Rulemaking1CEm Resource

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Subject: PR-51 Waste Confidence
Attachments: Comment of Michael Keegan with attachments.pdf

DOCKETED BY USNRC—OFFICE OF THE SECRETARY
SECY-067

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FRN#: 78FR56775
NRC DOCKET#: NRC-2012-0246
SECY DOCKET DATE: 12/20/13
TITLE: Waste Confidence—Continued Storage of Spent Nuclear Fuel
COMMENT#: 00930

Hearing Identifier: Secy_RuleMaking_comments_Public
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Subject: PR-51 Waste Confidence
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From:	mkeeganj@comcast.net
To:	RulemakingComments Resource
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Subject:	Comments Nuclear Waste Confidence - Site Specific - Partial
Date:	Saturday, December 21, 2013 12:03:15 AM
Attachments:	Site Specific Comments Nuclear Waste Confidence December 20, 2013.docx Fermi 2 Spent Fuel Pool Re-rack 11-99.pdf Palisades cask dangle summary report 4406.pdf Genevieve Cook - Connie Kline 1986 Besse document.pdf

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One of two with attachments. This one does not contain Davis-Besse 1972 photos.
Dear Rulemaking.Comments@nrc.gov

Waste Confidence Hearings - Written Comment Docket ID No. NRC–2012–0246

Comments of Michael J. Keegan, Coalition for a Nuclear Free Great Lakes – Site Specific Davis-Besse, Fermi, Perry, Palisades

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Over forevermore, indefinite storage of HLRWs on the Lake Erie shore, the dry casks will erode and release their contents. NRC assumes forevermore institutional control, including once per century complete replacement of the pads, inner canisters, and dry casks -- by using a "Dry Transfer System," DTS, as the pools will be dismantled during decommissioning. No price tag is given for this once per century complete replacement, forevermore, nor where that mysterious amount of money (infinite amount of money, by definition) would supposedly come from. NRC's assumption of forevermore institutional control comes amidst a US federal government shutdown.

As Arjun Makhijani has pointed out, in just the past couple-few hundred years, North America has seen multiple wars (the Revolutionary War, the War of 1812 -- including on the Lake Erie shores, where some of the worst battles took place, including in Monroe and Port Clinton -- the Civil War). Institutional control being maintained even just 300 or less years into the future is a huge domain assumption.

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Thank you

Michael J. Keegan
Coalition for a Nuclear Free Great Lakes
P.O. Box 463
Monroe, MI 48161

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P.O. Box 463
Monroe, MI 48161

Douglas R. Gipson
Senior Vice President, Nuclear Generation

Fermi 2
6400 North Dixie Hwy., Newport, Michigan 48166
Tel: 313.586.5201 Fax: 313.586.4172

Detroit Edison

10CFR50.92

November 19, 1999
NRC-99-0084

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington D C 20555-0001

Reference: Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Proposed Technical Specification Changes (License Amendment) –Design Features/Fuel Storage (Technical Specification 4.3) and Programs and Manuals/High Density Spent Fuel Racks (Technical Specification 5.5.13)

Pursuant to 10CFR50.90, Detroit Edison hereby proposes to amend the Fermi 2 Plant Operating License NPF-43, Appendix A, Technical Specifications (TS), to change: (1) the design features description of the fuel storage equipment and configuration to allow an increase in the spent fuel storage capacity and (2) the description of high density spent fuel racks program to clarify that surveillance program is applicable only to racks containing Boraflex as a neutron absorber.

Enclosure 1 provides a description and evaluation of the proposed TS changes. Enclosure 2 provides an analysis of the issue of significant hazards consideration using the standards of 10CFR50.92. Enclosure 3 provides the marked up pages of the existing TS to show the proposed changes and a typed version of the affected TS pages with the proposed changes incorporated. Enclosure 4 provides a licensing

report, which discusses the detailed technical evaluations performed to demonstrate the acceptability of this license change request. Please note that Enclosure 4 contains some information that is considered proprietary pursuant to 10CFR2.790. An affidavit is also included in this enclosure to attest to the proprietary nature of the material contained within the licensing report. In this regard, Detroit Edison requests that Enclosure 4 be withheld from public viewing. Enclosure 5 provides a non-proprietary version, as required by 10CFR2.790.

Detroit Edison has reviewed the proposed TS changes against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, nor significantly change the types or significantly increase the amounts of effluents that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, Detroit Edison concludes that the proposed TS changes meet the criteria provided in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement or an Environmental Assessment.

Detroit Edison requests that the NRC approve and issue these changes by January 30, 2001 with an implementation period of within 90 days following NRC approval.

Should you have any questions or require additional information, please contact Mr. Norman K. Peterson of my staff at (734) 586-4258.

Sincerely,

Enclosures

cc: A. J. Kugler
A. Vegel
NRC Resident Office
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

I, DOUGLAS R. GIPSON, do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

DOUGLAS R. GIPSON
Senior Vice President, Nuclear Generation

On this 19th day of November, 1999 before me personally appeared Douglas R. Gipson, being first duly sworn and says that he executed the foregoing as his free act and deed.

Notary Public

ROSALIE A. ARMETTA
Notary Public, Monroe County, MI
My Commission Expires Oct 11, 2003

AFFIDAVIT PURSUANT TO 10CFR2.790

I, Michael P. McNamara, being duly sworn, depose and state as follows:

(1) I am the Vice President, Nuclear Projects for Holtec International and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the revised pages to the document entitled "Licensing Report for Enrico Fermi 2 Spent Fuel Pool Rack Installation", Holtec Report HI-992154.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.790(a)(4), and 2.790(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b, 4.d, and 4.e, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to

know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed historical data and analytical results not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed using codes developed by Holtec International. Release of this information would improve a competitor's position without the competitor having to expend similar resources for the development of the database. A substantial effort has been expended by Holtec International to develop this information.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Michael P. McNamara, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 16th day of November, 1999.

Michael P. McNamara
Holtec International

Subscribed and sworn before me this 16th day of November, 1999.

MARIA C. PEPE
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2000

ENCLOSURE 1 TO
NRC-99-0084

FERMI 2 NRC DOCKET NO. 50-341
OPERATING LICENSE NO. NPF-43

REQUEST TO REVISE TECHNICAL SPECIFICATIONS:

DESIGN FEATURES - FUEL STORAGE
AND
PROGRAMS AND MANUALS- HIGH DENSITY SPENT FUEL RACKS

DESCRIPTION AND EVALUATION
OF THE PROPOSED CHANGES

DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE(S)

OVERVIEW:

The following is a request to amend Operating License NPF-43 by incorporating the proposed changes identified in Enclosure 3 into the Technical Specifications of Fermi 2 to increase the spent fuel storage capacity. Expansion of the fuel storage capacity in the Spent Fuel Pool (SFP) requires a change to the Design Features section of the Technical Specification. More specifically, Section 4.3, Fuel Storage, discusses the current storage capacity and design features of the existing racks, which ensure adequate design margin with respect to criticality. A request to amend TS 5.5.13, High Density Spent Fuel Racks, is also included to clarify that the surveillance program is only applicable to racks which utilize Boraflex as a neutron absorber.

BACKGROUND:

The SFP at Fermi 2 was reracked in 1982, prior to initial licensing and operation of the facility, with high-density freestanding racks using Boraflex as a neutron poison. At that time, the SFP was authorized to store 2305 fuel assemblies in 14 spent fuel racks, out of which two cells are reserved for the high density rack Boraflex surveillance program. An additional rack containing 35 cylindrical cells also currently exists to store control rods, up to 31 defective fuel canisters and other miscellaneous fuel related components. The SFP also contains four low density GE racks with a total of 80 storage locations. Therefore, the current configuration allows for storage of up to 2383 (2414 including the defective fuel locations) fuel assemblies.

The Fermi 2 reactor core holds 764 fuel assemblies. Current projections, based on expected future spent fuel discharges, indicate that loss of Full-Core-Discharge (FCD) capability will occur when new fuel is received for Cycle 9 in June 2001. Operation of Fermi 2 beyond loss of full-core-discharge capability is possible for Cycles 9 and 10 to provide approximately three additional years of operation until 2004. Fermi 2 operating license authorizes plant operations through March 20, 2025. The proposed change would increase the SFP storage capacity to permit continued plant operation until approximately 2015.

DESCRIPTION OF PROPOSED CHANGES:

Currently, Fermi 2 is proposing to expand the storage capacity in the SFP to accommodate a Full-Core-Discharge when new fuel is received for Cycle 9 in June 2001. This proposed modification will be accomplished by installing additional storage racks and replacing existing racks with high density racks in a three phased approach. The initial phase will add up to four racks to the SFP in open spaces to increase the storage capacity to 3146 assemblies. The second phase will remove the four GE racks, the existing defective fuel storage rack, and the 108 cell high density storage rack and install five new high density racks. This modification will increase the storage capacity to 3588 assemblies. The third phase will replace the remaining 13 existing

Boraflex racks with 14 new high density racks to increase the storage capacity to 4608 assemblies. The completed configuration represents a storage capacity increase of 2194 assemblies.

The new storage racks will be free standing and self supporting. The new modules will be separated by a gap of approximately 1.0 inch from one another and with greater gaps between the new racks and the existing racks.

As a means of providing additional storage space for miscellaneous components, the new storage racks are designed to accommodate two platforms, which may be each placed above, and supported by, an individual rack. The platforms weigh approximately 1,460 and 1,100 pounds, respectively, and may support up to five tons each (dry weight) of miscellaneous components. These platforms would be installed on an as needed basis.

The new racks will contain Boral as the active fixed neutron absorbing poison for primary reactivity control. The Boral absorbers are sized to sufficiently shadow the active fuel height of all fuel assembly designs stored in the pool. The proposed racks will allow fuel storage for enrichments up to 5.0 wt % U-235 with fuel of the highest anticipated reactivity and the pool flooded with unborated water at a temperature corresponding to the highest reactivity.

As there is no requirement for an in-service surveillance program for Boral, a clarification is being proposed to the Programs and Manuals section of the Technical Specifications. The existing requirement for the High Density Spent Fuel Rack Surveillance Program will only be applicable to the racks containing Boraflex.

To accommodate the proposed increase in capacity, the Fermi 2 Technical Specifications are required to be modified. The revised Technical Specification pages are provided in Enclosure 3. Enclosure 4 provides a report, which discusses the features of the new racks along with the evaluation methodologies used to establish adequate design margins with respect to structural, thermal and criticality performance.

SAFETY ASSESSMENT:

The planned expansion of the storage capacity involves the installation of additional fuel racks and removal of existing racks during three separate phases. Evaluations are performed to ensure that all possible fuel configurations remain safe under normal and accident conditions. The SFP thermal performance, criticality, and seismic response are re-analyzed considering the increased storage capacity. The results of these analyses have shown that the pool storage systems remain adequate to contain and cool the fuel in a subcritical condition.

With the expanded capacity, the systems available to provide cooling to the SFP will be required to remove an increased heat load while maintaining the SFP bulk temperature below the design limit of 150° F. The maximum heat load develops from the residual heat in the pool after the last full core discharge at the end of spent fuel pool storage capacity.

As plant operation continues, the SFP heat load increases due to the addition of spent fuel introduced into the SFP following each refueling operation. Due to the increase in decay heat load as the plant continues to operate and perform refueling operations, Fermi 2 has determined the maximum normal SFP bulk temperature, presently 125° F, will increase to a higher value of less than 150° F. Additionally, due to an increased SFP spent fuel inventory, time-to-boil values will decrease with a corresponding increase in boil-off-rates. The new time-to-boil and make-up requirements are consistent with the industry and within the plant design basis.

