of annual avg EC	%	3.93E-05	3.52E-05	7.62E-06	0.00	
C. PARTICULATES						
1. Particulates with half-life > 8 days	Ci	2.80E-06	2.36E-05	7.72E-06	7.09E-06	18.0
2. Average release rate for period	μCi/sec	3.59E-07	3.01E-06	9.71E-07	8.92E-07	
3. Percent of annual avg EC	%	8.47E-06	1.92E-05	2.38E-05	2.12E-05	
4. Gross alpha radioactivity	Ci	5.17E-07	2.01E-06	3.72E-06	2.07E-06	
D. TRITIUM						
1. Total Release	Ci	4.81E+00	1.53E+01	5.27E+00	4.93E+00	
2. Average release rate for period	μCi/sec	6.18E-01	1.95E+00	6.63E-01	6.20E-01	
3. Percent of annual avg EC	%	1.31E-03	4.15E-03	1.41E-03	1.32E-03	
E.						
1. Beta Airdose at Site Boundary Due to Noble Gases (ODCM App A III.C)	mrads	2.13E-04	2.14E-04	1.56E-06	0.00	
2. Percent limit	%	2.13E-03	2.14E-03	1.56E-05	0.00	
3. Gamma Airdose at Site Boundary Due to Noble Gases (ODCM App A III.C)	mrads	7.25E-05	7.82E-05	5.26E-07	0.00	
4. Percent limit	%	1.45E-03	1.56E-03	1.05E-05	0.00	
F.						
1. Maximum Organ Dose to Public Based on Critical Receptors (ODCM App A III.D)	mrem	5.64E-03	1.62E-02	5.47E-03	4.98E-03	
2. Percent of limit	%	7.52E-02	2.16E-01	7.29E-02	6.64E-02	

*NOTE: Data is reported for I-131 and I-133 only.

TABLE HP 10.5-2
PALISADES PLANT RADIOACTIVE
EFFLUENT REPORT
GASEOUS EFFLUENTS
January 1, 2001 to December 31, 2001

1. FISSION GASES	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
Argon-41	Ci	7.33E-04	<LLD	<LLD	<LLD
Krypton-85	Ci	<LLD	<LLD	<LLD	<LLD
Krypton-85m	Ci	<LLD	<LLD	<LLD	<LLD
Krypton-87	Ci	2.31E-04	<LLD	<LLD	<LLD
Krypton-88	Ci	<LLD	<LLD	<LLD	<LLD
Xenon-131m	Ci	4.29E-03	6.11E-03	<LLD	<LLD
Xenon-133	Ci	3.00E+00	2.83E+00	2.21E-02	<LLD
Xenon-133m	Ci	<LLD	<LLD	<LLD	<LLD
Xenon-135	Ci	2.21E-03	8.53E-02	<LLD	<LLD
Xenon-135m	Ci	4.44E-03	9.41E-04	<LLD	<LLD
Xenon-138	Ci	<LLD	<LLD	<LLD	<LLD
Total for Period	Ci	3.01E+00	2.92E+00	2.21E-02	<LLD

2. IODINES					
Iodine-131	Ci	2.22E-04	2.28E-04	5.69E-05	<LLD
Iodine-132	Ci	<LLD	1.00E-03	<LLD	<LLD
Iodine-133	Ci	3.30E-04	1.57E-04	<LLD	<LLD
Iodine-134	Ci	<LLD	<LLD	<LLD	<LLD
Iodine-135	Ci	<LLD	<LLD	<LLD	<LLD
Total for Period	Ci	5.52E-04	1.39E-03	5.69E-05	<LLD

TABLE HP 10.5-2

PALISADES PLANT RADIOACTIVE

EFFLUENT REPORT

3. PARTICULATES*	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
Chromium-51	Ci	<LLD	1.80E-06	<LLD	<LLD
Cobalt-58	Ci	<LLD	1.47E-05	<LLD	<LLD
Iron-59	Ci	<LLD	<LLD	<LLD	<LLD
Cobalt-60	Ci	<LLD	4.16E-07	<LLD	1.82E-06
Zinc-65	Ci	<LLD	<LLD	<LLD	<LLD
Strontium-89	Ci	9.75E-07	8.25E-07	9.95E-07	7.95E-07
Strontium-90	Ci	4.80E-07	5.15E-07	4.95E-07	4.65E-07
Niobium-95	Ci	<LLD	3.92E-07	<LLD	<LLD
Zirconium-95	Ci	<LLD	5.57E-07	<LLD	<LLD
Ruthenium-103	Ci	<LLD	1.29E-07	<LLD	<LLD
Cesium-137	Ci	<LLD	7.92E-07	1.51E-06	<LLD
Cerium-144	Ci	<LLD	<LLD	<LLD	<LLD
Net unidentified beta	Ci	1.34E-06	3.51E-06	4.72E-06	4.01E-06
Total		2.80E-06	2.36E-05	7.72E-06	7.09E-06

*Particulates with half-lives > 8 days.

