BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF: Docket 72-10

THE PRAIRIE ISLAND INDIAN COMMUNITY)
Petitioners)
)
)
vs)
)
UNITED STATES NUCLEAR REGULATORY COMMISSION)
Respondent

Petition

Pursuant to 10 CFR Part 2.206 of the Commission's regulations, the Prairie Island Indian Community petitions the Nuclear Regulatory Commission (NRC) to:

1) Determine that Northern States Power (NSP) violated the requirements of 10 CFR 72.122(l) by using its Materials License No. SNM-2506 for an Independent Spent Fuel Storage Installation (ISFSI) prior to establishing conditions for safely unloading the TN-40 dry storage containers.

2) Suspend Materials License No. SNM-2506 for cause under 10 CFR 50.100 until such time as all significant issues in the unloading process, as described herein, have been resolved, the unloading process has been demonstrated, and until an independent third party review of the TN-40 unloading procedure has been conducted.

3) Provide petitioners an opportunity to participate fully in the reviewing the unloading procedure for the TN-40 cask, hold hearings and allow petitioners to participate fully in these and any other procedures initiated in response to this petition.

4) Update the Technical Specifications (TS) for the Prairie Island ISFSI to incorporate mandatory unloading procedure requirements.

Facts

1. Northern States Power (NSP) owns and operates the Prairie Island Nuclear Generating Plant on Prairie Island in Minnesota.

2. The Prairie Island Nuclear Generating Plant is located next to the Prairie Island Indian Community.

3. The Prairie Island Indian Community is a federally recognized Indian tribe.

4. All government agencies and executive departments, including the NRC, have a Trust Responsibility towards Indian tribes. The United States Government's "trust responsibility" toward American Indians, which is the unique fiduciary and legal duty of the United States to assist Indians in the protection of their property and rights. This trust responsibility arises out of treaties, statutes, executive orders, legal precedence, the United States Constitution and the course of dealings between the United States Government and Indian tribes.

5. On April 29, 1994, President Clinton issued an Executive Memorandum laying out the principles for every executive department and agency to follow in their interactions with federally recognized Indian tribes. At the core of these principles is the premise that the United States Government has a unique legal relationship with Indian tribes. Moreover, the President's memorandum also stated that each executive department and agency shall consult with tribal governments prior to taking actions that affect tribal governments and shall assess the impacts their actions have on tribal trust resources.

6. On August 31, 1990, NSP submitted a license application to the NRC for an Independent Spent Fuel Storage Installation (ISFSI), pursuant to the requirements of Title 10, Part 72 of the Code of Federal Regulations (10 CFR 72). The application was assigned Docket #72-10. Included in the application were Technical Specifications and Safety Analysis Report (TSSAR) and an Environmental Report (ER).

7. NSP's ISFSI application consisted of two components: the cask (a TN-40 cask, designed by Transnuclear, Inc.) and the actual storage area (the concrete pad). The application provided general information regarding the ISFSI, the type of cask to be used, conformity to design criteria (as required by 10 CFR 72, Subpart F).

8. In their application, NSP stated that they planned to operate the ISFSI for the licensed life of the plant. The design basis life of each cask is twenty-five years, although the NRC has stated that waste can safely be stored in dry casks for up to one hundred years.

9. NRC regulations require that storage systems (i.e., the casks) be designed to allow ready retrieval of spent fuel for further processing or disposal (10 CFR 72.122(l)). In their application, NSP addressed this requirement by ensuring that "fuel criticality is prevented, cask integrity is maintained, and fuel is not damaged so as to preclude its

ultimate removal from the cask" (TSSAR, p. 3.2-1) and through a Decommissioning Plan described in the TSSAR (p. 4.6-1). Neither of these sections discusses the actual steps to be taken and the likely problems to be encountered when removing fuel from a TN-40 dry cask. With respect to decommissioning, there is just mention of removing waste from the TN-40 and shipping it in a licensed transportation cask.

10. The Technical Specifications (TS) document is issued by the NRC and provides general guidance regarding the safe receipt, possession, and storage of irradiated nuclear fuel at the ISFSI (i.e., design criteria, cask operating limits, surveillance requirements, etc.). For each issue there is a definition (i.e., the limiting condition), its applicability, action to be taken, surveillance requirements, and the basis for the specification. For example, two of the limiting conditions which would apply to unloading include the verification of the dissolved boron concentration of the spent fuel pool (e.g., greater than or equal to 1800 parts per million (PPM)) to ensure that spent fuel is subcritical (TS p. 3/4-3) and ensuring that the outside cask surface temperature is not greater than 250 ° F (121 °C), which ensures that fuel cladding will be protected against degradation (TS p. 3/4-5). The TS specifies that if the cask surface temperature is greater than 250 °F, it must be unloaded.

11. On July 10, 1992, NSP provided the NRC with information regarding the unloading of a TN-40 dry cask, as requested. The procedure described by NSP is as follows:

"Assuming that the spent fuel in the TN-40 cask will be transferred to a licensed transportation cask using a normal 'in pool' fuel transfer, the sequence of operations discussed in Section 5.1 of the ISFSI Safety Analysis Report (SAR) and in particular listed in Table 5.1-1, will be essentially performed in reverse." The letter lists the steps taken in Table 5.1-1 of the SAR, only in reverse. (Attachment A).

12. On July 28, 1992, the NRC issued a Finding of No Significant Impact (FONSI) based on its Environmental Assessment (EA) for the site. The EA referenced both the TS and the SAR included in NSP's application. The NRC found that no significant impacts from the construction of the ISFSI were to be expected. No impacts were expected from the operation of the ISFSI either.

With respect to radiological impacts, NRC staff expected that impacts from cask loading and preparation would be minimal. Table 5.3 in the EA (Attachment B) describes the steps that will be taken by NSP to receive, load, decontaminate, and store a TN-40 dry cask. There is no discussion of how a cask might be unloaded.

Table 6.1 in the EA summarizes the radiological occupational exposures expected to occur as a result of cask loading, decontaminating, and placement on the ISFSI (Attachment C). There is absolutely no mention or discussion of expected radiological occupational exposure during cask unloading.

With respect to cask decommissioning (i.e., at the end of service), the EA only mentions that fuel could be removed from the storage cask and placed in a certified transportation cask for shipment to a repository.

13. On October 19, 1993, NRC issued Materials License No. SNM-2506 to NSP for the ISFSI. Under the terms of the license, NSP is authorized to receive, possess, store, and transfer Prairie Island Nuclear Generating Plant spent fuel at the ISFSI in up to 48 TN-40 casks for a period of twenty years.

Along with the license, the NRC transmitted a Safety Evaluation Report (SER), which is an evaluation and review of the TSSAR submitted by the licensee. The review of the TSSAR addresses the handling, transfer, and storage of spent fuel in a TN-40 dry storage cask at the ISFSI. The SER discusses the design features of the TN-40 (e.g., 40 assemblies, enrichment factors, minimum cooling, etc.), protection against environmental conditions, natural phenomena, confinement barriers (i.e., protection of fuel cladding to ensure that degradation and gross rupture do not occur over the design life of the ISFSI), and criticality control (NRC regulations require that the spent fuel handling, transfer, and storage system be designed to be maintained subcritical (10 CFR 72.124(a)).

The maximum acceptable cladding temperature should not exceed 340 °C (644 °F), according to the SAR, during normal storage conditions to prevent cladding degradation and subsequent gross ruptures.

The SER states that the procedure to unload a TN-40 cask will be a reverse of the loading sequence. No mention is made of the potential safety issues that may be associated with cask unloading (cask reflooding, thermal shock, flash steam, etc.) and potential exposure to workers.

14. Within the ISFSI license (SNM-2506), a number of preoperational license conditions were specified:

A training exercise (dry run) of all TN-40 loading and handling activities must be conducted and shall include the following (but is not limited to);

a. Moving cask in and out of spent fuel pool area
b. Loading fuel assembly (using a dummy assembly)
c. Cask drying, sealing, and cover gas backfilling operations
d. Moving cask to, and placing it on, the storage pad
e. Returning the cask to the auxiliary building
f. Unloading the cask
g. Decontaminating the cask
h. All dry run activities shall be done using written procedures
I. The activities above shall be performed or modified and performed to show that each activity can be successfully executed before actual fuel loading.

As specified by the NRC, the above listed steps did not need to be performed in the order listed.

15. On March 6, 1995, NSP requested an exemption to the requirements of 10 CFR 72.82(e). 10 CFR 72.82(e) requires that "a report of the preoperational test acceptance criteria and requirements must be submitted to the NRC....at least 30 days prior to the receipt of spent fuel or high level radioactive waste." The purpose of this requirement is to allow the NRC 30 days to review and assess the licensee's ability to load (or unload) a cask. NSP requested that they be exempted from the 30 day waiting period following the submission of their preoperational test results (i.e., loading and unloading) and instead be allowed to wait only three days before they loaded the first cask.

16. On April 20, 1995, NSP transmitted to the NRC a report of their preoperational test results, pursuant to the requirements of 10 CFR 72.82(e). Within the correspondence were loading and unloading procedures and work orders requesting that certain aspects of each procedure be tested.

With regard to preoperational testing, NSP staff submitted a work order to internally test certain steps in the unloading procedure (D95.2) in February 1995 (Note: the work orders were included in the loading and unloading procedures). The request specified that steps 7.1 through 8.11 be tested (7.1 prepares the cask for transportation back to the Auxiliary Building, 8.11 requires the removal of the lid).