The Significant Hazards Consideration (SHC), contained herein and the attached Licensing Report (Enclosure 4) demonstrate the acceptability of the proposed increase in the SFP storage capacity and revisions to the Technical Specifications. The scope of the technical analysis supporting this evaluation focused mainly on the final configuration of the expanded storage space. Analysis of the transition to the final configuration involving some intermediate stages is also included in the evaluation.

MECHANICAL DESIGN EVALUATION:

The new fuel rack designs are evaluated with respect to the mechanical and material qualifications, neutron poison, fuel handling qualifications, fuel interfaces and accident considerations.

The principal construction materials for the new racks will be SA240 Type 304L stainless steel, and SA564-630 precipitation hardened stainless steel for the adjustable support spindles. The rack designs, material selection and fabrication process will comply with the applicable ASTM Standards A240, A276, A479, A564 and others, for nuclear service. The governing quality assurance requirements for fabrication of the racks meet or exceed 10CFR50, Appendix B requirements.

For primary nuclear criticality control in the new racks, a fixed neutron absorber will be integrated within the rack structure. The absorber, trade name Boral, is a boron carbide and aluminum-composite sandwich. Boral is chemically inert and has a long history of applications in the SFP environments where it has maintained its neutron attenuation capability under thermal loads. Boral is manufactured under the control of a quality assurance program, which meets or exceeds to the requirements of 10CFR50, Appendix B.

The installation of the new rack modules will preserve space for thermal expansion and seismic movement. The support legs on the racks will allow for remote leveling and alignment of the rack modules to accommodate variations in the floor flatness. A thick bearing pad will be interposed between the rack pedestals and the floor to distribute the dead load over a wider support area.

The rack structural performance with respect to the accidental drop impact and tensile loads, as well as the subcritical configuration, has been analyzed. The analyses included an accidental

drop of a fuel assembly during movement to a storage location and tensile loads (vertical and eccentric) on the rack arising from a stuck assembly in the storage cell. The results of analysis demonstrate that the stored spent fuel remains in a coolable and subcritical configuration. The storage rack structural integrity, and thus the fuel configuration, will be maintained. The fuel will retain its structural integrity and remain subcritical.

CRITICALITY CONSIDERATIONS:

The NRC guidelines and the ANSI standards specify that the margin of safety for criticality be maintained by having the maximum neutron multiplication factor, k_{eff} less than or equal to 0.95, including uncertainties, for all normal and accident conditions.

The new spent fuel racks are designed to maintain the required subcriticality margin when analyzed using conservative design criteria and assumptions. The racks are considered fully loaded with fuel of the highest acceptable reactivity, submerged in unborated water at a temperature corresponding to the highest reactivity. Neutron absorption in minor structural members is neglected, i.e., spacer grids are replaced by water. The criticality analyses are based upon the infinite multiplication factor (k_{inf}), i.e., lattice of storage racks is assumed infinite in all directions. No credit is taken for axial or radial neutron leakage, except in the assessment of certain abnormal/accident conditions where neutron leakage is inherent. The effects of calculational and manufacturing tolerances are evaluated and added in determining the maximum k_{inf} in the storage rack.

For reactivity control in the racks, Boral panels are used. The panels are sized to sufficiently shadow the active fuel height of all assembly designs stored in the pool. The panels are held in place and protected against damage by a stainless steel jacket, which is stitch welded to the cell walls. The panels are mounted on the exterior or on the interior of the cells, in an alternating pattern.

The analysis shows that the criterion of k_{eff} less than or equal to 0.95, including uncertainties, is always maintained under all normal conditions and postulated accidents. The accidents and malfunctions evaluated included a dropped fuel assembly on top of the fuel rack; impact on criticality of water temperature and density effects; and impact on criticality of eccentric positioning of a fuel assembly within the rack.

The new Boral racks are designed to a k_{inf} in the standard cold core geometry (SCCG) of 1.33, as discussed in Chapter 4 of the Licensing Report (Enclosure 4). However, in order to include additional criticality margin for the new racks, the Technical Specifications will remain consistent for all racks. Thus, the racks may store fuel with a maximum k_{inf} in the standard cold core geometry (SCCG) of 1.31. Refer to Technical Specification, Section 4.3.1 (a).

THERMAL-HYDRAULICS AND POOL COOLING:

A comprehensive thermal-hydraulic evaluation of the expanded storage capacity has been performed to analyze the thermal performance of the SFP and its cooling systems. The maximum allowable SFP bulk temperature is 150°F for all scenarios. The maximum local water temperature must remain below boiling at local saturation pressure and cladding temperatures must be limited such that nucleate boiling does not occur.

The calculation of the bounding long-term decay heat for thermal analysis of the pool was performed in accordance with the provisions of the USNRC Branch Technical Position ASB 9-2; "Residual Decay Energy for Light Water Reactors for Long Term Cooling". The determination of the decay heat took into account both the past discharges and the predicted future refueling cycles.

The evaluations considered the decay heat load from three separate discharge scenarios:

- A partial core discharge of 260 assemblies is considered, which corresponds to a condition that results in a loss of full core reserve in the SFP. The minimum decay time of the previously discharged fuel assemblies for this scenario is 18 months. The decay heat load coincident with the peak temperature for this scenario is 12.20 MBtu/hr, which occurs 72 hours after reactor shutdown.

- A normal full core discharge is considered, which produces a stored fuel inventory that conservatively exceeds the maximum possible inventory. The 764 discharged assemblies are separated into two distinct groups: 260 assemblies with a burnup of 50,000 MWD/MTU and 504 assemblies with a burnup of 33,333 MWD/MTU. The minimum decay time of the previously discharged fuel assemblies for this scenario is 18 months. The decay heat load coincident with the peak temperature for this scenario is 41.84 MBtu/hr, which occurs 160 hours after reactor shutdown.

- An emergency full core discharge is considered, which is the same as the normal full-core discharge case above, except for the cooling time considered. The 764 discharged assemblies are separated into two distinct groups: 260 assemblies with a burnup of 50,000 MWD/MTU and 504 assemblies with a burnup of 33,333 MWD/MTU. The minimum decay time of the previously discharged fuel assemblies for this scenario is 12 months. The decay heat load coincident with the peak temperature for this scenario is 42.37 MBtu/hr, which occurs 159 hours after reactor shutdown.

Under normal SFP operations, with a decay heat load of less than 15.83 MBtu/hr, two trains of the Fuel Pool Cooling and Cleanup System (FPCCS) provide sufficient cooling to maintain the SFP bulk temperature below 150°F. The 15.83 MBtu/hr decay heat load corresponds to a partial core discharge of 260 assemblies from the reactor shutdown 12 days previously and the remainder of the pool filled with background fuel assemblies.

During refueling conditions, supplemental cooling is normally provided to the SFP by the Residual Heat Removal (RHR) system. In fact, during fuel discharge to the pool, supplemental cooling from one division of the RHR system will normally be required to provide a satisfactory margin of safety to maintain the bulk pool temperature below 150°F as the decay heat load is rising in the pool. Subsequent to confirmation that once the SFP decay heat load has fallen below 15.83 MBtu/hr, supplemental cooling is no longer required to be provided to the SFP by the Residual Heat Removal (RHR) system.

In all scenarios, the cooling water that removes heat from the FPCCS and RHR heat exchangers is assumed to be at its design temperature and flow rate. In all cases analyzed, the heat transfer model conservatively accounted for an additional resistance from the fouling of the heat transfer surface in the heat exchangers and performance loss due to plugged tubes.

The local water temperature determinations are performed assuming that the pool is at its peak bulk temperature. The worst location was identified as the cell with the hottest assembly and the most restrictive flow arrangement. A conservative value for the axial peaking factor is used. The storage cell hydraulic resistance was based on the most hydraulically limiting fuel assembly type, the most restrictive water inlet geometry for cells located over the rack support pedestals, and the effects of partial outlet blockage due to a dropped fuel assembly lying over the top of the storage cells.

The bulk pool temperature analysis determined that the cooling systems have sufficient capacity to maintain the temperature below 150°F during, and subsequent to, all postulated fuel discharge scenarios.

The calculated maximum local water temperature is determined to be 169°F in the hottest channel and coincides in time with the highest pool bulk temperature. This water temperature is substantially below the 238°F local boiling temperature at the top of the racks. The maximum fuel cladding temperature is calculated to be 197.12°F. Therefore, nucleate boiling will not occur.

Complete loss of all forced pool cooling is not considered a credible event in the design basis, as stated in UFSAR section 9.1.3.3. Nevertheless, a loss-of-cooling event was analyzed for all discharge scenarios. The interruption of the cooling to the pool was assumed to occur coincident with the SFP peak decay heat generation. The analysis determined the time when the pool bulk water reaches boiling and the resultant maximum water loss rate from the surface. The calculated time to boil is 4.20 hours after the cooling is lost in the most severe scenario. However, this is acceptable because the corresponding boil-off rate is less than the makeup capacity of 100 gpm available from the condensate storage tanks, and additional sources of makeup including the fire protection system and category I systems which can be aligned to supply SFP makeup. Additionally, the 4.20 hour period allows sufficient time for the operators to intervene and line up an alternate source to remove the decay heat and replenish the pool inventory.

SEISMIC AND STRUCTURAL EVALUATION:

A complete re-evaluation of the mechanical and civil structures, to address the structural issues resulting from the expansion of the pool storage capacity, has been performed. The analysis considered the loads from seismic, thermal, and mechanical forces to determine the margin of safety in the structural integrity of the fuel racks, the liner, and the SFP located in the Reactor Building. The loads, load combinations, and acceptance criteria are based on the ASME Section III, Subsection NF, and on NUREG-0800, SRP Section 3.8.4, Appendix D.

a. The Storage Rack Evaluation:

The new high density racks are analyzed considering the configurations at the end of each of the three phases with the racks completely filled with fuel. Interim configurations and partially loaded racks are also evaluated considering single isolated racks. The seismic analysis is performed using a whole pool multi-rack analysis. The seismic load evaluations consider simulations of the Safe Shutdown Earthquake (SSE) and the Operating Basis Earthquake (OBE) in accordance with SRP 3.7.1. The rack modules are analyzed using a conservative assembly weight of 690 pounds. This weight is conservative since the actual weight of the assembly is 680 pounds. Two of the racks are also qualified for an additional optional storage function, i.e., the racks are designed to accommodate a storage platform, which has a capacity of storing up to five tons (dry). The 1,460 pound platform for rack B and a smaller 1,100 pound platform for rack G are movable, and can be installed on top of the racks by inserting its four support legs into empty storage cells.

The results indicate that the maximum seismic displacements do not result in any rack-to-wall or inter-rack impacts. The resultant member and weld stresses in the racks are all below the allowable stresses, with a safety factor of at least 1.05. This minimum calculated safety factor is associated with the pedestal support female thread shear stress. The minimum calculated safety factor associated with the cell membranes is 3.8. The minimum safety factor for the welds is 1.97. The racks will remain functional during and after all postulated loading conditions, including the SSE.

Fatigue analysis was performed on the storage racks to determine the cumulative damage factor resulting from twenty operating basis earthquakes followed by one design basis earthquake. This analysis showed that the factor of safety is greater than 2.6 for fatigue within the rack components.

The rack analysis provides pedestal-to-bearing pad impact loads resulting from lift-off and subsequent resettling during dynamic events. The pool floor stresses are determined for these impact loads to remain within allowable limits, even when considering the worst case pedestal location with respect to leak chases.

