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

JUN 20 2002

MEMORANDUM TO: James E. Dyer
Regional Administrator, RIII

FROM: Mary Jean Pool
Acting FOIA/PA Officer
Office of the Chief Information Officer

SUBJECT: REVIEW OF RECORDS RELATING TO A DIFFERING PROFESSIONAL VIEW

By memorandum dated May 28, 2002, you requested that my office coordinate the review of records, in accordance with NRC Management Directive 10.159, relating to a Differing Professional View for placement in the public domain. The review of the subject records has now been completed.

The records identified on Appendix A should be made publicly available by your region.

Attachments: As stated

APPENDIX A

NO.	DATE	DESCRIPTION/(PAGE COUNT)
1.	5/23/01	Memorandum to J Dyer from R Landsman, Subject: Differing Professional View Concerning the Startup of the Dry Cask Storage Loading Campaign at Dresden (3 pages)
2.	5/23/01	E-Mail from J Dyer to R Landsman, Subject: DPV Memo Dated 5/23/01 (1 page)
3.	7/11/01	Memorandum to J Dyer from R Landsman, Subject: Supplement to DPV Dated 5/23/01 (17 pages)
4.	7/13/01	Memorandum to R Landsman from J Dyer, Subject: Differing Professional View Concerning the Startup of the Cask Storage Loading Campaign at Dresden Units 2 and 3 (1 page)
5.	7/16/01	E-mail from R Landsman to B Berson, Subject: DPV (1 page)
6.	7/20/01	Memorandum to J Grobe from J Dyer, Subject: Ad Hoc Review Panel for Differing Professional View Concerning the Startup of the Cask Storage Loading Campaign at Dresden Units 2 and 3 (21 pages)
7.	4/2/02	Memorandum to J Dyer from J Grobe, Subject: Recommendation of AD Hoc Review Panel for Differing Professional View: Startup of Cask Storage Loading Campaign at Dresden Units 2 and 3 (15 pages)
8.	9/21/01	E-Mail from J Grobe to J Dyer, Subject: Dresden DPV - Update (4 pages)
9.	8/13/01	Letter to O Kingsley from C Pederson, Subject: NRC Inspection Report 07200037/2001-002; Dresden - Preparations for Spent Fuel Loading Into Dry Storage Casks (31 pages)
10.	12/28/01	Memorandum to J Zwolinski et al., from J Grobe, Subject: Differing Professional View Concerning Structural Issues Regarding the Dresden Reactor Building/125 Ton Crane and the Spent Fuel Cask Transfer Facility (5 pages)
11.	6/1/01	Memorandum to License File Nos. DPR-19, DPR-25 from B Jorgensen, Subject: Meeting with Exelon Regarding Dresden Unit 2/3 Reactor Building Crane Issues (40 pages)
12.	5/31/00	Certificate of Compliance for Spent Fuel Storage Casks (5 pages)

13.	11/2/01	Memorandum to M Dapas from J Grobe, Subject: Differing Professional View Regarding Structural Issues on the Dresden Reactor Building and 125 Ton Crane (2 pages)
14.	9/29/81	Design Report for Rector Building, Crane Bridge Girder Evaluation and Repairs, Dresden Nuclear Power Station, Units 2 and 3 (27 pages)
15.	3/16/98	Commonwealth Edison Company, Calculation NO. DRE98-0020 (18 pages)
16.	4/30/02	Memorandum to R Landsman from J Dyer, Subject: Resolution of Differing Professional View on Startup of Cask Storage Loading Campaign at Dresden Units 2 and 3 (17 pages)
17.	5/3/02	Letter to O Kingsley from C Pederson, Subject: Notification of a Potential Non-Compliance Issue (4 pages)
18.	6/15/01	Memorandum to C Pederson from E Brach, Subject: Facility Evaluation of Dresden Cask Transfer Facility (10 pages)
19.	2/4/02	Memorandum to J Grobe from W Brach, Subject: Response to Differing Professional View Structural Issues Regarding the Dresden Spent Fuel Cask Transfer Facility (4 pages)
20.	2/12/02	Dresden CTF Weld Documentation Requirements (1 page)
21.	6/15/01	Memorandum to C Pederson from C Carpenter, Subject: Staff Review of Dresden Reactor Building Crane Issues (6 pages)
22.	2/22/02	Memorandum to J Grobe from J Zwolinski, Subject: Response to Request for Headquarters Input on Differing Professional View Concerning Seismic/Structural Analysis for Dresden Units 2 and 3 Spent Fuel Cask Handling (7 pages)

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

May 23, 2001

MEMORANDUM TO: J. E. Dyer, Regional Administrator

FROM: Ross B. Landsman, Project Engineer, DNMS Ross B Landsman

SUBJECT: DIFFERING PROFESSIONAL VIEW CONCERNING THE STARTUP OF THE DRY CASK STORAGE LOADING CAMPAIGN AT DRESDEN UNITS 2 & 3

It is ironic that in 1976, we allowed the licensee to use the unacceptable Unit 2/3 Reactor Building Crane for handling fuel casks. This was temporary and only while the Unit 2 Reactor was shut down. If we didn't, it would impact the licensee's schedule for fuel handling (see licensee's letter dated 5/20/1976). It should be noted that they have loaded greater than 68 fuel casks between 1975 and 1984 with Unit 2 on-line which was contrary to technical specifications after the amendment on June 3, 1976. The four loaded prior to the amendment were contrary to their original license.

We gave them a temporary waiver on the installation of three planned modifications; one of which appears was never installed, the inching motor. Another issue we gave them a temporary bye on was the strength of the main cable (wire rope). The factor of safety was not acceptable. We told them to tell us when they would replace the rope with the appropriate rope. Twenty-six years have passed with the same unacceptable rope. NUREG-0554, paragraph K-1, requires a 10 to 1 factor of safety considering both static and dynamic loads. The factor of safety on the rope, on the lead line is only 6.564 to 1 based only upon a static load. A dynamic load increase of approximately 10% would reduce the factor of safety to under 6.0. Additionally, a critical item of NUREG-0554 is the hook which also requires a factor of safety of 10 to 1; it's at 8.5 to 1.

The required load cell was jumpered out of service before 1981. Without the load cell, the crane may have been overloaded numerous time during the 20 years.

We gave them a bye on the correct seismic design of the crane and supporting Reactor Building Superstructure because it was "not practicable." It would have required a new bridge and extensive modifications to the supporting structure. They told us the existing crane and support structure would be evaluated for OBE loads with AISC allowables used, and SSE loads with a maximum of 0.9Fy used for material strength. It should be noted they had to use greater than Fy for the building material strength to get the interaction coefficients at or equal to 1.0. These calculations were performed without lifted load included to determine if any revisions to beams would be required. See Inspection Report 07200037/2001-001 to see how well the required mods were implemented.

Even then, all these issues were contrary to Branch Technical Position APCSB 9-1 (the fore-runner of NUREG-0554 and NUREG-0012), and we still approved the crane (temporarily) and let them carry casks over safety related equipment unanalyzed. At least the reactor was supposed to be shut down.

Here we are, 25 years later, with the same unresolved issues, along with additional critical issues, letting them go again because of ...

In 1981, the crane was not load tested following extensive needed repairs of the girder as required by ANSI B-30.2. The licensee's calculations indicated that without the repairs, the main girders would be over stressed by 20%. Thus, necessitating repairs to restore it to Operability. Portions of the web plates were cut out and replaced along with the addition of cover plates over the bottom flanges. The licensee classified this as a minor repair and we are agreeing with them?

In 2001, in an attempt to restore the crane to Operability (because of years of neglect), a "major" (licensee's word) crane modification was performed which replaced all the crane controls including over 700 new electrical - terminations. Again, without the B-30.2 required load test and we are agreeing with them again?

On May 13, 1996, in their response to us from Bulletin 96-02, they re-affirmed their commitment to us not to carry heavy loads over safety related equipment while the reactor was at power because it's prohibited by Technical Specifications. They further stated that if such movements would be done in the future, they would demonstrate that they can safely shutdown the reactor as a result of a load drop inside the building. Later on in 1996, we allowed the licensee to remove all requirements and restrictions from the Technical Specifications concerning the Reactor Building Crane, and implement them through administrative procedures, which the amendment reviewer never saw (see Amendment dated June 28, 1996.) However, the requirement to not move heavy loads over safety-related equipment at power was conveniently never heard from again. When the inspector informed the licensee that they had a commitment (which did make it into the procedure), to demonstrate that they could shut down the reactor if there was a load drop, the licensee subsequently also deleted that commitment.

The licensee deleted the commitment "to demonstrate the capability of performing the actions necessary for safe shutdown in the presence of the radiological source term that may result from a breach of the dry fuel storage cask, damage to the fuel, and damage to safety-related equipment as a result of a load drop inside the facility." Their rational was that the Reactor Building Crane is single-failure-proof and thus a load drop analysis is not required to be performed.

Even though HQ concluded that the deficiencies noted above exist, they do not create an imminent threat of adequate protection, and no NRC action to intervene is required; there still is the unanswered question of does the proposed activity increase the consequences of an accident. HQ conclusion was based upon the fact that we issued a paper 25 years ago that said it was single-failure-proof. They also indicated that since the crane has operated for many years without dropping a load, i.e., the rope or repaired girder haven't failed, nor has an earthquake occurred during prior fuel cask handling, it must be ok.

Prior commission approval is required if the proposed change, test or equipment involves a change to commitments incorporated in the license or an unreviewed safety question exists.

Both of these are in effect here. Commitments made to the NRC have been deleted and there is an unreviewed safety question in moving the cask over the torus and other safety-related equipment while the reactor is at power.

An unreviewed safety question exists (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The proposed cask movement activities represent an unreviewed safety question that should be submitted for NRC review and approval per 10 CFR 50.59 and 50.90. This is based on the movement of loads heavier than those previously analyzed in the FSAR. This is also based on the fact that the load drop had not been previously evaluated, and on the possibility that a drop in the reactor building while the reactor is at power could result in consequences that are greater than those previously postulated in the FSAR.

Therefore, although the licensee had reduced the probability of dropping the cask, a load drop could result in an increase in the potential consequences, accordingly, as defined in 50.59(c), if an activity is found to involve an unreviewed safety question, an application for a license amendment must be filed with the commission pursuant to 50.90.

In summary, allowing Dresden to use the reactor building crane with an unacceptable rope, an untested crane, a crane or building structure that wouldn't support the load in an earthquake, all while Unit 2 is at power, does not meet the intent of our regulations and should be stopped. Furthemore, we stopped Oyster Creek in 1996 from doing the same thing with an identical non-single-failure-proof crane. Why is this different?

From: Jim Dyer, R3
To: Ross Landsman / R3
Date: Wed, May 23, 2001 3:52 PM
Subject: DPV Memo Dated 5/23/01

Ross,

I received your memo to me identifying your disagreement with the NRC decisions concerning the startup of dry cask storage loading activities at the Dresden Station. This memo identified several issues concerning the historical use of the crane as well as its current configuration and readiness to safely conduct cask loading activities.