In a later letter to NSP, however, the NRC noted that they did not require that the lid be removed under water and the cask to be filled with water to prevent any damage to the cask (Attachment D).

17. NRC staff have stated to Prairie Island Indian Community staff that they (the NRC) have no formal mechanism to approve or disapprove loading or unloading procedures. Any deficiencies to the procedures are identified through the NRC's inspection program. A Notice of Violation (NOV) is transmitted to the utility if deficiencies are identified and the utility must correct the problem.

18. On April 28, 1995, the NRC conducted a public exit interview with NSP to present findings relative to dry cask storage activities at the Prairie Island plant. Among other things, the availability of space in the spent fuel pool to hold assemblies from a TN-40, in the case of an emergency off-loading, was questioned (i.e., would there be enough space in the spent fuel pool).

19. On May 3, 1995, NSP provided to the NRC information regarding the unloading of a TN-40 cask. In their letter, NSP stated that they could unload a cask "if such action becomes necessary" (Attachment E).

The unloading procedure transmitted in this letter is comprised of 9 steps, which are essentially the reverse of the loading procedure in the SAR. The information was

requested by the NRC because of questions raised during the April 28, 1995 exit interview regarding the capacity of the spent fuel pool (after a planned outage) to store waste after the first cask has been loaded (i.e., if a cask needed to be unloaded).

20. On May 5, 1995, in response to NSP's submittal of information regarding the unloading of a TN-40 cask, the NRC found the plan for an unanticipated unloading prior to loading another cask would allow ready retrieval of the spent fuel for further processing or disposal as required by 10 CFR 72.122(l).

21. On May 11, 1995, the NRC granted NSP the 10 CFR 72.82(e) exemption. In the end, NSP waited 20 days before they loaded their first casks, instead of the requested 3 days.

22. On May 12, 1995, the NRC approved the preoperational test report and authorized NSP to load the first cask. The first cask was loaded immediately.

23. On June 30, 1995, the NRC issued an Inspection Report covering a variety of dry cask storage activities between January 24 and May 11, 1995 (Attachment D). Among other things, the inspection assessed NSP's performance relative to dry cask storage activities. The Inspection Report noted that the licensee (NSP) "did not complete review and approval of the unloading procedure until the day following the submission of the preoperational test report (emphasis added). Submission of this report [to the NRC] implied that the licensee was ready to load a cask with spent fuel and subsequently unload the cask, if necessary."

In that same Inspection Report, the NRC issued NSP a Notice of Violation (NOV) for not including certain technical specifications in their unloading procedure. The letter specifically cited NSP for omissions in the unloading plan and "overall poor planning for dry cask storage activities." The violations with regard to the unloading plan were technical specification deviations: 1) verification of boron concentrations not adequately specified (the concern is the potential for inadvertent criticality); and 2) verification of fuel integrity not adequately specified; no hold point was identified to ensure that work would not continue until results had been reviewed. NSP later corrected these omissions.

24. On May 30, 1996, at a briefing for the commissioners of the NRC, Andrew Kugler, Lead Project Manager, Dry Cask Storage, NRC, stated while the NRC has found dry cask loading procedures to be acceptable, the unloading procedures are far more complex (Attachment F). In loading a cask, the fuel has been characterized (i.e., its integrity verified), a loading dry run has been performed, and that licensees can take advantage of lessons learned from other licensees (in loading their casks).

Mr. Kugler stated that the older SAR's [Safety Analysis Reports] do not recognize this complexity and indicate that unloading would be the reverse of loading, which Kugler stated was not true. The unloading procedure in NSP's most recent SAR, and the SER for the ISFSI, is described as the reverse of unloading.

Of particular concern with respect to unloading, stated Mr. Kugler, are the potential condition of the fuel and issues associated with the reflooding of the cask with spent pool water: cask pressurization due to steam generation as colder spent fuel pool water is placed into the cask. Thermal shock to the fuel during reflooding and radiological exposure to workers during the operation from the venting the cask (venting will either be done directly to the pool or the ventilation system).

25. On November 8, 1996, NSP submitted to the NRC a revised TN-40 unloading plan.

26. The SAR for the Prairie Island ISFSI states that the fuel cladding may be as hot as 340 °C (644 °F) and spent fuel pool may be as cool as 110 °F. In the SAR, NSP stated that before a cask was returned to the spent fuel pool for unloading "cold water would be pumped into the cavity to reduce the temperature" and that steam might be produced when the water hits the cavity surface. Although the SAR recommended pumping cold water into the cask prior to immersion in the spent fuel pool, the TN-40 unloading procedure contains no such provision (see Attachment G).

27. The Prairie Island SAR also states that the fuel assemblies should be inspected for any physical damage which could potentially cause problems during removal from cask. The actual unloading procedure, however, contains no such requirement. The procedure only requires that the location of three of the forty assemblies be confirmed prior to off-loading (recall that NSP received a NOV for not including the fuel verification requirement in their procedures in June 1995). As Mr. Kugler mentioned on May 30, 1996, the potential condition of the spent fuel, with respect to reflooding, is a concern.

28. The SAR only discussed potential occupational exposure to radiation with respect to receipt of cask, cask loading, decontamination, removal to storage area, and periodic maintenance (Attachment H). There is no mention of potential radiological exposure to workers during cask unloading. The issues raised by Mr. Kugler, with regard to potential increases in radiological occupational exposure during cask unloading are not addressed (especially flash steam due to the hotter temperature of the inside cavity (644 °F) being exposed to the much cooler spent fuel pool water (110 °F).

29. Mr. Kugler stated on May 30, 1996 that there was "essentially no cask unloading experience" for licensees to look back on for lessons learned, like there is for cask loading. In a later letter to Dr. Mary Sinclair of Don't Waste Michigan, Mr. Kugler stated that there were, in fact, three instances where casks were unloaded during the loading evolution, due to problems encountered (Attachment I). In all three instances, however, the loading evolution had not yet been completed and none of the casks had moved beyond the decontamination area (i.e., none of the casks had been removed to the storage area).

30. A dry cask, after it is has been loaded and in storage/use, has never been unloaded

Petitioners Claim

1. The procedure to unload a TN-40 dry cask at the Prairie Island Nuclear Generating Plant has not been adequately evaluated or tested by either NSP or the NRC.

10 CFR 72.122(l) provides that:

Retrievability. Storage systems must be designed to allow ready retrieval of spent fuel or high-level radioactive waste for further disposal or storage.

Our interpretation of 10 CFR 72.122(l) is that the storage system (i.e., the cask) must be designed to allow ready retrieval (i.e., unloading) of spent fuel, not whether the spent fuel pool can accommodate spent fuel from a cask needing to be unloaded. This requirement has not been met because the neither the NRC nor NSP has completely demonstrated whether a TN-40 dry cask can be (or has been) unloaded after it has been sitting on the storage pad for a number of years. Thus the NRC requirement that storage systems be designed to allow ready retrieval of spent fuel has not been met because it has never been fully evaluated. No dry cask has ever been unloaded after it has been in use.

The question of "retrievability" was discussed at a public exit interview convened by the NRC in April 1995. The context in which the issue of retrievability was discussed, however, was wrong. Participants at the public exit interview were asking whether NSP would have enough space in spent fuel pool for the 40 assemblies they planned to load into a TN-40 cask right before a planned outage if an emergency warranted off-loading the first TN-40 cask. The NRC may recall that NSP's spent fuel pool was already full and there was a possibility that, after the planned outage, there would not be enough space in the spent fuel pool for the 40 assemblies (they would be short fifteen spaces). In response to these concerns, NSP assured the NRC that they would indeed have enough space in the spent fuel pool if they moved some non-fuel bearing components. In their letter, NSP calculated the number of spaces in the spent fuel pool they would have available, noting that they would be short by fifteen spaces, and offered two scenarios under which they might make space available. The NRC appeared satisfied that the requirements of 10 CFR 72.122 (l) had been met.

As stated above, NSP submitted their preoperational test acceptance criteria and results (and loading and unloading procedures) to the NRC before they were done testing it. Also as stated above, the NRC cited NSP for omissions in their unloading plan and noted that NSP "did not complete review and approval of the unloading procedure until the day following the submission of the preoperational test report. Submission of this report implied that the licensee was ready to load a cask with spent fuel and subsequently unload the cask, if necessary."

How could NSP have claimed they were ready to load, and subsequently unload a TN-40 cask, if necessary, if they had not completed a review and testing of their own

procedures? If these procedures were not fully tested, can the NRC be sure that 1) the licensee has the ability to unload a TN-40 cask and 2) these casks can be unloaded safely. How can members of the Prairie Island Indian Community feel safe that the casks can and will be unloaded should something go wrong with one of the casks? Many tribal members are fully aware of the dry cask situation at the Palisades plant (i.e., the problem with the VSC-24) and do not wish to see this occur at Prairie Island.