In addition to the seismic evaluations, the storage racks are also analyzed for all postulated structural accident conditions as described in UFSAR section 9.1.2.3. A fuel handling accident involving a fuel assembly dropped from the Refuel Bridge highest possible lift point would not compromise the integrity of the rack. Permanent deformation of the rack would be limited to the top region only. This is acceptable since the rack cross-sectional geometry at the active fuel height is not altered. Thus, the functionality of the rack is not affected.

In the event of a stuck fuel assembly in the rack, the resultant load on the members will not affect the rack structural integrity to maintain the fuel storage qualifications.

b. Spent Fuel Pool Structural Evaluation:

The SFP is located at the fifth floor of the Fermi 2 Reactor Building, north of the reactor drywell. The SFP consists of cast-in-place monolithic reinforced concrete interior and exterior walls and is designed as a seismic Class I stainless steel lined pool structure that provides space for storage of spent fuel assemblies.

The pool structure has been analyzed using a 3-D finite element model with rack pedestal and hydrodynamic loads, and seismic loads applied by developing a modal analysis and performing a quasi-static evaluation. The individual loads and load combinations used are in accordance with NUREG-0800, SRP Section 3.8.4 and based on the "ultimate strength" design method. The primary loads considered are:

- the dead weight of the concrete structure and steel liner, fully loaded racks, fully loaded overhead platforms and the water,
- quasi-static seismic loads consistent with the original plant design for the OBE and SSE cases,
- hydrostatic pressure force lateral to the walls,
- hydrodynamic coupling forces applied to the lower portion of the wall and water slosh and inertia pressures above the elevation of the top of the racks,
- bounding thermal loads from a full-core-discharge and a loss of cooling, producing the largest temperature gradient across the thickness of the wall and the slab,
- reactive forces due to live loads, and
- seismically induced rack pedestal loads.

In addition to the loads described above, the pool structure and liner are also analyzed for mechanical loads under accident conditions. Analyses are also performed on liner fatigue considering both temperature and seismic excitation. The results of the analyses performed on

the SFP and Reactor Building indicate that under all postulated loadings the structural components, floor slabs, pool walls, supporting walls, liner and its anchorages will be subjected to stresses or strains within acceptable limits.

RADIOLOGICAL CONSIDERATIONS:

Radiological consequences of the proposed change during the installation evolution and during normal and accident conditions in the SFP area of the Reactor Building are evaluated. The total dose to personnel during the installation phases is estimated to be less than 12 person-rem. This includes removal and cleaning of old racks, diving operations to remove underwater appurtenances, pool cleaning and rack installation. This dose is comparable to similar projects carried out at other facilities and allows compliance with the radiological limits of 10CFR20.

Low level solid radwaste will be generated by the removal of the existing SFP modules, as well as any interferences or SFP hardware that may have to be removed from the SFP to permit installation of the new spent fuel rack modules. These racks will be cleaned to the maximum extent possible prior to offsite shipment and volume reduction will be performed at a decontamination and processing facility prior to disposal and burial at a low level waste repository. Therefore, solid radwaste represented by these racks will be minimized. Thus, Fermi 2 does not expect that increasing the storage capacity of the SFP will result in a significant change in the generation of solid radwaste at Fermi 2.

There are no significant solid, gaseous or liquid radiological releases associated with this storage capacity increase effort.

The analysis of the fuel handling accident event shows that the racks remain intact and the resulting fuel damage maintains gas releases and the corresponding radiological dose below levels previously determined.

A rack drop involving radiological consequences is precluded, since all rack movement during removal and installation phases will follow safe load paths that prevent heavy loads from being transported over the stored spent fuel.

There has been no steady long-term increase of radiological conditions in the SFP resulting from the radionuclides within the fuel as more spent fuel is added to the pool. The radiological conditions within the building are typically dominated by the most recent batch of the spent fuel from a full-core-discharge. The radioactive inventory of the older fuel that will increase with the expanded storage capacity will be insignificant compared to that of the recent offload.

Since the new storage racks will be located in closer proximity to the SFP walls, an increase in the adjacent radiological doses is expected. Radiological analyses have shown that the dose levels adjacent to all pool areas will remain within acceptable levels.

ENCLOSURE 2 TO
NRC-99-0084

FERMI 2 NRC DOCKET NO. 50-341
NRC LICENSE NO. NPF-43

REQUEST TO REVISE TECHNICAL SPECIFICATIONS:
10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION

10 CFR 50.92 SIGNIFICANT HAZARDS CONSIDERATION

BASIS FOR SIGNIFICANT HAZARDS DETERMINATION

In accordance with 10CFR50.92, Detroit Edison has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve a SHC because they would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The following previously postulated accident scenarios are considered:

   a. A spent fuel assembly drop in the SFP

   b. Loss of SFP cooling flow

   c. A seismic event

   d. Misplaced fuel assembly

   The probability that any of the accidents in the above list can occur is not significantly increased by the modification itself. The probabilities of a seismic event or loss of SFP cooling flow are not influenced by the proposed changes. The probabilities of accidental fuel assembly drops or misplacement of a fuel assembly are primarily influenced by the methods used to lift and move these loads. The method of handling loads during normal plant operations is not changed, since the same equipment (i.e., Refuel Bridge) and procedures will be used. Since the methods used to move loads during normal operations remain the same as those used previously, there is no significant increase in the probability of an accident.

   During rack removal and installation, all work in the pool area will be controlled and performed in strict accordance with specific written procedures. Any movement of fuel assemblies required to support the modification (e.g., removal and installation of racks) will be performed in the same manner as during normal refueling operations. Spent Fuel shipping cask movements will not be performed during the modification period.

   Accordingly, the proposed modification does not involve a significant increase in the probability of an accident previously evaluated.

   The consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the SFP have been re-evaluated for the proposed change. The results show that the postulated accident of a fuel assembly striking the top of the storage racks will not distort the racks sufficiently to impair their functionality. The minimum subcriticality margin, k_{eff}

less than or equal to 0.95, will be maintained. The structural damage to the Reactor Building, pool liner, and fuel assembly resulting from a fuel assembly drop striking the pool floor or another assembly located within the racks is primarily dependent on the mass of the falling object and the drop height. Since these two parameters are not changed by the proposed modification, the structural damage to these items remains unchanged. The radiological dose at the exclusion area boundary will not be increased due to the changes. Thus, the results of the postulated fuel drop accidents remain acceptable and do not represent a significant increase in consequences from any of the same previously evaluated accidents that have been reviewed and found acceptable by the NRC.

The time to boil represents the onset of loss of pool water inventory and is commonly used as a gage for establishing the comparison of consequences before and after a reracking project. The heat up rate in the SFP is a nearly linear function of the fuel decay heat load. The fuel decay heat load will increase subsequent to the proposed changes because of the increase in the number of fuel assemblies stored in the spent fuel pool. The thermal-hydraulic analysis determined the maximum fuel decay heat loads and the corresponding time to boil conditions subsequent to complete loss of forced cooling. These results show that, in the extremely unlikely event of a complete failure of both the FPCCS and RHR System, there would be at least 4.20 hours available for corrective actions. The maximum water boiloff rate is less than 91 gpm. This is less than the normal makeup capacity of 100 gpm available from the condensate storage tanks, and additional sources of makeup are available. It has been determined that this duration provides sufficient time for the operators to provide alternate means of makeup (i.e., fire hoses) before the onset of pool boiling. Therefore, the proposed change represents no increase in the consequences of loss of pool cooling.

The consequences of a design basis seismic event are not increased. The consequences of this accident are evaluated on the basis of subsequent fuel damage or compromise of the fuel storage or building configurations leading to radiological or criticality concerns. The racks are analyzed in their new configuration and found safe during seismic motion. Fuel has been determined to remain intact and the storage racks maintain the fuel and fixed poison configurations subsequent to a seismic event. The structural capability of the pool and liner will not be exceeded under the appropriate combinations of dead weight, thermal, and seismic loads. The Reactor Building structure will remain intact during a seismic event and will continue to adequately support and protect the fuel racks, storage array, and pool moderator/coolant. Thus, the consequences of a seismic event are not increased.

A fuel misplacement accident represents a fuel assembly inadvertently lowered or dropped outside of and adjacent to a storage rack. The consequence of a fuel misplacement accident has been analyzed for the worst possible storage configuration subsequent to the proposed modification, and it has been shown that the consequences remain acceptable with respect to the neutron multiplication factor staying below 0.95 (i.e. the same acceptance criteria as used for normal conditions). Therefore, there is no increase in consequences.

Therefore, it is concluded that the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated.

Load drops were determined to be events that might represent a new or different kind of accident. The new loads that will be required during or subsequent to installation of the new racks include the rack modules, the overhead platforms, and the pool gates. Racks will not be allowed to travel over any racks containing fuel assemblies, thus a rack drop onto fuel is precluded. A construction accident of a rack dropping onto the pool floor liner is not a postulated event due to the defense-in-depth approach to be taken, as discussed in detail within Section 10.2 of the attached Licensing Report (Enclosure 4). A new temporary hoist and rack lift rig will be introduced to lift and suspend the racks from the bridge of the Reactor Crane. These temporary lift items are designed in accordance with the requirements of NUREG 0612 and ANSI N14.6. Nevertheless, the analysis of a rack dropping to the liner has been performed and shown to be acceptable. The integrity of the liner will be maintained and no loss of pool coolant would occur subsequent to a rack dropping to the liner. Since fuel integrity is maintained and significant loss of coolant does not occur, the drop of a rack is not considered a new type of accident.

A drop of a pool gate is also an extremely unlikely event. The new storage racks will not be located directly beneath the gates. However, the drop of a gate, weighing approximately 9500 pounds, onto racks containing irradiated fuel assemblies, and the drop of a gate onto the pool liner have been analyzed. The analysis performed for the drop of a pool gate onto fuel demonstrates that the number of fuel rods damaged (81) remains below the Fermi 2 fuel handling accident design basis (of 140 rods). The analysis performed for the drop of a pool gate onto the liner demonstrates that the liner would be locally ruptured. However, the underlying concrete slab remains intact and possible leakage would be confined to the leak chase system, which is monitored and controllable. The kinetic energy associated with the drop of the heaviest (1460 pound) overhead platform is enveloped by the kinetic energy associated with the gate drop. Therefore, the potential structural damage to fuel and the liner would be bounded by the results for the gate. Since the resulting fuel damage does not exceed the previously analyzed design basis condition and significant loss of coolant would not occur, the drops of a gate or an overhead platform are not considered a new type of accident.

The additional heat load resulting from additional storage of spent fuel has been evaluated for the possibility of creating a new or different kind of accident. The existing Fermi 2 SFP cooling system, has been shown by analysis, to be capable of removing the decay heat generated by the additional spent fuel assemblies. The pool coolant will not be significantly affected. Thus, the increased heat load does not create the possibility a new or different kind of accident.

No unproven technology has been utilized in the design, analysis or in the proposed installation methodology. The basic technology for the Fermi 2 spent fuel pool capacity increase is consistent with other license amendments (over 80) approved by the USNRC. This change has been evaluated in accordance with the USNRC position paper "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, April 14, 1978 and Addition dated January 18, 1979.

The proposed change does not alter the operating requirements of the plant or of the equipment credited in the mitigation of the design basis accidents. The proposed change does not affect the parameters required for safe fuel storage. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Involve a significant reduction in the margin of safety.

The function of the SFP is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as an earthquake or fuel assembly drop. The new rack design must meet all applicable requirements for safe storage and be functionally compatible with the SFP.

