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DOCUMENT NO. WEP04-08 **REVISION NO.** 0 **DATE:** June 4, 1996

PREPARED BY: T. J. Wenner **PAGE 1 OF 6**

TITLE: Action Plan To Address Fuel Loading Incident at Point Beach

Reviewed and Approved By: T. J. Wenner / for BAC 6/4/96
B. A. Chechelnitsky, Engineering Manager — Date

Reviewed and Approved By: T. J. Wenner 6/4/96
T. J. Wenner, Vice President Operations — Date

Reviewed and Approved By: [signature] 6/4/96
G.N. Dixon Jr., Vice President, Quality Assurance/Control — Date

Reviewed and Approved By: Art McS 6/4/96
A. J. McSherry, President — Date

FOR INFORMATION ONLY

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DUMS96-322 PDR

1.0 Purpose

The purpose of this document is to establish the actions to be completed during the investigation of the fuel loading incident of a VSC 24 MSB at Wisconsin's Point Beach Nuclear Plant on May 28, 1996. SNC will work with WEPCO personnel and will share all information collected and/or developed.

As this incident has been determined to be potentially reportable in accordance with 10CFR Part 21, Corrective Action Request CAR 96-07, has been issued for processing in accordance with QAP 15.2.

2.0 Reference(s)

2.1 Corrective Action Request CAR 96-07

3.0 Responsibility

The Vice President of Operations shall be responsible for ensuring implementation and completion of this action plan.

4.0 Schedule

The planned schedule for completion is June 30, 1996.

5.0 Documentation

All actions shall be documented in the form of written reports. Where analysis activities are required they shall be completed and documented in accordance with SNC's Design Control Procedure QAP 3.0.

6.0 Action(s)

6.1 Send letters to VSC clients formally advising them of the incident and advising them that their procedures should include venting of the air space in the MSB prior to performing any activity such as welding, cutting, grinding etc. on the shield lid.

6.2 Research the design process and data used to select the carbo zinc coating. Assemble a package containing this information for review/discussion with the NRC and clients.

6.3 Assess the potential sources for generating hydrogen in the MSB.

6.3.1 Reaction of carbo zinc with pool water

a. Determine how carbo zinc reacts with pool water, i.e. what chemical compounds will be produced.
b. Calculate the estimated rate of hydrogen generation.
c. Calculate how long it would take (at the calculated rate of hydrogen generation) to reach the hydrogen concentration in the air space necessary to cause ignition.
d. Compare this time with the actual time between shield lid placement and the time that the incident occurred.
e. Assess if the carbo-zinc reaction makes sense as the primary source (or a contributing source) for the hydrogen in the MSB.

6.3.2 Radiolysis

a. Estimate the hydrogen generation rate from radiolysis within the MSB.
b. Calculate how long it would take (at the calculated rate of hydrogen generation) to reach the hydrogen concentration in the air space necessary to cause ignition.
c. Compare this time with the actual time between shield lid placement and the time that the incident occurred.
d. Assess if the radiolysis makes sense as the primary source (or a contributing source) for the hydrogen in the MSB.

6.4 Evaluate observations from the incident and their possible consequences.

6.4.1 Evaluate effect of chemical reactions on boron concentration in the MSB.

a. Determine if boron concentration is reduced as a result of carbo zinc reaction.
b. Determine solubility of resultant compounds.

c. Determine effects, if any, on criticality analysis.

6.4.2 Evaluate effect of incident on fuel assemblies.

a. Visually inspect assemblies.
b. Determine loads (pressure) experienced by assemblies.
c. Evaluate effects of pressure on assemblies.

6.4.3 Evaluate white substance found in pool after MSB was unloaded.

a. Determine what substance is composed of.
b. Determine how it was produced.
c. Evaluate the effect/consequences of its presence.

6.5 Assess the effects of this incident on previously loaded casks.

6.5.1 Assess potential for continued hydrogen generation in casks that have been drained and dried.

a. From residual water.
b. From any residue generated during loading.

6.5.2 Evaluate possible formation of precipitate during the loading process from the pool water/carbo zinc reaction.

a. Determine if precipitate will form.
b. If so, determine which compounds are formed.
c. Determine effect on MSB performance.

6.5.3 Assess any residual effects from possible burn in previously loaded casks.

a. Evaluate effects of combustion products/residue on MSB performance.
b. Evaluate possible coating damage and effects on MSB performance.
c. Evaluate pressure stresses.

6.5.4 Evaluate the possibility and consequences of hydrogen contamination of shield lid weld.

6.6 Develop actions to preclude similar incidents.

6.6.1 Loading procedures

a. Use previously successful loadings at Palisades as a base.
b. Identify measures taken by Palisades which precluded a similar incident.
c. Investigate possible measures to prevent similar incidents.
   - Increase amount of water pumped out prior to welding.
   - Create negative pressure in air space prior to and during welding.
   - Check for combustible gas prior to and during welding.
   - Others as appropriate.

6.6.2 Design improvements - future casks

a. Investigate ways to mitigate hydrogen generation.
   - Possible installation of hydrogen "getters".
   - Investigate alternate coatings or application of a suitable topcoat.
b. Investigate ways of decreasing hydrogen generation rate.
   - Changes in paint procedure.
   - Possible acid wash.

6.6.3 Advise clients/NRC of actions necessary to prevent a similar incident.

6.7 Evaluate the suitability for future use of the MSB involved in the incident.

6.7.1 Structural integrity

a. Calculate the pressure that the MSB was subjected to.
b. Evaluate the effect of this pressure on the MSB.

6.7.2 Effects of possible residue

a. Check MSB for presence of residue.
b. Remove residue or evaluate to confirm that its presence does not affect MSB performance.

6.8 Complete 10CFR21 evaluation

6.8.1 Determine whether or not incident is reportable under 10CFR21.

6.8.2 Advise NRC and clients of the conclusion from this evaluation and the basis for this conclusion.

6.9 Assist utilities with evaluations required to address confirmatory action letters.

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TELECOPY TRANSMISSION NOTICE

TO: Roy CANIANO
Team Leader, VSC-24

TELECOPY NO: (414) 755-4374

DATE: 6/4/96

FROM: JOHN JANKOVICH
NRC Inspector

TELECOPY NO: (408) 438-5206)

SUBJECT:

PAGES: 6 + Cover

ROY,

- This is Sierra's Action Plan
- For your info re. design review meeting tomorrow.

J.J,

B/28

June 5, 1996
Note
For: Roy Caniano
AIT, Team Leader
From: John Jankovich
Inspection Team Leader
At Sierra Nuclear, Corp.

John P Jankovich

Subject: SUMMARY OF INSPECTION FINDINGS

The inspection at Sierra Nuclear, Corp., conducted between June 3-5, 1996 addressed the following areas of engineering design:

1. Classification of components in accordance with importance to safety
   - Procedure to address 10 CFR 72.144(b) was (and still is) insufficient
     - Has two categories only
     - Lacks criteria for classification
   - Coating was incorrectly classified as not-important-to-safety

   SEE DETAILS IN ATTACHMENT 1, ITEM 1.

2. Coating material: design basis and engineering analysis

   SEE DETAILS IN ATTACHMENT 2

3. Cleaning
   - ANSI Std. N45.2.1-1973 is specified by SNC
   - Classification "Class C" (called Level C) is incorrect because it permits considerable rust:
     - For carbon steel - whole surface
     - For alloy steel - 18 sq in/ft sq
   - WEPco representative (Jim Gill) said that they do not permit any rust. This issue must be further inspected at site (i.e., what WEPco specified?)

4. Engineering Design Process
   - Procedure - OK
   - Procedure requirements - not implemented, e.g., design verification missing

   SEE DETAILS IN ATTACHMENT 1, ITEM 4

5. Sierra's implementation of Part 21
   - SNC has an acceptable procedure, QAP-15.2
   - SNC learned of event from NRC on 5/28/96 (Phone call from Marissa Bailey)
   - SNC did not issue initial notification to NRC as required by 10 CFR 21.21(c)(3)
   - SNC considers that they have 60 days to complete their evaluation in accordance with 10 CFR 21.21(a)(1)

   SEE DETAILS IN ATTACHMENT 1, ITEM 2

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6. Sierra's support to licensee
   - SNC issued an action plan on 6/4/96
   - Drafted letter to clients on 6/3/96 advising them on event & procedural changes, final letter was faxed to clients on 6/5/96 predated to 5/30/96
   - Telephone communications with utilities between 5/28-31
   - Sent operations engineer to site at Point Beach
   - Issued Corrective Action Report (CAR 97-06, 5/28/96)
     - Description of adverse condition
     - Recommended corrective actions

   SEE DETAILS IN ATTACHMENT 1, ITEM 3

Attachment 1

1. Why was the paint not considered important to safety?

   Sierra Nuclear has a procedure (QAP-3.3, Specification, Selection and Qualification of Items and Services) that defines BASIC COMPONENT and COMMERCIAL GRADE ITEM. A BASIC COMPONENT is an item procured either as a safety-related item or as a Commercial Grade item which has been accepted and dedicated for safety related service. The COMMERCIAL GRADE ITEM is an item that meets three criteria: not subject to design or specification requirements that are unique to nuclear facilities; used in applications other than nuclear facilities, and ordered from suppliers on the basis of their published product description (e.g., by catalog number).

   In Fabrication Specification for the Multi-Assembly Sealed Basket (Document No. CMSB2-95-001) Section 3.4.10, identifies items not important to safety. The Fabrication Specification specifies the coating material as not important to safety. The inspection team requested Sierra Nucelar to provide the criteria used to assign an important to safety classification. Sierra Nuclear was also requested to provide the analysis documentation used to assign the not important to safety classification to the coating. Sierra Nuclear was unable to provide the classification criteria or the analysis documentation.

   Based on the above, the inspection team identified the following issues:

   1.1 The Sierra Nuclear procedure QAP-3.3 is inadequate to meet the requiremetns of 72.3 and 72.244 because it does not provide criteria for assigning important to safety classifications to components.

   1.2 Classifying the coating as not important to safety does not appear to be correct because (a) the basket itself to which the coating is applied was classified as important to safety, and (b) the hydrogen reaction and the potential impact on spent fuel chemistry could affect safety as addressed in 72.3, "Structures, systems, and components... ."

2. Was their Part 21 evaluation performed? Were we and the customers notified?

   Sierra Nuclear has Procedure QAP-15.2, Reporting of Defects and Noncompliance, that controls reporting defects and non-compliances. This procedure basically duplicates the reporting requirements specified in 10 CFR 21 with modifications for their operations. QAP-15.2 includes requirements specifying a final report submittal and notifying the users and then the NRC within two days provided they are not adequately informed of the situation. According to QAP-15.2 Section 4.1.1, when a potential 10 CFR 21 issue is identified, a Corrective Action Report is issued. Sierra Nuclear provided the inspection team with a copy of the Corrective Action Report (CAR No. 96-07). dated May 28, 1996. CAR No. 96-07 describes the event that occurred at Point Beach and describes

recommended corrective actions. The recommended corrective actions were to terminate all fuel loading activities, conduct a thorough investigation to determine the event root cause, advise all VSC-24 users about the event, and upon concluding the investigation, implement a corrective action. To demonstrate compliance with CAR No. 96-07, Sierra Nuclear provided conversation memos demonstrating VSC-24 user notifications.

Concerning the requirement of 72.21.21(c)(3) for initial notification of NRC, Sierra responded that the NRC called them first on 5/28/96 regarding the event and based on that phone call, they felt the NRC was adequately aware of the situation. Sierra Nuclear also indicated that upon completing a thorough event investigation, a report would be generated and transmitted to the NRC.

Based on the above, the inspection identified the following issue:

2.1 Sierra Nuclear did not notify the NRC of the potential 10 CFR 21 issue. This issue will remain open pending a ruling by NRC management.

3. Was SNC aware of MSB lid welding problems (moisture and blowouts) encountered at Palisades?

The inspection team discussed the occurrence of any other welding problems was discussed with representatives of Sierra Nuclear. Prior to the inspection team discussions, Sierra Nuclear was not aware that any other events had occurred.

4. Was the design process controlled in a manner that could have prevented this event?

Sierra Nuclear Procedure QAP-3.0, Design Control, controlled the design process. This procedure specifies personnel responsibilities, design process controls, computer software controls, drawing controls, design input controls, design verification, design quality assurance, completed design output documents controls, and design output document revisions/changes. The most recent procedure revision (Revision 5) appears to provide a detailed design control system.

The original revision (Revision 0) was issued July 1990. The original was used during the VSC-24 design development. Sierra used the original revision was used to verify whether the design process was controlled.

QAP-3.0, Revision 0, Section 3.1.3 specifies including a cover sheet for signatures by the responsible engineer, checkers, and project engineer. QAP-3.0, Revision 0, Section 3.1.5 specifies the Project Engineer (or the Project Manager) signing the Design Calculation Cover Sheet as the authenticating signature that authorized using the calculations in the design. In addition, QAP-3.0, Revision 0, Section 3.4.2 specifies documenting a method for verifying the design in the Project Plan and 3.4.3 specifies a report documenting the design verification is to be included in the project files.

The inspection team reviewed design calculation package "VSC-24 ANISN Neutron and Gamma Radiation Shielding Models", SNC No. WEP-109-.001.3, dated 2/14/89. The design package did not contain verfiication by either the Project Engineer or Project Manager. As a double check, the inspection team reviewed the project plan (Project No. WEP-01, Revision 02, dated 7/19/90) for the VSC-24 and VSC-17. Neither design plan include a reference to the design verification as specified in QAP-3.0. Sierra Nuclear was questioned regarding the project plan and responded that the design verification was performed through design checking and external design reviews with the final report being the SAR. Sierra Nuclear concluded that the Project Plan did not include this description, but this is what was intended and currently being done.

Based on the above, the inspection team identified the following issues:

4.1 The design control process, as performed previously was weak due to lack of design verification. This includes an authorizing signature verifying that the design calculations were acceptable.

4.2 The project plan did not provide the guidance and details called for by the procedure.

Attachment 2

Items submitted to Dr. J. Jankovich on June 5, 1996 by Dr. C. Interrante, while at SN on inspection.