The TN-40 dry cask is in use only at the Prairie Island Nuclear Generating Plant. There are no other plants currently employing a cask designed by Transnuclear Inc. The NRC has licensed the TN-24, but it is not in use anywhere in the country. In affidavit to the Minnesota Court of Appeals, Mr. Jon Kapitz, NSP's Project Manager for Dry Cask Storage, stated that a TN-24P storage cask was successfully unloaded as part of a project sponsored by the Department of Energy (DOE) and the Electric Power Research Institute (EPRI). (Note: this Affidavit was submitted in response to legal action initiated by the Prairie Island Indian Community). In 1987, the TN-24P was tested in a cooperative research program sponsored by the DOE, Virginia Power Company, and EPRI. The purpose of this research was to determine the thermal, shielding, and operational performance of the TN-24P storage cask (not whether it could be unloaded). The testing was conducted at the Idaho National Engineering Lab (INEL) using fuel that has been irradiated at the Surry plant in Virginia. The fuel was moved from Virginia in TN-8 transportation casks (which hold three PWR assemblies). Dry runs, to train personnel, were performed with nonirradiated fuel (i.e., dummy assemblies).

The transfer of spent fuel from the TN-8L into the TN-24P was done in the INEL hot shop in the air, via remote operation. That is, the TN-8 was not placed back into a spent fuel pool for transfer to the TN-24P. With respect to the performance of the TN-24P, the report stated that "the test demonstrated that the cask could be satisfactorily handled and loaded dry" (i.e., not in the pool). Therefore, the issues raised by Kugler, with respect to reflooding a hot cask (flash steam, pressure build-up, fuel integrity, etc.) would not have been addressed in report. This experience does not demonstrate that a fully loaded TN-40 dry cask can be safely unloaded after it has been out on the storage pad.

2) The NRC allowed NSP to load their first TN-40 cask, and subsequent casks, without a full evaluation of the unloading procedure. Both the SAR and SER for the ISFSI stated that unloading a cask is the reverse of loading a cask, implying that it was quite easy to do. NRC staff, however, have stated that cask unloading is quite complex and not the reverse of cask loading.

As stated above, NSP had not finished testing its own loading and unloading procedure before submitting its preoperational test report to the NRC, thereby declaring that they were ready to load, and unload a cask if necessary. According to the actual loading and unloading plan and preoperational test results, only part of the unloading procedure has been tested (refer to Fact No. 16). How can the Prairie Island Indian Community be assured that a cask can be unloaded if the procedure has never been fully tested before and it has never been done?

3. As stated above, NSP submitted a revised unloading procedure to NRC in November 1996. It is our belief that NSP's unloading procedure does not address the problems that likely would be encountered prior to and during the procedure. NRC staff have identified potential unloading problems that have not been addressed in NSP's unloading plan. Our analysis of the unloading plan identified several deficiencies:

a) Failed Fuel Considerations Although the revised plan contains a step to sample the internal gas for fission products (as an indication of fuel failure), there are no procedures on what to do if radioactive air concentrations indicate fuel with gross cladding defects. If the internal gas sampling indicates failed fuel, procedures should be developed to recover the fuel from the cask without contaminating the spent fuel pool. As the unloading procedure is currently written, the dry cask is placed in the pool for unloading, regardless of the radioactive concentration in the internal gas. This action may contaminate the fuel pool with microscopic irradiated fuel particles (fuel fleas), thereby creating a major radioactive hazard. (This has occurred at Southern California Edison's San Onofre 3 reactor).

There are no procedures in place to determine whether fuel in a TN-40 cask has gross cladding defects. During storage, the cladding is under pressure and may crack, due to the high internal temperature of the cask (up to 644°F). Over time, the cracks can become larger. This is a cumulative process; the longer fuel is in storage, the greater the likelihood of cladding degradation. If fuel with cladding defects is unloaded, the spent fuel pool may become contaminated and workers may be exposed to radiation. Plant personnel will have to decide whether or how fuel with cladding defects will be removed.

b) Venting of Radioactive Gases The unloading procedure does not indicate whether or how radioactive gases might be vented from the cask, if concentrations are greater than certain limits. Plant personnel will have to decide how to vent the cask. If the radioactive air is to be vented, is a permit required?

c) Radiation Monitors Before venting the cask, a stop-check must be instituted to verify that ventilation systems and radiation monitors are functioning. There is no such stop-check in NSP's unloading plan.

d) Steam Build-Up When water is pumped into the TN-40, it is very likely that steam will be created and pressure will build within the cask if the fuel cladding is greater than 212°F. Although NSP has included steps to cool down the cask and fuel cladding, it is not clear what the maximum cladding temperature will be prior to the addition of water into the cask. The steam overflow from the cask, which is directed into the spent fuel pool, is likely to be very hot. Warnings need to be posted regarding this hazard.

The pressure build-up also places certain stresses on the hose couplings and piping--this should be evaluated. As the unloading procedure is currently written, water is pumped into the cask at a rate not to exceed 10 psig and the temperature is kept below 240 °F, to

ensure that pressure not build up too quickly. The thermometer measuring the temperature should have a range greater than 50 °F to 300 °F; the maximum temperature should be 900 °F. The pressure gage should also be changed to one with a maximum pressure of 50 psig.

4. The Prairie Island Indian Community respectfully requests that the NRC thoroughly review the unloading procedure written by NSP to ensure that the technical issues raised by Mr. Kugler have been met. The NRC, as a federal agency, has special obligation to protect the trust resources of Indian tribes. Trust resources includes the health and safety of tribal members.

Conclusion

The public relies on the NRC to make decisions that these containers are safe, based on a complete evaluation. Neither the NRC nor NSP have fully evaluated the TN-40 unloading procedure, as evidenced by the lack of documentation regarding cask unloading and potential consequences in either the SAR or the SER. Both the SAR and SER implied that unloading a cask was very straightforward and just the reverse of loading a cask. It now appears that the NRC has somehow reversed itself (with respect to cask safety) by stating that there are now some concerns with regard to unloading a cask. NSP has not adequately demonstrated their ability to unload a TN-40 dry cask.

A dry cask has never been unloaded, there is no certainty that it can be done. The NRC has an obligation to the Prairie Island Indian Community to review the TN-40 unloading procedure to determine whether the issues raised by their own staff and within this petition have been fully evaluated and included in the unloading plan for a TN-40 cask.

Appendix of Attachments

Petition of the Prairie Island Indian Community

RE: Northern States Power procedure to unload a TN-40 dry cask

Attachment A

Table 5.1-1 of the Safety Analysis Report (SAR) and July 10, 1992 letter (NSP to NRC)

Attachment B

Table 5.3 of Environmental assessment (EA) for the ISFSI

Attachment C

Table 6.1 of the EA

Attachment D

June 30, 1995 letter from Greenman (NRC) to Watzl (NSP) and Notice of Violation
Relevant pages only

Attachment E

May 3, 1995 letter to NRC from NSP regarding cask unloading

Attachment F

Transcript from May 30, 1996 Commissioners meeting
Relevant pages only

Attachment G

Step 8.21 of NSP's unloading plan

Attachment H

Table 5.1-2 of the SAR for the ISFSI

Attachment I

June 18, 1996 letter to Dr. Mary Sinclair from Andrew Kugler

Attachment A

NSP

Northern States Power Company

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Minneapolis, Minnesota 55401-1927
Telephone (612) 330-5500

July 10, 1992

10 CFR Part 72

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND INDEPENDENT SPENT FUEL STORAGE INSTALLATION
Docket No. 72-10

Response to NRC Questions on TN-40 Cask Thermal Analysis and on Procedure for Unloading Cask

The attached information is provided in response to questions from the Lawrence Livermore National Laboratory Reviewers related to details of the TN-40 cask thermal analysis and the procedure that would be utilized for unloading a cask at decommissioning.

Please contact us if you have any questions with respect to the attached responses.

Thomas M Parker
Manager
Nuclear Support Services

c: Director, Office of Nuclear Material Safety and Safeguards, NRC
NMSS Project Manager, NRC
Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
Lawrence Livermore National Laboratory
J E Silberg
Prairie Island Independent Spent Fuel Storage Installation Service List

Attachment:

Response to NRC Questions on TN-40 Cask Thermal Analysis and on Procedure for Unloading Cask

NF04.1

3pp

Response to NRC Questions on TN-40 Cask Thermal Analysis and on Procedure for Unloading Cask

Question 1:

The shear stresses at basket plug welds caused by differential thermal expansion between aluminum and stainless steel are not self relieving (secondary) stresses and need to be combined with the primary stresses due to the inertia loads from the accident end drop.

Response:

The thermal analysis of the basket is described in Section 3.3.2.2 of the Prairie Island Independent Spent Fuel Storage Installation (ISFSI) Safety Analysis Report. That analysis was performed to determine the basket temperatures for the condition with maximum solar heating, maximum decay heat from the cask contents, and 100°F ambient air temperature. The temperatures from that thermal analysis were used directly in the ANSYS structural models to calculate the basket panel stresses due to differential thermal expansion. Stresses occur due to differences between the coefficients of thermal expansion of the 304 stainless, the aluminum and boral (see Section 4B of the ISFSI Safety Analysis Report for detailed ANSYS model descriptions). In order to calculate the maximum shear stresses at the 1/2" plug welds, it is conservatively assumed that the 1.38 in. diameter stainless plugs that penetrate the 1.5 in. diameter hole in the aluminum (and boral) plates are not centered. The plugs are assumed to be in contact initially (at 70°F) with the opposing sides of the two holes in the aluminum (the sides toward the center of the panel) so that the maximum interference of aluminum and steel will occur when the panel is heated. In this worst plug misalignment case with the highest temperature (530°F), seen by any portion of the basket, the weld shear stress could reach a maximum of 25,434 psi as shown on Figure 4B.6-6 of the ISFSI Safety Analysis Report.

A full length compartment wall (160 in. long) with a span length of 8.05 in. is evaluated for shear stresses at 1/2" plug welds due to a 50G end drop.