Detroit Edison has addressed the safety issues related to the expanded pool storage capacity in the following areas:

1. Material, mechanical and structural considerations

2. Nuclear criticality

3. Thermal-hydraulic and pool cooling

The mechanical, material, and structural designs of the new racks are reviewed in accordance with the applicable provisions of the USNRC position paper "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, April 14, 1978 and Addition dated January 18, 1979.. The rack materials used are compatible with the spent fuel assemblies and the SFP environment. The design of the new racks preserves the proper margin of safety during abnormal loads such as a dropped assembly and tensile loads from a stuck assembly. It has been shown that such loads will not invalidate the mechanical design and material selection to safely store fuel in a coolable and subcritical configuration.

The methodology used in the criticality analysis of the expanded SFP storage capacity meets the appropriate NRC requirements and the ANSI standards (GDC 62, NUREG 0800, Section 9.1.2, the OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, Reg. Guide 1.13, and ANSI ANS 8.17). The margin of safety for subcriticality is maintained by having the neutron multiplication factor equal to, or less than, 0.95, including uncertainties, under all accident conditions. This criterion is the same as that used

previously to establish criticality safety evaluation acceptance and remains satisfied for all analyzed accidents. Therefore, the accepted margin of safety remains the same.

The thermal-hydraulic and cooling evaluation of the pool demonstrated that the pool can be maintained below the specified thermal limits under the conditions of the maximum heat load and during all credible accident sequences and seismic events. The bulk pool temperature will not exceed 150°F during any conditions when forced cooling is available. The increase from the current maximum normal SFP bulk temperature of 125°F is not significant, because the existing racks and cooling system were previously evaluated for the 150°F condition, as stated in UFSAR sections 9.1.2.2.2 and 9.1.3.1, respectively. The maximum local water temperature in the hottest rack cell will remain below the boiling point. The fuel will not undergo any significant heat up after an accidental drop of a fuel assembly on top of the rack blocking the flow path. The time of 4.20 hours for the onset of pool boiling, subsequent to total loss of forced cooling allows sufficient time for the operators to intervene and line up alternate cooling paths and/or the means of inventory make-up before the onset of pool boiling.

Thus, it is concluded that the changes do not involve a significant reduction in the margin of safety.

The NRC has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (51FR7751, March 6, 1986) of amendments that are considered not likely to involve a SHC. The proposed changes for Fermi 2 are similar to Example (x): an expansion of the storage capacity of SFP when all of the following are satisfied:

(1) The storage expansion method consists of either replacing existing racks with a design that allows closer spacing between stored spent fuel assemblies or placing additional racks of the original design on the pool floor if space permits.

The Fermi 2 storage expansion modification involves replacement of the existing racks with a design that will allow a greater number of stored fuel assemblies and placement of additional racks in the SFP.

(2) The storage expansion method does not involve rod consolidation or double tiers.

The Fermi 2 storage expansion modification does not involve any fuel consolidation processes or storage of consolidated fuel. The racks will not be double tiered; no fuel assemblies will be stored above other assemblies.

(3) The k_{eff} of the pool is maintained less than, or equal to, 0.95.

The design of the new racks integrates a neutron absorber, Boral, within the racks to allow close storage of spent fuel assemblies while ensuring that k_{eff} remains less than or equal to 0.95 under all conditions.

(4) No new technology or unproven technology is utilized in either the construction process or the analytical techniques necessary to justify the expansion.

The rack vendor has successfully participated in the licensing of numerous other racks of a similar design. The construction process and the analytical techniques of the Fermi 2 modifications are the same as in the other completed pool storage capacity expansion projects. Thus, no new or unproven technology is used in the Fermi 2 storage expansion modification.

ENVIRONMENTAL CONSIDERATIONS:

Detroit Edison has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not significantly increase the types and amounts of effluents that may be released offsite nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, Detroit Edison concludes that the proposed changes meet the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an environmental impact statement.

CONCLUSION:

As discussed herein, the proposed changes to the Technical Specifications do not involve a SHC pursuant to 10CFR50.92. Storage expansion at the Fermi 2 SFP has been determined to be safe. Additionally, Detroit Edison has determined that this license amendment meets the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an environmental impact statement.

ENCLOSURE 3 TO
NRC-99-0084

FERMI 2

NRC DOCKET NO. 50-341
OPERATING LICENSE NPF-43

REQUEST TO REVISE TECHNICAL SPECIFICATIONS

DESIGN FEATURES
AND
PROGRAMS AND MANUALS

Attached is a mark-up of the existing Technical Specifications (TSs), indicating the proposed changes (Part 1) and a typed version of the TSs incorporating the proposed changes with a list of included pages (Part 2).

ENCLOSURE 3 - PART 1 TO
NRC-99-0084

PROPOSED TECHNICAL SPECIFICATION MARKED UP PAGES

INCLUDED PAGE(S):

4.0-2
5.0-19

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

c. ~~A nominal 6.22 inch center to center distance between fuel assemblies placed in the high density storage racks and a nominal 11.9 x 6.6 inch center to center distance between fuel assemblies placed in the low density storage racks.~~

REPLACE WITH INSERT #1

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 660 ft 11.5 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~2414~~ 4608 fuel assemblies.

INSERT # 1

The following nominal center to center distances between fuel assemblies placed in the various storage rack types, as applicable

Spacing (Inches)	Rack Type
6.22	High density storage racks that contain Boraflex as the neutron absorbing material
6.23	High density storage racks that contain Boral as the neutron absorbing material
11.9x6.6	Low density storage racks
10.5	Defective fuel assembly storage rack

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Primary Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.13 High Density Spent Fuel Racks

~~A program shall be provided which will assure that any unanticipated degradation of the high density spent fuel racks will be detected and will not compromise the integrity of the racks.~~

REPLACE WITH INSERT #2

INSERT # 2

A program shall be provided, for the high density storage racks containing Boraflex as the neutron absorber, which will ensure that any unanticipated degradation of the Boraflex will be detected and will not compromise the integrity of the racks.

ENCLOSURE 3 - PART 2 TO
NRC-99-0084

PROPOSED TECHNICAL SPECIFICATION REVISED PAGES

INCLUDED PAGE(S):

4.0-2
5.0-19

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

c. The following nominal center to center distances between fuel assemblies placed in the various storage rack types, as applicable

Spacing (inches)	Rack Type
6.22	High density storage racks that contain Boraflex as the neutron absorbing material
6.23	High density storage racks that contain Boral as the neutron absorbing material
11.9 x 6.6	Low density storage racks
10.5	Defective fuel assembly storage rack

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 660 ft 11.5 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4608 fuel assemblies.

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Primary Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.13 High Density Spent Fuel Racks

A program shall be provided, for the high density storage racks containing Boraflex as the neutron absorber, which will ensure that any unanticipated degradation of the Boraflex will be detected and will not compromise the integrity of the racks.

**Coalition for a Nuclear-Free Great Lakes * Don't Waste Michigan *
Nuclear Information and Resource Service (NIRS) ***

Summary Report

High-Level Atomic Waste Mishap at Palisades Nuclear Reactor Risks Radioactive Inferno with Casualty Potential of Thousands of Deaths Downwind

Based Upon U.S. Nuclear Regulatory Commission Freedom of Information Act (FOIA) Response Documents

April 4, 2006

Prepared by
Kevin Kamps
Nuclear Waste Specialist
NIRS
6930 Carroll Avenue, Suite 340
Takoma Park, MD 20912
Office 301.270.6477x14
Cell 240.462.3216
kevin@nirs.org
www.nirs.org

On March 18, 2006 the Detroit Free Press ran a front page article entitled "Nuclear safety left hanging as crane dangled fuel rods: Michigan incident got warning but no fine," by Hugh McDiarmid, Jr., Free Press Staff Writer. The article revealed a previously unreported October 2005 incident at the Palisades nuclear power plant on the Lake Michigan shoreline in southwest Michigan. According to a U.S. Nuclear Regulatory Commission (NRC) inspection report, a container weighing 110 tons, fully loaded with high-level radioactive waste, dangled for 55 hours from a stuck crane above the reactor's irradiated fuel storage pool. Plant personnel, lacking proper knowledge about the crane, and without permission from plant management, mishandled the crane's emergency brake, increasing the risk of the heavy load crashing, out of control, back down into the pool. The falling container could have severely damaged the pool, draining the cooling water. A radioactive waste fire could have followed, resulting in tens of thousands of cancer deaths from radiation exposure to a distance of 500 miles downwind, according to a separate NRC report.

Internal Palisades and NRC documents, received by NIRS via FOIA, reveal the mistakes that led up to this incident, and the potentially catastrophic consequences that could have resulted.

The Cask Dangle First Comes to Light

The Palisades nuclear power plant is located in Covert, Michigan, on the Lake Michigan shoreline. Living up to the name of its hometown – Covert -- Consumers Energy's Palisades nuclear power plant, with help from the NRC, managed to keep the public in the dark for months about an incident that could have led to a Chernobyl-scale radiation release on the Lake Michigan shoreline. Tens of thousands of people, out to a distance of 500 miles downwind, could have died immediately or due to later cancer, according to NRC reports.

Coalition for a Nuclear-Free Great Lakes, Don't Waste Michigan, and NIRS first learned of the cask dangle on December 21, 2005 while attending an NRC/Palisades technical meeting at NRC's Region III office in Lisle, Illinois. An NRC official revealed that, while lifting a fully loaded waste container out of its storage pool, Palisades experienced a brake engagement which left the cask suspended over the pool from October 11 to 13, 2005. It was also admitted that no event report had been published, thus having kept the public in the dark for over two months at that point.

NIRS filed a FOIA request on January 9, 2006. Although NRC stated that it would respond to the FOIA request in two to four weeks, the FOIA response was not received by NIRS until March 20th, over two months later.

However, a few days earlier, researchers from the Coalition for a Nuclear-Free Great Lakes uncovered an NRC quarterly inspection report issued January 25, 2005 (with an erratum dated February 2, 2006). This NRC inspection report revealed, at page 9:

"The [NRC] inspectors concluded that working outside the bounds of a work package on a crane with a suspended load that if dropped would damage the spent fuel pool warranted a safety significance determination…Had the load dropped, the spent fuel pool could have sustained severe damage. The inspectors were also aware that the individuals involved in the work activity were not fully knowledgeable of the crane's design, operation, and failure modes at the time the work occurred. In order to compensate for the gap in knowledge, the licensee [the owner, Consumers Energy, and operator, Nuclear Management Company] obtained telephonic support from the crane vendor. Therefore, the inspectors concluded working outside the bounds of the approved work package and manipulating the brake release represented an increase in the risk of a load drop. This increase in risk is directly associated with the reactor safety cornerstone objective of the spent fuel cooling system as a radiological barrier."(1)

In other words, the crashing cask, fully loaded with high-level radioactive waste and weighing 107 tons, could have cracked the bottom of the pool and drained out the cooling water. In a matter of hours or less, the years and decades worth of accumulated high-level radioactive wastes stored in the pool could have gotten so hot that it would have ignited into a radioactive conflagration.