EXHIBIT 2 C

ATTACHMENT 3

NUCLEAR MANAGEMENT COMPANY, LLC
PALISADES NUCLEAR PLANT

RADIOACTIVE EFFLUENT RELEASE REPORT
LIQUID EFFLUENTS - SUMMATION OF RELEASES

January - December 2001

2 pages follow

TABLE HP 10.5-3

PALISADES PLANT RADIOACTIVE

EFFLUENT REPORT

LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES

January 1, 2001 to December 31, 2001

A. FISSION AND ACTIVATION PRODUCTS	Units	Ist Qtr	2nd Qtr	3rd Qtr	4th Qtr	Est Total Error %
1. Total release (not including tritium, gases, alpha)	Ci	2.81E-06	2.45E-04	0.000	3.68E-05	26.0
2. Average release rate for period	μCi/ml	7.35E-14	9.68E-12	N/A	3.79E-12	
3. Percent of EC	%	1.29E-05	6.02E-04	N/A	3.91E-04	
B. TRITIUM						
1. Total Release	Ci	2.73E+01	9.87E+01	6.80E-03	3.70E+01	4.01
2. Average diluted concentration during period	μCi/ml	7.15E-07	3.90E-06	6.36E-10	3.82E-06	
3. Percent of EC	%	7.15E-02	3.90E-01	6.36E-05	3.82E-01	
C. DISSOLVED AND ENTRAINED GASES						
1. Total Release	Ci	0.000	0.000	0.000	0.000	N/A
2. Average diluted concentration during period	μCi/ml	N/A	N/A	N/A	N/A	
3. Percent of EC	%	N/A	N/A	N/A	N/A	
D. GROSS ALPHA RADIOACTIVITY (Total Release)	Ci	7.45E-08	2.13E-06	0.000	1.65E-07	
E. VOLUME OF WASTE RELEASED (Prior to Dilution)	Liters	1.02E+05	3.78E+05	0.000	1.70E+05	
F. VOLUME OF DILUTION WATER USED DURING PERIOD	Liters	3.82E+10	2.53E+10	1.07E+10	9.68E+09	
G. MAXIMUM DOSE COMMITMENT - WHOLEBODY	mrem	3.81E-05	2.55E-04	3.35E-08	3.89E-04	
Percent of ODCM App A III.H limit	%	2.54E-03	1.70E-02	2.23E-06	2.59E-02	
H. MAXIMUM DOSE COMMITMENT - ORGAN	mrem	3.70E-05	2.15E-04	3.35E-08	4.80E-04	
Percent of ODCM App A III.H limit	%	7.40E-04	4.30E-03	6.70E-07	9.60E-03	

TABLE HP 10.5-3
PALISADES PLANT RADIOACTIVE
EFFLUENT REPORT

LIQUID EFFLUENTS
January 1, 2001 to December 31, 2001

1. NUCLIDES RELEASED	Units	Ist Qtr	2nd Qtr	3rd Qtr	4th Qtr
Manganese-54	Ci	<LLD	<LLD	<LLD	<LLD
Cobalt-58	Ci	<LLD	2.23E-05	<LLD	<LLD
Iron-59	Ci	<LLD	<LLD	<LLD	<LLD
Cobalt-60	Ci	<LLD	3.44E-05	<LLD	<LLD
Zirconium-95	Ci	<LLD	<LLD	<LLD	<LLD
Strontium-89	Ci	3.57E-07	1.02E-06	<LLD	5.44E-07
Strontium-90	Ci	2.45E-06	1.97E-06	<LLD	1.62E-06
Silver-110m	Ci	<LLD	1.28E-04	<LLD	<LLD
Iodine-131	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-134	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-137	Ci	<LLD	<LLD	<LLD	3.46E-05
Niobium-95	Ci	<LLD	<LLD	<LLD	<LLD
Silver-110m	Ci	<LLD	<LLD	<LLD	<LLD
Antimony-125	Ci	<LLD	<LLD	<LLD	<LLD
Net unidentified beta	Ci	<LLD	5.72E-05	<LLD	<LLD
Fission & Activation Product Total	Ci	2.81E-06	2.45E-04	0.00	3.68E-05
Tritium	Ci	2.73E+01	9.87E+01	6.80E-03	3.70E+01
Grand Total	Ci	2.73E+01	9.87E+01	6.80E-03	3.70E+01

EXHIBIT 2 D

ATTACHMENT 2

Nuclear Management Company, LLC
Palisades Plant
Docket 50-255

RADIOACTIVE EFFLUENT RELEASE REPORT
GASEOUS EFFLUENTS - SUMMATION OF RELEASES

January - December 2002

3 Pages Follow

TABLE HP 10.5-2
PALISADES PLANT RADIOACTIVE
EFFLUENT REPORT

GASEOUS EFFLUENTS - SUMMATION OF RELEASES
January 1, 2002 to December 31, 2002

A. FISSION & ACTIVATION GASES	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr	Est Total Error %
1. TOTAL RELEASE	Ci	5.01E-01	3.20E+00	1.65E+00	3.26E+01	
2. Average release rate for Period	uCi/sec	6.44E-02	4.07E-01	2.08E-01	4.10E+00	8.41
3. Percent of annual ave EC	%	2.90E-05	1.71E-04	8.49E-05	1.71E-03	
B. IODINES						
1. Total Iodine *	Ci	2.61E-04	4.55E-04	6.81E-04	1.20E-03	
2. Average release rate for Period	uCi/sec	3.35E-05	5.79E-05	8.57E-05	1.50E-04	9.88
3. Percent of annual ave EC	%	1.60E-05	2.98E-05	4.31E-05	1.31E-04	
C. PARTICULATES						
1. Particulates with half-life > than 8 days	Ci	1.94E-04	4.88E-06	2.82E-06	8.32E-07	
2. Average release rate for Period	uCi/sec	2.50E-05	6.21E-07	3.55E-07	1.05E-07	15.97
3. Percent of annual ave EC	%	6.00E-05	2.22E-05	1.27E-05	3.75E-06	
4. Gross Alpha radioactivity	Ci	8.23E-07	1.75E-06	1.42E-06	7.17E-07	
D. TRITIUM						
1. Total release	Ci	4.87E+00	4.73E+00	4.79E+00	5.23E+00	
2. Average release rate for Period	uCi/sec	6.26E-01	6.02E-01	6.03E-01	6.58E-01	
3. Percent of annual ave EC	%	1.33E-03	1.28E-03	1.28E-03	1.40E-03	
E. SITE BOUNDARY DOSE						
1. Beta Airdose at Site Boundary Due to Noble Gases 2. (ODCM App A III.C)	mrads	3.55E-05	2.32E-04	1.38E-04	2.33E-03	
2. Percent limit	%	3.55E-04	2.32E-03	1.38E-03	2.33E-02	
3. Gamma Airdose at Site Boundary Due to Noble Gases (ODCM App A III.C)	mrads	1.22E-05	7.45E-05	3.17E-05	7.68E-04	
4. Percent limit	%	3.10E-04	1.49E-03	6.34E-04	1.54E-02	
F. ORGAN DOSE						
1. Maximum Organ Dose to Public Based on Critical Receptors (ODCM App A III.D)	mrem	5.70E-03	5.36E-03	5.66E-03	1.94E-02	
2. Percent limit	%	7.60E-02	7.15E-02	7.55E-02	2.59E-01	

* NOTE: Data is reported for I-131 and I-133 only.