Size of weld = 0.5" dia.

Number of welds = $2 \times 2 \times \frac{160}{8} = 80$ (each 8" spacing has two rows of plugs, each plug has two welds, one on each side)

$$\text{Shear area} = \frac{\pi}{4}(0.5)^2 \times 80 = 15.71 \text{ inches}^2$$

Wt. of aluminum = 2 x 8.05 x 0.25 x 160 x 0.105 = 67.62 lbs.

Wt. of boral = 1 x 7.50 x 0.075 x 160 x 0.0903 = 8.13 lbs.

Total Wt. of aluminum and boral = 75.75 lbs.

Assuming the 50 g compressive load is uniformly distributed to all of the 80 welds, the shear stress is

$$\tau = \frac{75.75 \times 50}{15.71} = 241\text{psi}$$

Appendix F to Section III of the ASME Code provides a basic 0.42 Su limit on the average primary shear stress across a section loaded in pure shear for Level D conditions. The combined shear stress in the weld due to differential thermal expansion and the end drop accident is 25,675 psi which is below the limit of 26,670 psi (0.42 Su) based on the temperature of 530°F at the worst panel location.

Based on the results of this analysis, it is concluded that the basket is structurally adequate for withstanding the combined loads due to thermal expansion and the 18 inch end drop accident, and it will properly support and position the fuel assemblies.

Question 2:

Describe the procedure for cask unloading prior to decommissioning.

Response:

Assuming that the spent fuel in the TN-40 cask will be transferred to a licensed transport cask using a normal "in pool" fuel transfer, the sequence of operations discussed in Section 5.1 of the ISFSI Safety Analysis Report, and in particular listed in Table 5.1-1, will be essentially performed in reverse.

The cask will be moved from the ISFSI back into the Auxiliary Building Rail Bay using the transporter. The weather cover will be unbolted and removed. The overpressure system will then be removed and the cavity gas sampled through the vent port.

After moving the cask into the fuel pool area, the cavity will be depressurized and the cask lowered into the spent fuel pool. With the cask lid at the pool surface, fill and drain lines will be connected to the lid drain and vent ports. Borated water will be slowly added to fill the cask and to gradually cool the fuel in the cask. When the cask is full, the fill and drain lines will be removed. The cask will then be lowered to the pool bottom where the lid would be removed making the fuel accessible for transfer.

TABLE 5.1-1

SEQUENCE OF OPERATIONS

A Receiving

1. Unload empty cask and separately packaged seals at plant site.

2. Inspect the following for shipping damage: exterior surfaces, sealing surfaces, trunnions, seals, accessible interior surfaces and basket assembly, bolts, bolt holes and threads, neutron shield vents.

3. Remove weather shield and install plug in neutron shield vent hole. (threaded hole in the top of the steel shell surrounding the resin which contains a pressure relief valve during storage).

4. Remove lid bolts and lid.

5. Install protective plate over cask body sealing area.

6. Obtain lid and lid seal from storage.

7. Attach lid seal to lid by means of six retaining screws.

8. Move to spent fuel pool area.

B. Spent Fuel Pool Area

1. Lower cask into cask loading pool.

2. Load preselected spent fuel assemblies into the 40 basket compartments.

3. Verify identity of the fuel assemblies loaded into the cask.

4. Remove protective plate from cask body flange.

5. Lower lid and place on cask body flange over the two alignment pins.

6. Lift cask to surface of pool and install lid bolts.

7. Connect drain line to quick-disconnect coupling in the drain port.

8. Bolt special adapter, with quick disconnect coupling, to vent port bolt holes.

9. Connect plant compressed air line to special adapter quick-disconnect coupling.

10. Pressurize cavity to force water from cavity through drain port to the spent fuel pool.

TABLE 5.1-1 (Continued)
SEQUENCE OF OPERATIONS

11. Disconnect plant compressed air line and drain line from their quick-disconnect couplings.

12. Move cask to the decontamination area.

C. Decontamination Area (Rail Bay)

1. Decontaminate cask until acceptable surface dose levels are obtained.

2. Torque lid bolts using the prescribed procedure.

3. Remove plug from neutron shield vent and install pressure relief valve.

4. Connect Vacuum Drying System (VDS) to vent port.

5. Evacuate cavity to remove remaining moisture using prescribed procedure.

6. Break vacuum by closing vacuum valve and opening air valve to admit dry air into the cavity.

7. Disconnect VDS at vent port and install vent port cover with seal and bolts.

8. Connect Vacuum-Backfill System (VBS) to quick-disconnect coupling in the drain port.

9. Evacuate cavity to 10 millibar and backfill with dry helium gas.

10. Pressurize cavity to about 2 atm with helium.

11. Disconnect VBS at the drain port quick-disconnect coupling and install drain port cover with seal and bolts.

12. Perform helium leak test of lid seals.

13. Remove overpressure port cover.

14. Install top neutron shield drum.

15. Torque the bolts using prescribed procedure.

16. Pressurize overpressure system, (seal interspaces), with Helium to a pressure of about 5.5 atm.

17. Perform leak test on overpressure system.

TABLE 5.1-1 (Continued)
SEQUENCE OF OPERATIONS

18. Check external surface temperatures using an optical pyrometer.

19. Check surface radiation levels.

20. Install protective cover with seal and bolts. (could be performed at storage area)

21. Load cask on transport vehicle.

22. Move cask to Storage Area.

D. Storage Area

1. Unload cask from transport vehicle.

2. Position cask in preselected location on storage pad.

3. Check for surface defects.

4. Connect pressure instrumentation to cask and to monitoring panel.

5. Check that pressure instrumentation is functioning.

6. Check surface radiation levels.

Concrete Pad

The storage casks will be stored in two parallel rows of 12 casks on each of two 216-foot long x 36-foot wide x 3-foot thick concrete pads. The two slabs will be positioned end to end with 40 feet in between. To improve foundation performance and earthquake safety, 3 feet of soil beneath each slab will be excavated and replaced with compacted structural fill. The pad elevation will be 693 feet 6 inches above mean sea level (msl) to preclude immersion of the cask seals during the probable maximum flood. They will be surrounded by a 17-foot high earthen berm.

5.3 ISFSI OPERATIONS

Fuel handling and cask loading operations in the Auxiliary Building will be done in accordance with requirements of the Prairie Island Nuclear Generating Plant 10 CFR Part 50 Operating Licenses DPR-42 (Unit #1) and DPR-60 (Unit #2). Cask transport and storage at the ISFSI will be subject to requirements of the Prairie Island ISFSI 10 CFR Part 72 License. The major steps associated with the placing of fuel in the Prairie Island ISFSI are presented in Table 5.3.

TABLE 5.3

ISFSI OPERATIONAL STEPS

A. RECEIVING

1. Unload empty cask and separately packaged seals at plant site.

2. Inspect the following for shipping damage: exterior surfaces, sealing surfaces, trunnions, seals, accessible interior surfaces and basket assembly, bolts, bolt holes and threads, neutron shield vents.

3. Remove weather shield and install plug in neutron shield vent hole.

4. Remove lid bolts and lid.

5. Install protective plate over cask body sealing area.

6. Obtain lid and lid seal from storage.

7. Attach lid seal to lid by means of six retaining screws.

8. Move to spent fuel pool area.

B. SPENT FUEL POOL AREA

1. Lower cask into cask loading pool.

2. Load preselected spent fuel assemblies into the 40 basket compartments.

3. Verify identity of the fuel assemblies loaded into the cask.

4. Remove protective plate from cask body flange.

5. Lower lid and place on cask body flange over the two alignment pins.

6. Lift cask to surface of pool and install lid bolts.

7. Connect drain line to quick-disconnect coupling in the drain port.

8. Bolt special adapter, with quick disconnect coupling, to vent port bolt holes.

9. Connect plant compressed air line to special adapter quick-disconnect coupling.

10. Pressurize cavity to force water from cavity through drain port to the spent fuel pool.

11. Disconnect plant compressed air line and drain line from their quick-disconnect couplings.

12. Move cask to the decontamination area.

C. DECONTAMINATION AREA (RAIL BAY)

1. Decontaminate cask until acceptable surface contamination levels are obtained.

2. Torque lid bolts using the prescribed procedure.

3. Remove plug from neutron shield vent and install pressure relief valve.

4. Connect Vacuum Drying System (VDS) to vent port.

5. Evacuate cavity to remove remaining moisture using prescribed procedure.

6. Break vacuum by closing vacuum valve and opening air valve to admit dry air into the cavity.

7. Disconnect VDS at vent port and install vent port cover with seal and bolts.

8. Connect Vacuum-Backfill System (VBS) to quick-disconnect coupling in the drain port.

9. Evacuate cavity to 10 millibar and backfill with dry helium gas.

10. Pressurize cavity to about 2 ATM with helium.

11. Disconnect VBS at the drain port quick-connect coupling and install drain port cover with seal and bolts.

12. Perform helium leak test of lid seals.

13. Remove over pressure port cover.

14. Install top neutron shield drum.

15. Torque the bolts using prescribed procedure.

16. Pressurize over pressure system with Helium to a pressure of about 5.5 ATM.

17. Perform leak test on over pressure system.

18. Check external surface temperatures using an optical pyrometer.

19. Check surface radiation levels.

20. Install protective cover with seal and bolts.

21. Load cask on transport vehicle.

22. Move cask to Storage Area.

D. STORAGE AREA

1. Unload cask from transport vehicle.

2. Position cask in preselected location on storage pad.

3. Check for surface defects.

4. Connect pressure instrumentation to cask and to monitoring panel.

5. Check that pressure instrumentation is functioning.

6. Check surface radiation levels.

The administrative procedures for the ISFSI will be the same as those used for the Prairie Island Nuclear Generating Plant. Any changes to these procedures will be reviewed and approved by the Station Operations Committee and Safety Audit Committee. Before startup and during the lifetime of the ISFSI, the cask monitoring instrumentation, the electrical system, the communications system, and the storage casks will be tested to ensure their proper functioning. The existing training program at the plant will be used to provide and maintain a well qualified work force for safe and efficient operation of the ISFSI. All personnel working in the fuel storage area will receive radiation and safety training and those actually performing cask and fuel handling functions will be given additional training in specific areas as required by the Radiation Protection program in effect at the Prairie Island Nuclear Generating Plant.