The Potentially Catastrophic Consequences Had the Cask Dropped

Another NRC report, NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," published February, 2001, examined just such heavy load drops causing the collapse of the waste storage pool floor. At page 3-16, NRC reports:

"The analysis exclusively considered drops severe enough to catastrophically damage the SFP [spent fuel pool] so that pool inventory [of cooling water] would be lost rapidly and it would be impossible to refill the pool using onsite or offsite resources. There is no possibility of mitigating the damage, only preventing it…The staff assumes a catastrophic heavy load drop (creating a large [cooling water] leakage path in the pool) would lead directly to a zirconium fire. The time from the load drop until a fire varies depending on fuel age, burn up, and configuration. The dose rates in the pool area before any zirconium fire are tens of thousands of rem per hour, making any recovery actions (such as temporary large inventory [of replacement cooling water] addition) very difficult. Based on discussions with [NRC] staff structural engineers, it is assumed that only spent fuel casks are heavy enough to catastrophically damage the pool if dropped."(2)

Given that Palisades is an operating reactor, the wastes in its storage pool are even hotter – radioactively and thermally – than wastes at a decommissioning, or permanently shut down, nuclear power plant. In fact, NRC has reported that "…the possibility of a zirconium fire leading to a large fission product release cannot be ruled out even many years after final shutdown..."(3)

Thus, these NRC reports reveal that a Chernobyl-scale nuclear catastrophe could have occurred on the Lake Michigan shoreline last October. NRC admits that once the cask had cracked the pool and drained the cooling water away, radiation doses near the pool would have killed any emergency responders who approached too near after just a few minutes exposure time. Firemen would have had to sacrifice their lives in any attempt to stop the quickly unfolding disaster. But NRC chillingly stated such an accident must be prevented in the first place, because once it starts, it is impossible to put the deadly radioactive genie back in the bottle.

Palisades' irradiated nuclear fuel rods are clad in zirconium metal. Zirconium, an ingredient in cluster bombs and old-fashioned camera flash bulbs, spontaneously combusts at a high-enough temperature. The thermal heat generated by radioactive decay occurring in Palisades' hotter wastes, more recently discharged from the reactor core, could initiate the fire, which would then likely spread to the entire waste inventory in the pool.

According to the U.S. Department of Energy's February 2002 Environmental Impact Statement for the proposed national high-level radioactive waste dump at Yucca Mountain, Nevada, Palisades currently has over 188 tons of highly radioactive nuclear fuel stored in its pool.(4). Palisades pool thus contains significantly more dangerous and deadly long-lasting radioactive poisons, such as Cesium-137, than were released by the

Chernobyl nuclear catastrophe in 1986.(5) Whereas the shorter lasting radioactive poisons that were present in Chernobyl's reactor core would have decayed away over the years and decades in the Palisades pool, the longer-lived radioactive poisons, such as Cesium-137 (hazardous for 10 to 20 "half lives," that is, 300 to 600 years), would still be present in very large quantities in the Palisades pool.

Alvarez et al. stated "Spent fuel recently discharged from a reactor could heat up relatively rapidly to temperatures at which the zircaloy fuel cladding could catch fire and the fuel's volatile fission products, including 30-year half-life [Cesium-137] would be released. The fire could well spread to older spent fuel. The long-term land-contamination consequences of such an event could be significantly worse than those from Chernobyl."(6)

Citing a United Nations report from 2000, the Alvarez report went on to state that:

"The damage that can be done by a large release of fission products was demonstrated by the April 1986 Chernobyl accident. More than 100,000 residents from 187 settlements were permanently evacuated because of contamination by [Cesium-137]. Strict radiation-dose control measures were imposed in areas contaminated to levels greater than [15 curies per square kilometer, or 555 kilo-Bequerels per square meter] of [Cesium-137]. The total area of this radiation-control zone is huge: [10,000 square kilometers], equal to half the area of the State of New Jersey. During the following decade, the population of this area declined by almost half because of migration to areas of lower contamination."(7)

10,000 square kilometers equals 3,800 square miles, nearly 7% of the total land area of the State of Michigan.

NRC goes on to report in NUREG-1738, in "Appendix 4: Consequence Assessment from Zirconium Fire," that tens of thousands of people could have then died, either promptly from radiation poisoning, or from latent cancers, up to 500 miles downwind. Table A4-7, "Mean [Average] Consequences for the Base Case," shows over 26,800 deaths possible. Table A4-15, assuming a larger population density per square mile, estimates a long-term consequence of 44,900 cancer fatalities downwind.(8)

Palisades and NRC report that the population and population density surrounding Palisades is relatively small. In 2000, 118,667 people were living within 20 miles of Palisades, for a density of 238 persons/square mile; 1,287,558 persons were living within 50 miles of the plant, for a density of 283 persons/square mile.(9) However, it must be pointed out that the bulk of Michigan's second largest city – Grand Rapids – lies outside that 50 mile zone. And the largest cities in Michigan and Illinois – Detroit and Chicago – fall within 500 miles of Palisades. These large populations would worsen casualty rates downwind of a major radiation release from Palisades.

Of course, not only could tens of thousands of people have died from radiation poisoning and cancer, but Michigan's entire tourism and agricultural industries could have been ruined, as

well. And Palisades is located on the shore of Lake Michigan, whose waters -- and the waters of the Great Lakes downstream into which Lake Michigan flows -- provide drinking water for millions of people in the U.S. and Canada. Thus, the consequences of a large radiation release from Palisades would be dire indeed.

Despite this, the NRC quarterly inspection report stated "because the actions by the worker did not result in any load motion and both crane brakes remained set, NRC management determined the finding to be of very low safety significance."(10)

Incredibly, NRC has let Palisades off with a slap on the wrist. It's not unlike the Davis-Besse nuclear power plant near-meltdown in 2002 near Toledo, in which the NRC's own inspector general reported that both NRC and the nuclear utility put company profits over public safety. In that case, we almost lost Toledo. In this case, we almost lost west Michigan, and Lake Michigan as well.(11)

Coalition for a Nuclear-Free Great Lakes also uncovered an NRC event report from October 12, 2005 – the exact timeframe for the cask dangle – revealing that "Portions of the Palisades Plant Process Computer (PPC) including the Emergency Response Data System (ERDS) became inoperable due to failure of a plant inverter…"(12)

This begs the question, could this computer failure have been yet another straw to break the camel's back that day, resulting in a radiological catastrophe downwind and downstream of Palisades?

Citing many of the NRC documents previously referenced, The Detroit Free Press reported the Palisades cask dangle, and its potentially catastrophic consequences, on March 18, 2006.

Revelations from NRC's FOIA Response to NIRS

NRC's "Partial" FOIA response, although dated March 8, 2006 – two full months after the FOIA request was made – did not reach NIRS until nearly two weeks later. NRC and Palisades internal documents reveal that many mistakes led up to the cask dangle, and also that other short cuts on safety could have made the incident event more dangerous.

For example, on October 6, 2005 – just five days before the cask dangle – NRC granted Palisades an exemption from "Criticality Accident Requirements" for loading of independent spent fuel storage installation casks. This despite NRC's admission that "NMC's [Nuclear Management Company's, Palisades operator] request for exemption…proposes to permit NMC to perform spent fuel loading, unloading, and handling operations related to dry cask storage without being subcritical under the most adverse moderation conditions feasible by unborated water." NRC also assumed that the spent fuel would be kept in "a geometrically safe configuration," and that "appropriate, conservative criticality margins during handling and storage of spent fuel" would be applied.(13)

But the cask dangle involved a container whose lid had not yet been bolted shut. If the cask had dropped into the pool, the waste within could have fallen out, forming a critical mass. The still-fissile components in the waste – uranium-235 and plutonium-239 – could have caused a nuclear chain reaction in the pool. This would be all the more likely if unborated water were added to the pool – such as to replenish cooling water in the event of a pool leak from the cask drop. Boron in the pool water serves as an anti-criticality measure.

Palisades upgraded its irradiated nuclear fuel storage pool crane to "single-failure-proof" in June, 2004, just 16 months before the cask dangle incident. NRC, in granting the crane upgrade, stated "[s]ince the new main hoist for the upgraded crane is of the single-failure-proof design, the cask drop analysis is no longer required for load drops from the main hoist. As a result of the impact-limiting pads previously installed in the spent fuel pool to protect the pool structure from the postulated transfer-cask-drop accident during dry fuel storage operations is being eliminated." Thus, as the crane was upgraded, the pool was allowed to lose a layer of protection against a cask drop. Was this to free up more space in the pool, so that more waste could be stored there? However, in allowing the impact-limiting pads to be removed, NRC was assuming that Palisades would provide proper "training and qualification of crane operators," as well as "inspection, testing and maintenance of cranes." But it was just such failures that led to the cask dangle.

An internal Palisades documents reveal that "[crane vendor] Ederer procedure 260 was apparently used a reference, but not followed completely to determine torque value for the EATL [energy absorbing torque limiter, an emergency brake]." It also revealed that "OE [operational experience] from Big Rock [a Consumer's Energy nuclear power plant in Charlevoix, MI that permanently shut down in August 1997] showed similar occurrences from an improperly set EATL that needed additional adjustments. This had occurred twice on a crane of the same design as Palisades."(14) So, despite previous company experience with just such crane malfunctions, the "lessons learned" were not enough to prevent the incident at Palisades. And despite a company pledge to NRC in its Final Safety Analysis Report, that "[t]he design and construction of the [irradiated fuel handling] system includes interlocks, travel and load limiting devices and other protective measures to minimize the possibility of mishandling or equipment malfunction that could cause damage to the fuel and potential fission product release,"(15) internal company documents reveal an alarming lack of understanding of the crane.

In an internal Palisades document tellingly titled "Intent of WO [Work Order] Task Exceeded During Troubleshooting," the team sent to inspect the stuck crane and its dangling cask admitted:

"The team members had all been trained to perform mechanical inspections of cranes. In this training, components are visually inspected and mechanically inspected using a number of techniques and tools. While being pre-job briefed, the team heard 'Go and perform a normal mechanical inspection you have been trained to perform." <u>The actual</u>

intent for the inspection approved by the Event Response Team was to perform a visual inspection on the mechanical components to determine if anything was broken. With this disconnect, the team performed a normal mechanical inspection which was outside the intended inspection approved by the Event Response Team…Although we all thought the information we were gathering was within the steps of the Work Order, we failed to consider the severity of the consequences if our troubleshooting caused the load to slip or fall into the Spent Fuel Pool. This is why we set up an Event Response Organization during problems like this – to allow an open forum to recommend tests and troubleshooting activities with full consideration of how these activities will affect the plant/health and safety of the public." [emphases added]

Despite the intention of Palisades management that "they wanted a visual inspection with no components touched," Palisades employee Chad Main wrote in a memo dated 10/11/05 entitled "Troubleshooting on Spent Fuel Pool Crane L-3" that, per crane manufacturer Ederer's instructions over the phone, "[t]o verify the emergency brake was set, Ederer recommended the nitrogen bottle valve be opened and the brake release moved very slowly to remove a small amount of tension on the brake mechanism. This was done and the actuator moved approximately 2 mm." Thus, Palisades workers partially overrode the emergency brake on the crane from which dangled a 107 ton cask fully loaded with high-level radioactive waste, which, if dropped, could have caused a radioactive inferno killing tens of thousands downwind.