As we discussed this morning, it is premature to start the DPV process on this issue before the NRC has come to a decision on these issues. We are focusing our current efforts on identifying the activities necessary for cask loading and will disposition the historical issues via the enforcement process. A Region III public meeting is being held on the subject today with the licensee and further inspection activities are planned in the near future. Therefore, as we agreed to earlier, I will hold your DPV until a NRC decision is made and then review with you the alternatives and whether you wish to proceed with establishing a DPV panel in accordance with MD 10.159. I discussed this approach with Jim McDermott, OHR, and he agreed with this deviation from the MD 10.159 timeline for establishing the DPV panel.

Additionally, I want to thank you for allowing me to provide copies of your memo to the NRC staff to better prepare for the public meeting today and facilitate further deliberations on the subject. I hope that you fully participated in today's public meeting and raised your concerns to the licensee and attending NRC staff.

Jim.

CC: Bruce Berson; James Caldwell; James McDermott; ...

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

July 11, 2001

MEMORANDUM TO: J. E. Dyer, Regional Administrator

FROM: Ross Landsman, Project Engineer, DNMS RBL

SUBJECT: SUPPLEMENT TO DPV DATED MAY 23, 2001

I agreed with you that the DPV was premature because RIII management had not made a decision on my issues. However, I wanted to issue it anyway so you could better understand the issues. Since that time, additional issues were identified during my inspection. These reinforced my reasons why the licensee should not use either the Unit 2/3 reactor building crane or the cask transfer facility (CTF) which I didn't have time to inspect before the original DPV.

These issues include:

- The reactor building is not designed for the 125 ton crane load, making the building unsafe to use.
- The cask lifting yokes do not meet our ANSI N14.6 requirements.
- The adequacy of the structural capability of the CTF cannot be determined based upon existing records.

These and other issues are included in my draft report which is attached.

Subsequently, RIII management allowed the licensee to load the first cask even though there were at least 15 violations of NRC requirements, some significant conditions adverse to quality. For example, the reactor building structural steel issue is identical to the one we issued in 1996, resulting in a $100,000 fine to Dresden and not allowing them to start up the units until the beams were brought back to within design allowables.

Please start the DPV process.

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Report Details[1]

1.0 General

This special inspection examined design, fabrication and testing of equipment for use in removing spent fuel from the Unit 2 fuel pool into components of the Holtec dry fuel cask system. The Unit 2/3 reactor building crane and the Cask Transfer Facility were examined in detail. Previously identified unresolved items were examined further and were determined to be acceptable items.

2.0 Handling of Heavy Loads

2.1 Unit 2/3 Reactor Building Crane

2.1.1 Review of Previously Identified Items

a. Inspection Scope

Licensee actions, responses or clarifications regarding an Unresloved Item (URI 07200037/2001-001(DNMS)) were examined. The URI consisted of a number of individual elements. These elements are addressed below in the same sequence as originally documented.

b. Observations and Findings

USFAR Commitments

- Safety Lugs

The previous inspection raised a question regarding whether safety lugs were installed on the Unit 2/3 crane trolley and bridge rails. The Updated Final Safety Analysis Report (UFSAR) in Sections 6.2.3.2.1 and 9.1.4.2.2 describes provisions made (safety lugs) to ensure the Unit 2/3 reactor building crane trolley and bridge do not become dislodged during an earthquake. During this inspection, the inspector was able to verify that safety lugs were in place on the trolley. The inspector could not determine whether safety lugs were in place on the bridge rails. The licensee subsequently determined that the bridge did not have the specified safety lugs. They had apparently never been installed.

The lack of specified safety lugs on the crane bridge rails, contrary to the description of the UFSAR, is considered a *de facto* design change. No safety evaluation was performed under 10CFR50.59 to establish that this change did not constitute an unreviewed safety question. 72.212(b)(4), requires in part, that prior to use of the general license, activities related to storage of spent fuel shall be evaluated for any unreviewed facility safety question, as provided under 50.59. Results of this determination must be documented.

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[1]NOTE: A list of acronyms used in this report is included at the end of these Report Details.

Contrary to the above, as of June 22, 2001, the licensee failed to document and evaluate that the Unit 2/3 reactor building crane bridge trolleys, as described in the UFSAR, do not have safety lugs installed.

- Wire-Rope Safety Factor

The previous inspection raised a question regarding the safety factor for the crane wire-rope. NRC's current guidance for crane cables is contained in NUREG-0554 and NUREG-0612, which were issued after Dresden Amendment No. 22 for Unit 2 and Amendment No. 19 for Unit 3, and recommend a safety factor of 10 to 1.0. A safety factor of 10 to 1.0 is not a requirement for Dresden Units 2 and 3.

It appeared the wire-rope on the Dresden Unit 2/3 crane had a safety factor of 8 to 1.0, per the UFSAR, Table 9.1-3, but the inspector found that it actually had a factor of 7.798 to 1.0 in the licensee's submittal.

10 CFR 50, Appendix B, Criterion III, requires in part, that measures shall be established for the identification and control of design interfaces. The design control measures must provide for verifying or checking the adequacy of design.

Contrary to the above, as of June 22, 2001, the licensee failed to state the correct wire rope safety factor in the UFSAR.

On June 3, 1976, in Amendment Number 22 for Unit 2 and Amendment Number 19 for Unit 3 the staff accepted the wire-rope static safety factor of 7.798 to 1.0 and the lead line safety factor of 6.564 to 1.0 even though it didn't meet the Branch Technical Position (BTP) 9-1 requirements. To compensate for this, the staff incorporated LCO and surveillance requirements in the Technical Specifications. Specifically, inspection requirements in accordance with ANSI B30.2. It also limited the fuel casks weight to 100 tons.

The NRC wrote to the licensee on January 30, 1976, that since the wire rope safety factors were not acceptable, provide a proposed inspection/replacement program for the wire rope. The licensee responded on March 2, 1976, that the ropes would be inspected and if required, replaced to assure compliance. Through the years, the licensee has been replacing the rope with like-for-like without ever considering replacing it with a rope that met 1976 standards or meets today's criteria of a 10.0 to 1.0 safety factor. The licensee is considering providing an additional safety enhancement by replacing the rope the next time with a 10.0 to 1.0 margin rope.

- Overload Protection

The previous inspection identified an issue relating to the apparent lack of overload protection on the Unit 2/3 crane hoist.

The initial licensee submittal in support of Amendment No. 22 for Unit 2 and Amendment No. 19 for Unit 3, Dresden Special Report No. 41, stated that a load sensing readout with high and low limit cut-offs will be provided as an overload protection feature. UFSAR section 9.1.4.2.2 states a digital-type weight indicator for the main hoist is provided. When the weight to be lifted is above the setpoint on the weight indicator, the control circuit for the slow speed motor will prevent its operation and the main hoist brakes will set.

Since initial installation of the load cell in 1976, a review of the history of the system showed it has been out-of-service because the license has been jumpering it out because of repeated problems with locking up the hoist, bypassing the "restricted mode" limitation in Technical Specification 3.10(F)1, making it outside the licensing basis.

The licensee was having so much trouble with the digital load limit setpoint disabling the crane, that it proceduralized it in procedure DFP 0800-20; how to jumper out the load cell signal in order to use the crane. It should be noted that the procedure even specified to use the "Control of Temporary System Alteration Procedure," DAP 07-04. DAP07-04 even cautions the user to not use a temporary alteration in lieu of a work order. It further requires a specific time frame that the temporary alteration may remain installed. The load cell was out for an undocumented number of years. The only document the licensee has uncovered to date that indicated operability of the load cell was an operability determination from December 13, 1991, which indicated that it has been out for many years.

72.150 requires in part, that activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures.

Contrary to the above, until December 13, 1991, the load cell was jumpered out-of-service without any documented evidence.

The plant lifted at least 68 fuel casks between 1976 and 1984 without the overload protection in place. As stated previously, there are no plant records to indicate the status of the load cell during that time frame.

It should be noted that the load cell was out-of-service in 1981 when the reactor head strongback hit the crane bridge girders rendering them around 22 percent overstressed at rated load. The actual impact load could not be determined because the load cell was out. This lack of indication and associated overload protection probably allowed overloading of the crane through the years during outages when heavy lift weights have apparently been calculated instead of measured.

72.168(b), requires in part, that measures to identify the operating status of systems shall be established to prevent inadvertent operation.

Contrary to the above, the license failed to control the crane in "restricted mode" while the load cell was jumpered out-of-service for at least 68 cask lifts.

During 1996, while re-base lining, the UFSAR to the current design requirements for the reactor building crane, the operability determination from 12/13/91 was uncovered that identified that the digital weight indicator has been out for many years and remained inoperable. In the operability determination, in the justification of its operability determination section, justification of its operability was based upon the crane's having another load limiting device through use of the crane's "over torque limit." The licensee is still attempting to ascertain what part of the crane this is. To meet the original safety intent of the digital weight indicator, engineering recommended that the indicator be restored to operating in 1991. As a temporary alternative, engineering believed that revising station procedures to add the requirement of additional supervising personnel during crane operation to ensure load hang ups do not occur would be sufficient. Site procedure DMP5800-18, Revision 07, has this statement incorporated into it.

A 10 CFR 50.59 Safety Evaluation was performed in 1996 to change USFAR Section 9.1.4.2.2 to include a statement that, "As an alternative to the digital load limiter, station procedures require supervising personnel to ensure load hang ups do not occur during reactor building crane operation." The 1991 operability was used as justification for the 50.59; the non-existent over-torque limit justification.

10 CFR 50.59 (d)(1), requires, in part, that the evaluation shall provide the basis for the determination that the change does not require a license amendment.

Contrary to the above, the stated basis, the "over torque limit," does not exist on the crane. Thus, the 50.59 basis was inadequate.

When the crane upgrade was installed in early 2001, a new digital overload protection system was installed and tested improperly to only a test load of 8000 pounds. The upper limit load trip was set according to drawing 12E-6510 to 110 percent of the 125 ton rated load until the inspector pointed out that it was set wrong. It should be set to the rated load.

During various dry runs the inspection noted that the digital indication of the load cell was reading a negative load with a weight on the main hook. The inspector was told that the load cell was accurate to within 50 pounds; and on another occasion, to within 1 percent of full scale or the actual reading - they weren't sure. The inspector was told that all this confusion on the load cell was because the plant relied on skill-of-the-craft to operate the crane and as a result, they weren't zeroing out the crane's load cell correctly. The training did not follow site training processes, nor was the operation of the load cell proceduralized.

72.144(d), requires in part, that indoctrination and training of personnel performing activities affecting quality, to ensure that suitable proficiency is achieved and maintained, shall be provided.

Contrary to the above, the licensee failed to provide the crane operators the appropriate training and written procedures for the operating load cell.

This training on zeroing out the load cell was supposed to fix the weight indication problem. Subsequently, the inspector noticed that the indicator was reading 5200 pounds when only the hi-trac lifting yoke was on the hook. It should be noted that the yoke has a calibrated weight of 3400 pounds. The total lifted weight of a full cask had been calculated as 199,394 pounds, using the lower (3400 lb.) yoke weight. If the yoke weight is actually 5200 lbs., the total lifted weight of a full cask will exceen 100 tons. The licensee was in the process of figuring out how to calibrate the new load cell system and indicated that they would not lift any cask until it was resolved. After the licensee thought it was resolved, the actual loaded cask (with fuel) indicated 215,000 pounds on the digital readout. This is slightly over the estimated calculated weight of 192,000 pounds. The licensee told the inspector that it's reading conservative; so it is alright to use the crane. Subsequently, on June 20, 2001, tests were conducted with a calibrated dynamometer that indicated the as-found load cell readings were up to 40 percent high as found. New calibrations were done on the load cell which set it slightly below (not conservatively) the actual test weight of 76,000 pounds.