TABLE 6.1

DESIGN BASIS OCCUPATIONAL ONE TIME EXPOSURES DURING CASK LOADING, TRANSPORT AND EMPLACEMENT[1]

Task	Time Required (hr)	No. of persons	Dose Rate (mrem/hr)	Dose (Person-rem)
Placement in pool[2]	2	3	5.0	0.03
Loading process	5	5	5.0	0.125
Removal from pool	5	5	30.0	0.75
Transfer to decontamination area	1	3	30.0	0.09
Processing of cask	6.5	2	30.0	0.39
Helium leak test	2	2	30.0	0.12
Decontamination	2	3	30.0	0.18
Install neutron shield, pressurize, test	3	2	30.0	0.18
Preparation for transport	1	3	30.0	0.09
Transfer of cask to ISFSI	1	3	20.0	0.06
Final cask emplacement	2	5	30.0	0.30
TOTAL				2.315

[1]Dose rates at 1 meter were utilized for all cases except cask transfer, when individuals will typically be at least 2 meters away from the cask.

[2]Steps from Table 5.3.

Attachment D

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

June 30, 1995

Mr. E. Watzl, Vice President
Nuclear Generation
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

Dear Mr. Watzl:

This refers to the special NRC inspection from January 24 through May 11, 1995, of dry cask storage activities at the Prairie Island site. This inspection was conducted by the resident inspectors, selected RIII based inspectors, and technical staff from the Office of Nuclear Reactor Regulation and the Office of Nuclear Materials Safety and Safeguards. The purpose of this inspection was to evaluate the acceptability of the as-built TN-40 cask and to assess your performance relative to dry cask storage including the preoperational testing activities.

We discussed the results of this inspection with you and other members of your staff at a public exit meeting on April 28, 1995. At that meeting we identified five items that required further resolution. You provided us with additional information for each of these items and we completed our review of the subject items during the next two weeks. On May 11, the NRC issued a schedular exemption from the requirements of 10 CFR Part 72.82(e) allowing you to submit the results of your preoperational test less than 30 days before the receipt of fuel at your onsite Independent Spent Fuel Storage Installation. On May 12 you loaded the first cask with spent fuel.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

Based on the results of this inspection, we concluded that you were ready to safely load spent fuel into the TN-40 dry storage cask and transport this cask to the onsite ISFSI. We also did not identify any safety concerns with the subject cask. However, one violation of NRC requirements was identified during the course of this inspection, as specified in the enclosed Notice of Violation (Notice). This violation pertained to cask handling, loading, and unloading activities that were not prescribed by procedures of a type appropriate to the circumstances.

Although 10 CFR 2.201 requires you to submit to this office, within 20 days of your receipt of this Notice, a written statement of explanation, we note that this violation had been corrected and those actions were reviewed during this inspection. Therefore, no response with respect to this violation is required. However, we are disappointed that NRC inspectors, rather than your own staff, identified these procedural deficiencies.

We also identified several weaknesses with your overall performance relative to dry cask storage activities. These weaknesses included: 1) poor oversight of vendor activities until late in the dry cask storage project; 2) lack of effective engineering involvement in vendor fabrication activities; 3) the ineffectiveness of your quality assurance organization in assessing vendor performance during the cask fabrication process; 4) the absence of a comprehensive plan for inspecting, auditing, and monitoring dry cask storage activities onsite, particularly those activities associated with the 10 CFR Part 50 license; and 5) overall poor planning for dry cask storage activities.

Based on the above weaknesses and as discussed at the exit meeting on April 28, we request that you provide us with a formal performance improvement plan documenting the specific corrective actions you have already taken and those you plan to implement to address the above weaknesses in dry cask activities. Please respond to this request within 30 days of the date of this inspection report. We will continue to evaluate the effectiveness of your corrective actions to improve your performance in dry cask activities during future NRC inspections.

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

The response requested by this letter is not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

Edward G. Greenman

Edward G. Greenman
Senior Oversight Manager
Region III Dry Cask Activities

Docket No. 50-282
Docket No. 50-306
Docket No. 72-10

Enclosures:
1. Notice of Violation
2. Inspection Report No. 50-282/95002; 50-306/95002; 72-10/95002(DRP)

See Attached Distribution

Distribution:

cc w/encl: Site General Manager, PINGP
John W. Ferman, Ph.D.,
Nuclear Engineer, MPCA
State Liaison Officer, State
of Minnesota
State Liaison Officer, State
of Wisconsin
Tribal Council
Prairie Island Dakota Community

NOTICE OF VIOLATION

Northern States Power Company — Dockets No. 50-282; 50-306; 72-10

Prairie Island Nuclear Plant — Licenses No. DPR-42; DPR-60; SNM-2506

During an NRC inspection conducted from January 24 through May 11, 1995, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, the violation is listed below:

10 CFR Part 72.142(b) requires a licensee to establish, maintain, and execute a quality assurance (QA) program with regard to an Independent Spent Fuel Storage Installation (ISFSI) that satisfies each of the applicable criteria of Subpart G, "Quality Assurance." In meeting the Part 72.142(b) requirement, 10 CFR Part 72.142(d) accepts a Commission-approved quality assurance program which satisfies the applicable criteria of Appendix B to 10 CFR Part 50. As such, the ISFSI Safety Analysis Report states that the previously approved Northern States Power QA program which satisfies applicable criteria of 10 CFR Part 50, Appendix B, will be applied to activities, structures, systems, and components of the ISFSI commensurate with their importance to safety.

Criterion V of Appendix B to 10 CFR Part 50 requires that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and that these activities be accomplished in accordance with the associated instructions, procedures, or drawings. Cask handling, loading, and unloading are activities affecting quality.

Contrary to the above, cask handling, loading, and unloading activities were not prescribed by approved procedures of a type appropriate to the circumstances as evidenced by the following examples:

1. Surveillance Procedure, SP 1077, "Special Lift Fixture for the TN-40 Cask," did not address dimensional checks of the special lifting device, as required.

2. Surveillance Procedure, SP 1075, "TN-40 Fuel Selection and Identification," did not incorporate the requirement of Technical Specification (TS) 4.1.2, which states that "before inserting a spent fuel assembly into a cask..., the identity of each fuel assembly shall be independently verified and documented."

3. Procedure D95.1, "TN-40 Cask Loading Procedure," specified in the prerequisites section that SP 1077 be performed 30 days prior to loading a cask. However, the TS 4.19-requirement to perform a visual inspection of the lifting device (lift beam and extension) and verify operability of the device 7 days prior to use, was not identified in D95.1. There also was no procedure identifying actions required to verify operability of the lifting device.

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4. Procedure D95.1, "TN-40 Cask Loading Procedure," did not include a step to perform radiation surveys of the cask surface before moving a cask to the ISFSI, as required by TS 4.6.1.

5. Procedure D95.2, "TN-40 Cask Unloading Procedure," did not adequately address the TS-requirement to sample the spent fuel pool for boron concentration within four hours of flooding the cask cavity for unloading the fuel assemblies.

6. Procedure D95.2, "TN-40 Cask Unloading Procedure" did not contain a hold point to ensure work would not continue until the results of the inner cask volume sample had been reviewed. This procedural hold point is important to ensure that an unplanned and unmonitored release path is not created while the cask is in the Auxiliary Building.

7. The licensee did not have a procedure for conducting 10 CFR Part 72.48 safety evaluations.

This is a Severity Level IV Violation (Supplement I) (50-282/95002-01; 50-306/95002-01; 72-10/95002-01(DRP)).

With respect to this violation, the inspection showed that steps had been taken to correct the identified violation and to prevent recurrence. Consequently, no reply to the violation is required and we have no further questions regarding this matter.

Dated at Lisle, Illinois
this 30th day of June 1995

- While the inspectors recognized that finalizing the loading and unloading procedures was contingent upon completion of the dry run and the subsequent incorporation of any lessons learned, there were many aspects of the procedures which should have been in place before the dry run. For example, Technical Specification requirements were not effectively incorporated into the loading and unloading procedures (paragraph 3.2). In addition, the licensee did not complete review and approval of the unloading procedure until the day following submission of the preoperational test report. Submission of this report implied that the licensee was ready to load a cask with spent fuel and subsequently unload the cask, if necessary.

- The licensee did not take a disciplined approach to inspecting the fuel designated for cask storage as evidenced by weaknesses identified by the inspectors during observation of fuel inspection activities (paragraph 7.3).

- Some weaknesses were noted with the licensee's documented basis for safety evaluation conclusions (paragraph 8.2).

operational checks of vehicle brakes, lifting equipment, turntables, jacks, and cask links.