TABLE HP 10.5-2

PALISADES PLANT RADIOACTIVE EFFLUENT REPORT

GASEOUS EFFLUENTS

January 1, 2002 to December 31, 2002

1. FISSION GASES	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
Argon-41	Ci	<LLD	<LLD	<LLD	8.68E-04
Krypton-85	Ci	<LLD	9.20E-02	3.56E-01	3.93E-01
Krypton-85m	Ci	<LLD	<LLD	9.85E-05	2.72E-04
Xenon-131m	Ci	<LLD	1.27E-03	6.54E-03	5.08E-03
Xenon-133	Ci	4.97E-01	3.10E+00	1.28E+00	3.22E+01
Xenon-135	Ci	1.46E-03	2.19E-03	2.55E-03	2.65E-03
Xenon-135m	Ci	2.72E-03	3.72E-03	4.65E-03	5.36E-03
Total for Period	Ci	5.01E-01	3.20E+00	1.65E+00	3.26E+01

2. IODINES	Units	1st Qtr	2nd Qtr	3rrd Qtr	4th Qtr
Iodine-131	Ci	8.11E-05	1.61E-04	2.32E-04	9.30E-04
Iodine-132	Ci	<LLD	<LLD	<LLD	<LLD
Iodine-133	Ci	1.80E-04	2.94E-04	4.49E-04	2.65E-04
Iodine-134	Ci	<LLD	<LLD	<LLD	<LLD
Iodine-135	Ci	<LLD	<LLD	<LLD	<LLD
Total for Period	Ci	2.61E-04	4.55E-04	6.81E-04	1.20E-03

TABLE HP 10.5-2

PALISADES PLANT RADIOACTIVE
EFFLUENT REPORT

GASEOUS EFFLUENTS

January 1, 2002 to December 31, 2002

*PARTICULATES	Units	1st Qtr	2nd Qtr	3rrd Qtr	4th Qtr
Chromium-51	Ci	<LLD	<LLD	<LLD	<LLD
Cobalt-58	Ci	1.06E-04	<LLD	<LLD	<LLD
Cobalt-60	Ci	8.35E-05	<LLD	<LLD	<LLD
Cobalt-57	Ci	6.53E-07	<LLD	<LLD	<LLD
Zinc-65	Ci	<LLD	<LLD	<LLD	<LLD
Strontium-89	Ci	<LLD	<LLD	<LLD	<LLD
Strontium-90	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-134	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-137	Ci	<LLD	<LLD	<LLD	<LLD
Net unidentified beta	Ci	2.45E-06	4.88E-06	2.82E-06	8.32E-07
Total for Period	Ci	1.94E-04	4.88E-06	2.82E-06	8.32E-07

* Particulates with half-lives > 8 days

EXHIBIT 2 E

ATTACHMENT 3

Nuclear Management Company, LLC
Palisades Plant
Docket 50-255

RADIOACTIVE EFFLUENT RELEASE REPORT
LIQUID EFFLUENTS - SUMMATION OF RELEASES

January - December 2002

2 Pages Follow

TABLE HP 10.5-3

PALISADES PLANT RADIOACTIVE EFFLUENT REPORT

LIQUID EFFLUENTS - SUMMATION OF RELEASES

January 1, 2002 to December 31, 2002

A. FISSION & ACTIVATION PRODUCTS	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr	Est Total Error %
1. Total release (not including tritium, gases, alpha)	Ci	9.59E-05	0.000	1.83E-04	7.48E-07	17.31
2. Average release rate for Period	uCi/ml	2.45E-12	N/A	4.93E-12	1.89E-14	
3. Percent of EC	%	3.13E-04	N/A	2.90E-04	3.78E-06	
B. TRITIUM						4.01
1. Total Release	Ci	4.17E+01	4.09E-02	4.90E+01	7.27E+01	
2. Average diluted concentration during period	uCi/ml	1.06E-06	1.03E-09	1.32E-06	1.84E-06	
3. Percent of EC	%	1.06E-01	1.03E-04	1.32E-01	1.84E-01	
C. DISSOLVED & ENTRAINED GASES						N/A
1. Total Release	Ci	0.000	0.000	0.000	0.000	
2. Average diluted concentration during period	uCi/ml	N/A	N/A	N/A	N/A	
3. Percent of EC	%	N/A	N/A	N/A	N/A	
D. GROSS ALPHA RADIOACTIVITY (Total Release)	Ci	3.09E-08	0.000	2.74E-06	3.74E-07	
E. VOLUME OF WASTE RELEASED (Prior to Dilution)	Liters	1.93E+05	0.000	2.10E+05	2.20E+05	
F. VOLUME OF DILLUTION WATER USED DURING PERIOD	Liters	3.92E+10	3.97E+10	3.71E+10	3.95E+10	
G. MAXIMUM DOSE COMMITTMENT – WHOLE BODY	mrem	8.47E-05	5.36E-08	1.73E-04	9.72E-05	
Percent of ODCM App A III. H limit	%	5.65E-03	3.57E-06	1.15E-02	6.48E-03	
H. MAXIMUM DOSE COMMITMENT - ORGAN	mrem	1.13E-04	5.36E-08	2.55E-04	9.69E-05	
Percent of ODCM App A III. H limit	%	2.26E-03	1.07E-06	5.10E-03	1.94E-03	