From discussions with a cognizant load cell calibration company, it was determined that because of hysteresis losses, creep, and non-linearity of load cells, the recommended calibrations should be done at 0, 50, 100, 50, and 0 percent of full scale output. National Institute of Standards and Technology (NIST) Handbook 44, requires a minimum test weight of 25 percent of rated load, and to capacity during initial verification.

72.162, requires in part, that the licensee establish a test program to ensure that all testing, required to demonstrate that components will perform satisfactorily in service, is performed.

Contrary to the above, the licensee still hasn't satisfactorily calibrated the new electronics for the load cell at the high end of its upper limit of 125 tons.

- Restricted Load Handling

The previous inspection identified a question about use of a slow-speed or "inching" motor when operating the crane in "restricted mode." The UFSAR, Section 9.1.4.2.2 described fuel cask handling above the 545-foot level of the reactor building as being a restricted load requiring handling in slow-speed with the fast-speed circuitry disabled. The slow-speed motor had malfunctioned in 1976, and it was never used even though the crane was operated in "restricted mode" on many occasions up to the present day. The cover letter to the Amendments, dated June 3, 1976, described the non-reliance on the slow-speed motor as an item for which the licensee had requested only a temporary waiver, until the end of August, 1976.

Further inspection showed this statement was in error. Prior to the date of the SER, the licensee requested an alternate approach, relying on a speed control circuit which could limit hoisting speed to five feet per minute, consistent with BTP 9-1. NRC approved the alternate approach within the SER itself. In the NRC's safety evaluation for the 1976 crane modification, the NRC noted that because CECo experienced problems with the slow speed motor installation, it could not complete the installation prior to planned fuel shipments in June 1976. Because it was consistent with BTP 9-1, the NRC did not make this part of the limiting conditions for operation in the technical specifications as the other two temporary component waivers were until August 30, 1076. Thus, the "inching" motor was never part of the single-failure-proof licensing basis for the crane.

As of June 22, 2001, USAR section 9.1.4.2.2 still refers to the crane having a slow speed (inching) motor in eight locations. The licensee never corrected the USAR from 1976 to delete the non-existent slow speed motor descriptions. The licensee also needs to revise the USAR to reflect the new digital electronic motor controllers.

Modifications to the Crane

The previous inspection identified a question regarding whether the crane had an "extensive repair" or a "major alteration" after it was damaged in 1981 and a load test was required. No load test had been performed. Subsequent observations by the inspector revealed that there are numerous smaller dings on the lower flanges of the girders, indicating that the girders have been hit numerous times because of various limits on the crane being temporarily jumpered out.

In 1981, the crane bridge box girders were damaged by impact and compression during an over-hoisting event involving the strongback for the reactor vessel head. The crane girders are built-up box beams approximately 8 ½ feet deep, 2 feet wide, and 113 feet long. The damaged surfaces were confined to the lower portion of the inside webs (buckling) and the inside portion of the bottom flanges (bending) on both the girders approximately 35 feet from one end. The Office of Nuclear Reactor Regulation (NRR) evaluated the licensee's actions to determine whether the crane had been extensively repaired.

The licensee's required repairs (to get back to the design basis stress allowables) consisted of a splice plate welded over the cutout portion of the damaged girder webs plate and welding cover plates to each girder's bottom flange. These repairs were made by the licensee and Nutech was contracted to perform the repair evaluation. On page 3.1 of their repair report, Nutech concluded that the west crane bridge girder damage was so severe that it would be 22 percent overstressed at rated capacity (125 tons) with the damaged area not fixed. Based on 22 percent reduction in load carrying capacity with the damaged section unaccounted for, the NRR staff concluded that the crane would be capable of lifting 100 tons without overstress.

The repairs made to the crane were determined to restore the capacity of the crane girders to the original design value (Page 3.2 of the Nutech report), thus the crane would be able to support a lift of about 125 tons without appreciable overstress. The staff notes that this is a conservative assessment since the damaged areas of the girder's bottom flanges were not removed, but instead covered. The crane girders and other critical structural components were designed to carry 125 tons with minimum safety factors.

Additionally, based on their analysis, Nutech concurred on Page 3.3 of their report that the repair work was not in the extensive repair or major alteration category as defined by the 1976 ANSI B30.2 code. If the repairs were found to be extensive, then ANSI B30.2 would require that the licensee conduct a load test on the crane.

Part of the licensing basis, Report No. 41, stated that the crane will be compatible with the requirements of ANSI B30.2. Technical Specification 4.10(F) stated that the crane will be adequately inspected in accordance with the accepted ANSI Standard B.30.2. B30.2, 1976, requires that prior to initial use, all extensively repaired, and altered cranes should be tested confirming the load rating of the crane.

Contrary to the above, the crane was not load testing following the repair which was needed to restore the crane to its design basis stresses.

Reactor Building Super-Structure

The previous inspection discussed a series of calculations from the 1960's forward which disclosed examples of various building structural members (roof trusses or columns) in overstress, either with or without design load assumed to be on the crane, and including or not including design-basis earthquake conditions.

During this inspection, substantial additional reviews of the licensing basis for the building, and of the possible performance of the building super-structure in seismic conditions, were performed. The Reactor Building is classified as Seismic Category I in the USAR Section 3.2.1, and is to be designed to accommodate the load conditions and stress criteria in section 3.8.4. Section 3.8.4.1.3 states the following load combinations will be used for Class I structures.

- D (dead load + live load) + E (OBE load)

- D + E' (SSE load)

Further, Sargent & Lundy (plant's engineers) Structural Standard SDS.E5.2, requires that the crane and trolley rated lifted live load, including impact load, longitudinal load and lateral load must be used in combination with the earthquake loads for seismic structures. Contrary to this, the licensee chose to define the live load to be without the crane rated load included in the above equations. This interpretation was assigned despite the fact that the analyzed loading combinations were unconventional, even for the 1970's. That is, the building was analyzed for "normal" loads (wind, snow, etc.) with full lifted live load on the crane, but the analyses for design basis earthquake conditions (Operating Basis Earthquake - OBE; and Safe Shutdown Earthquake - SSE) assumed no lifted load on the crane even though the UFSAR equations indicate so.

10 CFR 50, Appendix B, Criterion III, requires in part, measures shall be established to assure applicable regularity requirements and design basis, as specified in the license, are correctly translated into specifications. These measures shall include provisions to assure that deviations from such standards are controlled.

Contrary to the above, the rated live load of the crane (125 tons) was not incorporated into the original reactor building "D" load combination calculations. Therefore, the building never met Category I requirements. Even so, the licensee calculations without the lifted load included in the 1960s showed that the stresses in the support girders, support columns and several members of the roof truss were above the yield stress for the SSE loading. In some of the roof truss connections, the loading exceeded the ultimate capacity of the connections. Nothing was documented outside of the calculations to include any resolution of these issues.

Although there were no analyses of record documenting the performance of the crane, the building, and their interface, under design basis earthquake conditions and with a 100-ton load on the crane, some analyses of this type had been performed which were classified as "beyond the licensing basis." These also indicated the stresses would exceed yield for some structural members.

In 1973, new calculations were performed for the effects of the new heavier single-failure-proof trolley without lifted load. These calculations showed that the columns were overstressed by up to 30 percent for the SSE loading using 0.9 Fy as the allowable stress. The roof trusses and vertical bracing were not evaluated. Nothing was documented outside the calculations again to resolve this issue.

In 1975, new calculations were performed for the columns and the vertical bracing for the effects of the new trolley without lifted load. Modifications were designed to bring all elements within code allowable stresses. The modifications were subsequently not implemented. The Dresden calculation book carries the notation, "Project Canceled, calculation not approved," without explanation.

In 1998 in calculation DRE98-0013 the reactor building superstructure framing was examined again for the dry cask project. The crane support girders, interior building column members, as well as roof truss members again, had interaction coefficients above 1.0. This time they justified it in the calculations based on probabilistic considerations that the earthquake, most likely, would not occur when the crane is in use, and the crane is used only for a small fraction of the time. This probablistic rationale was not addressed in the USFAR.

In 1998, in calculation DRE98-0020, calculations again indicated members were overstressed. This time it was justified because of the small magnitude of overstress, 5 percent, which the calculations say are generally acceptable. They are not generally acceptable.

USAR, Table 3.8.11, states that stresses are to be held below 0.6 Fy for OBE and normal loads, and for SSE loads may exceed Fy in some elements only if the energy absorption capacity doesn't exceed the energy input. No energy balance was performed.

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to the appropriate levels of management.

Contrary to the above, from the 1960's until June 22, 2001, the licensee failed to promptly identify and correct known deficiencies in the Units 2 and 3 reactor building structural steel supporting the crane. Certain beams and columns exceeded the allowable stresses for Class I building type structures specified in Dresden Updated Final Safety Analysis report (USAR) Table 3.8-11, a significant condition adverse to quality. Ultimately, it was not possible to answer the question whether the building, the crane, and/or the load (a spent fuel cask containing 68 spent fuel assemblies) would fall.

In an eqrthquake, particularly one causing building deformation, the load would also be subjected to lateral forces. The crane yoke for holding the spent fuel cask load is not equipped with positive latching (see paragraph 2.2.b) capable of retaining the cask against significant lateral force. This suggests that a cask could slide off the yoke and fall, even if the crane and building do not fail catastrophically. This potential scenario also has not been analyzed.

Crane Inspections

The previous inspection identified that the crane manufacturer (Whiting Crane) had been performing annual inspections of the Unit 2/3 reactor building crane, and had been identifying discrepancies. These discrepancies were not being addressed by licensee corrective actions. The process to capture and act on vendor recommendations appeared ineffective.

Due to years of neglect of crane maintenance, numerous failures occurred to the crane during Unit 3's last outage. The failures included the main hoist brake, the trolley brake, the equalizing bar circuitry, and inverter on the trolley, a motor exciter, and trolley conductor shoe. To make the crane operational during the remainder of the outage, crane parts from the Unit 3 Turbine Building Crane were cannibalized.

Prior to the outage, an annual inspection was performed. In this inspection, five items were identified as needing correction and a work action request, AR990093876, was generated; but only one item was addressed, without addressing the other four. However, the action request was closed. Numerous needed repairs through the years have been documented on action requests, but canceled, with the justification, that the repairs were not required. These are examples of an inadequate process to capture and address crane issues. Correction of these items most likely would have prevented some of the crane failures during the outage. The licensee is addressing this issue, but not the fact that the action request was closed.

72.172, requires in part, that measures be established to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected.

Contrary to the above, the licensee addressed the fact that the corrective action process for the yearly crane inspection identified repairs was inadequate. However, the apparent cause evaluation (ACE) written to address this issue failed to address the fact that, AR990093876 for the five identified deficiencies during the last yearly inspection was closed as complete, without any work being performed on four of the five items, with no justification.