3.1.5 Surveillance Procedure, SP 1075, "TN-40 Fuel Selection and Identification"

The inspectors reviewed SP 1075 and the cask loading procedure, D95.1, to verify that selected Technical Specification (TS) requirements had been incorporated into procedures. Surveillance requirements for ensuring that fuel assemblies which satisfy the criteria of TS 3.1.1 would be loaded into the cask, are defined in TS 4.1.

TS 3.1.1(6) required that, "fuel assemblies known or suspected to have structural defects or gross cladding failures (other than pinhole leaks) sufficiently severe to adversely affect fuel handling and transfer capability shall not be loaded into the cask for storage." The licensee originally intended to visually inspect fuel assemblies designated for loading with binoculars to identify any "structural defects or gross cladding failures." The inspectors questioned the efficacy of this technique to provide a thorough inspection of the fuel. After further discussion with Region III staff on fuel inspection techniques, the licensee elected to use video recording equipment to perform the fuel inspection. The inspectors considered this a preferable method for identifying fuel anomalies and ensuring compliance with TS 3.1.1. The inspectors observed portions of the actual fuel inspection and identified weaknesses with the licensee's approach to this activity as discussed in paragraph 7.3.

During the review of SP 1075, the inspectors identified that the procedure did not incorporate the requirement of TS 4.1.2, which stated that "before inserting a spent fuel assembly into a cask..., the identity of each fuel assembly shall be independently verified and documented." The inspectors discussed the independent verification requirements of TS 4.1.2 with the licensee. Subsequently, the licensee revised SP 1075 to address independent verification of fuel assembly identification. Based on observations of the actual fuel inspection, the inspectors concluded that the licensee met all TS requirements for fuel identification. The failure to incorporate the requirements of TS 4.1.2 into SP 1075 is considered an example of a violation of Criterion V of Appendix B to 10 CFR Part 50 (50-282/95002-01; 50-306/95002-01; 72-10/95002-01(DRP)).

3.2 Loading and Unloading Procedures

The inspectors reviewed the loading (D95.1) and unloading (D95.2) procedures for technical adequacy and to determine if the lessons learned from the preoperational testing/dry run had been appropriately incorporated into the procedures.

3.2.1 D95.1, "TN-40 Cask Loading Procedure"

The original D95.1 procedure specified in the prerequisites section that SP 1077 be performed 30 days prior to loading a cask. However, the Technical Specification (TS) 4.19 requirement to perform a visual inspection of the

lifting device (lift beam and extension) and verify operability of the device 7 days prior to use, was not identified in D95.1. There was no procedure in existence identifying actions required to verify operability of the lifting device. This issue was identified by the inspectors. The inspectors verified that D95.1 had been updated to include the preoperational testing requirements of TS 4.19.

In addition, the original procedure did not include a step to perform radiation surveys of the cask surface before moving a cask to the ISFSI, as required by TS 4.6.1 to ensure compliance with TS 3.6.1. The inspectors discussed this issue with the licensee and verified that D95.1 was revised to include specific steps for performing TS-required gamma and neutron dose rate surveys.

The failure to incorporate the requirements of TS 4.19 into D95.1, to develop a procedure identifying actions required to verify operability of the lifting device, and to include a step for performing radiation surveys of the cask surface before moving a cask to the ISFSI as required by TS 4.6.1, are considered examples of a violation of Criterion V of Appendix B to 10 CFR Part 50 (50-282/95002-01; 50-306/95002-01; 72-10/95002-01(DRP)).

3.2.2 D95.2, "TN-40 Cask Unloading Procedure"

The inspectors identified that the final revised and approved D95.2 unloading procedure did not adequately address the TS-requirement to sample the spent fuel pool for boron concentration. Specifically, TS 4.2.1.2 required verification within four hours of flooding the cask cavity for unloading the fuel assemblies, that the dissolved boron concentration in the spent fuel pool water introduced into the cask cavity was greater than or equal to 1800 ppm. However, D95.2 required sampling four hours prior to lowering the cask in the pool. The inspectors noted that there may be some time delay between partially lowering the cask into the spent fuel pool and filling the cask. The subject TS requirement is important in that it increased the defense-in-depth for ensuring that there was not the potential for an inadvertent criticality. The inspectors verified that D95.2 was revised to incorporate the TS requirement.

The inspectors also verified that the D95.2 procedure contained specific steps for sampling the inner cask atmosphere to verify the integrity of the stored fuel. The inspectors noted that D95.2 did not contain a hold point to ensure work would not continue until the sample results had been reviewed. The inspectors considered this procedural hold point important to ensure that an unplanned and unmonitored release path would not be created while the cask was in the Auxiliary Building. The inspectors verified that D95.2 was revised to incorporate the subject hold point. The inspectors concluded that the final D95.2 procedure contained adequate guidance to ensure that the sampling and other unloading evolutions were performed in a manner that would maintain exposures to workers as-low-as-reasonably-achievable.

The failure of D95.2 to adequately address the TS-requirement for sampling the spent fuel pool and to include a hold point to ensure the results of the inner cask volume sample had been reviewed before allowing work to proceed, are

considered examples of a violation of Criterion V of Appendix B to 10 CFR Part 50 (50-282/95002-01; 50-306/95002-01; 72-10/95002-01(DRP)).

3.3 Emergency/Off-Normal Procedures

The Part 72 license required the licensee to develop an abnormal operating procedure (AOP) for a buried cask event. The inspectors asked the licensee if any other emergency/off-normal procedures were required in addition to alarm response procedures and the buried cask AOP. The inspectors reviewed the cask handling procedures to determine if contingency actions for abnormal events had been addressed.

The licensee does not have any procedures, in addition to the buried cask, which address off-normal events. The inspectors noted that step 5.0, of procedure D95.1, "TN-40 Cask Loading Procedure," stated that, "Should anything not look right during the performance of this procedure it is imperative that the issue be resolved prior to proceeding. All those involved in the performance of this procedure **SHALL** have their questions satisfactorily answered prior to having to perform their task." In addition, to this general precaution, the inspectors noted that D95.1 contained specific "hold points" at various steps in the procedure which required that the loading evolution be stopped and any abnormal condition evaluated before proceeding. The inspectors did not have any further concerns with this issue.

3.4 Conclusions

The licensee did not complete development of the loading and unloading procedures until the day following submission of the preoperational test report. Submission of this report implied that the licensee was ready to load a cask with spent fuel. While the inspectors recognized that finalizing these procedures was contingent upon completion of the preoperational testing evolution or "dry run" and the subsequent incorporation of any lessons learned, there were many aspects of the procedures which should have been in place before the dry run. For example, Technical Specification requirements were not effectively incorporated into the loading and unloading procedures.

Assuming procedural adherence, the final procedures in place for cask handling and loading were adequate to ensure that these evolutions would be conducted safely.

4.0 Audit Reports, Source Inspections, and Vender Records

4.1. Audit and Source Inspection Reports

The inspectors reviewed a sample of the licensee's audit and source inspection reports to determine if there were any issues that could affect the quality of the cask. This review included documentation pertaining to associated audit findings. The inspectors also reviewed several fabrication records to verify compliance with the design basis documents, including applicable industry standards. The following documentation was reviewed:

6.6 Instrument Calibrations

The inspectors reviewed the licensee's procedures for calibrating cask survey instruments and determined that the procedures were adequate to ensure proper calibrations. The inspectors will observe instrument calibrations and survey techniques during the actual cask loading evolution.

6.7 ISFSI Monitoring

The inspectors walked-down the ISFSI facility and ensured that TS-required thermoluminescent dosimeters were in place.

6.8 Radiation Protection (RP) Practices During Preoperational Testing

The inspectors observed RP practices during the loading dry run and noted that workers were kept informed of the radiological conditions and that RP personnel were prompt and thorough in performing dose rate surveys to monitor changing radiological conditions. The inspectors also considered the decontamination techniques used by the RP staff during the dry run adequate to ensure Technical Specification limits for surface contamination of the cask would not be exceeded.

6.9 Neutron Shield Performance

The NRC issued a violation in NRC Inspection Report 72-0010/94-212(NMSS) for inadequate control of special processes pertaining to the neutron shield resin pour during cask fabrication. Specifically, the data record sheet associated with the resin pour procedure indicated that the temperature of the resin mix before adding the catalyst was 63 degrees Fahrenheit rather than between 68 and 70 degrees as required by the procedure. In response to this violation, the licensee committed to perform a thorough survey of the cask following fuel load to verify that the integrity of the neutron shield was not affected by the procedure deviation. The inspectors reviewed the licensee's plans for surveying the neutron shield and determined that the survey techniques were adequate to confirm that the neutron shield was performing its design function.

6.10 Conclusions

With the exception of the procedural content problems discussed in paragraph 3.2, the licensee developed and implemented an effective radiological controls program for monitoring cask loading and unloading activities and storage in the ISFSI. Cask handling procedures and associated RWPs appropriately addressed items such as dosimetry requirements for workers, survey techniques and the use of calibrated instruments, required air sampling, protective clothing requirements, radiation and contamination area postings, and procedural hold points and work stoppage criteria.