The NMC document lists "Vague and Incorrect Guidance…Ineffective Communication…[and] Over Confidence" caused the human error despite the "sensitivity of the suspended load." The document concludes by saying "The team supervisory oversight was given two days off without pay due to not following the Event Response Team instruction."(16)

See on the next page a photo of the Palisades cask during its dangle over the irradiated nuclear fuel storage pool, dated 10.11.2005 at 15:03 Eastern Daylight Time.(17) "…the cask was approximately four feet out of the water,"(18) meaning that 11 feet remained underwater. The crane remained stuck and the cask dangled above the irradiated fuel storage pool for 43 hours, from 5:30 am on 10/11/2005 till 12:28 am on 10/13/2005.(see footnote (26) below, p. 6) Although Palisades site leadership, NRC, NMC's reactor fleet, and the Institute for Nuclear Power Operations (an industry self-policing regulatory body) received word about the cask dangle, the public was kept in the dark for over two months, and documentation was not made available to the public by the NRC till five months later.(19) The cask, weighing about 107 tons, contained 32 irradiated nuclear fuel assemblies, nearly 13 tons worth of high-level radioactive waste.(20)

The inspection team must not have read the work order carefully, because it clearly states "[r]emote visual to be performed" and "perform a visual inspection of mechanical equipment associated with the main hoist on l-3 crane." Further on, the Work Order makes explicit "Do not perform any movement of the L-3 crane" and a "WARNING" about the resetting of the emergency brake potentially causing "uncontrolled movement of the Main Hoist drum…" Further warnings that "caution and

conservatism during the test or evolution, particularly when uncertainties are encountered…[and] Verification that adequate margins of safety are to be maintained when interlocks and protection systems are bypassed," was also discussed, but went unheeded and were in fact violated. Despite admitting that "a MAE (maintenance avoidable error)" had occurred, Palisades inspection team answered "No" to the question "Are any Action Requests or Lessons Learned Warranted?"(21)

So what caused the emergency brake to engage in the first place? Palisades reports that "Prior to 10/11/2005 the EATL [energy absorbing torque limiter] was last set in August 2005…The as-left set-point was 175 ft-lbs. Review of the work order could not validate that Ederer Procedure 260 was utilized or the EATL was tested for repeatability after its setting was applied. The target set point was 186 +/- 18 ft-lbs…The low as-found EATL break away torque value confirms that the EATL is a credible cause for the emergency brake actuation. Discussions with the crane vendor Ederer indicated that EATL slippage from a 93 ton load would be equivalent to an EATL set-point of 121 ft-lbs, which is sufficiently close to the as-found set-point of 140 ft-lbs when consideration is given to the additional loading experienced due to the dynamic affects of a moving load."(22)

"Why did this occur?" NMC asks itself. Its answer: "Review of the work order from August 2005 shows that the break-away torque setting may not have been done correctly. It appears that they may have only adjusted the setting once, yet the procedure in the vendor's manual requires multiple evolutions."(23)

Despite the Ederer crane representative's improper instructions to manipulate the emergency brake during the cask dangle, an internal NMC document reveals that "[the crane] Vendor does not recommend adjusting the EATL with a load suspended" and suggested precautions and "conservative measures consistent with the nature of the load being handled." This document again mentioned that "Big Rock [Point nuclear power plant in Charlevoix, MI] experienced several emergency brake set incidences...The conclusion was that even when the EATL was set within the vendor recommended range (186+/-ft-lbs) the EATL caused emergency brake sets with the crane heavily loaded," apparently during emptying of the storage pool of high-level wastes into dry casks as part of Big Rock Point's decommissioning. However, news of those incidents was not, to this author's knowledge, ever made readily accessible to the public or the media by NRC or the company. And it appears the company did not learn lessons from Big Rock Point, at least not sufficiently enough to prevent a repeat of a cask dangle at Palisades. An important question regarding the Big Rock Point cask dangles is, was the Ederer crane there single-failure-proof, and were safety precautions such as the emergency brake overridden improperly as occurred at Palisades in Oct., 2006?(24)

NMC admitted that "The former L-3 crane…main hoist was not designed as single-failure-proof." But "In 2002/2003, NMC modified the L-3 crane to increase the rated load capacity to 110 tons and incorporate single-failure-proof technology." Luckily, this 2006 cask dangle had not occurred several years earlier, for a cask drop would have been much more likely then. Another important question to answer about the Big Rock

Point cask dangles in Nov. 2002 – before or after the crane at Big Rock was made single-failure-proof? Did Palisades upgrade its Palisades crane to single-failure-proof because of its cask dangles at Big Rock Point?(25)

In its "Root Cause Analysis Report: Crane operator heard loud noise during lift with L-3 crane," Nuclear Management Company admits that "[t]he EATL [Energy Absorbing Torque Limiter] is the last-line-of-defense for overload." It goes on:

"Completion of the annual PM crane inspection activities in August 2005 resulted in the EATL being adjusted. The 'as-found' condition was not recorded at that time. In a telephone interview the vendor representative, who was here in August, indicated the as found condition on August 5th was '…well over 200 ft-lbs.' The acceptable setting range of the EATL is 168-204 ft-lbs. With the acceptance criteria not met, the vendor, with the assistance of an inexperienced plant repair person, reset the EATL. The plant PM procedure and the referenced section of the vendor procedure procedure did not contain steps to reset the EATL. The vendor considered the activity to be routine as he had done it several times. Once the EATL had been set within the acceptance criteria, at 175 ft-lbs, the vendor did not proceed to recheck the torque setting as the procedure known to him did not require it, and in his experience, it was not required to verify the setting. According to the vendor, due to the torque imparted on the reduction gearing from the main hoist motor shaft to the hoist drum it requires two to three workers, working in unison on the torque wrench and motor hoist brake, to prevent kickback of the wrench in accomplishing this task."[emphases added](26) Note that only two workers, not three, each with significant gaps in their knowledge of proper procedure, were assigned to the job.

NMC concluded:

"Investigation into the cause for the EATL not being set at 175 ft-lbs, as was recorded in August, identified that plant procedure direction for checking the EATL setting was inadequate and that no direction existed for adjusting the EATL setting. The Plant's Administrative Service Management procedure had not been implemented to ensure a plant staff member understood the activities that the vendor was performing, and that the vendor was made aware of the plants (sic) process and expectations. Other factors also influenced the cause of the EATL being set incorrectly including: The word orders contain deficiencies. Work proceeded beyond what was detailed in the procedure. The plant staff was not knowledgable of the crane components and has relied on the vendor to complete the annual inspection activities. Additionally, error precursors including high heat and humidity and time pressure influenced the outcome."[emphasis added](27)

NMC reports that "A crane vendor representative noted that the EATL is the last-line-of-defense for overload." And the explanation for the 43 hour dangle before the cask was lowered back down to the pool floor included: "Several factors contributed to this longer than expected time including a general lack of knowledge related to L-3 operation and components related to its single-failure proof design…The system engineer and backup system engineer, who were involved with the shop testingt and acceptance testing

of the crane in 2002 and 2003, are no longer employed by NMC. Ownership of the crane in Outage Management is with its second owner since the beginning of 2004. Maintenance support is dependent on vendor support from Ederer Inc."[emphasis added](28)

A number of follow-on mistakes were made. The as-found EATL setting of "well over 200-pounds (sic)…was not recorded in the [Work Order], not was it entered into the corrective action system…Additionally, another Ederer representative brought to the plant on 10/12/05 agreed, based on his experience that repeating the EATL check is needed to verify its setting. A post maintenance test (PMT) of the EATL, as was specified in the work order, required a verification the EATL works properly. To complete the PMT a validation test should have included a retest of the EATL torque setting. The target set point for the EATL is 186 +/- 18 ft-lbs (168-204 ft-lbs). The work order summary does not maintain a PMT. There is no indication an adequate PMT was done." In addition, "there is no indication in the [Work Order] that the torque wrenches" used to check the EATL setting were properly calibrated.(29)

NMC goes on "The implication in the CAP [corrective action plan] is there is a relationship between setting the EATL and the failure of the diaphragm of the air canister. This is unclear communication that appears to be due to inadequate knowledge of the crane components…It is apparent that the repair of the air canister was believed to be, in part, repair of the EATL. This is not the case. This knowledge deficiency resulted in vague direction in the [Work Order]."(30)

Again referring to the faulty setting of the EATL in August 2005, NMC reports "No provisions were made fro placekeeping or step signoffs on the vendor procedure and neither were used to denote steps were completed…Since there is no specific direction in either of the [Work Order] documents the conclusion is that the EATL was set only by the experience of the vendor. Work outside the direction of a procedure or work order is not within our processes…the FIN repair-worker assigned to assist the Ederer representative on August 5 identified that he was assigned the…task…but had no knowledge of the crane brake operation as he had not previously worked on the equipment. This was a first time task for the FIN repair-worker. He sat down with the vendor and questioned what the task was, and got some understanding of the job, but was dependent on the vendor for direction.(31)

Citing more crane vendor errors both before and during the cask dangle incident, NMC reports "Deficiencies are evident in meeting all these service manager administrative requirements in August [2005] and additionally in March and during the latest vendor assistance from Ederer on October 12 and 13."(32)

The crane vendor and Palisades communicated poorly prior to the August 5, 2005 EATL setting. "This PMT step may not have provided clear direction to the vendor who was unfamiliar with the plant process requirements…This lack of clear direction points to a knowledge deficiency of the EATL operation…There is no indication the checklist was reviewed on August 5th when the EATL was reset. Interviews have indicated the PJB [pre-job briefts] was minimal on August 5…The as found condition of the EATL, per

interview with the vendor, was the setting '…was well over 200 ft-lbs.' This exceeded the acceptance criteria…Work activities did not stop at this point. The 'as-found' condition was not documented in the work order nor was an action request initiated. Work proceeded under the direction of the vendor representative to reset the EATL…this was outside the [Work Order] and procedure guidance that was directing the work activity."(33)

Heat, humidity, stress, and the desire to finish the job as soon as possible in order to leave for vacation contributed to the errors:

"The temperature and humidity up on the crane trolley near the ceiling was said to be very hot and it was humid…The vendor indicated it was hot and that he perspired heavily while working on the crane trolley that day. There were no ice-vests worn. The FIN repair-worker indicated it was extremely hot and humid and protective clothing was an issue from a heat stress standpoint. The personnel involved did not initiate any actions to address heat stress that would have addressed ice vest requirements or stay time restrictions if any were needed…The vendor representative had been onsite for two days prior to commencing work on the crane. On August 3rd the crane was not in the correct spot to access and work was stopped that day. On August 4th the failed diaphragm, in the air canister that failed on March 17, was replaced. On August 5th work was done on the crane that included replacing the rebuilt air canister and checking and resetting the EATL with the vendor representative. Additionally, on Friday August 5th the vendor had a plane reservation to return home and it was perceived he was anxious to leave on time as his vacation was to begin."[emphases added](34)

NMC admits that "Dependence on vendor experience due to plant staff lack of knowledge may be prevalent in other cases."(35)

Incredibly, "Because BRP [Big Rock Point] is not an operating unit it did not submit any operating experience to this issue." This despite a cask dangle at Big Rock on an Ederer crane on Nov. 6, 2002. In addition, "On August 15, 2003 while attempting to lift the BRP Reactor Vessel and place it in a shipping container the Containment Building Crane, an Ederer X-SAM single-failure-proof crane, malfunctioned."(36)

Insuring that the public could not demand that industry learn from these repeated mistakes, NMC reports that "There was no report required to the NRC."(37) This, when false fire alarms and plant management personnel changes are required to be reported to NRC!