1. AIT Questions in Item 1.

1.1 Did the SNC design review include a corrosion engineer? Dr Massey answered this on June 3, 1996. No, a corrosion engineer was not involved. Neither was an environmental effects specialist. None of the clients nor team members were concerned about the environment, except for utilities who said that if iron is bare in pools, it could come off and build up on fuel pool filters. Also, iron could somehow go into primary system and become radioactive, as iron 59 or perhaps some other iron isotope."

1.2 Should the designers have anticipated the hydrogen generation in the SFP? "There are no hydrogen recombiners in the pool. There was bubbling in the pool but this was regarded as trapped air, not hydrogen gas evolution. This question of hydrogen generation never came up. Therefore, fire or explosion due to hydrogen were never considered."

2. AIT Questions in Item 2.

2.1 What was the design basis for the paint selection? A set of specifications was developed, referred to as the "Multi-Assembly sealed basket (MSB) Coating Specification." Dr. Massey indicated that these specifications were based upon the experience of SN team members and their understanding of (1) requirements of utilities, (2) pool water, (3) plant SFARs, (4) temperature and pressure and atmosphere [environment], and (5) the SN calculations on radiation. Items 2.1 through 2.6 give the technical requirements. Item 2.7 on thickness is another requirement, but it was not specifically enumerated to me as a design requirement. Dr. Massey indicated further that Carboline had already been tested extensively and was already being used in nuclear power industry, so it was a primary candidate for coating these parts/components. [Carboline 11 does have a 2 to 3 mil thickness recommended for application and this is interpreted by SN as a minimum recommended thickness.]

2.2 What environmental compatibility study was done? Dr. Massey indicated, on June 3, 1996, as follows: (1) Oak Ridge National Laboratories and others had conducted tests of this material. Specifics were not readily available or offered by him at that time. The purpose of the coating was to be prevention of putting iron into solution [and thereby into the SFP system]. No thought was given to zinc as an ion that would be put into solution. (2) Carboline had shown that spray with Boric acid had no effect. See Carboline Data Sheet p2 Analytical...and ORNL report (12-17-96??); (3) Carboline was approved for use in the SFP at Palisades, (4) ORNL did tests (same ORNL Report) on design basis accident (DBA) steam injection.

2.3 How did SN address the paint specification prohibition against the use of zinc? On June

4, 1996, Tom Wenner indicated that "it is interpreted as an indication of long-term, not short-term performance."

3. AIT Questions in Item 8.
Reportedly the paint was qualified at Oak Ridge. What were the qualifications requirements? These are the specifications discussed under Item 2, above.

4. AIT Questions in Item 9.
The SAR, pgs 1-8 and1-9 state the purpose of the paint was to protect fuel pool chemistry." What was meant by this statement?
"The zinc paint keeps the steel components from contacting the pool, so as to prevent the introduction of iron ions into the SFP. All basket internals, as well as the shell [of the MSB], are coated with the Carboline 11 to ensure that this happens. All of these components (tubes, shell and ring bands) are fabricated from a 516 Grade 70 carbon steel [or A36 or an equvalent] in an 'all welded' construction. None of the utilities permit the exposure of iron in the pool. The prohibition on iron was general knowledge from the utility companies and not associated with any one power company"

This is different from important to safety: Art McSherry indicated that the NRC definition of important to safety is what was used to determine that the zinc coating / paint was not important to safety. He indicated that zinc is present solely to protect the pool water chemistry. Whereas, the NRC definition involves the structures [structural components] and the containment boundary, factors that affect criticality, peak fuel cladding temperature and shielding. The paint is involved in none of these items important to safety.

5. AIT Questions in Item 11.
Why does Carboline data sheet prohibit immersion? See answer in Item 2.3 above.

6.. AIT Questions in Item 12.
What are licensees criteria for cleaning before shipping?
"Licensee's procurement specifications related to cleaning are given in Section 3.10 of their Procurement Specification for the Fabrication of the Multi-Assembly Sealed Basket, Safety Classification: Important to Safety, WMSB-94-001, revision 0, November 1994. It requires the following:

Section 3.10.1-- All metal surfaces of the MSB assembly shall have a surface cleanliness of classification of Level C or better as defined in Section 2 of ANSI N45-2.1.

ANSI N45-2.1, indicates "...cleanliness classification for an item shall be specified in accordance with paragraph 3.1 of this standard."

Section 3.1.3, class C is given as "an intermediate level of cleanliness generally applicable to closed service-water systems that cool components containing reactor coolant, engineered safety systems, and other high-integrity systems. Surfaces shall meet

class B except : 1. Thin uniform rust films are acceptable on carbon steel surfaces. 2. Scattered areas of rust are permissible provided that the area of rust does not exceed 15 square inches in any 1 square foot of corrosion resistant alloys. 3. Flush-test filters may exhibit considerable rust staining.

[This level of cleanliness is regarded by the NRC inspectors to be inappropriate, as no bare iron components are permitted in the SFP. Carbon steels are coated with a zinc paint to prevent such iron contact with the pool. As rust is permitted under a Class C cleanliness level, this level permits bare iron, as corroded surfaces of iron. This level violates the requirement that exposure iron to the SFP is unacceptable.]

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TELECOPY TRANSMISSION NOTICE

TO: ROY CANIANO
Team Leader, VSC-24

TELECOPY NO: (414) 755-4374

DATE:

FROM: JOHN JANKOVICH
NRC Inspector

TELECOPY NO: (408) 438-5206)

SUBJECT:

PAGES:

ROY,

• For your use on 6/6/96 as input from the Sierra inspectors

J. J.

6/6/96

ISFSI PHASE II
DRAFT ACTION PLAN

I. Determination and Characterization of the Source of Hydrogen

II. Detailed Inspection of Fuel Removed from Multi-assembly Sealed Basket (MSB) #3

III. Justification for the Continued Use of MSBs #1, #2, and #3

IV. Development of Short-Term Plan for MSB #3

V. Development and Implementation of Inspection Plan for MSB #3

VI. Procedure/Process Change Identification for Loading and Unloading

VII. Safety Analysis Report (SAR) Review

VIII. Review of Weights and Rigging Methods

IX. Review of 72.48s

13/30

Determination and Characterization of the Source of Hydrogen -- (Ed Hinshaw)

1. Preliminary investigation into the source(s) of hydrogen (H_2)
   - Conduct preliminary evaluations into the interaction of spent fuel pool (SFP) water and zinc coating.
   - Determine theoretical H_2 generation rate.
   - Verify that radiolysis is not a significant contributor.
   - Assess the potential for any contribution from RX-277.
   - Verification that Carbo-Zinc 11 was used for Multi-assembly sealed basket (MSB) #3 through review of the vendor painting procedure and quality assurance (QA) records.
   - Take appropriate samples from the MSB prior to, during, and after draindown.
   - Determine whether the H_2 burn was a contributor to precipitate formation.

2. Conduct daily update conference calls with Sierra Nuclear Cornoration (SNC), and coordinate Wisconsin Electric (WE) and SNC evaluations.

3. Conduct discussions with Carboline, and retrieve reference information as necessary.

4. Obtain chemical/structural analysis of floating residue and crud layer on bottom of shield lid.
   - Sample sent to Argonne National Laboratory on 5/31.
   - Use X-Ray crystallography (confirmation of structure).
   - Follow-up as necessary with Inductively Coupled Plasma (ICP) analysis; Fourier Transformed Infrared (FTIR) analysis; Gas Chromatography/Mass Spectroscopy (GC/MS) analysis; Carbon/Hydrogen/Nitrogen (CHN) thermal analysis; Auger electron analysis.

5. Perform necessary WE calculations for contaminants taking into account the following:
   - Effect of residence time of SFP water in MSB on paint structure.
   - Effect of residence time of SFP water on MSB materials.
   - Precipitates

6. Obtain information through discussions (and possibly additional testing) with other users
   - Ventilated Storage Cask (VSC-24) users (Palisades, Arkansas Nuclear One)
     - discuss what they may have experienced that is similar
     - discuss procedure differences that may explain why PBNP experienced a hydrogen burn, but not Palisades
   - Users of other dry storage systems

6/6/96

Detailed Inspection of Fuel Removed from MSB #3 -- (Bill Hennessy)

1. Conduct a detailed inspection of all 24 fuel assemblies that were removed from MSB #3. This is in addition to the initial inspection that occurred when the MSB was unloaded, and will serve to validate those initial results.

2. Each assembly will be placed into the periscope inspection location. All four faces of each assembly will be inspected using a high resolution camera. The following items will be specifically checked during the inspection:

    - Evidence of physical damage to the fuel assembly nozzles, grids, guide tubes, or fuel rods.

    - Accumulation or deposition of the precipitate on any part of the fuel assembly. The top of the top nozzle is to be checked before engaging the fuel handling tool on the assembly.

    - Any unusual stains or marks which may indicate trauma to the fuel assembly, especially near the top. Such stains could include marks, evidence of unusual chemical reactions, or shiny spots on the assemblies.

3. The inspections will be videotaped, and results will also be documented using the appropriate forms.

4. After the inspection is completed, the fuel assemblies will be returned to their original storage locations.

Justification for the Continued Use of MSBs #1, #2, and #3 -- (Mike Baumann)

1. Potential for H_2 generation in MSBs #1 and #2.
   - Evaluate whether the Carboline - Zn/boric acid reaction stops once water is removed.
   - SNC reviewing potential amount of water remaining following vacuum drying and inert gas backfill.

2. Consider possible measurement of H_2 generation following draindown of MSB.

3. Evaluate the possible plate-out of the precipitate on interior surfaces of MSB (retrograde solubility), and the potential for continued H_2 production.

4. Evaluate impact of the effects of the precipitate on the fuel and MSB, considering the following (using expertise of Westinghouse and SNC as appropriate):
   - Evaluation of potential impact of precipitate on fuel integrity.
   - Evaluation of potential impact of precipitate on heat transfer.
   - Evaluation of potential impact of precipitate on criticality.

5. Structural considerations
   - Sargent & Lundy (S&L) performing initial calculation for pressure transient caused by H_2 burn in MSB #3; Stevenson & Assoc. to perform more detailed calculation.
   - S&L performing calculation to determine ASME code allowables for the pressure transient in the MSB.
   - WE to analyze the stress on the MSB due to the event, possible damage, and expected location of the damage. Factor this information into the MSB #3 inspection plan.

6. Evaluating alternative coatings / need for coating.
   - Data on Dimetcote 6 and Everlube 812 & 823.
   - Determine the basis for the use of Carbo-Zinc 11, or equivalent

Development of Short-Term Plan for MSB #3 -- (Mike Baumann, Mike Holzmann)

1. Determine needs for liquid, gas, and solid sampling prior to, during, and after MSB draindown (being finalized).

2. Develop detailed plan for draindown and drying of the MSB (being finalized).

3. Develop detailed radiological control plan for activities related to sampling, draindown, and drying of the MSB (being finalized).

Development and Implementation of Inspection Plan for MSB #3 -- (Pat Keily)

1. Provide dimensional inspection data from the manufacture of the MSB.

2. Incorporate information from the identification of the high-stress MSB locations into the inspection plan.

3. Develop visual and dimensional inspection plan which addresses inspections, acceptance criteria, and records (being finalized). This will include the following:

   - Direct visual examination of 100% of the MSB exterior.

   - Visual inspection of random areas of the MSB interior, as well as areas of highest stress. Use either a video probe, fiberscope, or boiler camera.

   - Magnetic particle examination of any MSB exterior area exhibiting damage to the paint.

   - Dimensional inspection of the MSB shell and shield lid commensurate with the dimensional inspections performed by the vendor.

   - Direct visual examination of the shield lid and drain tube.

   - Liquid penetrant or magnetic particle examination of the drain tube in the area surrounding the threaded connection.

4. Develop plan for ultrasonic inspection method if indicated by engineering analysis.

5. Review inspection results and decide upon further action as necessary.

Procedure/Process Change Identification for Loading and Unloading of VSCs -- (Rick Wood)

1. Coordinate information/procedures with other VSC-24 users. Compare for consistency, and determine best practices for loading and unloading.

2. Review loading and unloading procedures for sections that would be affected by the measures decided upon to eliminate/minimize the risk from H_2 generation. Potential items to consider for these procedure/process changes are:

   - Maintain a vent or purge on the MSB from removal from the SFP until sealed, and determine appropriate cover gas as necessary.
   - Check for flammable gas before welding.
   - Maintain the vent or purge during welding.
   - Update component weights.
   - Evaluate ways to passivate the coating.
   - During unloading, provide a continuous purge until the MSB is completely reflooded.

3. Ensure that new/revised procedures address the first three points of the confirmatory action letter from the NRC.

SAR Review -- (Tom Malanowski)

1. Phase I
   - Assemble copies of all screenings and evaluations performed pursuant to 10 CFR 72.48 requirements.
   - Review screenings and evaluations for VSC generic vs. individual cask-specific changes.
   - Identify VSC SAR descriptions/information altered as a result of the generic change.
   - Determine means to annotate/supplement the information that is revised with the correct information.
   - Prepare and distribute update.
2. Phase II
   - Based on results from the Process Improvement Team for maintaining/updating the PBNP Final Safety Analysis Report (FSAR), determine best means to identify needed changes on an ongoing basis.
   - Determine update frequency and process.
   - Implement changes.

Review of Weights and Rigging Methods -- (Bruce Sasman)

1. Review the load calculations of the components for the ISFSI. The load calculations need to be reviewed to verify that the weights and configurations lifted are the same.

2. Review the rigging purchased to verify that it was purchased for the correct loads.

3. Verify that the rigging used is the rigging purchased for that purpose and as specified in the procedure(s).

4. Verify that all of the above meet or exceed the commitments made for NUREGs 0612 and 0554, and consistent with the response to NRC Bulletin 96-02.

Review of 72.48s -- (Harv Hanneman)

1. Select a three-person team to conduct the required review with knowledge and experience in the safety evaluation process (50.59s and 72.48s). The team has been tentatively selected.