7.0 Pre-operational Testing (Loading and Unloading Dry-Run)

The NRC license for the ISFSI required the licensee to conduct pre-operational testing to demonstrate cask handling capabilities before loading the first

cask with spent fuel. The inspectors observed and/or reviewed several pre-operational testing activities. These included: cask arrival and receipt inspection; transport vehicle pre-operational testing; cask transport to/from the ISFSI storage pad/Auxiliary Building; cask pressure monitoring system pre-operational testing; cask vacuum drying, helium backfill, and seal performance testing; fuel inspection; placement of the cask in the spent fuel pool and simulated fuel loading; and cask removal from the spent fuel pool and subsequent decontamination. The removal of the cask lid under water and the filling of the cask with water were two evolutions that were not demonstrated by the licensee during dry run activities. These exemptions were approved by the NRC to prevent any unnecessary damage to the lid seating surface during the dry run and did not affect the licensee's ability to demonstrate unloading. a was performed to demonstrate that the transporter and cask would not tip over during cask transport should a seismic event occur. However, the subject SE did not address the consequences of a tip-over accident in the Auxiliary Building rail bay.

The inspectors discussed this issue with the licensee and with representatives from NMSS. Based on these discussions and the results of a previous analysis involving the loss of all cask confinement barriers during a spent fuel shipping cask handling accident, the inspectors concluded that if a release of radioactivity occurred due to a tip-over event in the Auxiliary Building, the release would be substantially less than 10 CFR Part 100 guidelines. Thus, the inspectors agreed with the licensee's "no" response to the subject question. However, the documented basis was incomplete in that it did not address the consequences of a cask tip-over event within the Auxiliary Building.

While the inspectors noted some weaknesses with the quality of SE No. 344, the inspectors determined that the licensee's conclusion that operation of an ISFSI would not create an unreviewed safety due to an adverse impact on reactor plant operations, was valid.

7.1 Seal Performance Test

The inspectors reviewed the licensee's methodology for performance testing of the cask seals. The lid sealing system was designed with three sets of double O-rings: one set on the circumference of the main lid and one set on the flange covers for each of the vent and drain ports. The spaces between the O-rings for the lid and each flange were interconnected via drilled channels to the overpressure (OP) port. The OP port was connected to the OP tank which was designed to apply helium pressure to the volume of space between all of the O-rings. Should inner-seal leakage occur, helium would leak from the OP tank into the cask (because cask pressure was lower than OP tank pressure). Should outer-seal leakage occur, helium would leak from the OP tank to the environment. OP tank pressure would be monitored on the storage pad and an alarm generated if tank pressure was low. The pressure monitoring equipment was prepared and tested prior to cask transport. After cask placement, the pressure monitoring system would be installed on the cask and tested via a surveillance procedure. Completion of these activities was documented in D95.1.

8.4 QA Overview of Dry Cask Storage Activities

After receipt of the first TN-40 cask at the site, the inspectors determined that the licensee did not have a comprehensive plan to inspect, audit, or monitor dry cask activities onsite, in particular, those activities that interface with the Part 50 license. The inspectors identified several issues that should have been identified by the licensee. After discussion with the inspectors, the licensee developed an "Integrated Dry Cask QA Assessment Plan," which provided direction for the Nuclear Quality Department in the inspection, audit, and surveillance of dry cask storage activities. Once established, the licensee's quality verification efforts were effective in identifying issues with the dry cask storage project which required resolution by the line organization.

8.5 Retrievability

On May 3, 1995, the licensee submitted on the docket, correspondence that addressed the ability to unload the first TN-40 cask following completion of the May 1995, Unit 2 refueling outage and prior to receipt of the second cask onsite. The NRC's Office of Nuclear Material Safety and Safeguards responded on May 5, 1995 to the licensee and stated that the plans described in the May 3 letter to address unanticipated unloading of a cask before another cask had been loaded, would allow ready retrieval of the spent fuel for further processing or disposal as required by 10 CFR Part 72.122(l).

8.6 Exit Interview

The inspectors met with the licensee representatives denoted in paragraph 8.7 during the inspection period and at the conclusion of the inspection on April 28, 1995. The inspectors summarized the scope and results of the inspection, and discussed the likely content of this inspection report. The licensee acknowledged the information and indicated that some of the information disclosed during the inspection could be considered proprietary in nature.

8.7 Persons Contacted

Northern States Power Company

#E. Watzl, Vice President Nuclear Generation
#M. Wadley, Plant Manager
#K. Albrecht, General Superintendent, Engineering
G. Lenertz, General Superintendent, Maintenance
#D. Schuelke, General Superintendent, Radiation Protection and Chemistry
J. Sorensen, General Superintendent, Plant Operations
J. Goldsmith, General Superintendent, Nuclear Generation Services Engineering
#T. Amundson, Director, Generation Quality Services
#P. Kamman, Generation Quality Services
#J. Hill, Manager, Generation Quality Services
#J. Bystrzycki, General Superintendent, Project Management

Attachment E

Northern States Power Company
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East
Welch, Minnesota 55089

10 CFR Part 72

May 3, 1995

RECEIVED
MAY 09 1995

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND INDEPENDENT SPENT FUEL STORAGE INSTALLATION
Docket No. 72-10
Materials License No. SNM-2506

Information Related to Unloading of TN-40 Cask

The attached information is provided in response to questions raised during the April 28, 1995 public meeting on the Prairie Island Independent Spent Fuel Storage Installation. The questions were related to the ability of NSP to completely unload the first TN-40 cask following completion of the May 1995 Unit 2 refueling outage and prior to receipt of the second cask onsite. As shown in the attached assessment, NSP will have the capability to completely unload fuel from the first TN-40 cask back into the spent fuel pool in a timely manner, following the May 1995 Unit 2 refueling outage, if such action becomes necessary.

We have made no new Nuclear Regulatory Commission commitments in this letter or the attachment. Please contact Gene Eckholt (612-388-1121) if you have any questions related to the information provided.

Michael D Wadley for

Roger O Anderson
Director
Licensing and Management Issues

c: Director, Office of Nuclear Material Safety and Safeguards, NRC
NMSS Project Manager, NRC
Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
J E Silberg
Prairie Island Independent Spent Fuel Storage Installation Service List

Attachment: Assessment of Capability to Unload TN-40 Cask Following May 1995 Unit 2 Refueling Outage

9505100025 3pp.

ASSESSMENT OF CAPABILITY TO UNLOAD TN-40 CASK FOLLOWING MAY 1995 UNIT 2 REFUELING OUTAGE

The Prairie Island spent fuel pool is designed and licensed to store 1386 fuel assemblies. Due to inaccessible locations and other non-fuel bearing components, the practical storage capacity is normally considered 1362. After the May 1995 Unit 2 refueling outage there will be a total of 1377 spent fuel assemblies on site, 1337 in the pool and 40 in the first TN-40 cask. Using the practical storage capacity of 1362, this would leave 25 spaces available in the spent fuel pool that could be used for storage of spent fuel from a cask. Thus, 15 additional pool locations would be required to completely unload a TN-40 cask back into the spent fuel pool.

However, 15 of the non-fuel bearing components noted above can be temporarily relocated as described below to provide the 40 pool locations required to unload a TN-40 cask. These non-fuel bearing components could be relocated by either of the following processes;

1. Relocation to Temporary Pool Location:

   Move non-fuel bearing components to a temporary location in the pool (most likely the fuel transfer canal). A conceptual design of the hardware required for this temporary storage has been developed. We estimate the required hardware could be fabricated and the non-fuel bearing components relocated to their temporary locations in approximately 1 working week. This would adequately support any credible situation requiring cask unloading.

or,

2. Relocation to TN-40 Cask:

   Even though the TN-40 cask being returned to the spent fuel pool may not be qualified to hold spent fuel, it quite possibly could still safely hold irradiated non-fuel bearing components. If this is the case, as the TN-40 cask is being unloaded, the required non-fuel bearing components could be relocated, on a temporary basis, into the TN-40 cask. The cask would then be removed from the pool until another cask is available to remove the spent fuel. Following loading of the replacement cask, the non-fuel bearing components would be relocated back to the spent fuel pool.

Conceptually, the following basic operations would be required to perform these options:

<u>Relocation to Temporary Pool Location</u>:

1. Bring cask back from the ISFSI into the plant Auxiliary Building.

2. Move cask into the spent fuel pool and remove lid.

3. Fabricate the hardware necessary to temporarily relocate non-fuel bearing components to the transfer canal.

4. Relocate non-fuel material to transfer canal.

5. Off-load fuel from the cask into spent fuel racks.

6. Remove cask from the spent fuel pool.

7. Repair existing cask or provide replacement cask.

8. Load the repaired or replacement cask with fuel and return it to the ISFSI.

9. Relocate the non-fuel bearing components back into spent fuel racks.

Relocation to TN-40 Cask:

1. Bring cask back from the ISFSI into the plant Auxiliary Building.

2. Move cask into spent fuel pool and remove lid.

3. Remove fuel assemblies from the cask, place them back in the spent fuel racks, and relocate the required non-fuel bearing components into the cask.

4. Replace lid and remove cask from spent fuel pool.

5. Relocate fuel from the spent fuel pool into a replacement cask.

6. Move the loaded replacement cask to the ISFSI.

7. Place the original cask back into the spent fuel pool.

8. Relocate the non-fuel bearing components back into the spent fuel pool.

9. Remove the empty cask from the spent fuel pool and repair if possible.

In summary, either of the options described above would allow NSP to completely unload fuel from the first TN-40 cask back into the spent fuel pool in a timely manner, following the May 1995 Unit 2 refueling outage, if such action becomes necessary.

Next slide, please.

[Slide.]