During this inspection, repairs were performed for each of the four vendor-identified discrepancies from the latest inspection, or the affected equipment was replaced as part of the digital control system modification. No previously-identified material discrepancies remained when the crane was made available to the fuel storage project for heavy load handling.

c. Conclusions

2.1.2 New Identified Items

a. Inspection Scope

The inspection included a review of additional Unit 2/3 reactor building crane issues.

b. Observations and Findings

Seismic Crane Qualification

Dresden Special Report No. 41, dated November 8, 1974, stated that the entire crane trolley and existing bridge girders will be reviewed for the revised new trolley weights in conjuction with the lifted load requirements to establish compliance with Crane Manufacturers Association of America (CMAA) permissible stress ranges. Design values for the Operational Basis Earthquake (OBE) will be based on AISC Code requirements of 0.60 times the minimum yield strength of the material (Fy) and 0.90 Fy for the Design Basis Earthquake (DBE), the Safe Shutdown Earthquake (SSE). In the interim, the NRC issued Branch Technical Position APCSB 9-1 in April 1975, which required that the crane be classified as Seismic Category I and should be capable of retaining the maximum design load during an SSE. The design rated load plus operational and seismically-induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for the design of the bridge.

Dresden Special Report No. 41, Supplement A, dated June 3, 1975, stated that the crane was identified as Safety Class II in the plant operating license and it is not practical to consider reclassifying the crane as Seismic Class I. This is because it would require a new bridge and extensive modifications to the bridge trackway (the Rx Building Superstructure). It further stated that the bridge and trolley will be analyzed with only static lifted loads considered. These are all not consistent with the Branch Technical Position.

Furthermore, during the evaluation and repairs from the 1981 event, NUTECH stated in their report that the original governing design codes state that the most restrictive code allowables shall be used. This crane was built to the CMAA (1975) stress ranges which only allow for 17,600 pounds per square inch tension and compressive stresses. A note in the 1981 calculations indicates that at CECO's direction, to use AISC allowables of 21,600 pounds per square inch stresses. This had a major impact on the acceptability (extensive versus minor) of the existing crane and damage repairs evaluation results. at the CMAA stress limits, the crane would have been approximately 50 percent overstressed instead of 22 percent.

As of June 22, 2001, the licensee has not been able to locate any evaluations for the seismic stresses on the trolley as committed to in Supplement A.

The crane bridge girders were only checked for static lifted loads in 1974 without any pendulum or swinging loads. Calculations indicate the girders were acceptable for the non-seismic loading conditions for the new trolley additional weight of 25,000 pounds. The calculations indicate that the girders were acceptable for SSE static loading. However, the calculations indicate that the girders were 2 percent overstressed for OBE static loading and 6 percent with the added pendulum loads.

72.170 requires in part, that measures shall be established to control materials, parts, or components that do not conform to their requirements in order to prevent their inadvertent use. These measures must include procedures for identification and documentation.

Contrary to above, the bridge girders were found to be overstressed but weren't documented as non-conforming and have continued to be used.

NRC Bulletin 96-02

When the Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," was developed, the NRC policy in the area of handling heavy loads at the time was set forth in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and the Standard Review Plan (SRP). BTP 9-1 was not part of the policy at the time, having been replaced previously in 1979 and 1980 by the two NUREGs. The staff has determined that the provisions of the BTP were subsumed into the NUREGs.

In the May 13, 1996 licensee's response to Bulletin 96-02 the licensee stated that they currently have no plans for any movement of dry storage casks over safety-related equipment while the reactor is at power. However, if such movements were planned in the future, they would demonstrate the capability of safe shut-down in the presence of the radiological source term that may result from a breach of the dry storage cask, damage to the fuel, and damage to safety-related equipment as a result of a load drop inside the facility.

This condition was incorporated into site procedure DMP 5800-18, Revision 7. After discovering this, the inspector requested the evaluation that the licensee could safely shutdown the plant. After being told numerous times that the evaluation is forthcoming, the inspector was given a Commitment Change Evaluation form, 2001-002, dated after the inspector's request, that deleted the commitment. The rationale used was that the reactor building crane is single-failure-proof and dropping the cask is a non-credible event so they do not have to have an evaluation.

c. Conclusions

2.2 Cask Transfer Facility (CTF)

a. Inspection Scope

Because removing fuel from the Unit 2/3 reactor building required the use of another heavy load device, the Cask-Transfer Facility (CTF), the inspection evaluated various attributes including whether the design of the CTF satisfied all requirements in accordance with the Certificate of Compliance (CofC).

b. Observations and Findings

The inspector reviewed various aspects of the CTF including single-failure-proof-design, fabrication, and ANSI N14.6 requirements.

Single-Failure-Proof Design

The concept of a CTF initially appeared in the Hi-Storm TSAR. As the NRC considered the concept, additional detail regarding the design of a CTF was needed if it was to be included in the CofC for the Hi-Storm system. In response, Holtec expanded Section 2.3.3.1 of the Hi-Storm TSAR to include sketchy design features that any CTF would have to meet. However, the CofC does not address the CTF.

TSAR, Section 2.3.3.1 addresses single-failure-proof design in several places: Section A.iii defines the CTF Structure as the "stationary, anchored" portion of the CTF; Section A.iii also defines Single-Failure-Proof as a device wherein all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria in NUREG-0612. This definition was refined by Section 203.12 of the licensee's CTF purchase specification, which required the CTF to be able to withstand a failure of any one part or component without resulting in an uncontrolled lowering of the load.

The TSAR (Section 2.3.3.1.C.ii) states that 3 main portions of the CTF shall comply with NUREG-0612 guidance: the connector bracket, Hi-Trac lifter, and MPC lifter. These three portions are defined rather broadly but are clear enough to permit distinction between these portions and the stationary CTF structure itself. The licensee in their purchase specification (Section 203.6) stated that the CTF structure includes the fixed-position vertical structural members that support the cask.

NUREG-0612, Section 5.1.6 invokes NUREG-0554 for the design of cranes. It defines single-failure-proof as a system that is designed so that a single failure will not result in the loss of the system to safely retain the load. By issuing the CofC, the NRC approved TSAR Section 2.3.3.1; however, Section 3.5 of Appendix B specified the Hi-Trac lifter and the MPC lifting device must be designed, fabricated, operated, tested, inspected, and maintained in accordance with NUREG-0612. The NUREG-0612 requirements include the requirements of NUREG-0554, which it invokes. These appear to apply to the Hi-Trac lift system, including all load bearing components and parts, which includes the lift platform itself, and not just the cask lifting yoke pieces, and the screw jacks, (the Hi-Trac lifter).

72.146 requires, in part that measures shall be established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions.

Contrary to the above, the safety factor for the lift platform beam does not meet single-failure-proof criteria of NUREG-0554.

Cask Transfer Facility Fabrication

NUPIC performed a joint audit of Holtec's implementation of its quality assurance program at its subcontractor Omni Fabricators on May 22 to May 26, 2000. The audit found the following:

- Omni was not on the approved supplier list.

- Omni did not have a quality assurance program that conformed to the requirements of Parts 71 or 72, nor to the requirements of 10 CFR 50 Appendix B.

- Fabrication process controls were less than effectively implemented. Weaknesses were identified associated with special processes (i.e., welding), work process control documents, and the lack of documented instructions and procedures for controlling activities that have an impact on quality.
- Holtec, consistent with its quality program, had extended its quality program to Omni, but Holtec's oversight and implementation of its quality program at Omni's facility were considered to be ineffective at the time of the audit.

- Weld procedures still needed to be qualified, as did welders.

Based on the audit findings, a follow up audit was conducted August 29-31, 2000. The "limited-scope" audit assessed the adequacy of corrective action implementation in response to just the five documented audit findings. The corrective actions included: revising the incorrect procedures and records identified in the audit; developing procedures that the audit team found missing; and providing training to personnel on the new procedures. Corrective actions were narrowly scoped, only looked at past work, and did not address programatic issues. The finding that Holtec had not been effectively implementing its quality assurance program at Omni was not re-examined. The audit did not look at any work activity at Omni. The audit report stated the number of qualified welders was very limited; weld procedures still needed to be qualified, as did welders and employee experience at Omni with demands of a quality program mandated by the NRC was essentially non-existent. Even so, the audit findings did not lead to a work stoppage.

Subsequently, condition reports were being written at the Dresden site for items received which were manufactured at Omni, which included the transfer cask (Hi-Trac) with its various ancillary equipment, and the CTF. Condition reports identified: incomplete and/or inaccurate CofC information, missing documentation, fit up problems with various pieces of equipment, and corrective actions that clearly demonstrated lack of compliance to the quality program. However, the licensee apparently did not examine the actual fabrication records for compliance.

The inspector requested various welding and inspection records for specific welds on the CTF. The records received consisted of weld data and inspection data transferred cumulatively by the Omni QA Manager, in weld groups, according to the size of welds. The date recorded for the transferred data was the latest date of performance for a particular group of welds. This resulted in all the welds for the whole CTF being signed off by the QA Manager on the same day. Holtec's procedure for control of shop travelers, HSF-316, does not address consolidation of data by the QA Manager. The reason given for the data transfer was the poor condition of the shop traveler paperwork and drawings after being exposed to the shop environment (dirt, grease, etc.). When the original data was requested by the inspector, it could not be provided because it had been discarded.

Without individual weld data, if there is a condition adverse to quality involving a specific welder, all that welder's work is suspect. Nonconformance Report 46, dated September 12, 2000, identified Welder K making a 5/16 inch stitch welds incorrectly. The faulty weld was repaired; however, other similar welds may have been made inside of eight closed boxes making up the vertical tower structure of the CTF. These welds are now inaccessible. Welder identity and the fabrication sequence are non-existent; thus, the other seven welds are questionable, because documentary evidence that the other welds are acceptable is also non-existent.

Licensee and vendor oversight of fabrication at Omni involved multiple, full-time QC inspectors being assigned to examine every aspect of the job. Weekly reports by the Holtec Users Group inspector constitute a generalized record of activities inspected, and indicate all the work was being performed by qualified and certified welders and NDE technicians, using qualified procedures, with no specific details given. This canned statement appears on every report with no other details. As a result of this approach, the structural adequacy of the CTF cannot be determined based on specific records of quality verification.

10 CFR Part 72.154, "Control of Purchased Material, Equipment and Services", requires that measures shall be established to ensure that purchased material conform to the procurement documents. They must include objective evidence of quality furnished by the contractor. Documentary evidence that the material and equipment conform to the procurement specifications shall be available for life of the ISFSI. The evidence shall be sufficient to identify the specific requirements met by the equipment.

Contrary to the above, in the case of the CTF, the 5/16 inch stitch welds could not be individually verified to be conforming to the design drawings.

ANSI N14.6

The CofC requires that the CTF be designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." NUREG-0612 requires that the special lifting devices should satisfy the guidelines of ANSI N14.6. N14.6 requires that (1) special lifting devices that require remote engagement with the shipping container shall be provided with lead-in guides and sufficient clearance between the container attachment points and the lifting hook to allow simple motion engagement; (2) that the means of attaching the special lifting device to the shopping container shall be addressed during the design to ensure the security of the attachment method under load; (3) the actuating mechanism used shall securely engage or disengage; (4) load-carrying components that may become inadvertently disengaged shall be fitted with a retaining latch; and (5) engagement indication be provided whenever it is difficult to observe the attachment points between the special lifting device and the shipping container.