2. Train the team on the design and licensing-basis of the VSC-24 dry cask system, including the SAR. Training has agreed to put together existing training materials, which could be used by the team for some type of self study or other training to be arranged.

3. Assemble copies of all 72.48 screenings and safety evaluations, a list of all Engineering Change Requests (ECRs) for the VSC-24, and the appropriate licensing/design documents and procedures for review and/or reference.

4. The team will review all 72.48s (both screenings and safety evaluations) primarily focusing on identification of problems with internal consistency or "disconnects," but also documenting any noted discrepancies between the 72.48s and other documents such as ECRs or the SAR (to be coordinated with the SAR review by Malanowski). A list of discrepancies will be developed. Each 72.48 will be reviewed by at least two of the three team members.

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO-III-96-033C

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by Region III staff (Lisle, Illinois) on this date.

Facility	Licensee Emergency Classification
Wisconsin Electric Power Co.	Notification of Unusual Event
Point Beach 1 2	Alert
Two Rivers,Wisconsin	Site Area Emergency
Dockets: 50-266,50-301	General Emergency
	X Not Applicable

Subject: UNIDENTIFIED GAS IGNITED DURING SPENT FUEL CASK WELDING (THIRD UPDATE)

The NRC Augmented Inspection Team (AIT) has completed its onsite review of the ignition of hydrogen gas during welding of a shield lid on a VSC-24 spent fuel storage cask on May 28, 1996. The team held an exit meeting June 7, 1996, with the licensee. The meeting was open to public observation.

Following the event, the VSC-24 cask was returned to the spent fuel storage pool and the fuel assemblies were unloaded. The cask was then moved back to the adjacent decontamination area. There was no apparent damage to the cask components or the spent fuel.

Among the findings of the AIT, which were presented at the exit meeting: The source of the hydrogen gas was a chemical reaction between boric acid in the spent fuel pool water and zinc in a coating on the carbon steel interior of the cask. Hydrogen accumulated in an air space under the shield lid and ignited when technicians began welding the lid in place.

There were possible precursor events at Point Beach during the welding of another cask on May 20, including a minor combustible gas burn. Most of the weld had been completed, and welders were grinding a portion of the weld before finishing the weld. During the grinding, the welders observed a small blue flame at the unwelded portion of the gap between the cask wall and the shield lid. This flame lasted about 30 seconds. The welders incorrectly attributed the flame to residue of a cleaning solvent, and the incident was not documented.

A review of video tapes of the cask loading process at Point Beach and also at the Palisades Nuclear Plant, which also uses the VSC-24 cask, showed visible bubbles rising from the grid portion of the casks. The significance of these bubbles, apparently hydrogen, was not recognized at either facility.

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The licensee's response to the event was thorough, and appropriate precautions were taken.

In addition to the AIT inspection at the Point Beach site, the NRC Office of Nuclear Materials Safety and Safeguards conducted an inspection at Sierra Nuclear Co., the cask designer. The separate inspection at the Sierra Nuclear facility determined that the potential reaction between the zinc-based coating and the water in the spent fuel pool had not been recognized by the cask designers.

Confirmatory Action Letters were issued by NRC regional offices on May 31, 1996, to the three utilities using or preparing to use the VSC-24 casks. The letter confirms that, prior to any cask loading or unloading, the utilities will assess the potential for generation of combustible gases and take steps to minimize the potential generation and ignition, and have procedures to respond to a gas ignition.

There has been news media interest in the event and the exit meeting. The State of Wisconsin will be provided with this updated information.

This information is current as of 10 a.m. on June 7, 1996.

Contact: ROY CANIANO (708)829-9904 JACK GROBE (708)829-9701

RALD TIMES REPORTER

JUNE 7, 1996 HOME DELIVERY 41¢ SINGLE COPY 50¢

Nuke plant: 1st fire not reported

Utility officials saw gas burn while loading previous cask

By MARYBETH AJACK
Herald Times Reporter
Staff Writer

TWO CREEKS — A gas fire in a nuclear waste storage cask May 28 at Point Beach Nuclear Plant might have been prevented if Wisconsin Electric Power Co. workers had correctly diagnosed a similar but smaller fire the previous week, Nuclear Regulatory Commission officials said today.

The NRC held a press conference at the Two Creeks Town Hall near the plant.

While loading a cask about two weeks ago workers noticed a bluish-white flame but thought it was leftover cleaning fluid burning, NRC inspection team leader Roy Caniano said. The flame burned for 30 or 40 seconds.

Moisture on the outside of the cask also should have tipped off workers that hydrogen gas had built up inside, he said.

The NRC is taking the May 28 incident "very seriously," Caniano said.

In the May 28 incident the fire, which one independent scientist called an explosion, was strong enough to slightly tip the 1,400-pound steel lid of the cask. The 24 highly radioactive fuel rod assemblies inside the cask were not damaged and no radiation was released, Wisconsin Electric and NRC officials said.

The investigation is continuing, Caniano said.

The second, more serious fire, which led Wisconsin Electric to suspend cask-loading, probably was caused when acidic water from the plant's underground waste storage pool interacted with the zinc lining of the cask, producing hydrogen, NRC officials said previously.

The steel lid was being welded shut when the flame ignited the hydrogen.

Two of the 18-foot-tall casks, which are manufactured by Sierra Nuclear Corp. of Scotts Valley, Calif., have been loaded and installed on a concrete pad about one-quarter mile from the plant on the shore of Lake Michigan.

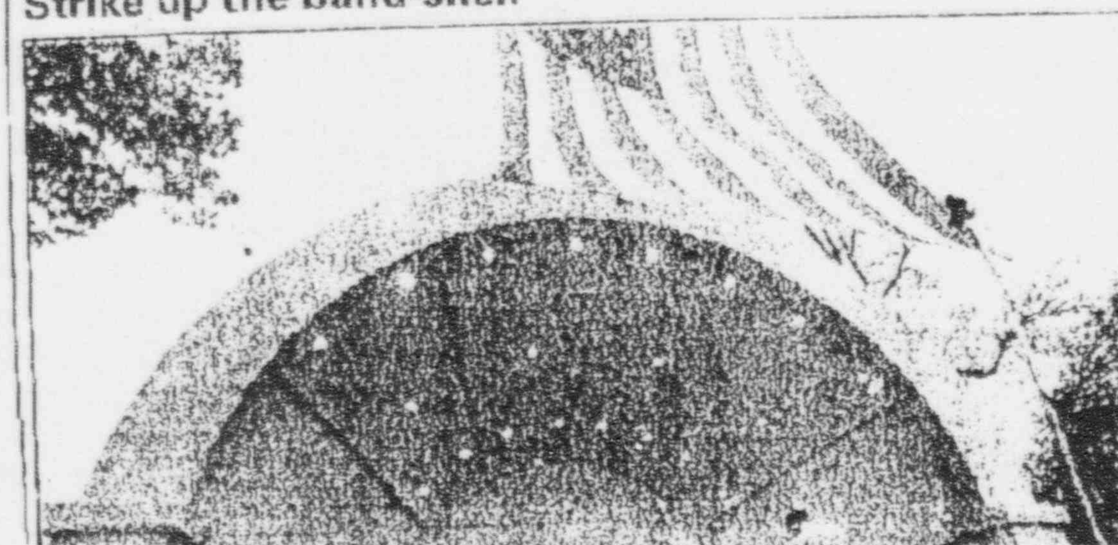
Child care in schools endorsed

By CINDY STONE
Herald Times Reporter
Staff Writer

MANITOWOC — The Board of Education's Curriculum Committee voted Thursday to allow the YMCA to implement before- and after-school child care in one or two of the school district's buildings on a one-year pilot basis.

If the full school board gives its approval and the program is implemented, the school board will evaluate the program after the year, with the possibility of expanding it to

Strike up the band-shell



House approves welfare waiver

By BOB VITALE
Herald Times Reporter
Washington Bureau

WASHINGTON — Final federal go-ahead for Wisconsin's welfare overhaul — and the political points that come with it — has come down to a race between Senate Republicans and the Clinton White House.

After nearly four hours of debate Thursday on the Wisconsin

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POINT BEACH

Utility criticized in fire

MILWAUKEE JOURNAL SENTINEL SATURDAY, JUNE 8, 1996

Wisconsin Electric misread flames

By LEE BERGQUIST
of the Journal Sentinel staff

Two Creeks — Federal regulators on Friday criticized the owner of the Point Beach nuclear power plant, saying that a fire there May 28 could have been avoided had plant personnel correctly assessed the cause of a similar fire eight days earlier.

"There were a number of aspects to this that were disturbing," said Jack Grobe, deputy director of division reactor safety for the Midwest region of the Nuclear Regulatory Commission. "This was preventable."

The occurrence of an earlier fire was disclosed in Two Creeks Friday at a meeting attended by NRC officials and representatives of Point Beach's owner, Wisconsin Electric Power Co. of Milwaukee. Both fires were connected to the loading of spent nuclear fuel rods into steel and concrete storage containers.

The fire May 28 was a brief burn that employees described as a loud pop or two that was powerful enough to upend the 3-ton lid of a fuel container, leaving the lid slightly ajar so that one edge stuck up about an inch.

Opponents of Wisconsin Electric's use of storage containers have called the burn an explosion, although the NRC said it was not.

The fire May 28 occurred when plant crews were welding the lid to the top of a storage container, igniting hydrogen inside, officials said. The same thing had occurred on May 20, but, at the time, the cause of the combustion was incorrectly diagnosed.

In both incidents, no one was injured, and no radiation was released.

Officials of both the government agency and Wisconsin Electric agreed that the cause of both fires can be traced to the interaction of acidic water and a zinc coating inside the container. Mixing the water and zinc creates hydrogen gas, which is highly combustible.

After the incident May 28, the spent nuclear fuel was removed from the storage container and returned to a storage area inside the plant.

The NRC said Point Beach's response to the May 28 fire generally was thorough and that it took appropriate precautions.

However, NRC officials have decided that no more spent fuel can be stored in containers of this design at Point Beach and other affected plants until the utility and others can decide how to avert future mishaps.

Robert E. Link, vice president of nuclear power at Wisconsin Electric, said he had no idea how long that process would take. He estimated that Point Beach has enough spent-fuel storage space left inside the plant to last about a year. The spent fuel remains highly radioactive for some 10,000 years.

Point Beach is located north of Two Rivers on the shore of Lake Michigan in Manitowoc County. It was built in 1970. It is one of only two nuclear power plants in the state, the other one being the Kewaunee nuclear power plant just a few miles north of Point Beach.

Wisconsin Electric is using the containers to store 24 used fuel rods per container because its storage pool in the plant is running out of space. The utility has filled two containers, and officials at the NRC said they feel confident that old fuel in those casks is being stored safely.

The utility, like other utilities across the country with nuclear power plants, is waiting for the federal government to build a permanent repository for spent fuel. A proposed site in Nevada is currently years behind schedule.

Meeting at the town hall in Two Creeks, NRC officials said Wisconsin Electric should have diagnosed potential problems early on. Wisconsin Electric officials did not dispute that claim.

As part of its findings, the NRC said it discovered that crews watched the first burn of a container during the welding on May 20. What they observed was a "small blue flame" in a space at the unwelded portion of the lid that lasted for about 30 seconds, the NRC said in a preliminary analysis released Friday.

"The welders incorrectly attributed the flame to residue of a cleaning solvent, and the incident was not documented."

Also, a review of videotapes at Point Beach and at Palisades nuclear plant in South Haven, Mich., the only two power plants in the United States now using containers of the same design, showed "visible bubbles rising" from inside the container during the loading of the fue[l]

"The significance of thes[e] bubbles, apparently hydroger[n] was not recognized at either fa[cility]."

NRC officials also said the[y] were troubled that the designe[r] of the container, Sierra Nuclea[r] Co. of Scotts Valley, Calif., di[d] not recognize the potentia[l] chemical reaction between zinc based coating and acidic wate[r] from the spent fuel pool tha[t] stays in the container until it i[s] welded shut.

After it is welded shut, th[e] water is discharged and the con[tainer] is filled with helium gas.

Possible fines against th[e] utility will be reviewed later, th[e] NRC said.

The mishap angered som[e] nearby residents. Several at[tended] Friday's meeting and ex[pressed] fear about the safety o[f] the plant, asking how such trou[bles] might affect the value o[f] their properties.

Bob Ozanne of Two Creek[s] told the NRC officials that be[cause] storing spent fuel outsid[e] of power plants is relativel[y] new, "we certainly feel lik[e] guinea pigs."

The Citizens' Utility Board, [a] Madison-based advocacy grou[p] that has fought Wisconsin Elec[tric]'s storage plan for years, sai[d] it was pleased that Wisconsi[n] Electric was taken to task.

"This explosion clearly show[s] that the NRC and WEPC[o] failed to do their homewor[k] when they certified that the cas[k] was safe," said David Merril[l], CUB's executive director. "W[e] can't risk the safety of Lak[e] Michigan and the local commu[nity] based on false claims [of] safety."

JUNE 8, 1996 — HOME DELIVERY — SINGLE COPY

Cask fire 'very disturbing'

Federal official says Point Beach incident preventable

By MARYBETH AJACK
Herald Times Reporter
Staff Writer

TOWN OF TWO CREEKS — The events surrounding the May 28 gas burn at Point Beach Nuclear Power Plant are "very disturbing," according to a Nuclear Regulatory Commission official.

NRC and Wisconsin Electric Power Co. personnel met with the public Friday at Two Creeks Town Hall to discuss the gas burn at the utility's nuclear plant. The incident happened during a mechanical welding procedure to seal a cask used to store highly-radioactive spent fuel.

"We should have seen what was going to happen. This could have been prevented," said John Grobe, deputy director of reactor safety for the NRC. "A number of things about this are very disturbing and disappointing."

The NRC is concerned because Point Beach officials did not fully investigate anomalies resulting from the sealing of another cask. They had also figured the wrong weight of the shield lid of the cask. They said it weighs 4,400 pounds, but the NRC discovered it actually weighs 6,39[illegible] pounds.