TABLE HP 10.5-3

PALISADES PLANT RADIOACTIVE EFFLUENT REPORT

LIQUID EFFLUENTS

January 1, 2002 to December 31, 2002

NUCLIDES RELEASED	Units	1^{st} Qtr	2^{nd} Qtr	3rrd Qtr	4^{th} Qtr
Manganese-54	Ci	<LLD	<LLD	<LLD	<LLD
Cobalt-58	Ci	<LLD	<LLD	<LLD	<LLD
Cobalt-60	Ci	4.15E-05	<LLD	1.12E-04	<LLD
Zirconium-95	Ci	<LLD	<LLD	<LLD	<LLD
Silver-110m	Ci	<LLD	<LLD	<LLD	<LLD
Strontium-89	Ci	<LLD	<LLD	<LLD	<LLD
Strontium-90	Ci	6.95E-07	<LLD	7.35E-07	7.48E-07
Cesium-134	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-137	Ci	<LLD	<LLD	6.87E-05	<LLD
Iodine-131	Ci	<LLD	<LLD	<LLD	<LLD
Antimony-125	Ci	<LLD	<LLD	<LLD	<LLD
Net unidentified beta	Ci	5.37E-05	<LLD	<LLD	<LLD
Fission & Activation Products Total	Ci	9.59E-05	0.00	1.83E-04	7.48E-07
Tritium	Ci	4.17E+01	4.09E-02	4.90E+01	7.27E+01
Grand Total	Ci	4.17E+01	4.09E-02	4.90E+01	7.27E+01

EXHIBIT 2 F

ATTACHMENT 2

RADIOACTIVE EFFLUENT RELEASE REPORT
GASEOUS EFFLUENTS – SUMMATION OF RELEASES

JANUARY – DECEMBER 2003

3 Pages Follow

TABLE HP 10.5-2
PALISADES PLANT RADIOACTIVE EFFLUENT REPORT

GASEOUS EFFLUENTS - SUMMATION OF RELEASES
January 1, 2003 to December 31, 2003

A. FISSION & ACTIVATION GASES	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr	Est Total Error %
1. Total Release	Ci	6.07E+01	3.05E+00	4.96E-01	7.42E-01	
2. Average release rate for Period	uCi/sec	7.81E+00	3.88E-01	6.23E-02	9.33E-02	5.57
3. Percent of annual ave EC	%	3.27E-03	1.48E-04	2.82E-05	4.15E-05	
B. IODINES						
1. Total Iodine *	Ci	1.86E-03	8.78E-04	4.00E-04	3.53E-04	
2. Average release rate for Period	uCi/sec	2.39E-04	1.12E-04	5.04E-05	4.44E-05	8.46
3. Percent of annual ave EC	%	2.15E-04	1.02E-04	2.18E-05	1.90E-05	
C. PARTICULATES						
1. Particulates with half-life > than 8 days	Ci	8.28E-04	1.32E-04	1.42E-06	4.38E-07	
2. Average release rate for Period	uCi/sec	1.06E-04	1.68E-05	1.79E-07	5.51E-08	7.89
3. Percent of annual ave EC	%	5.28E-05	2.03E-05	6.34E-06	1.97E-06	
4. Gross ALPHA Radioactivity	Ci	4.20E-07	6.16E-07	5.86E-07	4.56E-07	
D. TRITIUM						
1. Total release	Ci	5.39E+00	6.43E+00	5.25E+00	5.16E+00	
2. Average release rate for Period	uCi/sec	6.93E-01	8.18E-01	6.60E-01	6.49E-01	
3. Percent of annual ave EC	%	1.47E-03	1.74E-03	1.40E-03	1.38E-03	
E. SITE BOUNDARY DOSE						
1. Beta Airdose at Site Boundary Due to Noble Gases (ODCM App A III.C)	mrads	4.36E-03	2.64E-04	3.78E-05	5.32E-05	
2. Percent limit	%	4.36E-02	2.64E-03	3.78E-04	5.32E-04	
3. Gamma Airdose at Site Boundary Due to Noble Gases (ODCM App A III.C)	mrads	1.44E-03	5.41E-05	1.12E-05	1.78E-05	
4. Percent limit	%	2.88E-02	1.08E-03	2.24E-04	3.56E-04	
F. ORGAN DOSE						
1. Maximum Organ Dose to Public Based on Critical Receptors (ODCM App A III.D)	mrem	3.09E-02	1.55E-02	5.66E-03	5.52E-03	
2. Percent limit	%	4.12E-01	2.07E-01	7.55E-02	7.36E-02	

* **NOTE**: Data is reported for I-131 and I-133 only.

TABLE HP 10.5-2

PALISADES PLANT RADIOACTIVE EFFLUENT REPORT

GASEOUS EFFLUENTS

January 1, 2003 to December 31, 2003

1. FISSION GASES	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
Krypton-85	Ci	8.33E-01	7.75E-01	4.34E-02	1.26E-02
Krypton-87	Ci	<LLD	<LLD	2.50E-04	<LLD
Krypton-88	Ci	<LLD	<LLD	<LLD	<LLD
Xenon-131m	Ci	1.14E-02	9.13E-02	<LLD	<LLD
Xenon-133	Ci	5.98E+01	2.18E+00	4.47E-01	7.24E-01
Xenon-133m	Ci	<LLD	5.84E-04	<LLD	<LLD
Xenon-135	Ci	9.55E-02	1.81E-03	1.54E-03	1.65E-03
Xenon-135m	Ci	2.94E-03	2.24E-03	3.32E-03	3.63E-03
Xenon-138	Ci	<LLD	<LLD	<LLD	<LLD
Total for Period	Ci	6.07E+01	3.05E+00	4.96E-01	7.42E-01

2. IODINES	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
Iodine-131	Ci	1.51E-03	7.23E-04	1.03E-04	8.95E-05
Iodine-132	Ci	1.06E-05	1.34E-04	<LLD	<LLD
Iodine-133	Ci	3.46E-04	1.55E-04	2.97E-04	2.63E-04
Iodine-135	Ci	4.65E-06	<LLD	<LLD	<LLD
Total for Period	Ci	1.87E-03	1.01E-03	4.00E-04	3.53E-04