The licensee's purchase specification requires that the lifting yokes (on the CTF and the Unit 2/3 crane) be designed in accordance with ANSI N14.6. Holtec's lift yoke design criteria also specify design in accordance with ANSI N14.6. However, during a review of the criteria, the inspector could not determine how the lifting yokes meet the intent of the ANSI N14.6 requirements stated above. The lifting yoke employs two air-operated swing arms with circular cut-outs which slip over the cask lift trunnions. No physical locks or latches are provided, nor is any type of flag provided as an indication of engagement.

72.146 requires in part, that measures shall be established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, the Unit 2/3 crane yoke and the CTF Hi-Trac lifting yoke, do not meet ANSI N14.6 requirements.

Subsequent to the inspector questions, Holtec revised the design criteria to include statements that "each cask lifting trunnion is equipped with an end cap, which allows for visual verification that the lift yoke arms are properly engaged," and "because the design of the lift yoke shall ensure that the lift yoke arms hang plumb to engage the lifting trunnions, there is no need to provide an additional security device to maintain attachment. There are no credible loads that would apply a side load between the cask lifting trunnion and lift yoke arm. Therefore, the security of the attachment method under load in all handling positions is assured."

The inspector determined that the design does not ensure that the lift yoke arms hang plumb. The arms are adjustable and could engage the trunnions at an angle, thereby increasing the stress concentrations on the trunnions. During "dry run" testing, the arms were made plumb, not by design or site procedures, but by engineering inspection and adjustment. The licensee is considering a procedure step requiring verification of the yoke arms being plumb.

c. Conclusions

3.0 Management Meetings

The inspector presented the inspection results to licensee management daily during the ongoing inspection and at a special exit meeting on June 7, 2001, and during a subsequent phone call on June 22. The licensee acknowledged the findings presented. The licensee did not identify any of the documents or processes reviewed as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Dale Ambler, Regulatory Assurance Manager
Robert Fisher, Unit 2-3 Station Manager
Paul Planing, Unit 1 Manager
Nate Leech, Dry Cask Storage Project Manager
Bob Rybak, Regulatory Assurance
Ken Ainger, Regulatory Services
Joe Sipek, Nuclear Oversight Manager
Dave Schupp, Operations
Preston Swafford, Site Vice President
Tom Luke, Site Engineering Director
Chip Cerovac, Training
Dave Williams, Electrical Maintenance Superintendent
Ken Bowman, Operations Manager,\
Joe Kotowski, Operations Supervisor
Pete Scardigno, Site Project Manager

Illinois Department of Nuclear Safety

Rick Zuffa, Dresden Resident Inspector

NRC - Region III

Marc L. Dapas, Deputy Director, DNMS
Bruce L. Jorgensen, Chief, Decommissioning Branch, DNMS

The inspector also interviewed other licensee personnel in the course of the inspection.

From: Jim Dyer
To: Ross Landsman
Date: Wed, May 23, 2001 3:52 PM
Subject: DPV Memo Dated 5/23/01

Ross,

I received your memo to me identifying your disagreement with the NRC decisions concerning the startup of dry cask storage loading activities at the Dresden Station. This memo identified several issues concerning the historical use of the crane as well as its current configuration and readiness to safely conduct cask loading activities.

As we discussed this morning, it is premature to start the DPV process on this issue before the NRC has come to a decision on these issues. We are focusing our current efforts on identifying the activities necessary for cask loading and will disposition the historical issues via the enforcement process. A Region III public meeting is being held on the subject today with the licensee and further inspection activities are planned in the near future. Therefore, as we agreed to earlier, I will hold your DPV until a NRC decision is made and then review with you the alternatives and whether you wish to proceed with establishing a DPV panel in accordance with MD 10.159. I discussed this approach with Jim McDermott, OHR, and he agreed with this deviation from the MD 10.159 timeline for establishing the DPV panel.

Additionally, I want to thank you for allowing me to provide copies of your memo to the NRC staff to better prepare for the public meeting today and facilitate further deliberations on the subject. I hope that you fully participated in today's public meeting and raised your concerns to the licensee and attending NRC staff.

Jim.

CC: Bruce Berson; James Caldwell; James McDermott; ...

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

July 13, 2001

MEMORANDUM TO: Ross B. Landsman, Project Engineer, DNMS

FROM: J. E. Dyer, Regional Administrator JEDyer

SUBJECT: DIFFERING PROFESSIONAL VIEW CONCERNING THE STARTUP OF THE CASK STORAGE LOADING CAMPAIGN AT DRESDEN UNITS 2 & 3

This memorandum acknowledges receipt of your July 11, 2001, memorandum supplementing your Differing Professional View (DPV) dated May 23, 2001, and requesting initiation of the DPV process concerning the startup of the dry cask storage loading campaign at Dresden Units 2 & 3. You had previously agreed that the filing of the DPV on May 23 was premature and could be held in abeyance.

In accordance with Management Directive 10.159, Differing Professional Views or Opinions, I will be appointing the DPV Ad Hoc Review Panel chairperson and one technically qualified member in the next few days. Please provide a list of individuals from which the panel chairperson can select the third panel member to Bruce Berson as soon as possible.

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From: Ross Landsman / R3
To: Bruce Berson / R3
Date: 7/16/01 2:22PM
Subject: DPV

As we discussed, try Pat Hiland.

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

July 20, 2001

MEMORANDUM TO: John A. Grobe, Director
Division of Reactor Safety

FROM: J. E. Dyer *JEDyer*
Regional Administrator

SUBJECT: AD HOC REVIEW PANEL FOR DIFFERING PROFESSIONAL VIEW CONCERNING THE STARTUP OF THE CASK STORAGE LOADING CAMPAIGN AT DRESDEN UNITS 2 AND 3

This memorandum is to confirm our conversation regarding the Differing Professional View (DPV) concerning the startup of the cask storage loading campaign at Dresden Units 2 and 3 (copy attached). In accordance with Management Directive 10.159, Differing Professional Views or Opinions, you have been appointed as the chairperson for the ad hoc review panel. Additionally, John Jacobson, DRS, Region III, has been appointed as a technically qualified member of the panel.

This memorandum also confirms that for your other panel member, you have selected Pat Hiland from the Division of Nuclear Materials Safety, Region III, as requested by the employee submitting the DPV.

You are to conduct the review of this DPV in accordance with Management Director 10.159. You should complete your review and forward your recommendation to me by August 31, 2001. I understand that this schedule has been discussed with the employee submitting the DPV and this date is acceptable.

Attachment: As stated

cc w/attach: J. Jacobson, RIII
P. Hiland, RIII

cc w/o attach: J. McDermott, OD/HR
C. Pederson, RIII

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

July 11, 2001

MEMORANDUM TO: J. E. Dyer, Regional Administrator

FROM: Ross Landsman, Project Engineer, DNMS RBL

SUBJECT: SUPPLEMENT TO DPV DATED MAY 23, 2001

I agreed with you that the DPV was premature because RIII management had not made a decision on my issues. However, I wanted to issue it anyway so you could better understand the issues. Since that time, additional issues were identified during my inspection. These reinforced my reasons why the licensee should not use either the Unit 2/3 reactor building crane or the cask transfer facility (CTF) which I didn't have time to inspect before the original DPV.

These issues include:

- The reactor building is not designed for the 125 ton crane load, making the building unsafe to use.
- The cask lifting yokes do not meet our ANSI N14.6 requirements.
- The adequacy of the structural capability of the CTF cannot be determined based upon existing records.

These and other issues are included in my draft report which is attached.

Subsequently, RIII management allowed the licensee to load the first cask even though there were at least 15 violations of NRC requirements, some significant conditions adverse to quality. For example, the reactor building structural steel issue is identical to the one we issued in 1996, resulting in a $100,000 fine to Dresden and not allowing them to start up the units until the beams were brought back to within design allowables.

Please start the DPV process.

Report Details[1]

1.0 General

This special inspection examined design, fabrication and testing of equipment for use in removing spent fuel from the Unit 2 fuel pool into components of the Holtec dry fuel cask system. The Unit 2/3 reactor building crane and the Cask Transfer Facility were examined in detail. Previously identified unresolved items were examined further and were determined to be acceptable items.

2.0 Handling of Heavy Loads

2.1 Unit 2/3 Reactor Building Crane

2.1.1 Review of Previously Identified Items

a. Inspection Scope

Licensee actions, responses or clarifications regarding an Unresloved Item (URI 07200037/2001-001(DNMS)) were examined. The URI consisted of a number of individual elements. These elements are addressed below in the same sequence as originally documented.

b. Observations and Findings

USFAR Commitments

- Safety Lugs

The previous inspection raised a question regarding whether safety lugs were installed on the Unit 2/3 crane trolley and bridge rails. The Updated Final Safety Analysis Report (UFSAR) in Sections 6.2.3.2.1 and 9.1.4.2.2 describes provisions made (safety lugs) to ensure the Unit 2/3 reactor building crane trolley and bridge do not become dislodged during an earthquake. During this inspection, the inspector was able to verify that safety lugs were in place on the trolley. The inspector could not determine whether safety lugs were in place on the bridge rails. The licensee subsequently determined that the bridge did not have the specified safety lugs. They had apparently never been installed.

The lack of specified safety lugs on the crane bridge rails, contrary to the description of the UFSAR, is considered a *de facto* design change. No safety evaluation was performed under 10CFR50.59 to establish that this change did not constitute an unreviewed safety question. 72.212(b)(4), requires in part, that prior to use of the general license, activities related to storage of spent fuel shall be evaluated for any unreviewed facility safety question, as provided under 50.59. Results of this determination must be documented.

[1]NOTE: A list of acronyms used in this report is included at the end of these Report Details.

Contrary to the above, as of June 22, 2001, the licensee failed to document and evaluate that the Unit 2/3 reactor building crane bridge trolleys, as described in the UFSAR, do not have safety lugs installed.

- Wire-Rope Safety Factor

The previous inspection raised a question regarding the safety factor for the crane wire-rope. NRC's current guidance for crane cables is contained in NUREG-0554 and NUREG-0612, which were issued after Dresden Amendment No. 22 for Unit 2 and Amendment No. 19 for Unit 3, and recommend a safety factor of 10 to 1.0. A safety factor of 10 to 1.0 is not a requirement for Dresden Units 2 and 3.

It appeared the wire-rope on the Dresden Unit 2/3 crane had a safety factor of 8 to 1.0, per the UFSAR, Table 9.1-3, but the inspector found that it actually had a factor of 7.798 to 1.0 in the licensee's submittal.

10 CFR 50, Appendix B, Criterion III, requires in part, that measures shall be established for the identification and control of design interfaces. The design control measures must provide for verifying or checking the adequacy of design.

Contrary to the above, as of June 22, 2001, the licensee failed to state the correct wire rope safety factor in the UFSAR.