The cask involved in the May 28 incident was the third of 12 that the state Public Service Commission has given Wisconsin Electric permission to load with spent fuel from the plant's storage pool.

A report by the NRC states the gas burn, which caused the lid to move several inches, was caused when the zinc-based coating on the cask interacted with boric acid in the spent nuclear fuel storage pool. The reaction produced hydrogen that accumulated in an air space under the shield lid and ignited when technicians began welding the lid into place. Zinc and silicon residue was also found inside the tank.

NRC inspector Roy Caniano said a review of tapes of the cask being loaded at Point Beach and another at Palisades Nuclear Power Plant in Michigan, which uses the same type of cask, showed "visible bubbles rising" from a portion of the cask.

"The significance of these bubbles, apparently hydrogen, was not recognized at either facility," he said.

NRC officials determined that no radiation was released during the incident and that the nuclear fuel rods inside were not damaged.

(See *Nuclear*, Page A-2)

HTR
6/8/96

Herald Times Reporter
Manitowoc-Two Rivers, Wis.

HTR 6/18/96

Nuclear

(Continued from Page A-1)

The NRC report is preliminary and any penalties will be issued following the completion of the investigation.

Grobe said Wisconsin Electric chemists and the manufacturer of the casks, Sierra Nuclear Corp. of Scotts Valley, Calif., should have predicted the chemical reaction and the gas burn.

"If methodical evaluation had been conducted, the gas burn would have never happened," he said.

Bob Link, vice president of nuclear power for Wisconsin Electric, said he agreed with the NRC's report and the plant was taking the necessary steps to make sure a similar incident won't happen.

"Our investigation of the burn agrees with what the NRC has said. We share the hindsight on the issue that we should have seen the precursors to this and that it could have been prevented," he said.

Both the NRC and utility investigations showed correct diagnosis of moisture near the cask and a small blue flame during the welding of a cask the previous week could have prevented the May 28 gas burn.

"We had several opportunities to diagnose the presence of hydrogen in the cask and it was missed," Caniano said.

Despite assurances from NRC and Wisconsin Electric officials, several area residents voiced their displeasure with Point Beach's use of a dry cask storage system.

"A lot of us local residents feel like we are being used as guinea pigs here," said Bob Ozanne of Two Creeks. "All of this awful publicity about this blunder has reduced the value of our property. Who would want to buy a home here now?"

Grobe said Point Beach was still extremely safe and denied area residents were being used as "guinea pigs."

"I assure you we are not conducting an experiment here. It's not possible to prevent all mistakes. We're only human," he said.

David Merritt, executive director of the Citizen's Utility Board, said the May 28 incident is a prime reason to close the plant down.

"This explosion clearly shows that the NRC and WEPCO failed to do their homework when they certified that this cask was safe," he said. "We can't risk the safety of Lake Michigan and the local community based on false claims of safety."

The NRC will use the Point Beach incident to prevent gas burns at other nuclear power plants, said Dr. Carl Paperiello, director of the NRC office of nuclear material safety and safeguards. Plants in Michigan and Arkansas also use Sierra Nuclear casks to store spent nuclear fuel.

Point Beach has suspended the loading of casks until all causes of the gas burn are identified and corrected, Link said. He said the plant has enough storage room to operate the plant for another year.

"Once the root causes of this have been completely analyzed, we will continue using the dry cask storage system," Link said.

So far, two casks have been loaded and installed on a concrete pad about one-quarter mile from the plant.

NRC AUGMENTED INSPECTION TEAM EXIT MEETING

JUNE 7, 1996
10:00 AM to 2:00 PM
TWO CREEKS TOWN HALL

Agenda

1.	Introduction	J. Grobe/R. Caniano	10:00-10:15 AM
2.	General Summary	R. Caniano	10:15-10:30 AM
3.	Model/Description of Event	T. Kobetz	10:30-11::00 AM
4.	Team Member Inspection Results	Team Members	11:00-1:00 PM
5.	General Public Participation	J. Grobe	1:00-2:00 PM

Additional Information

NRC Representatives: C. Paparellio, M. Farber, J.Grobe, A. Kugler, J. Strasma, J. Jankovich, T. Kobetz, J. Davis, P. Narbut, R. Paul, C. Withee

WE Representatives: R. Link, G. Maxfield, A. Cayia, J. Olson, G. Krieser

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POINT BEACH AIT EXIT MEETING
PUBLIC QUESTIONS AND COMMENTS

June 12, 1996

1. Fawn Shillinglaw - She mentioned that in her brief for the Wisconsin PSC she asked about a memo that stated that the Carbo Zinc 11 coating was only approved for the first two casks and she expressed a concern about why additional analysis was needed. She also asked these questions in a letter to Andy Kugler in March 1996 and another letter to Allen Hansen in May 1996. She has not received an answer to either letter. She also expressed concerns that changes are being made to the generic cask designs after certification. She asked why no one checked on the coating during the design reviews? She stated she felt like we are experimenting with dry cask storage.

   She also expressed concern that the CAL talks about minimizing problems from hydrogen evolution rather than eliminating them. She stated that hydrogen generation should not be allowed. She also expressed her concern that the response to the CAL and our "formal" review would be done in such a way that the public would not have access.

   Jack Grobe responded that the two weeks mentioned in the CAL is for our review of the licensee response. The response and the results of our review will be placed in the PDR. He also stated that we are not running an experiment; the NRC monitors licensee activities for safety. This particular event had no safety consequences. However, we must learn from events like this and take corrective actions. We will also be looking back at the adequacy of our design review process.

   Carl Paperiello responded that we are already looking at changes to our review process to help prevent a recurrence. We missed the coating interaction with the fuel pool water; we should have caught it. We will also look back at other cask designs for other system/component interactions.

2. Dennis Dums (Citizens Utility Board) - Dennis asked, in relation to the CAL, (1) whether the NRC will formally review the licensee response and issue a formal approval/denial, (2) whether the public will be given an opportunity to comment on the response and review, and (3) if we could guarantee that no documents associated with the CAL prepared by WEPCo would be kept secret. He also expressed a concern that the time allotted for the NRC review of the licensee response was not adequate.

   Carl Paperiello responded that we would review the information submitted by WEPCo and make a written response of approval or denial. The documents associated with this process will be placed in the PDR. Carl committed to allow enough time (two weeks was agreed upon) after the documents are placed in the PDR for the public to look at them. The only exception for the public availability of the documents would be proprietary information.

   Dennis questioned the process for withholding information as proprietary and asked if we expected any of the information associated wit the CAL to be proprietary.

   Jack responded that information submitted as proprietary is reviewed by the staff to ensure it meets the applicable criteria. He stated he did not foresee any proprietary information associated with the CAL, but he could not guarantee it.

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Dennis askeo, in light of this event, whether the NRC should do a complete re-review of the design of the VSC-24 cask before any more are used.

Carl responded that we would not necessarily need to do that and he would not make that decision in this meeting.

Jack responded that we do intend to take a broader look at cask design issues. He indicated we expect the licensees and vendors to do the same.

Dennis asked if we intended to perform a formal review of the placement of the shield lid on the wall in the fuel pool and the placement of the MTC/MSB combination on top of the VCC. He indicated he was concerned about these in case of a seismic event. He asked if the evaluation of these should have been done in the SAR/SER?

Tim Kobetz responded that we had reviewed the licensee safety evaluation for the placement of the shield lid on the wall.

(NOTE: These evaluations would be done under 50.59 and were not included in the scope of the cask SAR.)

Dennis asked about the statement in the CAL that expressed the possibility of a more serious event.

Jack responded that we did not have any specific more severe consequences in mind. The statement related more to the unexpected nature of the event.

Bob Link (Point Beach) stated that the licensee intended to maintain an open process with all documents sent to the NRC placed in the PDR and also through press releases.

3. Bob Ozanne stated he feels like a guinea pig. He heard us discuss design inadequacies and that WEPCo doesn't implement the process well. He asked if NRC should require that installation of these systems only be done by experienced individuals. He also asked how serious the delay would be pending permission to continue. He asked if it would be best to just shut the plant down. He pointed out that bad publicity at the plant hurts real estate values in the area and thus is a major concern to residents. He expressed the hope that we could find a resolution that truly solves the problems.

Jack responded that the length of the delay is not a concern for us; safe resolution is. He pointed out that activities involving humans will invariably involve mistakes. We must act to limit the risk from human error, not to eliminate all mistakes. Although this event did not cause harm to anyone, we must learn from the mistake and take action.

Bob Link added that safety is their #1 goal. An evaluation of the problems and corrective actions will be completed prior to proceeding. He stated he did not expect the delay to cause any long-term problems.

4. Richard Ralph (ComEd) asked whether licensees could expect any further guidance as a result of this event. He also asked whether it would affect current licensing reviews.

   Carl Paperiello responded that we will likely issue further guidance, although when and in what form is not decided. We will include the lessons-learned in our current licensing reviews. This will probably add more questions to the review.

5. Pat Dupuis asked how many would die due to nuclear waste, drinking water from Lake Michigan.

   Jack stated none based on no radiological releases.

   Pat Dupuis asked, on behalf of her children, whether we could give a 100% guarantee of safety for the time the waste is here.

   Jack responded that he could give a 100% guarantee that we would not allow the plants or the ISFSIs unless we believed they were safe.

6. Mickey Maricque asked whether NRC should require an independent review of cask designs.

   Jack responded that NRC performs a review/evaluation of the safety analysis report versus our safety standards. However, we missed the coating issue.

   Mickey Maricque also asked about the difference in the level of RP response after the event.

   Jack responded that NRC inspects the RP activities at the plant on a routine basis and we have not identified any concerns with their RP activities.

7. Don Taylor asked, in light of the March 4, 1996, TIME magazine article, how we could expect to have any credibility with the public.

   Carl Paperiello responded that the agency has been reviewing the issues raised by the article and making the results public. We agree that we are responsible for correcting our problems.

   Don Taylor stated he wanted to be sure we are doing our job.

8. Alexander Hopp, Two Creeks Town Attorney asked what they can expect from the resident inspectors. Shouldn't they be observing all cask loading activities?

   Jack responded that the residents were not on site at the time of the event but the Senior Resident Inspector responded to the site immediately. The licensee prepares the procedures and we inspected these and observed the dry runs full-time.However, we can't be on site observing all activities. We do perform deep backshift coverage on a regular basis. We try to observe a variety of activities.

Alexander Hopp stressed that cask loading should be a high priority. The licensee missed the precursors, we might have caught it.

Carl Paperiello responded that we must prioritize our inspection effort. If we observed all cask loading activities we would not be watching something else. However, he indicated we would take a look at our inspection program.

Tim Kobetz added that we did observe most of the first two loads, but did not see and/or recognize the precursors.

Alexander Hopp asked for reassurance that the fuel was not damaged.

Carl Paperiello responded that an individual standing on the lid did not feel the event. The pressure required to lift the lid is relatively low because of the large surface area. Inspections of the fuel did not reveal any damage.

Alexander Hopp asked about the hot particle mentioned in the exit.

Jack responded by explaining the basics of contamination control and that we were not satisfied with how the licensee controlled the contamination in the decontamination area. The problem was internal to the plant.

Alexander Hopp asked about QC in light of the confusion over the weight of the lid.

Jack explained that the original shield lid design was two parts, a 2000# plate and a 4400# section that includes the shielding. The licensee modified the design to just one piece weighing 6400#. However, the licensee missed updating some procedures which still reflected the old 4400# weight.

Alexander Hopp asked if any aspect of the event was upsetting to NRC - did we raise hell?

Jack responded that we don't "raise hell" because that would be unprofessional. But we have tried to make it very clear to the licensee how disappointed we are in their performance. This event was preventable in light of the precursors. The lack of rigor in following up on problems encountered earlier contributed to this event.

9. Gary Kunz indicated he has four daughters and he is concerned for their safety. He asked if the two loaded casks could boil through because of radiolysis.

Jack explained in basic terms what radiolysis is and why it isn't an issue in the cask (spent fuel with minimal fission activity). Our root cause review looked at radiolysis because we did not want to leave any possibility unreviewed. The casks on the pad are dry, therefore, the issues of radiolysis and hydrogen generation are removed. To ensure this, the casks are pumped down, a vacuum is drawn and held to remove remaining water, and the casks are inerted with helium prior to sealing.

10. Fran Ozanne asked how long the study would take and whether there is enough room in the pool if future loading is delayed.

    Jack responded that the event evaluation will take as long as it takes to ensure proper resolution.

    Bob Link added that there should be enough room for one more year of operation. The delay could impact the steam generator replacement effort planned for this fall's outage. The licensee hopes to resolve the issues in time to prevent any problems.

11. Pat Dupuis returned to the microphone to ask whether there were plans to re-use the cask involved in this event.

    Bob Link responded that they would only do so if they could be sure it was safe.

    John Jankovich added that the Sierra Nuclear Corp. action plan included an activity to look at how current casks can or will be used.

This summary of the questions and answers from the Point Beach AIT exit represents the combined notes from Andy Kugler (NRR) and Fritz Sturz (NMSS/SFPO).

Roy

June 27, 1996

CAL 3-96-006A

Mr. T. Palmisano, General Manager
Palisades Nuclear Generating Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: SUPPLEMENT TO CONFIRMATORY ACTION LETTER 3-96-006

Dear Mr. Palmisano:

On May 28, 1996, an event occurred involving a VSC-24 spent fuel cask at the Point Beach Nuclear Plant. During that event, a hydrogen gas ignition occurred during the welding of the shield lid on a VSC-24 multi-assembly sealed basket (MSB). The gas ignition displaced the shield lid, leaving it in place but tipped at a slight angle. Although the gas ignition caused no injuries, no radiological releases, and no apparent damage to the spent fuel or to the storage cask, the staff was concerned that such an event had not been evaluated and could have resulted in more severe consequences.

Therefore, on May 29, 1996, an NRC Augmented Inspection Team (AIT) was formed and sent to Point Beach to investigate the event. In addition, on June 3, 1996, we issued Confirmatory Action Letter (CAL) 3-96-006 to you, confirming your commitment to assess the potential for such an event at your facility, to implement compensatory actions to minimize the potential for the generation and ignition of explosive gases, and to have procedures in place to respond in the event of a gas ignition.