TABLE HP 10.5-2

PALISADES PLANT RADIOACTIVE EFFLUENT REPORT

GASEOUS EFFLUENTS

January 1, 2003 to December 31, 2003

3. PARTICULATES*	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
Chromium-51	Ci	2.86E-04	<LLD	<LLD	<LLD
Manganese-54	Ci	1.88E-05	<LLD	<LLD	<LLD
Cobalt-58	Ci	3.62E-04	1.17E-04	<LLD	<LLD
Cobalt-60	Ci	4.14E-05	9.38E-06	<LLD	<LLD
Niobium-95	Ci	5.47E-05	1.80E-06	<LLD	<LLD
Ruthenium-103	Ci	8.88E-06	<LLD	<LLD	<LLD
Strontium-89	Ci	<LLD	<LLD	<LLD	<LLD
Strontium-90	Ci	<LLD	2.40E-07	6.70E-07	3.75E-07
Cesium-134	Ci	8.49E-08	<LLD	<LLD	<LLD
Cesium-137	Ci	9.57E-07	<LLD	<LLD	<LLD
Zirconium-95	Ci	5.17E-05	1.11E-06	<LLD	<LLD
Cobalt-57	Ci	<LLD	1.46E-07	<LLD	<LLD
Net unidentified beta	Ci	3.33E-06	2.42E-06	7.50E-07	6.30E-08
Total for Period	Ci	8.28E-04	1.32E-04	1.42E-06	4.38E-07

* Particulates with half-lives > 8 days

EXHIBIT 2 G

ATTACHMENT 3

RADIOACTIVE EFFLUENT RELEASE REPORT
LIQUID EFFLUENTS – SUMMATION OF RELEASES

JANUARY – DECEMBER 2003

2 Pages Follow

TABLE HP 10.5-3

PALISADES PLANT RADIOACTIVE
EFFLUENT REPORT

LIQUID EFFLUENTS - SUMMATION OF RELEASES
January 1, 2003 to December 31, 2003

A. FISSION & ACTIVATION PRODUCTS	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr	Est Total Error %
1. Total release (not including tritium, gases, alpha)	Ci	2.09E-04	5.40E-04	0.000	1.45E-03	
2. Average release rate for Period	uCi/ml	8.25E-12	1.60E-11	N/A	3.75E-11	14.16
3. Percent of EC	%	7.37E-04	8.89E-04	N/A	2.42E-03	
B. TRITIUM						
1. Total Release	Ci	5.87E+01	9.21E+01	5.57E-02	4.67E+01	
2. Average diluted concentration during period	uCi/ml	2.32E-06	2.72E-06	1.39E-09	1.21E-06	
3. Percent of EC	%	2.32E-01	2.72E-01	1.39E-04	1.21E-01	4.01
C. DISSOLVED & ENTRAINED GASES						
1. Total Release	Ci	0.000	0.000	0.000	0.000	
2. Average diluted concentration during period	uCi/ml	N/A	N/A	N/A	N/A	N/A
3. Percent of EC	%	N/A	N/A	N/A	N/A	
D. GROSS ALPHA RADIOACTIVITY (Total Release)	Ci	<LLD	8.30E-07	0.000	<LLD	
E. VOLUME OF WASTE RELEASED (Prior to Dillution)	Liters	1.79E+05	3.49E+05	0.000	1.79E+05	
F. VOLUME OF DILLUTION WATER USED DURING PERIOD	Liters	2.53E+10	3.38E+10	4.01E+10	3.86E+10	
G. MAXIMUM DOSE COMMITTMENT – WHOLE BODY	mrem	2.49E-04	3.92E-04	7.31E-08	6.49E-04	
Percent of ODCM App A III. H limit	%	1.66E-02	2.61E-02	4.87E-06	4.33E-02	
H. MAXIMUM DOSE COMMITMENT - ORGAN	mrem	3.05E-04	5.02E-04	7.31E-08	1.04E-03	
Percent of ODCM App A III. H limit	%	6.10E-03	1.00E-02	1.46E-06	2.08E-02	

TABLE HP 10.5-3

PALISADES PLANT RADIOACTIVE EFFLUENT REPORT

LIQUID EFFLUENTS
January 1, 2003 to December 31, 2003

NUCLIDES RELEASED	Units	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
Manganese-54	Ci	<LLD	2.35E-05	<LLD	<LLD
Cobalt-58	Ci	<LLD	1.20E-04	<LLD	3.31E-04
Cobalt-60	Ci	1.16E-04	1.03E-04	<LLD	6.38E-04
Zirconium-95	Ci	<LLD	<LLD	<LLD	<LLD
Silver-110m	Ci	<LLD	<LLD	<LLD	<LLD
Strontium-89	Ci	<LLD	<LLD	<LLD	<LLD
Strontium-90	Ci	2.86E-06	1.03E-05	<LLD	2.86E-05
Cesium-134	Ci	<LLD	<LLD	<LLD	7.51E-05
Cesium-137	Ci	3.76E-05	1.37E-04	<LLD	1.84E-04
Iodine-131	Ci	<LLD	<LLD	<LLD	<LLD
Antimony-125	Ci	<LLD	9.76E-05	<LLD	<LLD
Net unidentified beta	Ci	5.22E-05	4.91E-05	<LLD	1.91E-04
Fission & Activation Products Total	Ci	2.09E-04	5.40E-04	0.0	1.45E-03
Tritium	Ci	5.87E+01	9.21E+01	5.57E-02	4.67E+01
Grand Total	Ci	5.87E+01	9.21E+01	5.57E-02	4.67E+01

EXHIBIT 2 H

Inventory of Radionuclides for the Great Lakes

Nuclear
Task
Force

International
Joint
Commission

December 1997

International Joint Commission
United States and Canada

United States Government Code of Federal Regulations (CFR)
10 CFR 835 (1995) *Occupational Radiation Protection*

40 CFR 51 (1993) *Requirements for Preparation, Adoption and Submittal of Implementation Plans*, as amended

UNSCEAR (1977)
Sources, Effects and Risks of Radioactivity; Report to the UN General Assembly with Appendices; United Nations, New York.