On June 3, 1976, in Amendment Number 22 for Unit 2 and Amendment Number 19 for Unit 3 the staff accepted the wire-rope static safety factor of 7.798 to 1.0 and the lead line safety factor of 6.564 to 1.0 even though it didn't meet the Branch Technical Position (BTP) 9-1 requirements. To compensate for this, the staff incorporated LCO and surveillance requirements in the Technical Specifications. Specifically, inspection requirements in accordance with ANSI B30.2. It also limited the fuel casks weight to 100 tons.

The NRC wrote to the licensee on January 30, 1976, that since the wire rope safety factors were not acceptable, provide a proposed inspection/replacement program for the wire rope. The licensee responded on March 2, 1976, that the ropes would be inspected and if required, replaced to assure compliance. Through the years, the licensee has been replacing the rope with like-for-like without ever considering replacing it with a rope that met 1976 standards or meets today's criteria of a 10.0 to 1.0 safety factor. The licensee is considering providing an additional safety enhancement by replacing the rope the next time with a 10.0 to 1.0 margin rope.

- Overload Protection

The previous inspection identified an issue relating to the apparent lack of overload protection on the Unit 2/3 crane hoist.

The initial licensee submittal in support of Amendment No. 22 for Unit 2 and Amendment No. 19 for Unit 3, Dresden Special Report No. 41, stated that a load sensing readout with high and low limit cut-offs will be provided as an overload protection feature. UFSAR section 9.1.4.2.2 states a digital-type weight indicator for the main hoist is provided. When the weight to be lifted is above the setpoint on the weight indicator, the control circuit for the slow speed motor will prevent its operation and the main hoist brakes will set.

Since initial installation of the load cell in 1976, a review of the history of the system showed it has been out-of-service because the license has been jumpering it out because of repeated problems with locking up the hoist, bypassing the "restricted mode" limitation in Technical Specification 3.10(F)1, making it outside the licensing basis.

The licensee was having so much trouble with the digital load limit setpoint disabling the crane, that it proceduralized it in procedure DFP 0800-20; how to jumper out the load cell signal in order to use the crane. It should be noted that the procedure even specified to use the "Control of Temporary System Alteration Procedure," DAP 07-04. DAP07-04 even cautions the user to not use a temporary alteration in lieu of a work order. It further requires a specific time frame that the temporary alteration may remain installed. The load cell was out for an undocumented number of years. The only document the licensee has uncovered to date that indicated operability of the load cell was an operability determination from December 13, 1991, which indicated that it has been out for many years.

72.150 requires in part, that activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures.

Contrary to the above, until December 13, 1991, the load cell was jumpered out-of-service without any documented evidence.

The plant lifted at least 68 fuel casks between 1976 and 1984 without the overload protection in place. As stated previously, there are no plant records to indicate the status of the load cell during that time frame.

It should be noted that the load cell was out-of-service in 1981 when the reactor head strongback hit the crane bridge girders rendering them around 22 percent overstressed at rated load. The actual impact load could not be determined because the load cell was out. This lack of indication and associated overload protection probably allowed overloading of the crane through the years during outages when heavy lift weights have apparently been calculated instead of measured.

72.168(b), requires in part, that measures to identify the operating status of systems shall be established to prevent inadvertent operation.

Contrary to the above, the license failed to control the crane in "restricted mode" while the load cell was jumpered out-of-service for at least 68 cask lifts.

During 1996, while re-base lining, the UFSAR to the current design requirements for the reactor building crane, the operability determination from 12/13/91 was uncovered that identified that the digital weight indicator has been out for many years and remained inoperable. In the operability determination, in the justification of its operability determination section, justification of its operability was based upon the crane's having another load limiting device through use of the crane's "over torque limit." The licensee is still attempting to ascertain what part of the crane this is. To meet the original safety intent of the digital weight indicator, engineering recommended that the indicator be restored to operating in 1991. As a temporary alternative, engineering believed that revising station procedures to add the requirement of additional supervising personnel during crane operation to ensure load hang ups do not occur would be sufficient. Site procedure DMP5800-18, Revision 07, has this statement incorporated into it.

A 10 CFR 50.59 Safety Evaluation was performed in 1996 to change USFAR Section 9.1.4.2.2 to include a statement that, "As an alternative to the digital load limiter, station procedures require supervising personnel to ensure load hang ups do not occur during reactor building crane operation." The 1991 operability was used as justification for the 50.59; the non-existent over-torque limit justification.

10 CFR 50.59 (d)(1), requires, in part, that the evaluation shall provide the basis for the determination that the change does not require a license amendment.

Contrary to the above, the stated basis, the "over torque limit," does not exist on the crane. Thus, the 50.59 basis was inadequate.

When the crane upgrade was installed in early 2001, a new digital overload protection system was installed and tested improperly to only a test load of 8000 pounds. The upper limit load trip was set according to drawing 12E-6510 to 110 percent of the 125 ton rated load until the inspector pointed out that it was set wrong. It should be set to the rated load.

During various dry runs the inspection noted that the digital indication of the load cell was reading a negative load with a weight on the main hook. The inspector was told that the load cell was accurate to within 50 pounds; and on another occasion, to within 1 percent of full scale or the actual reading - they weren't sure. The inspector was told that all this confusion on the load cell was because the plant relied on skill-of-the-craft to operate the crane and as a result, they weren't zeroing out the crane's load cell correctly. The training did not follow site training processes, nor was the operation of the load cell proceduralized.

72.144(d), requires in part, that indoctrination and training of personnel performing activities affecting quality, to ensure that suitable proficiency is achieved and maintained, shall be provided.

Contrary to the above, the licensee failed to provide the crane operators the appropriate training and written procedures for the operating load cell.

This training on zeroing out the load cell was supposed to fix the weight indication problem. Subsequently, the inspector noticed that the indicator was reading 5200 pounds when only the hi-trac lifting yoke was on the hook. It should be noted that the yoke has a calibrated weight of 3400 pounds. The total lifted weight of a full cask had been calculated as 199,394 pounds, using the lower (3400 lb.) yoke weight. If the yoke weight is actually 5200 lbs., the total lifted weight of a full cask will exceen 100 tons. The licensee was in the process of figuring out how to calibrate the new load cell system and indicated that they would not lift any cask until it was resolved. After the licensee thought it was resolved, the actual loaded cask (with fuel) indicated 215,000 pounds on the digital readout. This is slightly over the estimated calculated weight of 192,000 pounds. The licensee told the inspector that it's reading conservative; so it is alright to use the crane. Subsequently, on June 20, 2001, tests were conducted with a calibrated dynamometer that indicated the as-found load cell readings were up to 40 percent high as found. New calibrations were done on the load cell which set it slightly below (not conservatively) the actual test weight of 76,000 pounds.

From discussions with a cognizant load cell calibration company, it was determined that because of hysteresis losses, creep, and non-linearity of load cells, the recommended calibrations should be done at 0, 50, 100, 50, and 0 percent of full scale output. National Institute of Standards and Technology (NIST) Handbook 44, requires a minimum test weight of 25 percent of rated load, and to capacity during initial verification.

72.162, requires in part, that the licensee establish a test program to ensure that all testing, required to demonstrate that components will perform satisfactorily in service, is performed.

Contrary to the above, the licensee still hasn't satisfactorily calibrated the new electronics for the load cell at the high end of its upper limit of 125 tons.

- Restricted Load Handling

The previous inspection identified a question about use of a slow-speed or "inching" motor when operating the crane in "restricted mode." The UFSAR, Section 9.1.4.2.2 described fuel cask handling above the 545-foot level of the reactor building as being a restricted load requiring handling in slow-speed with the fast-speed circuitry disabled. The slow-speed motor had malfunctioned in 1976, and it was never used even though the crane was operated in "restricted mode" on many occasions up to the present day. The cover letter to the Amendments, dated June 3, 1976, described the non-reliance on the slow-speed motor as an item for which the licensee had requested only a temporary waiver, until the end of August, 1976.

Further inspection showed this statement was in error. Prior to the date of the SER, the licensee requested an alternate approach, relying on a speed control circuit which could limit hoisting speed to five feet per minute, consistent with BTP 9-1. NRC approved the alternate approach within the SER itself. In the NRC's safety evaluation for the 1976 crane modification, the NRC noted that because CECo experienced problems with the slow speed motor installation, it could not complete the installation prior to planned fuel shipments in June 1976. Because it was consistent with BTP 9-1, the NRC did not make this part of the limiting conditions for operation in the technical specifications as the other two temporary component waivers were until August 30, 1076. Thus, the "inching" motor was never part of the single-failure-proof licensing basis for the crane.

As of June 22, 2001, USAR section 9.1.4.2.2 still refers to the crane having a slow speed (inching) motor in eight locations. The licensee never corrected the USAR from 1976 to delete the non-existent slow speed motor descriptions. The licensee also needs to revise the USAR to reflect the new digital electronic motor controllers.

Modifications to the Crane

The previous inspection identified a question regarding whether the crane had an "extensive repair" or a "major alteration" after it was damaged in 1981 and a load test was required. No load test had been performed. Subsequent observations by the inspector revealed that there are numerous smaller dings on the lower flanges of the girders, indicating that the girders have been hit numerous times because of various limits on the crane being temporarily jumpered out.

In 1981, the crane bridge box girders were damaged by impact and compression during an over-hoisting event involving the strongback for the reactor vessel head. The crane girders are built-up box beams approximately 8 ½ feet deep, 2 feet wide, and 113 feet long. The damaged surfaces were confined to the lower portion of the inside webs (buckling) and the inside portion of the bottom flanges (bending) on both the girders approximately 35 feet from one end. The Office of Nuclear Reactor Regulation (NRR) evaluated the licensee's actions to determine whether the crane had been extensively repaired.

The licensee's required repairs (to get back to the design basis stress allowables) consisted of a splice plate welded over the cutout portion of the damaged girder webs plate and welding cover plates to each girder's bottom flange. These repairs were made by the licensee and Nutech was contracted to perform the repair evaluation. On page 3.1 of their repair report, Nutech concluded that the west crane bridge girder damage was so severe that it would be 22 percent overstressed at rated capacity (125 tons) with the damaged area not fixed. Based on 22 percent reduction in load carrying capacity with the damaged section unaccounted for, the NRR staff concluded that the crane would be capable of lifting 100 tons without overstress.

The repairs made to the crane were determined to restore the capacity of the crane girders to the original design value (Page 3.2 of the Nutech report), thus the crane would be able to support a lift of about 125 tons without appreciable overstress. The staff notes that this is a conservative assessment since the damaged areas of the girder's bottom flanges were not removed, but instead covered. The crane girders and other critical structural components were designed to carry 125 tons with minimum safety factors.

Additionally, based on their analysis, Nutech concurred on Page 3.3 of their report that the repair work was not in the extensive repair or major alteration category as defined by the 1976 ANSI B30.2 code. If the repairs were found to be extensive, then ANSI B30.2 would require that the licensee conduct a load test on the crane.

Part of the licensing basis, Report No. 41, stated that the crane will be compatible with the requirements of ANSI B30.2. Technical Specification 4.10(F) stated that the crane will be adequately inspected in accordance with the accepted ANSI Standard B.30.2. B30.2, 1976, requires that prior to initial use, all extensively repaired, and altered cranes should be tested confirming the load rating of the crane.