The AIT investigation of the Point Beach event raised additional concerns about the use of the VSC-24 cask that were not known to the staff at the time of the issuance of CAL 3-96-006. Specifically, once the MSB had been returned to the spent fuel pool, a white precipitate was observed under the shield lid when it was removed from the MSB. The precipitate's impact on the ability of the cask to safely store spent fuel over a 20-year period has not been evaluated. As a result of the AIT findings, the staff is preparing a generic communication to licensees and cask vendors based on both the potential for the generation and ignition of explosive gases during all phases of cask operation and the long term effects of any such reactions on the ability of the cask to safely store fuel.

Pursuant to a telephone conversation between J. Grobe, and R. Fenech on June 26, 1996, it is our understanding that you will take the following actions prior to loading or unloading a VSC-24 cask with spent fuel or placing a VSC-24 cask into the spent fuel pool:

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1. You will provide the response required in the forthcoming generic communication.

2. Upon completion of the above action, you will not load or unload a VSC-24 cask or place a VSC-24 cask into the spent fuel pool until the NRC has reviewed and accepted your response and verified any subsequent actions taken in response to the forthcoming generic communication.

Item 4 of the June 3, 1996 CAL, included a 14-day advance notice to NRC prior to loading or unloading a VSC-24 cask with spent fuel or placing a VSC-24 cask into the spent fuel pool. This provision is superseded by item 2 above. However, nothing in this supplement precludes you from submitting any information you have developed as a result of the June 3, 1996 CAL.

Pursuant to Section 182 of the Atomic Energy Act, 42 U.S.C. 2232, you are required to:

1) Notify me immediately if your understanding differs from that set forth above;

2) Notify me if for any reason you cannot complete the actions within the specified schedule and advise me in writing of your modified schedule in advance of the change; and

3) Notify me in writing when you have completed the actions addressed in this Confirmatory Action Letter.

Issuance of this Confirmatory Action Letter does not preclude issuance of an order formalizing the above commitments or requiring other actions on the part of the licensee; nor does it preclude the NRC from taking enforcement action for violations of NRC requirements that may have prompted the issuance of this letter. In addition, failure to take the actions addressed in this Confirmatory Action Letter may result in enforcement action.

The responses directed by this letter are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), and your response will be placed in the NRC Public Document Room (PDR). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the

specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Sincerely,

/s/ A. B. Beach

Hubert J. Miller
Regional Administrator

Docket No. 50-255

cc: Robert A. Fenech, Vice President,
Nuclear Operations
R. W. Smedley, Manager, Licensing Dept.
James R. Padgett, Michigan Public
Service Commission
Michigan Department of
Public Health
Department of Attorney General (MI)

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Caniano

July 1, 1996

EA No. 96-215

Mr. Robert Link
Vice President, Nuclear Power
Wisconsin Electric Power Company
231 West Michigan Street - P379
Milwaukee, WI 53201

SUBJECT: NRC REGION III AUGMENTED INSPECTION TEAM REVIEW OF THE MAY 28, 1996, HYDROGEN GAS IGNITION DURING DRY CASK STORAGE WELDING OPERATIONS REPORTS NO. 50-266/96005; 50-301/96005

Dear Mr. Link:

On May 30 through June 7, 1996, an NRC Augmented Inspection Team (AIT) conducted an inspection at your Point Beach 1 & 2 Nuclear Plants. The inspection focused on the circumstances surrounding an unanticipated hydrogen gas ignition which occurred on May 28, 1996, during welding of the shield lid on the multi-assembly sealed basket of a VSC-24 spent fuel cask. Specifically, the AIT focused on the radiological significance of the event, your response to the event, your root cause investigation and generic implications related to the event.

The AIT was composed of Messrs. R. Caniano (Team Leader), T. Kobetz, and R. Paul, of this office; C. Withee and P. Narbut of the Office of Nuclear Materials Safety and Safeguards (NMSS); and J. Davis of the Office of Nuclear Reactor Regulation (NRR). A separate inspection was conducted by several other NRC inspectors at Sierra Nuclear Corporation. A copy of that inspection report will be forwarded to you when it is issued.

The enclosed copy of our AIT report identifies areas examined during the inspection. At the conclusion of the inspection, the AIT discussed its findings with you and others of your staff during a public meeting on June 7, 1996.

The AIT concluded that there were no offsite radiological consequences as a result of this event. The AIT further concluded that there were no measurable releases of radioactivity from the cask and no unanticipated exposures to your staff.

Your management and staff response during and following the event was good which enabled you to successfully return the cask to the spent fuel pool. There were, however, several weaknesses identified with regard to updating of cask unloading procedures, safety evaluations, and rigging practices. These are indicative of a weakness in your staff's ability to evaluate conditions which may have been outside the VSC-24 safety analysis.

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B/36

In addition, we are concerned that your staff had missed opportunities to identify that combustible gas was being generated during cask loading operations. We are concerned that these opportunities had not been properly documented in accordance with your condition reporting system. Had they been documented, a common root cause may have been identified and this event may have been prevented.

It is not the responsibility of an AIT to determine compliance with NRC rules and regulations or to recommend enforcement actions. These aspects will be reviewed during subsequent inspections.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures and your response to this letter will be placed in the NRC Public Document Room (PDR).

We will gladly discuss any questions you have concerning this inspection. Your cooperation with us is appreciated.

Sincerely,

Original signed by A. B. Beach/for

Hubert J. Miller
Regional Administrator, Region III

Docket Nos. 50-266; 50-301; 72-005

Enclosure: Inspection Reports
No. 50-266/96005(DRS);
No. 50-301/96005(DRS)

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- Your staff did not always generate condition reports of unusual occurrences during previous cask loading operations.

- Your design change process failed to identify the increased shield lid weight resulting in the use of undersized rigging.

While none of these weaknesses by themselves pose a significant safety concern, collectively they indicate a weakness in your staff's ability to evaluate conditions which may not be bounded by the VSC-24, Safety Analysis Report.

Even though the safety consequences of this event were minimal, we are concerned that your staff had missed opportunities to identify that combustible gas was being generated during cask loading operations. It is disappointing that these opportunities had not been properly documented in accordance with your condition reporting system. Had they been documented, a common root cause may have been identified and this event may have been prevented.

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In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures and your response to this letter will be placed in the NRC Public Document Room (PDR).

We will gladly discuss any questions you have concerning this inspection. Your cooperation with us is appreciated.

Sincerely,

Hubert J. Miller
Regional Administrator, Region III

Docket Nos. 50-266; 50-301; 72-005

Enclosure: Inspection Reports
No. 50-266/96005(DRS);
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cc w/encl: G. J. Maxfield, Plant Manager
Virgil Kanable, Chief
Boiler Section
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C. A. Paperiello, NMSS

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

AUGMENTED INSPECTION TEAM

Docket No: 50-266, 50-301, 72-005
License No: DPR-24; DPR-27

Report No: 50-266/301-96005

Licensee: Wisconsin Electric Power Company
231 West Michigan Street - P379
Milwaukee, WI 53201

Facility: Point Beach Nuclear Plant Units 1 and 2

Dates: May 30 through June 7, 1996

Inspectors: R. Caniano, Team Leader, Chief, Plant Support Branch 2
T. Kobetz, Senior Resident Inspector, Point Beach
R. Paul, Senior Radiation Specialist
J. Davis, Materials Engineer, NRR
C. Withee, Senior Criticality and Shielding Engineer
Spent Fuel Program Office (SFPO), NMSS
P. Narbut, Senior Nuclear Safety Inspector, SFPO, NMSS

Roy J Caniano

R. Caniano, Team Leader
Chief, Plant Support Branch 2
Division of Reactor Safety

John A. Grobe

J. Grobe, Deputy Director
Division of Reactor Safety

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EXECUTIVE SUMMARY

On May 28, 1996, after loading a VSC-24 ventilated storage cask with spent fuel, an unanticipated hydrogen gas ignition occurred inside the cask during welding of the shield lid. The gas ignition, which was heard by plant technicians, displaced the shield lid in the upward direction approximately 3 inches, and cocked it at a slight angle. The shield lid is approximately 9 inches thick, 5 feet in diameter, and weighs slightly less then 6,400 pounds.

There was no evidence of damage to the spent fuel in the cask as a result of the gaseous ignition. The Augmented Inspection Team (AIT) concluded that there were no offsite radiological consequences as a result of this event. During this event, all possible station release pathways to the public were monitored with no indication of abnormal releases. The AIT further concluded that there were no measurable releases of radioactivity from the cask and no unanticipated radiation exposures to the staff. There were no personnel injuries.

The licensee's actions during and following the event including management oversight were good. However, the inspectors identified several weaknesses in unloading procedures, safety evaluations, corrective actions and rigging practices.

The licensee has concluded and the AIT agrees, that the source of the hydrogen was an electrochemical reaction of zinc in the Carbo Zinc 11 coating when in contact with the borated water in the spent fuel pool (SFP). The coating is used to prevent corrosion of the multi-assembly sealed basket (MSB). At the conclusion of the AIT inspection the licensee had not fully completed their root cause investigation. However, the licensee believes that opportunities were missed to identify that the electrochemical reaction of the coating with borated water would result in the generation of hydrogen. Those opportunities occurred during the initial design, design review and design specification for the VSC-24 cask.

In addition, the AIT concluded that the licensee had several opportunities to identify the generation of gas inside of the MSB during previous cask loading operations due to several noted abnormalities. However, the abnormalities were not documented, were not thoroughly evaluated, and were not viewed collectively. This is of particular concern because the licensee had direct indications that combustible gas was being produced.

The AIT determined that the potential generic implications of the event extend beyond the use of the VSC-24 system. Consideration should be given to reviewing the adequacy of the chemical compatibility evaluations conducted during design reviews for all dry cask storage designs. Consideration should also be given to determining the suitability of Carbo Zinc 11 and other similar coatings used in nuclear applications, where there is the potential to expose them to boric acid.

REPORT DETAILS

1.0 Purpose of the Augmented Team Inspection

Following initial review of the May 28, 1996, incident involving the unanticipated ignition of a combustible gas during welding of a shield lid on a VSC-24 spent fuel cask, an NRC Augmented Inspection Team (AIT) was formed to examine the circumstances surrounding the event. The AIT Charter (Attachment 1) consisted of evaluations of the licensee's response to the event including the radiation protection consequences of the event to both the plant staff and the general public, the effectiveness of the licensee's root cause investigation, and determination of any potential generic implications of the event.

2.0 Background and Summary of the Event

2.1 Background

Point Beach Nuclear Plant (PBNP) began storing spent nuclear fuel at an Independent Spent Fuel Storage Installation (ISFSI) in December 1995. PBNP utilizes the VSC-24 ventilated storage cask system which it operates under a general license in accordance with 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste" and Certificate of Compliance 1007.

PBNP completed loading its first VSC-24 on December 19, 1995. The second cask load was completed on May 26, 1996. PBNP began loading its third cask on May 26, 1996, following completion of the second cask load.

The VSC-24 cask was loaded with spent fuel, removed from the spent fuel storage pool about 4:10 p.m. on May 27, and placed in the cask decontamination area in the auxiliary building. The cask remained mostly filled with water; about 30 gallons had previously been removed to create an air space under the shield lid to facilitate welding. A welding machine was then installed to weld the shield lid in place.

On May 28 at 2:45 a.m., hydrogen gas ignited unexpectedly inside the cask during initiation of welding the shield lid. The gas ignition, which was heard by plant technicians, displaced the shield lid in the upward direction approximately 3 inches and cocked it at a slight angle. The shield lid is approximately 9 inches thick, 5 feet in diameter and weighs slightly less then 6,400 pounds.

Immediately following the event the licensee performed radiation surveys and determined there was no evidence of damage to the spent fuel in the cask. In addition, based on the normal radiation levels and visual inspections, the licensee determined the fuel was still adequately contained inside of the MSB. Continuous air measurements in the cask decontamination area showed no measurable radioactivity indicating there was no threat to the plant staff or general public. There were no personnel injuries. Upon removal from the cask, more extensive visual

inspections of the fuel were performed, confirming that no damage had occurred.

To alert other utilities of the event, on May 31, 1996, the NRC issued Information Notice 96-34, "Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket." In addition, on June 3, 1996, the NRC issued Confirmatory Action Letters (CALs) to all licensees either currently using or preparing to use the VSC-24 system (Attachment 3). The purpose of the CALs was to ensure that the applicable licensees were aware of the event and document their agreement: (1) to assess the potential for the event, (2) take compensatory measures to minimize the potential of a similar event, and (3) develop response procedures if needed. The CALs indicated that once these actions had been completed, the licensees must notify the appropriate NRC Regional office 14 days prior to loading or unloading a VSC-24 cask. On June 27, 1996, supplemental CALs were issued to the applicable licensees informing them of a planned issuance of a generic communication based on additional concerns (Attachment 4). The CALs also documented an agreement that no cask loading or unloading activities would be performed until licensees had responded to the forthcoming generic notification and NRC accepted the response.

2.2 Licensee Response to the Event [Charter Item No. 3]

The inspectors assessed the licensee's actions during and following the event, including their immediate response to the event, implementation of emergency plans and procedures, event reporting, followup actions, and management response. These assessments were based on the following information:

- Interviews with staff and management directly involved in the loading of all three casks at PBNP and the recovery from the event.

- Observations by onsite NRC inspectors during the recovery from the event.

- A review of licensee loading and unloading procedures..

Overall, licensee response to and recovery from the event was good. Specifically:

- Health Physics was quick to respond and evaluate the radiological consequences of the event. Radiation and airborne contamination levels were monitored with no increases noted.

- Operations, Engineering and Maintenance supervision were also quick to respond to the event. Their immediate focus was to ensure that the cask was in a safe condition and make preparations to safely return it to the spent fuel pool.