UNSCEAR (1982)
Sources, Effects and Risks of Ionizing Radiation; Report to the UN General Assembly with Appendices; United Nations, New York.

UNSCEAR (1988)
Sources, Effects and Risks of Ionizing Radiation; Report to the UN General Assembly with Appendices; United Nations, New York.

UNSCEAR (1993)
Sources, Effects and Risks of Ionizing Radiation; Report to the UN General Assembly with Appendices.; United Nations, New York.

Wahlgren, M.A., Robbins, J.A., and Edgington, D.N. (1980)
Plutonium in the Great Lakes. In: *Transuranic Elements in the Environment*, W.C. Hanson (ed.); United States Department of Energy, Washington, D.C. pp. 659–683.

Yan, N.D., Mackie, G.L., and Boomer, D. (1989)
Chemical and biological effects correlates of metal levels in crustacean zooplankton from Canadian Shield Lakes: a multivariate analysis; *Sci. Total Environ.* 87/88: 419–458.

This generic letter supplement only requires information from the addressees under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). Therefore, the staff has not performed a backfit analysis. The information required will enable the NRC staff to determine whether licensees are complying with the requirements of

10 CFR 50.60, 10 CFR 50.61, Appendices G and H to 10 CFR Part 50 and any associated license conditions, and licensee commitments related to GL 88-11 and GL 92-01, Revision 1. The staff is not establishing a new position for such compliance in this generic letter supplement. Therefore, this generic letter supplement does not constitute a backfit and no documented evaluation or backfit analysis need be prepared.

Federal Register Notification

A notice of opportunity for public comment was not published in the *Federal Register* because the NRC needs to receive the responses to the generic letter in an expeditious manner. However, comments on the technical issue(s) addressed by this generic letter may be sent to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555.

Paperwork Reduction Act Statement

The information collections contained in this request are covered by the Office of Management and Budget clearance number 3150-0011, which expires July 31, 1997. The public reporting burden for this collection of information is estimated to average 600 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needs, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, D.C., 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

Compliance with the following request for information is voluntary. The information would assist the NRC in evaluating the cost of complying with this GL supplement.
(1) the licensee staff time and costs to perform requested record reviews and developing plans for inspections;
(2) the licensee staff time and costs to prepare the requested reports and documentation;
(3) the additional short-term costs incurred as a result of the inspection findings such as the cost of the corrective actions or the costs of down time; and
(4) an estimate of the additional long-term costs that will be incurred as a result of implementing commitments such as the estimated costs of conducting future inspections and repairs.

If you have any questions about this matter, please contact the technical contacts listed

relevant data;
(3) a determination of the need for use of the ratio procedure in accordance with the established Position 2.1 of Regulatory Guide 1.99, Revision 2, for those licensees that use surveillance data to provide a basis for the RPV integrity evaluation; and
(4) a written report providing any newly acquired data as specified above and (1) the results of any necessary revisions to the evaluation of RPV integrity in accordance with the requirements of 10 CFR 50.60, 10 CFR 50.61, Appendices G and H to 10 CFR Part 50, and any potential impact on the LTOP or P-T limits in the technical specifications or (2) a certification that previously submitted evaluations remain valid. Revised evaluations and certifications should include consideration of Position 2.1 of Regulatory Guide 1.99, Revision 2, as applicable, and any new data.

Required Response

All addressees are required to submit the following written responses providing the information described above:
(1) within 90 days from the date of this generic letter, a written response to part (1) of the information requirement specified above; and
(2) within 6 months from the date of this generic letter, a written response to parts (2), (3), and (4) of the information requirement above.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy to the appropriate regional administrator.

The NRC recognizes the potential difficulties (number and types of sources, age of records, proprietary data, etc.) that licensees may encounter while ascertaining whether they have all of the data pertinent to the evaluation of their RPVs. For this reason, 90 days is allowed for the initial response.

The information obtained from the licensees as a result of Revision 1 to GL 92-01 has been entered into a computerized reactor vessel integrity database (RVID), which will be made publicly available in the third quarter of 1995. The NRC intends to hold a public meeting on this GL supplement within 30 days of its issuance and a public workshop on RPV integrity, addressing the RVID and other RPV integrity issues, in the third quarter of 1995.

Related Generic Communications
(1) NRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity," March 6, 1992.
(2) NRC Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," July 12, 1988.

Backfit Discussion

NRC Staff Generic Evaluation of RPV Structural Integrity Data for PTS Events

The staff is assessing the generic implications of chemical composition variability with regard to the current methodology for ensuring protection against PTS events for PWRs. The staff considers that the larger variability observed in recent reviews could be applicable to other reactor vessels and may, therefore, reduce the margins of safety provided by the PTS screening criteria. The staff will evaluate this concern as part of its review of plant-specific evaluations and longer-term reassessment of the PTS rule.

To provide assurance that all PWRs will maintain adequate protection against PTS events while the PTS rule is being reassessed, the staff has assessed all of the PWR RPVs using generic values of chemistry and increased margin terms to account for potentially larger chemical composition variability. It should be noted that such analyses are considered conservative evaluations, that were performed to determine whether an immediate safety concern exists for this issue and whether there is adequate time to perform a more rigorous assessment of the issue. As stated in the previous section, based upon currently available information, the staff believes that the Palisades vessel will exceed the PTS screening criteria before any other PWR. However, because of the importance of RPV integrity and the potential impact of additional, unreviewed data on RPV evaluations, the staff believes that this issue needs to be resolved on an expedited basis.