Internal NRC emails show that NRC officials were aware of the potential of a cask drop. Magdalena Gryglak wrote to Jamnes Cameron on 10/11/05 that "We were just briefed on the potential cask drop…Based on some older documents, before the crane was upgraded to single failure proof crane, the licensee determined that if the cask were to be dropped, there would be significant damage to the pool and flooding could result…" It seems that the removal of the impact-limiting pads from the bottom of the pool, mentioned above, would only make such pool damage worse.(38)

An NMC document reveals that "All unnecessary personnel were removed from the Spent Fuel Pool Floor," during the cask dangle.(39)

NRC's Mary Jane Ross-Lee, emailing Eric Benner and Carla Roque-Cruz "Re: Palisades," on 10/11/05, shows that Palisades cover up of the incident begin immediately: "the licensee was trying to find out if this event is reportable or not." It seems in NRC's estimation it was not, given how they helped the company keep the public in the dark for months on end. And, despite Ross-Lee assuring her NRC colleagues that "the cask has its own source of cooling," Nuclear Management Company felt the need to "Take temperature of the cask every 4 hours," as well as to "Collect sample and determine boron concentration of SFP [spent fuel pool] and cask every 48 hours...[and] Develop plan to sample cask for Boron concentration." Apparently, NRC's exemption on boron concentration safeguards granted just five days earlier lowered safety margins during this cask dangle incident. NMC also initiated evaluating "the need to whether to refill cask with Spent Fuel Pool Water," apparently to insure adequate boron concentration to prevent nuclear criticality, and cool enough water to prevent waste fuel overheating. Other precautions were taken as well, such as ordering that "No Continuous Work allowed in Auxiliary Building." NMC also had personnel "Analyze Worst Case Condition – Dropped Load" and prepared a "L-3 [crane] Contingency: Preparations for Potential Damage w/Heavy Load Drop."(40)

In order to ensure that they could observe the lowering of the cask once the stuck crane was addressed, NRC officials made sure that all shifts at Palisades would be covered by an NRC official even throughout the wee hours of the night. However, they failed to report the incident for many months, keeping the public in the dark.(41)

Internal NRC emails also expressed concern about boron concentrations. "They should continue to monitor pool boron concentrations…and maintain provisions to identify, mitigate and terminate the consequences of a boron dilution accident as required by our exemption…Keep us informed, especially if there are any plans to add makeup water to the cask." A boron dilution accident could occur if unborated water were added to the pool or dry cask in order to maintain cooling. The risk, however, would be that the unborated water would provide sufficient neutron moderation that a nuclear chain reaction could occur in the still-fissile waste.(42)

It is not entirely clear why NRC reported the cask was "13 feet off the pool floor" during the dangle. Most pools are around 40 feet deep, so the bottom of the cask in that typical situation would have been 29 feet above the pool floor. The higher above the pool floor, the more force the cask would have delivered to the pool floor if dropped. NRC patted itself on the back, saying "The Region-based inspectors and the resident inspectors have been working very well together to provide coverage of this issue." Did they mean "covering up" of this issue, because they kept it quiet for months.(43)

On Wednesday, March 29 the Cook nuclear power plant dropped a 35 ton missile block 15 feet onto the reactor cavity floor. Again, NRC held that the incident was not

reportable. It would not have been reported by NRC until the next quarterly inspection report about Cook. But an anonymous source notified Dave Lochbaum of Union of Concerned Scientists. Lochbaum wrote to the NRC Region III Office of Public Affairs:

"Good Day:

An industry colleague informed me about an incident that happened recently at DC Cook Unit 2 that's making the rounds inside the industry. I find zero information about it on the NRC's website.

Can you confirm any or all of the following:

1) On March 29, 2006, a heavy load was dropped at Unit 2 during refueling.

2) The heavy load was a 35-ton missile shield.

3) The load dropped onto the reactor cavity floor.

4) The load was dropped either because of a rigging problem or a crane failure.

5) A "stop work" was issued by the company in response to the incident.

6) NRC Region III has had more heavy load drops than any other NRC region in the past 12 months."(44)

It is still not clear how much damage was done to the Cook nuclear power plant by this heavy load drop.

In conclusion, with the 20th anniversary of the Chernobyl nuclear catastrophe approaching on April 26th, 2006, it is very sobering to realize that Palisades came all too close to a catastrophic radiation release due to near-drop of a 107 ton cask onto its waste storage pool floor. And that the Cook nuclear power plant, just 30 miles south of Palisades on the Lake Michigan shoreline, actually did drop a heavy load near its reactor vessel, with as-yet incomplete damage assessments.

When combined with the near melt down at Davis-Besse nuclear power plant near Toledo in 2002, it seems that by the grace of God, or by sheer luck, the Great Lakes region has dodged a Chernobyl-scale catastrophe on its very shores.

References:

(1) The NRC quarterly inspection report, covering October 1 through December 31, 2005, is entitled "NRC Inspection Report 05000255/2005012" and is available upon request from Kevin Kamps, NIRS, 301.270.6477x14.
(2) Document available upon request from Kevin Kamps, NIRS, 301.270.6477x14.
(3) NRC NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," published February, 2001, Executive Summary, page Roman numeral x.

(4) DOE's Feb. 2002 Final Environmental Impact Statement for Yucca Mountain is viewable online. See Table A-7, "Proposed Action spent nuclear fuel inventory," and Table A-8, "Inventory Module 1 and 2 spent nuclear fuel inventory," at http://www.ocrwm.doe.gov/documents/feis_2/vol_2/apndx_a/index2_a.htm Calculations, based upon the above tables, showing Palisades' current waste pool inventory available upon request from Kevin Kamps, NIRS, 301.270.6477x14.

(5) Robert Alvarez, Jan Beyea, Klaus Janberg, Jungmin Kang, Ed Lyman, Allison Macfarlane, Gordon Thompson, and Frank N. von Hippel, "Reducing the hazards from stored spent power-reactor fuel in the United States, Science & Global Security, Vol. 11, No. 1, 2003, page 1. See the report at: http://www.princeton.edu/%7Eglobsec/publications/pdf/11_1Alvarez.pdf They report, at page 6: "Inventories of Cs-137 in spent-fuel storage pools. The spent-fuel pools adjacent to most power reactors contain much larger inventories of [Cs-137] than the 2 MegaCuries (MCi) that were released from the core of Chernobyl 1000-Megawatt electric (MWe) unit #4 or the approximately 5 MCi in the core of a 1000-MWe light-water reactor. A typical 1000-MWe pressurized water reactor (PWR) core contains about 80 metric tons of uranium in its fuel, while a typical U.S. spent fuel pool today contains about 400 tons of spent fuel…Furthermore, since the concentration of [Cesium-137] builds up almost linearly with burnup, there is on average about twice as much in a ton of spent fuel as in a ton of fuel in the reactor core."

(6) Alvarez et al., page 6, citing "Exposures and effects of the Chernobyl accident," Annex J in Sources and Effects of Ionizing Radiation, United Nations, 2000, p.472-475. See http://www.unscear.org/pdffiles/annexj.pdf.

(7) Alvarez et al., p. 6, see immediately above at footnote (6).

(8) NRC NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," published February, 2001, Appendix 4, Consequence Assessment from Zirconium Fire; Table A4-7, p. A4-9; Table A4-15, p.A4-15.

(9) NRC NUREG-1437, Supplement 27, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Regarding Palisades Nuclear Plant, Draft Report for Comment, Feb. 2006, Section 2.2.8.5, Demography, p.2-56, citing Nuclear Management Company, LLC, Applicant's Environmental Report-Operating License Renewal Stage, Palisades Nuclear Plant. Docket No. 50-255. Covert, Michigan (March, 2005).

(10) The NRC quarterly inspection report, covering October 1 through December 31, 2005, entitled "NRC Inspection Report 05000255/2005012," p.9.

(11) Hubert T. Bell, NRC Inspector General, "NRC's Regulation of Davis-Besse Regarding Damage to the Reactor Vessel Head," Case No. 02-03S, Dec. 30, 2002, at http://www.nrc.gov/reading-rm/doc-collections/insp-gen/2003/02-03s.pdf .

(12) NRC event report, Event Number 42053, Event Date 10/12/2005, Event Time 17:09 EDT. This condition persisted till 05:00 10/13/2005, or nearly 12 hours.

(13) Letter from L. Mark Padovan, Project Manager, Section 1, Project Directorate III, Division of Licensing Project Management, Office of Nuclear

Reactor Regulation, NRC, to Mr. Paul A. Harden, Site Vice President, Nuclear Management Company, LLC, Palisades Nuclear Plant, dated Oct. 6, 2006, see especially p. 4.

(14) Palisades "L-3 SFP [Spent Fuel Pool] Crane Operation Action Plan," undated.

(15) FSAR Chapter 9 – Auxiliary Systems, Fuel Handling and Storage Systems, Revision 24, Page 9.11-1 of 9.11-26, undated.

(16) Nuclear Management Company, A/R 01000753: Intent of WO Task Exceeded During Troubleshooting, 4 pages, undated.

(17) Pictures of Palisades cask, 7 pages, undated.

(18) NMC, Work Order: L-3 Contingency Dry Fuel Storage 200, 35 pages, undated.

(19) NMC, "Root Cause Evaluation Charter: CAP/RCE0100065901 Crane Operator Heard Loud Noise during Lift with L-3 Crane," 1 page, 10/17/2005.

(20) See Footnotes (4) and (19) above.

(21) See footnote (18) above.

(22) NMC, Manager Sponsor Darrel Turner, "Validation of Cause/PMT Load Test Considerations w/handwritten notes," 1 page, undated.

(23) NMC, A/R 01000980, "Break-away Torque Setting out of Spec (Low) for L-3 SFP Crane," 1 page, undated.

(24) NMC, "L-3 Spent Fuel Pool Suspended Load Recovery," 2 pages, undated.

(25) NMC, "Maintenance History of L-3, Spent Fuel Pool Crane," 2 pages, undated; also see footnote (26) below, p. 14.

(26) NMC, Root Cause Analysis Report, NCE0100065901, "CAP01000659, Crane operator heard loud noise during lift with L-3 crane," 58 pages, undated.

(27) See footnote (26), p. 3.

(28) See footnote (26) pgs. 6-7.

(29) See footnote (26), p. 8.

(30) See footnote (26), p. 9.

(31) See footnote (26), p. 10.

(32) See footnote (26), p. 11-12.

(33) See footnote (26), p. 11.

(34) See footnote (26), p. 12.

(35) See footnote (26), p. 16.

(36) See footnote (26), p. 18.

(37) See footnotes (26), p. 20.

(38) Email, M. Gryglak to J. Cameron, Subject: Palisades crane, 2 pages, 10/11/05.

(39) Action Request Report, Number 01000626, Crane operator heard loud noise during lift with L-3 crane, 1 page, 10/11/2005.

(40) NRC email, and NMC, Event Response Plan: L-3 Spent Fuel Pool Crane, 5 pages, 10/12/05.

(41) Email, J. Ellegood to C. Lipa/J. Cameron/L.M. Padovan, Subject: Schedule for lowering load, 2 pages, 10/12/2005.

(42) Email from Robert Taylor to Mark Padovan regarding Palisades Dry Storage Cask Suspended in Air, 1 page, 10/13/05.

(43) Email, J. Cameron to M. Phillips, Subject: FOR YOUR ACTION: Palisades EDO Bullet, 3 pages, 10/14/05.

(44) Email from Dave Lochbaum, UCS, to NRC RIII OPA, 3/30/2006; Hugh McDiarmid, Jr., "Concrete shield falls in nuclear plant mishap," March 31, 2006.

Exhibit - E

Sept. 8, 1986

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE ADMINISTRATIVE JUDGE HELEN F. HOYT, ESQ.

In the matter of: TOLEDO EDISON CO. and CLEVELAND ELECTRIC ILLUMINATING CO. (Davis-Besse Nuclear Power Station, Unit I)	))))))	Docket No. 50-346

DAVIS-BESSE HEARING ON SITE DISPOSAL OF SLUDGE-RESIN LOW LEVEL RADIOACTIVE WASTES

FINDING OF FACTS:

1. In 1968-9 TEC traded the small Darby Marsh for the 954 acre Navarre Marsh.
2. TEC did not inform the Department of the Interior that the Navarre Marsh was part of the Ottawa County Wildlife Refuge Sanctuary and an International Bird Flyway.