Contrary to the above, the crane was not load testing following the repair which was needed to restore the crane to its design basis stresses.

Reactor Building Super-Structure

The previous inspection discussed a series of calculations from the 1960's forward which disclosed examples of various building structural members (roof trusses or columns) in overstress, either with or without design load assumed to be on the crane, and including or not including design-basis earthquake conditions.

During this inspection, substantial additional reviews of the licensing basis for the building, and of the possible performance of the building super-structure in seismic conditions, were performed. The Reactor Building is classified as Seismic Category I in the USAR Section 3.2.1, and is to be designed to accommodate the load conditions and stress criteria in section 3.8.4. Section 3.8.4.1.3 states the following load combinations will be used for Class I structures.

- D (dead load + live load) + E (OBE load)
- D + E' (SSE load)

Further, Sargent & Lundy (plant's engineers) Structural Standard SDS.E5.2, requires that the crane and trolley rated lifted live load, including impact load, longitudinal load and lateral load must be used in combination with the earthquake loads for seismic structures. Contrary to this, the licensee chose to define the live load to be without the crane rated load included in the above equations. This interpretation was assigned despite the fact that the analyzed loading combinations were unconventional, even for the 1970's. That is, the building was analyzed for "normal" loads (wind, snow, etc.) with full lifted live load on the crane, but the analyses for design basis earthquake conditions (Operating Basis Earthquake - OBE; and Safe Shutdown Earthquake - SSE) assumed no lifted load on the crane even though the UFSAR equations indicate so.

10 CFR 50, Appendix B, Criterion III, requires in part, measures shall be established to assure applicable regularity requirements and design basis, as specified in the license, are correctly translated into specifications. These measures shall include provisions to assure that deviations from such standards are controlled.

Contrary to the above, the rated live load of the crane (125 tons) was not incorporated into the original reactor building "D" load combination calculations. Therefore, the building never met Category I requirements. Even so, the licensee calculations without the lifted load included in the 1960s showed that the stresses in the support girders, support columns and several members of the roof truss were above the yield stress for the SSE loading. In some of the roof truss connections, the loading exceeded the ultimate capacity of the connections. Nothing was documented outside of the calculations to include any resolution of these issues.

Although there were no analyses of record documenting the performance of the crane, the building, and their interface, under design basis earthquake conditions and with a 100-ton load on the crane, some analyses of this type had been performed which were classified as "beyond the licensing basis." These also indicated the stresses would exceed yield for some structural members.

In 1973, new calculations were performed for the effects of the new heavier single-failure-proof trolley without lifted load. These calculations showed that the columns were overstressed by up to 30 percent for the SSE loading using 0.9 Fy as the allowable stress. The roof trusses and vertical bracing were not evaluated. Nothing was documented outside the calculations again to resolve this issue.

In 1975, new calculations were performed for the columns and the vertical bracing for the effects of the new trolley without lifted load. Modifications were designed to bring all elements within code allowable stresses. The modifications were subsequently not implemented. The Dresden calculation book carries the notation, "Project Canceled, calculation not approved," without explanation.

In 1998 in calculation DRE98-0013 the reactor building superstructure framing was examined again for the dry cask project. The crane support girders, interior building column members, as well as roof truss members again, had interaction coefficients above 1.0. This time they justified it in the calculations based on probabilistic considerations that the earthquake, most likely, would not occur when the crane is in use, and the crane is used only for a small fraction of the time. This probablistic rationale was not addressed in the USFAR.

In 1998, in calculation DRE98-0020, calculations again indicated members were overstressed. This time it was justified because of the small magnitude of overstress, 5 percent, which the calculations say are generally acceptable. They are not generally acceptable.

USAR, Table 3.8.11, states that stresses are to be held below 0.6 Fy for OBE and normal loads, and for SSE loads may exceed Fy in some elements only if the energy absorption capacity doesn't exceed the energy input. No energy balance was performed.

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to the appropriate levels of management.

Contrary to the above, from the 1960's until June 22, 2001, the licensee failed to promptly identify and correct known deficiencies in the Units 2 and 3 reactor building structural steel supporting the crane. Certain beams and columns exceeded the allowable stresses for Class I building type structures specified in Dresden Updated Final Safety Analysis report (USAR) Table 3.8-11, a significant condition adverse to quality. Ultimately, it was not possible to answer the question whether the building, the crane, and/or the load (a spent fuel cask containing 68 spent fuel assemblies) would fall.

In an eqrthquake, particularly one causing building deformation, the load would also be subjected to lateral forces. The crane yoke for holding the spent fuel cask load is not equipped with positive latching (see paragraph 2.2.b) capable of retaining the cask against significant lateral force. This suggests that a cask could slide off the yoke and fall, even if the crane and building do not fail catastrophically. This potential scenario also has not been analyzed.

Crane Inspections

The previous inspection identified that the crane manufacturer (Whiting Crane) had been performing annual inspections of the Unit 2/3 reactor building crane, and had been identifying discrepancies. These discrepancies were not being addressed by licensee corrective actions. The process to capture and act on vendor recommendations appeared ineffective.

Due to years of neglect of crane maintenance, numerous failures occurred to the crane during Unit 3's last outage. The failures included the main hoist brake, the trolley brake, the equalizing bar circuitry, and inverter on the trolley, a motor exciter, and trolley conductor shoe. To make the crane operational during the remainder of the outage, crane parts from the Unit 3 Turbine Building Crane were cannibalized.

Prior to the outage, an annual inspection was performed. In this inspection, five items were identified as needing correction and a work action request, AR990093876, was generated; but only one item was addressed, without addressing the other four. However, the action request was closed. Numerous needed repairs through the years have been documented on action requests, but canceled, with the justification, that the repairs were not required. These are examples of an inadequate process to capture and address crane issues. Correction of these items most likely would have prevented some of the crane failures during the outage. The licensee is addressing this issue, but not the fact that the action request was closed.

72.172, requires in part, that measures be established to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected.

Contrary to the above, the licensee addressed the fact that the corrective action process for the yearly crane inspection identified repairs was inadequate. However, the apparent cause evaluation (ACE) written to address this issue failed to address the fact that, AR990093876 for the five identified deficiencies during the last yearly inspection was closed as complete, without any work being performed on four of the five items, with no justification.

During this inspection, repairs were performed for each of the four vendor-identified discrepancies from the latest inspection, or the affected equipment was replaced as part of the digital control system modification. No previously-identified material discrepancies remained when the crane was made available to the fuel storage project for heavy load handling.

c. Conclusions

2.1.2 New Identified Items

a. Inspection Scope

The inspection included a review of additional Unit 2/3 reactor building crane issues.

b. Observations and Findings

Seismic Crane Qualification

Dresden Special Report No. 41, dated November 8, 1974, stated that the entire crane trolley and existing bridge girders will be reviewed for the revised new trolley weights in conjuction with the lifted load requirements to establish compliance with Crane Manufacturers Association of America (CMAA) permissible stress ranges. Design values for the Operational Basis Earthquake (OBE) will be based on AISC Code requirements of 0.60 times the minimum yield strength of the material (Fy) and 0.90 Fy for the Design Basis Earthquake (DBE), the Safe Shutdown Earthquake (SSE). In the interim, the NRC issued Branch Technical Position APCSB 9-1 in April 1975, which required that the crane be classified as Seismic Category I and should be capable of retaining the maximum design load during an SSE. The design rated load plus operational and seismically-induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for the design of the bridge.

Dresden Special Report No. 41, Supplement A, dated June 3, 1975, stated that the crane was identified as Safety Class II in the plant operating license and it is not practical to consider reclassifying the crane as Seismic Class I. This is because it would require a new bridge and extensive modifications to the bridge trackway (the Rx Building Superstructure). It further stated that the bridge and trolley will be analyzed with only static lifted loads considered. These are all not consistent with the Branch Technical Position.

Furthermore, during the evaluation and repairs from the 1981 event, NUTECH stated in their report that the original governing design codes state that the most restrictive code allowables shall be used. This crane was built to the CMAA (1975) stress ranges which only allow for 17,600 pounds per square inch tension and compressive stresses. A note in the 1981 calculations indicates that at CECO's direction, to use AISC allowables of 21,600 pounds per square inch stresses. This had a major impact on the acceptability (extensive versus minor) of the existing crane and damage repairs evaluation results. at the CMAA stress limits, the crane would have been approximately 50 percent overstressed instead of 22 percent.

As of June 22, 2001, the licensee has not been able to locate any evaluations for the seismic stresses on the trolley as committed to in Supplement A.

The crane bridge girders were only checked for static lifted loads in 1974 without any pendulum or swinging loads. Calculations indicate the girders were acceptable for the non-seismic loading conditions for the new trolley additional weight of 25,000 pounds. The calculations indicate that the girders were acceptable for SSE static loading. However, the calculations indicate that the girders were 2 percent overstressed for OBE static loading and 6 percent with the added pendulum loads.

72.170 requires in part, that measures shall be established to control materials, parts, or components that do not conform to their requirements in order to prevent their inadvertent use. These measures must include procedures for identification and documentation.

Contrary to above, the bridge girders were found to be overstressed but weren't documented as non-conforming and have continued to be used.

NRC Bulletin 96-02

When the Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," was developed, the NRC policy in the area of handling heavy loads at the time was set forth in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and the Standard Review Plan (SRP). BTP 9-1 was not part of the policy at the time, having been replaced previously in 1979 and 1980 by the two NUREGs. The staff has determined that the provisions of the BTP were subsumed into the NUREGs.

In the May 13, 1996 licensee's response to Bulletin 96-02 the licensee stated that they currently have no plans for any movement of dry storage casks over safety-related equipment while the reactor is at power. However, if such movements were planned in the future, they would demonstrate the capability of safe shut-down in the presence of the radiological source term that may result from a breach of the dry storage cask, damage to the fuel, and damage to safety-related equipment as a result of a load drop inside the facility.

This condition was incorporated into site procedure DMP 5800-18, Revision 7. After discovering this, the inspector requested the evaluation that the licensee could safely shutdown the plant. After being told numerous times that the evaluation is forthcoming, the inspector was given a Commitment Change Evaluation form, 2001-002, dated after the inspector's request, that deleted the commitment. The rationale used was that the reactor building crane is single-failure-proof and dropping the cask is a non-credible event so they do not have to have an evaluation.

c. Conclusions

2.2 Cask Transfer Facility (CTF)

a. Inspection Scope

Because removing fuel from the Unit 2/3 reactor building required the use of another heavy load device, the Cask-Transfer Facility (CTF), the inspection evaluated various attributes including whether the design of the CTF satisfied all requirements in accordance with the Certificate of Compliance (CofC).

b. Observations and Findings

The inspector reviewed various aspects of the CTF including single-failure-proof-design, fabrication, and ANSI N14.6 requirements.

Single-Failure-Proof Design

The concept of a CTF initially appeared in the Hi-Storm TSAR. As the NRC considered the concept, additional detail regarding the design of a CTF was needed if it was to be included in the CofC for the Hi-Storm system. In response, Holtec expanded Section 2.3.3.1 of the Hi-Storm TSAR to include sketchy design features that any CTF would have to meet. However, the CofC does not address the CTF.