Consideration of All Data Relevant to Reactor Pressure Vessel Integrity

As described previously, another result of recent reviews was that the staff became concerned that licensees might not necessarily have all of the data pertinent to the evaluation of the structural integrity of their RPVs. This is particularly true where the RPV fabricator holds, or has held, the applicable data to be proprietary in nature. Such data include, but are not limited to: chemical composition, heat treatment, plate and forging manufacturing process records, RPV fabrication records, all mechanical property data (tensile, impact, fracture toughness), and surveillance data. Sources of data that licensees should reexamine include material test reports from the steel producer, weld wire manufacturer, RPV fabricator, independent testing laboratories, and nuclear steam supply system (NSSS) vendor. Licensees are encouraged to work closely with their respective vessel owners groups and NSSS vendor groups to ensure that all sources of information pertinent to the analysis of the structural integrity of their RPVs have been considered. The information submitted in response to this generic letter should be considered to be public information.

Required Information

Addressees are required to provide the following information:
(1) a description of those actions taken or planned to locate all data relevant to the determination of RPV integrity, or an explanation of why the existing data base is considered complete as previously submitted;
(2) an assessment of any change in best-estimate chemistry based on consideration of all

information on the two key aspects of RPV structural integrity of primary concern to the NRC: PTS and USE. With respect to USE, licensees of all plants were able to demonstrate compliance with the Appendix G requirements either through consideration of applicable data or through equivalent margins analyses. With regard to PTS, only two plants (Beaver Valley 1 and Palisades) were projected to exceed the PTS screening criteria of 10 CFR 50.61 before the end of operating life (EOL). As stated previously, based on data and analyses submitted for GL 92-01, Revision 1, and other recent reviews (e.g, Ref. 2), the staff has determined that not all licensees were aware of all the information pertinent to the analysis of the structural integrity of their RPVs. In addition, recent reviews have indicated larger-than-expected variabilities in weld chemical composition, which have, in turn, highlighted the extreme sensitivity of RPV embrittlement estimates to small changes in the chemical composition of beltline materials.

Recent NRC Staff Evaluations of RPV Structural Integrity Data for PTS Events

The staff issued a safety evaluation report to the licensee for Palisades on the variability of reactor vessel weld properties for the Palisades reactor vessel on April 12, 1995 [Ref. 2]. The staff agreed with the licensee's best-estimate analysis of the chemical composition of the reactor vessel welds and concluded that continued operation through Cycle 14 (late 1999) was acceptable. As discussed previously, while performing the evaluation, the staff noted larger variability in the chemical composition of the welds compared to that assumed for the development of the PTS rule. The staff evaluated the implications of this larger variability on the PTS rule generic margins for the Palisades vessel using the same analytic methods as those used in formulating the rule. The staff has reviewed the other PWR vessels and, based upon currently available information, believes that the Palisades vessel will reach the PTS screening criteria by late 1999, before any other PWR.

On March 27 and 28, 1995, the staff reviewed the Asea Brown Boveri-Combustion Engineering proprietary RPV data-base. The most significant information reviewed concerned the Kewaunee RPV. The particular concern was the impact of data generated subsequent to the response to GL 92-01, Revision 1, on the plant's PTS evaluation. The staff met with the licensee for Kewaunee (April 13, 1995) to discuss issues related to consideration of all appropriate chemical composition data in addition to the applicable surveillance program data. In that meeting, the licensee presented its plant-specific surveillance program results and some new information related to the chemical composition variability in the RPV welds. Based upon this information, the licensee believes that the Kewaunee vessel will not exceed the PTS screening criteria before EOL. The staff has not completed its review of the new information on the Kewaunee vessel. However, based on the new vessel specific surveillance data, chemical composition data and the greater margin to the PTS screening criteria (300°F for the limiting Kewaunee circumferential weld compared to 270°F for the limiting Palisades axial weld), the staff believes that the Kewaunee vessel will not exceed the PTS screening criteria before the Palisades vessel. A key aspect of the Kewaunee review is the determination of the need for use of the ratio procedure in accordance with the established Position 2.1 of Regulatory Guide 1.99, Revision 2, by licensees using surveillance data.

2/17/94

EXHIBIT 3 A

Dear Dr. Selin,

I'm glad the Palisades resident got your ear about my concern over the lack of seismic requirements for the spent fuel dry cask storage pad and more importantly, the foundation material. I discovered this while I was investigating an allegation from Mary Sinclair on Palisades. I've discussed this issue with Fritz Sturz, one of the authors of Part 72 the licensing requirements for the casks. Initially, it was ok as is until I requested it in writing. Enclosed is their response to this issue. It's still not addressing the generic issue of lack of seismic requirements in the regulations. It seems that Subpart E, of Part 72, covers this point quite well for storage casks that are not pre-approved.

However, if you use NRC-approved casks under Subpart K, the regulations are silent about the foundation material or the pad. Actually, its the consequences that might occur from an earthquake that I'm concerned about. The casks can either fall into Lake Michigan or be buried in the loose sand because of liquefaction This event might be in the publics mind in view of what just happened in Southern California. It is apparent to me that NMSS doesn't realize the catastrophic consequences of their continued reliance on their current ideology. (emphasis added-MPs)

Please call.

Dr. Ross Landsman

RIII 708-829-9609

EXHIBIT 3 B

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE SECRETARY

In the Matter of Nuclear Management Company, LLC Palisades Nuclear Power Station Regarding Renewal of Facility Operating License No. DPR-20 for an Additional 20 Year Period	Docket No. 50-255

DECLARATION OF Dr. Ross Landsman,
Retired U.S. Nuclear Regulatory Commission
Nuclear Safety Engineer and Palisades Dry Cask Storage Inspector

Under the penalty of perjury, I, Ross Landsman, declare that the following statements are true and correct to the best of my knowledge and belief:

1. My name is Ross Landsman. I am a retired U.S. Nuclear Regulatory Commission Region III Nuclear Safety Engineer and Palisades Dry Cask Storage Inspector. I live at 9234 North Lowell, Skokie, Illinois.

2. NRC Region III, where I formerly worked before recently retiring, has requested assistance from NRC Headquarters division of Nuclear Reactor Regulation (NRR), in coordination with the division of Nuclear Materials Safety and Safeguards (NMSS) Spent (sic) Fuel Project Office (SFPO), in order to resolve questions involving the licensing basis for the Palisades Nuclear Power Plant and the appropriateness of the licensing basis to the seismic design of the Palisades ISFSI.