  For this particular cask load the licensee was allowed 55 hours, after the MSB was removed from the SFP, to completely drain down the MSB. Generally, the time limit is calculated for each cask load in accordance with the Certificate of Compliance. The

purpose of the time limit is to prevent boiling in the MSB and is based on the heat output of the fuel being stored in each cask. If the MSB cannot be completely drained during this period, it must be returned to the SFP within the allowed time.

To preclude further ignitions, the licensee suspended all hot work in the area and continuously monitored the gap between the shield lid and the MSB for hydrogen.

- The duty shift supervisor was quickly notified of the circumstances. He then promptly notified Senior PBNP Management and the NRC Senior Resident Inspector. A review of the licensee's evaluation of event classification and NRC notification indicate that the licensee properly classified the event in accordance with its procedures and NRC requirements.

- On May 28, 1996, PBNP Management promptly began to develop an action plan for the following activities:

    - Determine the source and type of the gas.

    - Recover the shims that had been displaced during the event and restore the lid to the correct position.

    - Update unloading procedures to take into account the post-event conditions so that the cask could be safely returned to the spent fuel pool before the time limit expired.

- Senior licensee managers provided around-the-clock coverage until the cask was returned to the SFP. The cask was successfully returned to the SFP at 8:23 p.m. on Wednesday, May 29, approximately 52 hours after being removed from the pool. The licensee consciously proceeded slowly to ensure it did not experience any further unanticipated conditions. However, delays were encountered when the licensee determined that the shield lid actually weighed more than previously calculated (see Section 2.3).

  Following the return of the cask to the SFP, the AIT observed unloading of 16 of the 24 fuel assemblies. A visual inspection of the fuel assemblies, by the licensee, did not show any signs of damage. Once all visual inspections were complete the licensee confirmed that no physical damage had occurred to any of the 24 assemblies.

2.3 Weaknesses Identified During Licensee Response to the Event

The inspectors identified the following weaknesses in the licensee's preparations for loading the cask and subsequent recovery from the event:

2.3.1 Inadequate Design Change Process

Prior to returning the cask to the spent fuel pool, the licensee determined that it had miscalculated the weight of the shield lid. PBNP

VSC-24 loading and unloading procedures noted the weight of the shield lid as 4,429 pounds when it actually weighed 6390 pounds (determined during a subsequent weight test by the licensee). This weight discrepancy indicates a weakness in the engineering evaluations and design change process for dry cask storage activities.

2.3.2 Inappropriately Sized Rigging

As a result of the shield lid weight discrepancy, the licensee identified an additional concern. Since the weight of the shield lid had been underestimated, the rigging used to move the lid over the spent fuel pool for all previous lifts, had been undersized. Although the safety factor of the rigging was slightly greater than 10-to-1, PBNP did not meet commitments to the NRC for a safety margin of 11-to-1 that takes into account the dynamic loading of the crane.

During the investigation of the above issue, the AIT identified another rigging weakness. The rigging used to lower the cask into the ventilated concrete cask for the first two cask loads was also undersized when taking into account the possibility that in an uncontrolled crane lift, the cask could actually lift the MSB transfer cask (MTC) as discussed in the Safety Analysis Report (SAR). The weight of the MTC had not been taken into account when sizing the lifting slings contrary to the evaluation in the SAR.

2.3.3 Insufficient Safety Evaluation in Accordance with 10 CFR 72.48

The Management Supervisory Staff verbally approved a safety evaluation in which the shield lid was to be weighed in place on the cask. This was required to determine the actual weight of the lid to ensure that future movements utilized appropriately sized rigging. The method was to use the primary auxiliary building crane to lift the lid during weighing. This method involved administrative controls that relied on worker communication to ensure that the lid was not inadvertently removed from the cask by an uncontrolled crane lift.

However, the safety evaluation did not include supporting information, based on experience or dry runs, that communications between workers would be sufficient to ensure the shield lid would not be inadvertently removed from the MSB and expose the workers to the spent fuel; or that the dose consequence of removing the shield lid from the MSB was not an unreviewed safety question. The inspectors concluded that the safety evaluation did not fully address the following two conditions, as required by 10 CFR Part 72.48:

- "If the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased" or

- "If a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created,"

After the inspectors identified this issue, the licensee reevaluated its methods to weigh the lid and developed a new approach which did not

require the crane to be energized. This prevented the possibility of crane failure from inadvertently removing the shield lid from the cask.

2.3.4 Unloading Procedures not Properly Updated

The inspectors noted examples where the licensee had not updated unloading procedures in a timely manner. The AIT identified that the unloading procedures had not been updated to reflect the fact that the shield lid would have to be placed on the spent fuel pool divider wall to install shorter lifting slings to manipulate the lid outside of the pool. The change and updated safety evaluation were not performed prior to removal of the shield lid. This resulted in a delay in which the shield lid had to remain suspended above the spent fuel pool while engineering staff re-evaluated the increased weight of the shield lid on the spent fuel pool divider wall.

2.3.5 Conclusions

These findings indicate several weaknesses in the licensee's evaluation of conditions which could result in circumstances not bounded by the VSC-24 SAR.

3.0 Root Cause Investigation [Charter Item No. 4]

The inspectors assessed the licensee's root cause investigation including potential source(s) of hydrogen generation, and the initial evaluation of the licensee's corrective actions. These assessments were based on the following information:

- Interviews with staff and management directly involved in the root cause investigation of the event at PBNP.
- Participation in the technical discussions between the licensee and design and fabrication vendors.
- Independent NRC research of the materials involved.

The licensee's root cause investigation was being performed in two phases. PBNP contacted and obtained onsite support from the VSC-24 vendor, Sierra Nuclear Corporation, and the two other NRC licensed-users of the cask, Palisades and Arkansas Nuclear One. All three organizations participated in the PBNP root cause investigation. Phase 1 was nearing completion at the end of the AIT investigation and included:

- Determining the root cause of the event.
- Identifying the source of the combustible gas.
- Reviewing welding procedures, processes and work practices, including purging during welding as is the normal practice for primary side welding activities, positioning of the person performing the welding and sampling for combustible gases.

- Evaluating operating experience at other utilities using the VSC-24 MSB and Carbo Zinc 11 coating.

- Review of precursors to this event.

Phase II, which was also in progress at the conclusion of AIT included:

- Performing a detailed investigation into the source of hydrogen.

- Performing a detailed inspection of fuel removed from the MSB effected by the event.

- Justifying continued use of the MSB affected by the event and the two MSBs which were previously loaded and stored at the ISFSI. This includes further inspection of the MSB effected by the event.

- Reviewing the procedure change process for loading and unloading.

- Reviewing of the VSC-24 Safety Analysis Report.

- Reviewing of the VSC-24 component weights and rigging methods.

- Reviewing of existing safety evaluations performed in accordance with 10 CFR Part 72.48.

The AIT concluded that the licensee's investigation scope was thorough and encompassed the appropriate safety related issues.

3.1 Identification of the Source of Hydrogen

The licensee investigated several possible causes for the presence of gas that ignited during the tack welding of the lid on the third MSB.

The licensee concluded that the primary source of hydrogen in the cask was the electrochemical reaction of the zinc in the Carbo Zinc 11 coating when in contact with the borated water in the SFP. The licensee proposed that none of the other sources of hydrogen considered could produce sufficient hydrogen to account for the amount of hydrogen present in the cask. The AIT agreed with the licensee's conclusion.

The following is a breakdown, by possible cause, considered by the licensee and the reason the cause was accepted or rejected:

- Generation of hydrogen from the Carbo Zinc 11 primer in the cask

  Preliminary calculations and measurements by the licensee indicated that the primary source of hydrogen was corrosion of the Carbo Zinc 11 primer. The gas coming from the Ca. o Zinc 11 primer was confirmed to be hydrogen, both in the cask and in laboratory experiments where Carbo Zinc 11 coated coupons were immersed in borated water. Gas generation from the laboratory samples was readily apparent and was analyzed and confirmed to be hydrogen by the licensee. Calculations conducted by the licensee

predicted hydrogen concentrations in the range of the concentrations measured at the space between the shield lid and the borated water.

- Corrosion of the Zircalloy cladding

  Preliminary calculations of the quantity of hydrogen produced from corrosion of the fuel cladding indicate that the amount of hydrogen generated would be an order of magnitude lower than the amount observed. Furthermore, the licensee demonstrated that hydrogen could be generated even when fuel had been removed from the cask.

- Radiolytic production of hydrogen from the water in the cask

  Preliminary calculations indicate that radiolysis of water would not produce hydrogen in the concentrations observed. As further evidence that radiolysis of water was not responsible for the generation of hydrogen, the hydrogen content in the spent fuel pool was 0.1 cc/kg as compared to 16 cc/kg in the storage cask. Since there was fuel with higher heat output in the spent fuel pool, the concentration of hydrogen should be higher in the spent fuel pool if radiolysis is the primary source of hydrogen. Furthermore, the licensee demonstrated that hydrogen could be generated even when fuel had been removed from the cask.

- Defective Carbo Zinc 11 coating

  No coating defects were identified that could cause a release of hydrogen. The certified material test report (CMTR), the coating application procedures and the coating check-off lists were reviewed and no unusual occurrences were identified.

- Contamination of the Carbo Zinc 11 coating

  No contaminants were identified that could cause the release of hydrogen during the review of the CMTR or the check-off list.

- Improper application of the Carbo Zinc coating

  No evidence could be found that the Carbo Zinc 11 was improperly applied based on the review of the application procedures and the coating check-off list.

- Inadvertent introduction of combustible gas during the loading process

  No evidence was uncovered that any combustible gas was inadvertently introduced to the cask during the loading process.

3.2 Production of Precipitates

Once the MSB had been returned to the SFP, a white precipitate was observed under the shield lid when it was removed from the MSB. Most of the precipitate floated to the top of the SFP; however, some remained

below the surface of the water. The precipitate was foam-like in the water but quickly dried and formed a powder when removed from the water. Samples were analyzed, by Argonne National Laboratory for the licensee, to determine the source of the precipitate. The initial analysis, using X-ray diffraction, identified boric oxide and silicate. Additional chemical analysis indicated that the following elements were the most prevalent:

Element	Concentration
Boron	4.01%
Iron	2.08%
Silica	1.16%
Zinc	14.9%
Aluminum	0.02%

The licensee and NRC discussed this precipitate with Carboline, the manufacturer of Carbo Zinc 11. Carboline indicated that precipitates of zinc are common due to the reactivity of the zinc. The common reactions result in the formation of zinc hydroxide or zinc oxide.

An unanticipated constituent of the precipitate included zinc silicate which would not normally be expected since all of the silicate would be chemically bound in the coating. However, silicate is present in the spent fuel pool from the Boroflex in the fuel racks and was available to react with the zinc. Additional discussions with the PBNP Chemistry Department showed that the silicate level in the cask increased with time. Palisades' representatives indicated that they also noted an increase in silicate levels in the SFP during cask loading. In addition, the boron level in the cask did not decrease with time. Therefore, the boron identified in the precipitate is thought to be the result of borated water and not from a reaction with the zinc.

Carboline stated that the precipitate would not normally float. Additional testing revealed that soap was present in the precipitate. The soap was used during the MSB decontamination process. This appears to be the reason the precipitate floated to the surface of the SFP.

Carboline conducted a study to determine if hydrogen would be generated during dry cask storage. The study indicated that there is almost no hydrogen generation from the coating when the coating is dry. This is a preliminary indication that hydrogen should not develop after an MSB has been drained of water, vacuum dried, and filled with helium in preparation for storage at the independent spent fuel storage installation.

3.3 Conclusions

The occurrence of the hydrogen ignition event made it clear that appropriate attention had not been given to the chemical compatibility of the basket's zinc coating and the spent fuel pool's boric acid conditions. Neither the designer nor the licensee considered the potential for the chemical reaction in their reviews. Several indications that might have alerted the involved parties were available.

The root causes being considered by the licensee are deficiencies in the design, the design review, the design specifications, and the

independent review of the design. The licensee has concluded and the AIT agrees, that during one or more of the design steps, the fact that hydrogen would be generated from the electrochemical reaction of the Carbo Zinc 11 coating and spent fuel pool water should have been identified.

4.0 <u>Event Description and Comparison to Pervious Cask Loading Activities</u> [Charter Items No. 1 and No. 2]

The event description and sequence of events for all three casks loaded by PBNP were independently developed and validated by the inspectors. These events were then compared by the inspectors in an effort to identify any differences between the first two cask loads and the third load. Any differences were then reviewed to see if they identified precursors to the event or identified potential root causes of the event. The following information was used by the inspectors during these tasks:

- A review of the licensee's formal and informal logs.
- Interviews with staff and management directly involved in the loading of all three casks at PBNP including recovery from the event.
- Observations by onsite NRC inspectors during the recovery from the event.
- A review of licensee loading and unloading procedures.

The sequence of events studied consisted of those operations determined to be relevant to the root cause of the event. This comparison identified some anomalies which, in retrospect, are indicative of the conditions thought to have contributed to the hydrogen ignition.

The tasks associated with cask loading which were relevant to this inspection were:

- Filling the cask with borated water.
- Placing the cask into the SFP.
- Loading 24 spent fuel assemblies into the MSB.
- Placing the shield lid on the MSB.
- Removing the MTC and MSB from the SFP
- Draining 30 gallons of water from below the shield lid of the MSB.
- Decontaminating the MTC and MSB.
- Welding the shield and structural lids on the MSB.
- Completing drain down of the MSB.

The actual time-sequence of events for all three cask loading operations was taken mainly from entries in informal logs supplemented by some events being recorded in the control room log and the procedures check-off list. The informal logs were kept by the project managers to aid in identifying procedure changes to improve future loading operations. The time and date chronology of the relevant events for all three loading operations is given in Attachment 2, Table 1. Attachment 2, Table 2, gives the elapsed time of the relevant loading operations.

4.1 Possible Indications of Hydrogen During Previous Cask Loads

The following sections highlight the differences and abnormalities noted during the three cask loads at PBNP. In addition, although not part of the AIT Charter, several team members had discussions with the representatives from both Palisades and Arkansas Nuclear One, the other licensees using the VSC-24 cask. Both indicated that, in retrospect, there may have been similar, minor, indications of gas being generated inside the MSB during their loading operations.