3. TEC promised to use about 200 acres of the marsh for the Davis-Besse Nuclear Power Plant and to preserve the rest of the marsh as a wildlife refuge, although TEC already had plans to build 5 reactors on this site along Lake Erie shores.

4. TEC also promised at the construction license hearing that there would be no waste disposal on the site. TEC's operating license (Docket No. 50-346---NPF-3) speaks of the reprocessing of spent fuel and of closed plans to package and ship low level wastes to a licensed landfill within 300 miles. It also states there would be no effluent on site. Again these waste plans are stated in the Davis-Besse FSAR-1973 and also the FES-co-1973.

5. In 1979 TEC had applied for Limited Advance Work Authority for Units II & III to be built just south of Unit I. These were cancelled in 1981, but account for two of the assortment of ponds on the site. TEC's borrow pits, numerous ponds, and drainage ditch have a run-off into the Toussaint River, which empties into Lake Erie. We wonder if the settling basin does too.

6. In 1985 we learn from the Federal Register (Oct.9, 1985) that TEC planned a shallow site burial of its sludge-resin waste from its settling ponds--also that TEC already had received NRC permission since it was ruled as having no significant environmental impact. This arrangement with the NRC was made by correspondence and telephone calls. Again there was no mention of the special environmental issues. Ohio people were also shocked to learn that there was a settling pond on the Davis-Besse site already authorized and operating.

7. Widespread objections of countless Ohio organizations and individuals to this crude site disposal plan led to Governor Celeste's request that the State of Ohio be an intervenor in the hearing through the Office of the Attorney General and the Ohio Department of Natural Resources.

8. The hearing was scheduled for the last week in June-1986, but TEC was granted a delay because of a strike among its employees. Since the Davis-Besse plant had not been operating for over a year following the June-1985 serious malfunction, people thought it a bit odd that employees would be striking for fringe benefits at this time. The hearing was rescheduled for the first week in August with a tour Aug.4 of the Davis-Besse site for the intervenors.

Coalition for Safe Electric Power

9. The tour included a view of the incredibly primitive settling ponds and of a field marked with red-flagged stakes to indicate the general location of the burial site. Intervenors were provided no view of the dikes nor were they clearly labeled on the map. NRC's Judges Hoyt and Kline greeted the intervenors.

10. Ohioans objections to the site burial were many:
   a. Danger of further contamination of Lake Erie after 25 years of cleanup efforts.

   b. Danger to millions who depended on Lake Erie for water supplies and to northwestern Ohioans whose essential ground water might be contaminated.

   c. Endangerment of Ohio's environmental resources, since this area had many profitable industries which might be adversely affected--especially the recreational, such as camping, sport-fishing and boating.

   d. Destruction of state wildlife preserves from loss of vegetation and loss of wildlife habitats for many native species, including several birds on the endangered species list. Endangerment of migrating birds. Radiation in the food chain.

   e. Destruction of fish life since their spawning reefs are off-shore in this area. Reduction and contamination of fish with losses to the very profitable fish industry.

   f. Shock that the NRC would consider a marsh or floodplane area suitable for a low level waste dump and crude settling ponds.

   g. Disgust that TEC is now pretending that the Navarre Marsh is only 400 acres east and north of the Davis-Besse site.

11. Our intervening group had been informed by area fishermen that the Davis-Besse dikes, after years of pounding waves, were in very deteriorated condition. We made that one of our contentions. Fishermen later informed us that TEC had repaired and upgraded the dikes during the summer, before the hearing but after our contentions had been received.

12. The first day of the hearing Mr. Anthony Celebrezze, Ohio Attorney General, and Mr. Joseph Sommer, Director of the Ohio Department of Natural Resources, made very effective presentations of the reasons for Ohio concerns. Then Mrs. Virginia Aveni, Deputy Director of Ohio EPA explained the responsibilities of Ohio EPA with respect to TEC's plans.

13. The main issues debated in the hearing were naturally:
   a. The unsuitable geology and hydrology of the Davis-Besse site for a low level shallow waste burial. The danger of both lake and groundwater contamination.

   b. The local threat to public health and the effect on the environment---the wildlife refuge, migrating birds, the loss of commercial enterprises and the loss of state recreational areas.

   c. The destructive effect of frequent violent storms, winds and flooding, and their effect on erosion of the burial site and consequent migration of its hazardous contents.

   d. The chemistry and durability of the resins used to absorb the radionuclides from the secondary demineralizing system. The possible reaction of other substances with the resins (Ex. kiln dust) and release of radionuclides into the environment.

   e. The lack of testing and necessary specifics to assure safety in the construction and operation of the burial cells.

The State of Ohio was represented by 3 Deputy Attorneys General and 4 witnesses from the Ohio Department of Natural Resources, whose work was very clearcut and effective. TEC had 13 witnesses- all well rehearsed- including several from its own staff. For most of the public present this was a first experience with an NRC hearing.

14. The State contended that TEC's geology studies were quite inadequate and revealed a limited understanding of indications of soil types, permeability, and water flow patterns at the Davis-Besse site. State also thought TEC should have made a thorough hydrology study.

15. Mr. Van Kley busily pried details out of TEC's witnesses bit by bit until Mr. Hendron admitted that TEC's geology findings and hydrological observations were done in 1970 relating to the construction of the Davis-Besse nuclear plant.

16. The State testified that there had been a major advancement the last decade in both knowledge in the fields of both geology and hydrology and understanding of soil indications. Greatly improved instruments and equipment were in use with advanced techniques. Also the process of deep excavation in the past usually smeared evidence of sand and gravel layers, of cracks, of soil permeability, and of tiny water flow pathways. Bore logs were frequently deceptive where parts of the core were missing.

17. TEC contended that State's observations and references to studies of experts made their statements generic rather than specifically applicable to the Davis-Besse area in both geology and hydrology.

18. State pointed out the similarity of till, glaciolacustrine, clay and sand patterns of soils for the whole Great Lakes area, and especially for Ottawa County with its widespread marsh areas. State reviewed evidence of early glacial movements in soil patterns. State contended there was an upper till aquifer which, when saturated, drained into Lake Erie, the Navarre Marsh, and the Toussaint River. Also State cited indications of drainage pathways-some lateral and then down into the ground water and bedrock lower aquifer. Mr. Pavey insisted that by all indications, the water in the glacial sediments connected to the bedrock--that the fluctuations were there. State cited "The Soil Survey of Ottawa County" by Gordon and Huebner in support of its findings of cracks, fractures, thin seams, lenses, and former tree root flow paths (from the early forests) to account for drainage down to the ground water aquifer from the till above. Even one of TEC's own boring roles (B-125 from 1974--ATEC Assn.,Inc.) documented the presence of sand layers.

19. TEC insisted there was no upper aquifer and that layers of till and glaciolacustrine lacked any permeability. State cited inadequate TEC testing and the deceptiveness of faulty bore cores with parts missing.

20. State contended that the whole of northwestern Ohio depended on the same ground water bedrock aquifer system, which included the entire Ottawa Marsh area. TEC stated that well water in the Davis-Besse area was unpleasant from sulfur content. Several systems for sulfur removal were said to be successfully in use among those dependent on the well water.

21. Both sides agreed the limestone-dolomite bedrock was highly permeable. Also the ground water levels were responsive to weather, seasons, lake levels, river levels, and marshlands. When high northeast winds raised the Lake Erie water levels at the west end, the ground water levels also rose. After the storm, the flow of both was gradually reversed. TEC verified the extent of the groundwater system and its permeability from the wide radius affected by its dewatering procedures in early 1970's. State observed that ground water was released into Lake Erie through the permeable

bedrock that extended out into Lake Erie.

22. State challenged TEC's report of the average frequency of flooding incidents in the Ottawa area. When the number of incidents was averaged over a 25 year period, the problem appeared less menacing. This method hid the fact that with the present unprecedented high levels of Lake Erie, about 3/4 of the flooding episodes had occurred during the preceding 6 or 8 years. In other words, flooding had markedly increased with high lake levels. Prognostications about future lake levels differed. State's source anticipated a gradual continued level increase through 1994. TEC's witnesses are sure that lake levels will recede back to normal average levels by 1994. Time will tell. The Davis-Besse site was flooded in both 1972 and 1973. Also following 4 northeast windstorms between 1968 and 1973, the Davis-Besse shoreline receded 20--60 feet. Shore property erosion is of major concern these days.

23. TEC plans 6 burial cells--a plan presented just before the hearing. The first is a triple cell unit, next a double unit, and finally a single cell about 162 by162 ft. The cells are southeast of the plant and close to the easterly marsh, and not far from the Toussaint, which empties into Lake Erie. TEC plans to fill one cell every 5 years. Each will be shaped like a square bowl with a 4 ft. liner composed of 2'6" of compacted clay, then a plastic membrane liner, plus a foot of a leachate collection layer topped with 6" of clay. The sloping sides will be supported by a clay dike with rip-rap. The burial depth will be about 8 1/4 ft. to hold approximately 34,000 ft.^{3} of the 5 year accumulation of sludge-resin waste. The topper will be 4 ft. of clay in the center tapering in all directions to 2 ft. at the edges and overlapping the top of the outward sloping dike. The waste is to be mixed with cement kiln dust, which should harden the waste to compact clay consistency.

24. We don't know how the wastes will be pumped out of the settling pond. Or what or how they'll dispose of the water that comes with it. We don't know how they plan to mix 34,000 ft.^{3} of sludge with cement kiln dust or decide how much kiln dust is needed. We don't know the kind or thickness of the membrane liner or whether it will split under the weight of a bulldozer or backhoe--even with large rubber tires. We don't know whether the clay liner will hold up either or whether it will crack when dried out.

25. TEC estimates 8 or 9 days to transfer the waste from settling basin to burial cell and mix it with kiln dust. Then the waste will be exposed for a couple days to the weather while the cap is being applied. We don't know what the chemical reaction will will be between the kiln dust and the resin or how much radiation could be released from the resin to migrate about the area. We don't know the effect of ground or water-contained chemicals reacting with the resin. We don't know how long the cell will remain intact under winds, storms, and flooding. We're not sure about the impulses of furry burrowing creatures. We don't know whether TEC is creating another "BATHTUB EFFECT." Ohioans would be grateful if the NRC would call a halt to it.

Northern Ohioans would be eternally grateful also if the NRC would take a good hard look at the even more hazardous primitive settling basin on the Davis-Besse site to see if they honestly still think "no significant environmental impact."

Sept.8,1986
Original received by
Judge Helen F. Hoyt
Sept.8,1986
by Federal Express

Respectfully submitted on behalf of Intervenors:
Consumers League of Ohio
Save Our State from Nuclear Waste
Arnold Gleisser
Genevieve S. Cook

Genevieve S. Cook

CERTIFICATE OF SERVICE

Judge Helen F. Hoyt, Esq.
Administrative Judge
U.S. Nuclear Regulatory Commission
Atomic Safety and Licensing Board
4350 East-West Highway---4th Floor
Bethesda, Maryland 20814

Jay E. Silberg
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N.W.
Washington, D.C. 20036

Charles A. Barth, Esq.
Office of Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket & Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Ohio Assistant Attorneys General
Jack A. Van Kley
Sharon Sigler
Edward Lynch
Environmental Enforcement Section
30 East Broad St.
Columbus, Ohio 43215

Terry J. Lodge
618 North Michigan St. Suite 105
Toledo, Ohio 43624

Genevieve S. Cook