3. On August 4, 2004, on behalf of the NRC, I completed an inspection of design and operational activities associated with the newly constructed Palisades ISFSI pad. The results of this inspection were documented in NRC Inspection Report No. 07200007/2004-002 (DNMS). As a result, I identified two issues, characterized as violations by me in the draft report but changed to unresolved items (URI) by my boss in the final report to allow Palisades to go ahead and load fuel instead of stopping them even though it was not safe. These two issues were associated with the licensee's translation of the safe shutdown earthquake (SSE) from the reactor site to the ISFSI pad (URI 072007/2004-002-1) and its assessment of the sub-surface bearing stability beneath the ISFSI pad (URI 0720007/2004-002-2). After the final report was issued, I wrote a Differing Professional Opinion (DPO) on this issue but the agency (the NRC) would not accept it based on the fact that there was no issue to disagree about since the NRC has not made a decision on my issues yet because they changed them from violations to

unresolved items. I informed them that they did make a decision and let Palisades load casks over my objections. They turned me down again because I was retiring and officially couldn't brother them any more, but the point is, the pad is not safe to hold any loaded casks.

4. During an inspection of the 2004 ISFSI installation, I reviewed the licensee's seismic calculations associated with the ISFSI pad and the irradiated fuel canisters. I determined that the licensee performed the ISFSI pad SSE calculations assuming a seismic horizontal acceleration of 0.2g in the free-field and at the ISFSI pad ground surface elevation of 623 feet. The licensee stated its understanding that the seismic horizontal acceleration value of 0.2g was approved by the NRC at the time of initial reactor plant licensing. The licensee further stated its understanding that the 0.2g horizontal acceleration value was applicable for SSE seismic calculations associated with any location and at any elevation on the plant site. I noted that the licensee performed a soil-structure interaction, seismic assessment for the ISFSI pad using the SSE seismic horizontal acceleration of 0.2g. The soil-structure interaction assessment results indicated that the irradiated fuel canisters would experience 0.25g horizontal acceleration during an SSE. The irradiated fuel canister seismic horizontal acceleration design limit is 0.25g.

5. While reviewing the licensee's calculations, I noted significant differences between the elevation and subsurface soil composition of the reactor plant and the 2004 ISFSI pad. Specifically, the reactor containment building was constructed, following the removal of the soil/sands overburden, at a ground surface elevation of 590 feet on compacted glacial till. The 2004 ISFSI pad was constructed, without the removal of the soils/sands overburden, at a ground surface elevation of 625 feet on sands that the licensee mechanically compacted. The licensee estimated that the compacted glacial till soil layer, at the location of the 2004 ISFSI pad, was at an elevation of 560 to 570 feet.

6. Based upon the subsurface soil composition and elevation differences between the reactor plant site and the 2004 ISFSI site, I determined that the licensee's application of the 0.2g horizontal acceleration value that the ISFSI site was non-conservative. Specifically, the inspectors noted that the calculated SSE seismic horizontal acceleration would likely be larger at the ISFSI compared to the reactor plant site due to the increased site elevation and the approximately 50 to 60 feet of mechanically compacted sands present on top of the compacted glacial till material at the ISFSI site. In addition, I concluded that the soil-structure interaction calculation results were non-conservative, which if revised to incorporate a larger horizontal acceleration value based on the increased ISFSI pad elevation and the soil profile differences, would likely result in a seismic horizontal acceleration value in excess of the irradiated fuel canister design limit.

7. Additionally, correspondence between the NRC and the licensee, dated December 1966, telephone call between R. Maccary (Atomic Energy Commission, AEC) and H. Wahl (Bechtel for the licensee), indicates that the NRC considered SSE to be defined as having a horizontal acceleration, at the bedrock, of 0.15g with an amplification factor of 1.25, producing a 0.2g ground acceleration. This demonstrates the NRC's understanding

13. Revision 0 of the Final Safety Analysis Report indicated that a 0.2g surface acceleration was used for the SSE. Licensee calculations of the seismic adequacy of those structures housing safety-related components were all performed at the grade elevation of 590 feet. This was also the ground surface elevation since the overburden of sand dunes was removed prior to construction.

14. NRR and NMSS have been requested by NRC Region III to respond to each of the following questions:

a. During initial licensing of the Palisades Nuclear Power Plant, did the NRC anchor the horizontal acceleration for seismic evaluations at the "ground surface" of the reactor building, elevation 590 feet and on top of the compacted glacial till, or the "ground surface" of the general plant site, any elevation and with any combination of soil structures intervening between the "ground surface" and the underlying bedrock?
b. During Initial licensing of the Palisades Nuclear Power Plant, did the NRC consider that the seismic horizontal acceleration would be amplified from its value at the bedrock to the value used at the "ground" surface due to the type and thickness of the intervening soil between the bedrock and the "ground surface"?
c. Does the NRC expect, based upon the regulations in 10CFR72.212(b)(2)(i)(B) and 10CFR72.212(b)(3), a licensee to incorporate new information and technology into its assessment of the continued appropriateness and re-application of the previous reactor plant seismic siting and design criteria for the design and construction of an ISFSI pad?
d. Irrespective of the previous answers, should the NRC require the licensee to demonstrate that the irradiated fuel canister seismic design is appropriate, using ISFSI pad-specific seismic data, given that the calculated ISFSI horizontal acceleration is at the canister design limit without consideration of the increases expected due to the site-specific soil profile and elevation?

15. Regarding intra-NRC coordination on these questions, NRC Region III staff spoke with NRR staff and others on April 29, 2005. NRR agreed to accept this issue as a Task Interface Agreement and to respond to this request 30 days after receipt, but at least prior to the next dry cask loading campaign, because the pad is not safe to hold the irradiated fuel. The Task Force Agreement Number is 2005-06.

16. Upon request, I would be happy to identify the more than one dozen references referred to in the preparation of this declaration.

/s/ Dr. Ross Landsman
[Signature]

Date: 9-15-2005