4.1.1 First Cask Load

- Prior to actually loading spent fuel into the first cask, it was used in two pre-operational test "dry run" operations to satisfy Certificate of Compliance requirements. In each of these two evolutions, the cask was flooded with borated water. The cask was flooded a third time when the actual loading operation began. It is likely that the reaction rate of the boric acid and zinc decreased during each of these evolutions resulting in a reduced hydrogen production rate during the actual loading operation. However, the licensee did not observe any gas production in the MSB during these activities.

- One event of note during the loading of the first cask was the presence of a small amount of moisture around the vent port which was removed with rags. Although not considered at the time, this moisture may have been the first indication of potential gas pressure buildup in the cask forcing water out of the vent. The other loading operations associated with the first cask were uneventful.

4.1.2 Second Cask Load

- During the welding operations for the second cask load, the week of May 20, 1996, there were also possible indications of gas buildup in the cask but the significance was not recognized by the licensee. The root pass on the shield lid weld was stopped about 3 inches from closure because of the possibility of porosity at the start of the weld caused by an initial high welding machine current. The welders began to grind the area in question to smooth the two weld ends for the final closure weld. The time between ending the weld and grinding was 2 to 3 minutes. During the grinding operation the welder noticed a small blue flame that burned for 30 to 40 seconds. The intensity of flame was light and as such was not noticeable unless one put a glove or other item behind the flame. When the flame was noticed, the licensee staff

present rationalized that it was probably the result of residual cleaner or solvents left from the decontamination or other cleaning procedures. The weld was finished successfully. It was later determined; however, that no flammable solvents or cleaners were used in the decontamination process.

- During the subsequent welding operations on the structural lid of the second cask, it was noticed that water was seeping up past the drain port foreign material exclusion plug. The licensee suspected that thermal expansion of the water during cask warm up was pushing up the drain tube and causing this seepage. The engineering staff made a rough calculation and it was concluded that thermal expansion of the water could be the cause of the seepage. This moisture caused the initial seal weld to fail examination. The area was vacuum dried and the weld was successfully completed. In addition, the gas pressure in the cask was vented at least twice to relieve the potential pressure for forcing water up the drain tube.

4.1.3 Third Cask Load

- In the third cask loading operation, the shield lid had a tighter fit than for the first two casks. This fit could have retained more hydrogen inside the MSB. It is possible that this overall set of conditions created a hydrogen concentration distribution that supported the hydrogen ignition. The welder, who was kneeling on the shield lid at the time of the hydrogen ignition, jumped off of the lid when he heard the sound. However, he did not feel anything or notice any flash, heat, smoke or odor of a burn which supports the hypothesis that the combustible gas was hydrogen.

- The licensee and inspectors also noted a white foamy precipitate that was discovered on the underside of the shield lid when the third cask was unloaded. This effect was not observed when the shield lid was removed after either of the dry runs (see Section 3.2 for further discussions on the precipitate).

4.1.4 Time Available for Gas Generation

A final consideration in the differences between the three cask loading operations was the time available for gas to generate and collect. Two starting points were considered as possible times for gas collection to begin. These were when the shield lid was placed on the cask and when the cask had 30 gallons of water pumped from it. The time interval between the shield lid placement and the beginning of the tack welds was 17.5 hours, 14.25 hours and 11.9 hours for loads one, two and three, respectively. The corresponding time interval from the 30 gallon pump down to tack weld was 13.25 hours, 10.3 hours and 10.4 hours. The third cask was loaded more quickly than the first two operations and thus may have provided less time to generate and collect hydrogen.

The AIT could not conclude if these time differences contributed to the collection and concentration of hydrogen within the MSB.

4.1.5 Condition Reporting System not Always Utilized

The inspectors noted a weakness in the licensee's use of its condition reporting system. Condition reports were not initiated for either of the following issues:

- During completion of the root weld of the shield lid during loading of the second cask on May 22, 1996, the welders noted a small blue flame while grinding a portion of the weld. The flame appeared over one of the shims in an area that had not yet been welded. The welders brought this to the attention of their supervisors and Engineering. However, those involved rationalized that the flame was the result of igniting residual cleaning fluid or other solvent from the surface of the cask and/or lid.

- In addition, later, during welding of the structural lid on the second cask, the licensee's staff observed water seepage from the cask drain line onto the top of the shield lid. The staff rationalized that the leakage was due to a pressure build-up under the lid from water expansion due to heat generated by the fuel. However, this also could be an indication that some type of gas was collecting underneath the shield lid.

Although these appear to be separate issues, they do, in hindsight, appear to be indications of gas build-up inside of the cask. However, since no condition reports were issued, no further staff or management evaluation of the conditions was conducted.

4.2 Conclusions

The licensee did not recognize that these abnormalities indicated combustible gas was being generated. Therefore, the licensee failed to aggressively pursue these precursors either individually or collectively. Had the licensee recognized and further evaluated these conditions, this gas ignition event may have been prevented.

5.0 Determination of Whether Appropriate Attention was given to the Condition of Systems and Components Associated with Dry Cask Evolutions, Including Compatibility of the Dry Cask with Spent Fuel Pool Conditions. [Charter Item No. 5]

The inspectors' assessment of the appropriateness of the attention given to the systems and components was based on the following:

- Reviewing fabrication and receipt inspection records.

- Reviewing inspection procedures.

- Interviewing licensee personnel involved in fabrication oversight.

5.1 Oversight of MSB Fabrication

The inspectors examined the licensee's overview and control of the MSB during the entire fabrication and receipt inspection processes. The inspectors concluded that the licensee had given appropriate attention

to the oversight of the fabricators throughout the MSB fabrication process. The licensee had essentially full-time Quality Assurance representation at the fabrication sites and documented its oversight through the use of hold points in the construction documents. Licensee representatives independently verified important parameters such as cleanliness, weld size, and material certifications. Their verifications included detailed parameters such as coating thickness. The licensee's receipt inspection process also independently verified that the casks received on-site had been maintained in a quality condition.

5.2 Compatibility of the MSB with the Spent Fuel Pool

As stated earlier, the occurrence of the hydrogen ignition event made it clear that appropriate attention had not been given to the chemical compatibility of the basket's zinc coating and the spent fuel pool's boric acid conditions. Neither the designer nor the licensee considered the potential for the chemical reaction in their reviews. Several indications that might have alerted the involved parties were available.

First, the coating manufacturer's specification sheet stated that the coating was not recommended for immersion in acids. Secondly, the licensee studied the incompatibility of the coating and the spent fuel pool boric acid in a study they conducted in July 1995. The study noted the potential for the introduction of dissolved zinc and a zinc borate precipitate. Hydrogen generation was not addressed. Since hydrogen is routinely used to maintain reactor coolant chemistry, chemistry personnel only focussed on the effect that zinc would have on the reactor coolant system and did not consider the production of hydrogen during dry cask storage activities. The study was a result of the licensee's policy of having the Chemistry Department review any new item that might introduce chemical contaminants into the spent fuel pool.

Unfortunately, the study focused only on potential reactor plant effects and not on the potential effects on dry cask storage operations.

Review of the Safety Analysis Report for the Ventilated Storage Cask System showed that considerable attention was given to corrosion considerations for long term dry storage, but consideration of the temporary condition of the cask being immersed in borated water was, likewise, not assessed.

The inspectors concluded that insufficient attention was given to the chemical compatibility of the dry cask with spent fuel pool conditions.

6.0 Determination of the Potential Generic Implications of the Event. [Charter Item No. 6]

The inspectors' assessment of the potential generic implications of the event was based on the following:

- Review of the circumstances of the event as described in this report.

- Discussions with AIT team members, licensee personnel and management, and NRC staff and management.

The team identified the following potential generic issues. It was noted that other generic issues might be identified as the NRC continued to assess the event subsequent to the AIT inspection.

6.1 Chemical Compatibility Studies for all Currently Licensed Dry Cask Systems

Consideration should be given to review the adequacy of the chemical compatibility evaluations conducted during design reviews for all cask designs. Appropriate focus should be given to the cask interaction with facility environments. Items to consider are:

- The amount and acceptable limits of zinc, silica and other unanticipated materials introduced into the spent fuel pool.

- Identify materials that may migrate to the reactor coolant system after being introduced into the spent fuel pools.

Examples of oversights identified from this event included the generation of hydrogen from a zinc coating and borated water reaction, and the possibility of water intrusion in the shield lid's shielding material in some shield lid designs. The shield lid design used at PBNP is seal welded such that water cannot intrude into the shielding material. This is not the case for the shield lid design used by the other two user utilities.

6.2 The Suitability of Carbo Zinc 11 and Similar Products for Use in Nuclear Applications

Consideration should be given to review the suitability of this coating in other nuclear applications, where there is a potential for exposure to boric acid.

6.3 The Adequacy of Cleanliness Controls During Cask Fabrication and Use

Consideration should be given to generically define cask cleanliness controls for all cask designs. This should include the potential for the presence of chemical reaction of residues and other foreign material on fuel and storage cask components. The term "potential" for the presence of material is used because the PBNP casks have some residue; however, it is not clear whether other cask systems would have unanticipated materials.

AT PBNP, the casks were not cleaned with water before immersion. In addition, soap was knowingly introduced into the cask during the decontamination process. Neither the licensee or the AIT determined what affect these conditions had on the formation of precipitate in the cask.

6.4 The Adequacy of Foreign Material Exclusion (FME) Controls During Cask Loading and Unloading Operations

Consideration should be given to assess what potential hazards exist from the introduction of foreign material into the cask during loading and unloading operations. Determination of specific FME controls should be made.

7.0 Evaluation of the Technical Support by the Vendor for the Prevention of Similar Events. [Charter Item No. 6]

The inspectors' assessment of the technical support by the vendor for the prevention of similar events was based on the following:

- Interviews with site personnel including the vendor representative.

- Discussions with the NRC team leader conducting an independent inspection at Sierra Nuclear Corporation on June 3-5, 1996, (Reference Inspection Report No. 72-1007/96-204).

After the event, the cask designer, Sierra Nuclear Corporation, dispatched an operations engineer to provide technical expertise and assist in the unloading of the cask. Additionally, frequent telephone conference calls were held between the Wisconsin Electric staff and Sierra Nuclear Corporation designers to discuss technical issues pertinent to the analysis of the event. Staff from two other nuclear plants (Palisades and Arkansas Nuclear One) using the cask design also participated in the telephone conferences. An NRC inspection team at Sierra Nuclear noted that Sierra Nuclear had developed a detailed evaluation and action plan in response to the event.

Overall, the inspectors considered the technical support provided by Sierra Nuclear for this event to be adequate. The adequacy of the vendor's support to prevent similar events will be determined by an NRC review of further activities conducted by Sierra Nuclear in response to this event.

8.0 Dry Cask Loading Activities and Radiological Protection Performance As Result of Dry Cask Hydrogen Ignition. [Charter Item Nos. 7 and 8]

The inspectors' assessment concerning radiation protection and precautions taken as part of the dry cask loading and unloading activities and the radiological consequences to the public and staff from the hydrogen ignition event were based on the following reviews:

- Radiological surveys.
- Air sample results (inplant and vent effluent).
- Chemistry sample results.
- Radiation Protection (RP) and Chemistry personnel Interviews
- Direct field observations.

The inspectors concluded that as a result of this event there were no offsite radiological consequences. The assessment indicated that during cask loading operations and the unloading of the third cask good

radiological controls were implemented. These controls were part of the radiological protection and chemistry programs developed for those evolutions. The program included appropriate radiation protection and emergency procedures, sound work practices, employee training, monitoring techniques, and ALARA initiatives.

The inspectors further determined that as a result of the May 28, 1996 event, there were no measurable releases of radioactivity from the cask, and radiation exposure to the staff was equivalent to normal work activities associated with the evolution. For this event, all possible station release pathways to the public were monitored with no indication of abnormal releases. In addition, radiation protection and chemistry technicians responded to the event in a timely manner and implemented good radiological controls and measurements. Following the event it was noted that increased radiological surveys were routinely performed as were increased collections of gas and cask samples.

One weakness was identified concerning poor contamination controls around the cask decontamination (decon) area. Specifically, following the transfer of the cask to the decon area on June 4, 1996, surveys on the MSB, on the lid of the MTC, and on the floor of the decon area indicated some increased loose levels of mixed fission and corrosion products. Because of the nature of the area, it was also likely to have radioactive "hot" particles on and around the cask. For work in that area (chemistry samples, inspection, etc) only booties, gloves, and one set of protective clothing (PC) coveralls were required. When exiting the area, the coveralls were not required to be removed. Given the nature of the work, and the possibility of hot particles, the practice of using only one set of PCs and allowing the PCs to be worn after exiting the area increased the probability of transporting particles and loose contamination. In addition, it was also noted that a fan used to dilute hydrogen produced in the MSB was set up near the top of the MSB and was installed without the knowledge of radiation protection personnel. Although no contamination was identified, the fan had the potential to blow some of the loose contamination off the lid into the decon area. These practices may have contributed to a hot particle shoe contamination on a person who was located outside of the contaminated area. Although there was very little dose from the hot particle, the licensee did not maintain sufficient controls to ensure particles of this type remain in controlled areas.

9.0 Exit Meeting

The team met with licensee representatives (identified below) during a public meeting on June 7, 1996, and summarized the purpose of the AIT, AIT charter items, and inspection findings. The team discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the team during the inspection.