MR. KUGLER: In terms of the procedures hemselves, the inspectors have found the loading procedures o be acceptable. There are a number of factors that implify the preparation of loading procedures as compared o unloading procedures. During loading process you've haracterized the fuel; you know what condition it is in as ou put it into the cask. Also you can take advantage of essons learned from other licensees and from the dry runs hat the licensee performs on site.

For the unloading procedures, what we are finding s that they are more complex than the loading procedures.

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nfortunately some of the older SARs fail to recognize this nd tend to indicate that unloading is simply the reverse of oading, which is not true. For one thing, licensees need o consider the potential condition of the fuel when they go o unload it. Depending on the situation, the fuel may have een in the cask for decades, and they need to evaluate the ondition of the fuel to the extent possible before they tart unloading it.

We do put an inert environment into these casks to revent oxidation of the fuel. Assuming that that nvironment has been maintained, the fuel should be in good ondition when they go to unload it, but they need to valuate.

There are also issues associated with the eflooding of the cask. During the unloading process we ave to refill the cask with water. There are some issues ssociated with that such as cask pressurization due to team generation as you put cold water onto the hot fuel. lso the consideration of any thermal shock to the fuel as ou are reflooding it, and also radiological protection for he workers during that phase, because you will be venting he cask. Generally they are going to direct that venting ither to the pool or to a ventilation system, but they need o consider that.

In addition, there is essentially no cask

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nloading experience for them to look back on for lessons earned. So they don't have that information available to hem as compared to loading procedures.

In addition to the working group activities, the taff has been putting increased emphasis on our inspection ctivities in this area. The procedures for the recently uilt facilities have been inspected during the reoperational phase using the new inspection procedures hat Bill Travers had mentioned. These inspections were a oint effort between the regions, NRR and NMSS. We asically pool our resources and our expertise to perform hose inspections.

We plan to continue those inspections for all uture facilities.

We are also taking a look back at some of the old acilities and looking at what inspections have been erformed there to determine whether we feel that we have ocumented well enough that those procedures have been nspected. If we determine that these older facilities were ot well documented, we are going back and take a look at hem as well and do further inspections in those locations.

That is all I planned to say on loading and nloading. If there are no questions, I will turn it over o Charlie to talk about NRR staff initiatives.

NOTE:	While the Aux Building Crane is moving the cask into the Spent Fuel Pool, the crane switch will be in "CRITICAL" position. In this condition, the crane will be unable to move more than 1 inch east or west once it passes the roof slot centerline and is within 6 feet either side of the enclosure.

8.17 Place the Aux Building Crane in the "CRITICAL" mode.

8.17.1 Turn the key switch on the crane controls to the "CRITICAL" position.

_____ _____
Rigger Date

8.17.2 Key switch on the crane controls verified in the "CRITICAL" position by a rigger different from the rigger who changed the key switch position.

_____ _____
Rigger Date

8.17.3 Place the key in the key cabinet in the Maintenance Supervisor's Office. _____

8.18 Within four (4) hours prior to loading fuel, verify SFP boron concentration is >2280 ppm. Two samples must be drawn and analyzed independently by two separate individuals. Record sample results on the Cask Loading Report, Appendix A. _____

8.19 Open the spent fuel pool enclosures roof hatches. _____

8.20 Raise the cask to the 755' level and position it over the pool. (Figure 10) _____

8.21 Slowly lower the cask into the spent fuel pool while spraying the cask and lift beam with demineralized water to provide a film of clean water on the cask surfaces. _____

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TABLE 5.1-2

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR CASK HANDLING OPERATIONS

Operation	No. of Personnel	Time (min)	Avg. Distance (ft) from Cask
Receiving			
1. Unloading (A1)	*	*	*
2. Inspection (A2 through A7)	*	*	*
3. Transfer to cask loading pool (A8)	*	*	*
Cask Loading Pool			
4. Lower cask into pool (B1)	*	*	*
5. Load fuel (B2 through B4)	5	*	*
6. Place lid on cask (B5)	5	*	*
7. Lift cask to pool surface (B6)	5	30	5
8. Install lid bolts (B6)	5	120	3
9. Drain cavity (B7 through B11)	5	90	6
10. Transfers to decontamination area (B12)	3	60	10
Decontamination Area			
11. Decontaminate cask (C1, C2)	3	120	3
12. Remove vent plugs	2	30	5
13. Drying, evacuating, backfilling (C3 through C13)	2	480	5
14. Install top neutron shield C14)	2	15	3
15. Install pressure transducers (C15 through C17)	2	30	5
16. Pressurize interspace (C18)	*	*	*
17. Check leakage (C19)	2	30	5
18. Check surface temperature (C20)	2	30	5
19. Check surface dose rate (C21)	2	30	3
20. Install protective cover (C22)	2	30	5
21. Load on transport vehicle (C23)	3	60	5
22. Transfer to storage area C24)	3	60	10

TABLE 5.1-2 (Continued)

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR CASK HANDLING OPERATIONS

Operation	No. of Personnel	Time (min)	Avg. Distance (ft) from Cask
Storage Area			
23. Unload from vehicle position in location (D1, D2, D3)	5	60	5
24. Check surface dose rate (D6)	5	30	3
25. Connect pressure instrumentation (D4, D5)	5	30	5
Periodic Maintenance			
1. Visual surveillance (NA)	2	15	5
2. Repair surface defects (NA)	2	60	3
3. Instrument testing and calibration (NA)	2	180	5
4. Instrument repair (NA)	2	60	3
Major Maintenance (once in 20 years)			
1. Replace cask lid seals	3	1950**	8

* No measurable dose associated with this activity. Therefore, the number of personnel, time and distance are not significant.

\# Parenthetical information corresponds to Table 5.1-1 activity numbers.

** Total time to transfer cask to spent fuel pool, replace lid seals, and return cask to ISFSI pad.

Initial File
50-255
72-7
72-1007

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

June 18, 1996

Dr. Mary Sinclair
Don't Waste Michigan
P.O. Box 1002
Monroe, Michigan 48161

Dear Dr. Sinclair:

In a letter to you dated December 28, 1995, I stated that the staff was not aware of any U.S. Nuclear Regulatory Commission (NRC) approved dry storage casks that had been unloaded by NRC reactor licensees. Since that letter, the Wisconsin Electric Power Company, the licensee for the Point Beach plants, unloaded a cask after a hydrogen ignition event that occurred when the licensee initiated welding for the closure of the shield lid. In addition, I have recently learned of two past cases in which licensees have unloaded casks after identifying problems during the loading process. In all three cases, the licensee had not completed the loading process and none of the affected casks were moved beyond the decontamination area. The purpose of this letter is to provide you with information related to these activities.

The first case occurred at the Surry Power Station in Virginia in November 1986. The licensee had loaded the first Castor V/21 cask, a design that uses two bolted lids with seal rings. The inner lid had been installed and the cask was drained, but the seal ring failed the helium leak test required for this type of closure. The licensee moved the cask back to the spent fuel pool, reflooded it, and lowered it into the cask pit. When the lid was removed, the licensee determined that the seal ring had shifted out of its groove due to hydrodynamic forces as the lid was placed onto the cask. The licensee revised the loading procedure to lower the lid more slowly and the problem has not recurred.

The second case happened at the H.B. Robinson Nuclear Plant in February 1989. The licensee had loaded the first NUHOMS-7P cask and moved it to the decontamination area. In preparation for welding the inner (shield) lid, the licensee performed a survey of the radiation dose rates above the lid. The licensee found dose rates that were higher than those predicted and decided to move the cask back to the spent fuel pool and unload it as part of the investigation of the dose rates. The cask had not yet been drained and so issues associated with reflooding a cask were not applicable. The licensee determined that variation in the dose rates was within the accuracy of the methods used for dose prediction and well within the limits in the technical specifications. The licensee then proceeded to load the cask.

Information concerning these two cases was not widely known within the NRC staff because the problems that led the licensees to unload the casks were not safety-significant. In both cases the licensees found the problems through appropriate testing or monitoring and took prompt, conservative corrective actions.

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

June 20, 1996

Attachment I

M. Sinclair

The event at Point Beach occurred on May 28, 1996. When the licensee struck a spark with the automated welding machine to begin welding the shield lid, a hydrogen burn occurred that generated sufficient pressure in the cask to displace the lid somewhat, leaning it cocked at an angle. As in the case at Robinson, the cask had not been drained (except for a small area near the top of the cask to facilitate welding). After investigating the situation and determining a safe course of action, the licensee straightened the lid, filled the air space at the top of the cask with water, and moved it back into the spent fuel pool. The licensee then removed the shield lid and unloaded the fuel. The NRC staff, including an Augmented Inspection Team, monitored licensee activities during this event. The investigation of this event and its implications is continuing. The NRC has issued confirmatory action letters (CALs) to the licensees for the independent spent fuel storage installations at Point Beach, Palisades, and Arkansas Nuclear One[1] (licensees using the VSC-24 cask). The CALs document the agreement of these licensees to refrain from loading or unloading casks until after they have completed the actions discussed in the CALs and contacted the NRC.

I trust that this information will be helpful to you. I apologize that the information contained in my December 28, 1995, letter was not completely accurate. If you have any questions, please contact me.

Sincerely,

ORIGINAL SIGNED BY:

Andrew J. Kugler, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

cc: Mr. Richard W. Smedley
Consumers Power Company

See next page

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[1] The licensee for this facility has not yet loaded any casks.

REISSUED