Attachments:
1. A.I.T. Charter
2. Chronology of Operations
3. Confirmatory Action Letter dated 6/3/96
4. Confirmatory Action Letter dated 6/27/96

PERSONNEL PARTICIPATING IN THE EXIT MEETING

Wisconsin Electric Power Company

F. Link, Vice President, Nuclear Power
G. Krieser, Manager, Industry and Regulatory Services Section
G. Maxfield, Plant Manager, Point Beach
F. Cayia, Production Manager, Point Beach
A. Riemer, Manager, Nuclear Engineering

U. S. Nuclear Regulatory Commission

C. Paperiello, Director, Office of Nuclear Materials Safety and Safeguards
J. Grobe, Deputy Director, Division of Reactor Safety, Region III
R. Caniano, Chief, Plant Support Branch 2, Division of Reactor Safety
T. Kobetz, Senior Resident Inspector, Point Beach
R. Paul, Senior Radiation Specialist
J. Davis, Materials Engineer, NRR
C. Withee, Senior Criticality and Shielding Engineer, SFPO, NMSS
P. Narbut, Senior Nuclear Safety Inspector, SFPO, NMSS

Augmented Inspection Team Charter - Point Beach Nuclear Plant

Examine the circumstances surrounding the dry cask ignition event at the Point Beach Nuclear Plant on May 28, 1996, including but not limited to the following:

1. Develop and validate a chronological sequence of events and activities for the dry cask evolution, detailing events just prior to and immediately after the gas ignition.

2. Compare this sequence of events to other dry cask loading evolutions at Point Beach to identify any anomalies associated with this particular load.

3. Evaluate the licensee's actions during and following the event; including their immediate response to the event, implementation of emergency plans and procedures, event reporting, followup actions, and management response.

4. Evaluate the extent of the licensee's analysis and determination of the root cause for the event, including potential source(s) of hydrogen generation, and the initial evaluation of appropriate corrective actions.

5. Determine if appropriate attention was given to the condition of systems and components associated with dry cask evolutions, including compatibility of the dry cask with spent fuel pool conditions.

6. Determine any potential generic implications of the event. Evaluate the technical support by the vendor for the prevention of similar events.

7. Evaluate the adequacy and appropriateness of radiation protection precautions taken by the licensee as part of the dry cask loading activity.

8. Evaluate the radiation protection consequences of the event to both the plant staff and the general public.

Table 1

CHRONOLOGY OF RELEVANT DRY CASK OPERATIONS
(TIME : DATE)

EVENT	LOAD #1	LOAD #2	LOAD #3
(Serial #)	(MSB-01)	(MSB-03)	(MSB-02)
Start MSB fill	0911; 12/11/95	1800e; 5/20/96	2145; 5/26/96
MSB into SFP	1030	2215	2300e
Load fuel	1235-0030; 12/12	0100-1330; 5/21	0100e-1142; 5/27
Shield lid in place	0415	1645	1450
MSB breaks SFP surface	0820	2035	1610
30 gal. Pump down	0830e	2041	1620e
MSB out of SFP	0900	2100	1645
Tack welds begin	2145	0700e; 5/22	0245; 5/28
Shield weld complete	0330 : 12/13	1230	*
Weld structural lid	0900-1530	2300-1100e:5/23	*
Structural lid seal weld	1640-1715	0405e[1]; 5/25	*
Drain MSB	2145-2230	1600e-1641; 5/23	*

e - Time of event not recorded; estimate based on adjacent recorded events.
* - Not performed
1 - Drained down prior to completing structural lid seal weld
MSB - Multi-Assembly Sealed Basket
SFP - Spent Fuel Pool removed from the spent fuel pool.

Table 2

CHRONOLOGY OF RELEVANT DRY CASK OPERATIONS
ELAPSED TIME (HRS:MINS)

EVENT	LOAD #1	LOAD #2	LOAD #3
(Serial #)	(MSB-01)	(MSB-03)	(MSB-02)
Start MSB fill	0	0	0
MSB into SFP	1:19	4:15	1:15
Load fuel	15:19; 12 hr	19:30; 12.5 hr	13:57; 10.7 hr
Shield lid in place	19:04	22:45	17:05
MSB breaks SFP surface	23:09	26:35	18:25
30 gal. Pump down	23:19	26:41	18:35
MSB out of SFP	23:49	27:00	19:00
Tack welds begin	36:34	37:00	29:00
Shield weld complete	42:19	42:30	*
Weld structural lid	54:19; 6.5 hr	65:00; 12 hr	*
Structural lid seal weld	56:04	106:00[1]	*
Drain MSB	61:19	70:00	*

* - Not performed
MSB - Multi-Assembly Sealed Basket
SFP - Spent Fuel Pool
1 - Drained down prior to completing structural lid seal weld

Attachment 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

June 3, 1996

CAL No. RIII-96-0005

Mr. Robert Link
Vice President, Nuclear Power
Wisconsin Electric Power Company
231 West Michigan Street - P379
Milwaukee, WI 53201

SUBJECT: CONFIRMATORY ACTION LETTER

Dear Mr. Link:

On May 28, 1996, an event occurred involving a VSC-24 spent fuel storage cask at your facility. During that event, a hydrogen gas ignition occurred during the welding of the shield lid on a VSC-24 multi-assembly sealed basket (MSB). The gas ignition displaced the shield lid, leaving it in place but tipped at a slight angle.

The VSC-24 multi-assembly transfer cask (MTC), a shielded lifting device used to transfer the MSB loaded with spent fuel for storage, had been placed in the cask decontamination work area in the auxiliary building. Approximately 30 gallons of spent fuel pool water had been drained to facilitate welding of the shield lid by creating an air space below the lid. The hydrogen gas ignition occurred during the initiation of the shield lid welding, approximately 11 hours after the loaded MTC had been removed from the spent fuel storage pool.

Although the gas ignition caused no injuries, no radiological releases, and no apparent damage to the spent fuel or to the storage cask, the staff is concerned that such an event has not been evaluated and could potentially result in more severe consequences. Therefore, on May 29, 1996, an NRC Augmented Inspection Team (AIT) was formed and sent to Point Beach to investigate the event. The objectives of the AIT are to identify and communicate both the facts of the event and any potential generic safety concerns and to document the findings and conclusions of the onsite inspection.

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Attachment 3

Pursuant to a telephone conversation between Fred Cayia, Production Manager, and Jack Grobe, Deputy Director, Division of Reactor Safety, and other members of my staff on June 3, 1996, it is our understanding that you have taken or will take the following actions prior to loading or unloading a VSC-24 cask with spent fuel or placing a VSC-24 cask into the spent fuel pool:

(1) You will assess the potential for the generation and ignition of explosive gases during all phases of operation of the VSC-24 system;

(2) You will have compensatory actions in place to minimize the potential for the generation and ignition of explosive gases;

(3) You will have procedures in place to respond in the event of a gas ignition and the applicable personnel trained and briefed accordingly; and

(4) Upon completion of the above actions, and 14 days prior to loading or unloading a VSC-24 cask with spent fuel or placing a VSC-24 cask into the spent fuel pool, you will contact William L. Axelson, Director, Division of Reactor Projects, at (708) 829-9600.

Pursuant to Section 182 of the Atomic Energy Act, 42 U.S.C. 2232, you are required to:

1) Notify me immediately if your understanding differs from that set forth above;

2) Notify me if for any reason you cannot complete the actions within the specified schedule and advise me in writing of your modified schedule in advance of the change; and

3) Notify me in writing when you have completed the actions addressed in this Confirmatory Action Letter.

Issuance of this Confirmatory Action Letter does not preclude issuance of an order formalizing the above commitments or requiring other actions on the part of the licensee; nor does it preclude the NRC from taking enforcement action for violations of NRC requirements that may have prompted the issuance of this letter. In addition, failure to take the actions addressed in this Confirmatory Action Letter may result in enforcement action.

The responses directed by this letter are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

Attachment 3

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and your response will be placed in the NRC Public Document Room (PDR). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Sincerely,

Hubert J. Miller
Regional Administrator

Docket No. 50-266
Docket No. 50-301

cc: G. J. Maxfield, Plant Manager
Virgil Kanable, Chief, Boiler Section
Cheryl L. Parrino, Chairman,
Wisconsin Public Service Commission
State Liaison Officer

Attachment 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

June 27, 1996

CAL 3-96-005A

Mr. Robert Link, Vice President
Nuclear Power
Wisconsin Electric Power Company
231 West Michigan Street - P379
Milwaukee, WI 53201

SUBJECT: SUPPLEMENT TO CONFIRMATORY ACTION LETTER 3-96-005

Dear Mr. Link:

On May 28, 1996, an event occurred involving a VSC-24 spent fuel cask at your facility. During that event, a hydrogen gas ignition occurred during the welding of the shield lid on a VSC-24 multi-assembly sealed basket (MSB). The gas ignition displaced the shield lid, leaving it in place but tipped at a slight angle. Although the gas ignition caused no injuries, no radiological releases, and no apparent damage to the spent fuel or to the storage cask, the staff was concerned that such an event had not been evaluated and could have resulted in more severe consequences.

Therefore, on May 29, 1996, an NRC Augmented Inspection Team (AIT) was formed and sent to Point Beach to investigate the event. In addition, on June 3, 1996, we issued Confirmatory Action Letter (CAL) 3-96-005 to you, confirming your commitment to assess the potential for another such event at your facility, to implement compensatory actions to minimize the potential for the generation and ignition of explosive gases, and to have procedures in place to respond in the event of a gas ignition.

The AIT investigation of the Point Beach event raised additional concerns about the use of the VSC-24 cask that were not known to the staff at the time of the issuance of CAL 3-96-005. Specifically, once the Point Beach MSB had been returned to the spent fuel pool, a white precipitate was observed under the shield lid when it was removed from the MSB. The precipitate's impact on the ability of the cask to safely store spent fuel over a 20-year period has not been evaluated. As a result of the AIT findings, the staff is preparing a generic communication to licensees and cask vendors based on both the potential for the generation and ignition of explosive gases during all phases of cask operation and the long term effects of any such reactions on the ability of the cask to safely store fuel.

Pursuant to a telephone conversation between J. Grobe and you on June 26, 1996, it is our understanding that you will take the following actions prior to loading or unloading a VSC-24 cask with spent fuel or placing a VSC-24 cask into the spent fuel pool:

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1. You will provide the response required in the forthcoming generic communication.

2. Upon completion of the above action, you will not load or unload a VSC-24 cask or place a VSC-24 cask into the spent fuel pool until the NRC has reviewed and accepted your response and verified any subsequent actions taken in response to the forthcoming generic communication.

Item 4 of the June 3, 1996 CAL, included a 14-day advance notice to NRC prior to loading or unloading a VSC-24 cask with spent fuel or placing a VSC-24 cask into the spent fuel pool. This provision is superseded by item 2 above. However, nothing in this supplement precludes you from submitting any information you have developed as a result of the June 3, 1996 CAL.

Pursuant to Section 182 of the Atomic Energy Act, 42 U.S.C. 2232, you are required to:

1) Notify me immediately if your understanding differs from that set forth above;

2) Notify me if for any reason you cannot complete the actions within the specified schedule and advise me in writing of your modified schedule in advance of the change; and

3) Notify me in writing when you have completed the actions addressed in this Confirmatory Action Letter.

Issuance of this Confirmatory Action Letter does not preclude issuance of an order formalizing the above commitments or requiring other actions on the part of the licensee; nor does it preclude the NRC from taking enforcement action for violations of NRC requirements that may have prompted the issuance of this letter. In addition, failure to take the actions addressed in this Confirmatory Action Letter may result in enforcement action.

The responses directed by this letter are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), and your response will be placed in the NRC Public Document Room (PDR). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the

specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Sincerely,

Hubert J. Miller
Regional Administrator

Docket No. 50-255

cc: Robert A. Fenech, Vice President, Nuclear Operations
R. W. Smedley, Manager, Licensing Dept.
James R. Padgett, Michigan Public Service Commission
Michigan Department of Public Health
Department of Attorney General (MI)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PRIORITY ROUTING

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FILE h

August 8, 1996

Ms. Fawn Shillinglaw
1952 Palisades Drive
Appleton, Wisconsin 54915

SUBJECT: YOUR LETTER TO THE STAFF OF THE NUCLEAR REGULATORY COMMISSION REGARDING DRY CASK STORAGE AT THE POINT BEACH NUCLEAR PLANT

Dear Ms. Shillinglaw:

As the lead manager for dry cask issues in the Office of Nuclear Reactor Regulation (NRR), Nuclear Regulatory Commission (NRC), I am responding to a letter which you forwarded to Allen Hansen of my staff on May 14, 1996.

You asked several questions about the serial numbers and loading order of the multi-assembly sealed basket (MSB) casks and the ventilated concrete casks (VCCs) used at the independent spent fuel storage installation at the Point Beach Nuclear Plant in Two Creeks, Wisconsin. The serial numbers assigned by the licensee are for identification purposes only. The casks are interchangeable and are not matched by common serial numbers.

You asked several questions about the coating on the MSB. The coating was tentatively approved for MSBs 1 and 2 only due to concern for potential introduction of contaminates into the spent fuel pool. Wisconsin Electric Power Company, the licensee for the Point Beach Nuclear Plant, later determined that there was no significant risk of pool contamination, and approved the coatings for subsequent casks. Additional information on the coatings is provided in the Augmented Inspection Team Report (96-005) issued by NRC dated July 1, 1996. This report provides details of the special inspection that was conducted at Point Beach following the cask event on May 28, 1996, including a discussion of MSB coating qualification and performance.

You posed additional questions about MSB transfer cask (MTC) test deficiencies identified during the loading of the first MSB at Point Beach. (Your letter refers to Palisades, but based on the issues addressed, I assume that you meant to refer to Point Beach.) Please see my letter to you dated May 3, 1996, for a complete discussion of this issue.

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I trust that the information I have provided answers your questions. Please contact Allen Hansen of my staff at (301) 415-1390 if you need further assistance.

Sincerely,

Gail H Marcus

Gail H. Marcus, Director
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-266,
50-301 and 72-5

cc: See next page

Point Beach Nuclear Plant
Wisconsin Electric Power Company

Unit Nos. 1 and 2

cc:

Ernest L. Blake, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Mr. Gregory J. Maxfield, Manager
Point Beach Nuclear Plant
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Index of /RIII/ers/dry.cask

A/40

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total 21982
-rw-r--r--   1 ers      riii        182883 Jun 12 10:23 1-2.gif
-rw-r--r--   1 ers      riii         33447 Aug 27 17:24 1-2.jpg
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