TSAR, Section 2.3.3.1 addresses single-failure-proof design in several places: Section A.iii defines the CTF Structure as the "stationary, anchored" portion of the CTF; Section A.iii also defines Single-Failure-Proof as a device wherein all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria in NUREG-0612. This definition was refined by Section 203.12 of the licensee's CTF purchase specification, which required the CTF to be able to withstand a failure of any one part or component without resulting in an uncontrolled lowering of the load.

The TSAR (Section 2.3.3.1.C.ii) states that 3 main portions of the CTF shall comply with NUREG-0612 guidance: the connector bracket, Hi-Trac lifter, and MPC lifter. These three portions are defined rather broadly but are clear enough to permit distinction between these portions and the stationary CTF structure itself. The licensee in their purchase specification (Section 203.6) stated that the CTF structure includes the fixed-position vertical structural members that support the cask.

NUREG-0612, Section 5.1.6 invokes NUREG-0554 for the design of cranes. It defines single-failure-proof as a system that is designed so that a single failure will not result in the loss of the system to safely retain the load. By issuing the CofC, the NRC approved TSAR Section 2.3.3.1; however, Section 3.5 of Appendix B specified the Hi-Trac lifter and the MPC lifting device must be designed, fabricated, operated, tested, inspected, and maintained in accordance with NUREG-0612. The NUREG-0612 requirements include the requirements of NUREG-0554, which it invokes. These appear to apply to the Hi-Trac lift system, including all load bearing components and parts, which includes the lift platform itself, and not just the cask lifting yoke pieces, and the screw jacks, (the Hi-Trac lifter).

72.146 requires, in part that measures shall be established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions.

Contrary to the above, the safety factor for the lift platform beam does not meet single-failure-proof criteria of NUREG-0554.

<u>Cask Transfer Facility Fabrication</u>

NUPIC performed a joint audit of Holtec's implementation of its quality assurance program at its subcontractor Omni Fabricators on May 22 to May 26, 2000. The audit found the following:

- Omni was not on the approved supplier list.

- Omni did not have a quality assurance program that conformed to the requirements of Parts 71 or 72, nor to the requirements of 10 CFR 50 Appendix B.

- Fabrication process controls were less than effectively implemented. Weaknesses were identified associated with special processes (i.e., welding), work process control documents, and the lack of documented instructions and procedures for controlling activities that have an impact on quality.
- Holtec, consistent with its quality program, had extended its quality program to Omni, but Holtec's oversight and implementation of its quality program at Omni's facility were considered to be ineffective at the time of the audit.

- Weld procedures still needed to be qualified, as did welders.

Based on the audit findings, a follow up audit was conducted August 29-31, 2000. The "limited-scope" audit assessed the adequacy of corrective action implementation in response to just the five documented audit findings. The corrective actions included: revising the incorrect procedures and records identified in the audit; developing procedures that the audit team found missing; and providing training to personnel on the new procedures. Corrective actions were narrowly scoped, only looked at past work, and did not address programatic issues. The finding that Holtec had not been effectively implementing its quality assurance program at Omni was not re-examined. The audit did not look at any work activity at Omni. The audit report stated the number of qualified welders was very limited; weld procedures still needed to be qualified, as did welders and employee experience at Omni with demands of a quality program mandated by the NRC was essentially non-existent. Even so, the audit findings did not lead to a work stoppage.

Subsequently, condition reports were being written at the Dresden site for items received which were manufactured at Omni, which included the transfer cask (Hi-Trac) with its various ancillary equipment, and the CTF. Condition reports identified: incomplete and/or inaccurate CofC information, missing documentation, fit up problems with various pieces of equipment, and corrective actions that clearly demonstrated lack of compliance to the quality program. However, the licensee apparently did not examine the actual fabrication records for compliance.

The inspector requested various welding and inspection records for specific welds on the CTF. The records received consisted of weld data and inspection data transferred cumulatively by the Omni QA Manager, in weld groups, according to the size of welds. The date recorded for the transferred data was the latest date of performance for a particular group of welds. This resulted in all the welds for the whole CTF being signed off by the QA Manager on the same day. Holtec's procedure for control of shop travelers, HSF-316, does not address consolidation of data by the QA Manager. The reason given for the data transfer was the poor condition of the shop traveler paperwork and drawings after being exposed to the shop environment (dirt, grease, etc.). When the original data was requested by the inspector, it could not be provided because it had been discarded.

Without individual weld data, if there is a condition adverse to quality involving a specific welder, all that welder's work is suspect. Nonconformance Report 46, dated September 12, 2000, identified Welder K making a 5/16 inch stitch welds incorrectly. The faulty weld was repaired; however, other similar welds may have been made inside of eight closed boxes making up the vertical tower structure of the CTF. These welds are now inaccessible. Welder identity and the fabrication sequence are non-existent; thus, the other seven welds are questionable, because documentary evidence that the other welds are acceptable is also non-existent.

Licensee and vendor oversight of fabrication at Omni involved multiple, full-time QC inspectors being assigned to examine every aspect of the job. Weekly reports by the Holtec Users Group inspector constitute a generalized record of activities inspected, and indicate all the work was being performed by qualified and certified welders and NDE technicians, using qualified procedures, with no specific details given. This canned statement appears on every report with no other details. As a result of this approach, the structural adequacy of the CTF cannot be determined based on specific records of quality verification.

10 CFR Part 72.154, "Control of Purchased Material, Equipment and Services", requires that measures shall be established to ensure that purchased material conform to the procurement documents. They must include objective evidence of quality furnished by the contractor. Documentary evidence that the material and equipment conform to the procurement specifications shall be available for life of the ISFSI. The evidence shall be sufficient to identify the specific requirements met by the equipment.

Contrary to the above, in the case of the CTF, the 5/16 inch stitch welds could not be individually verified to be conforming to the design drawings.

ANSI N14.6

The CofC requires that the CTF be designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." NUREG-0612 requires that the special lifting devices should satisfy the guidelines of ANSI N14.6. N14.6 requires that (1) special lifting devices that require remote engagement with the shipping container shall be provided with lead-in guides and sufficient clearance between the container attachment points and the lifting hook to allow simple motion engagement; (2) that the means of attaching the special lifting device to the shopping container shall be addressed during the design to ensure the security of the attachment method under load; (3) the actuating mechanism used shall securely engage or disengage; (4) load-carrying components that may become inadvertently disengaged shall be fitted with a retaining latch; and (5) engagement indication be provided whenever it is difficult to observe the attachment points between the special lifting device and the shipping container.

The licensee's purchase specification requires that the lifting yokes (on the CTF and the Unit 2/3 crane) be designed in accordance with ANSI N14.6. Holtec's lift yoke design criteria also specify design in accordance with ANSI N14.6. However, during a review of the criteria, the inspector could not determine how the lifting yokes meet the intent of the ANSI N14.6 requirements stated above. The lifting yoke employs two air-operated swing arms with circular cut-outs which slip over the cask lift trunnions. No physical locks or latches are provided, nor is any type of flag provided as an indication of engagement.

72.146 requires in part, that measures shall be established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, the Unit 2/3 crane yoke and the CTF Hi-Trac lifting yoke, do not meet ANSI N14.6 requirements.

Subsequent to the inspector questions, Holtec revised the design criteria to include statements that "each cask lifting trunnion is equipped with an end cap, which allows for visual verification that the lift yoke arms are properly engaged," and "because the design of the lift yoke shall ensure that the lift yoke arms hang plumb to engage the lifting trunnions, there is no need to provide an additional security device to maintain attachment. There are no credible loads that would apply a side load between the cask lifting trunnion and lift yoke arm. Therefore, the security of the attachment method under load in all handling positions is assured."

The inspector determined that the design does not ensure that the lift yoke arms hang plumb. The arms are adjustable and could engage the trunnions at an angle, thereby increasing the stress concentrations on the trunnions. During "dry run" testing, the arms were made plumb, not by design or site procedures, but by engineering inspection and adjustment. The licensee is considering a procedure step requiring verification of the yoke arms being plumb.

c. Conclusions

3.0 Management Meetings

The inspector presented the inspection results to licensee management daily during the ongoing inspection and at a special exit meeting on June 7, 2001, and during a subsequent phone call on June 22. The licensee acknowledged the findings presented. The licensee did not identify any of the documents or processes reviewed as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Dale Ambler, Regulatory Assurance Manager
Robert Fisher, Unit 2-3 Station Manager
Paul Planing, Unit 1 Manager
Nate Leech, Dry Cask Storage Project Manager
Bob Rybak, Regulatory Assurance
Ken Ainger, Regulatory Services
Joe Sipek, Nuclear Oversight Manager
Dave Schupp, Operations
Preston Swafford, Site Vice President
Tom Luke, Site Engineering Director
Chip Cerovac, Training
Dave Williams, Electrical Maintenance Superintendent
Ken Bowman, Operations Manager,\
Joe Kotowski, Operations Supervisor
Pete Scardigno, Site Project Manager

Illinois Department of Nuclear Safety

Rick Zuffa, Dresden Resident Inspector

NRC - Region III

Marc L. Dapas, Deputy Director, DNMS
Bruce L. Jorgensen, Chief, Decommissioning Branch, DNMS

The inspector also interviewed other licensee personnel in the course of the inspection.

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

May 23, 2001

MEMORANDUM TO: J. E. Dyer, Regional Administrator

FROM: Ross B. Landsman, Project Engineer, DNMS

SUBJECT: DIFFERING PROFESSIONAL VIEW CONCERNING THE STARTUP OF THE DRY CASK STORAGE LOADING CAMPAIGN AT DRESDEN UNITS 2 & 3

It is ironic that in 1976, we allowed the licensee to use the unacceptable Unit 2/3 Reactor Building Crane for handling fuel casks. This was temporary and only while the Unit 2 Reactor was shut down. If we didn't, it would impact the licensee's schedule for fuel handling (see licensee's letter dated 5/20/1976). It should be noted that they have loaded greater than 68 fuel casks between 1975 and 1984 with Unit 2 on-line which was contrary to technical specifications after the amendment on June 3, 1976. The four loaded prior to the amendment were contrary to their original license.

We gave them a temporary waiver on the installation of three planned modifications; one of which appears was never installed, the inching motor. Another issue we gave them a temporary bye on was the strength of the main cable (wire rope). The factor of safety was not acceptable. We told them to tell us when they would replace the rope with the appropriate rope. Twenty-six years have passed with the same unacceptable rope. NUREG-0554, paragraph K-1, requires a 10 to 1 factor of safety considering both static and dynamic loads. The factor of safety on the rope, on the lead line is only 6.564 to 1 based only upon a static load. A dynamic load increase of approximately 10% would reduce the factor of safety to under 6.0. Additionally, a critical item of NUREG-0554 is the hook which also requires a factor of safety of 10 to 1; it's at 8.5 to 1.

The required load cell was jumpered out of service before 1981. Without the load cell, the crane may have been overloaded numerous time during the 20 years.

We gave them a bye on the correct seismic design of the crane and supporting Reactor Building Superstructure because it was "not practicable." It would have required a new bridge and extensive modifications to the supporting structure. They told us the existing crane and support structure would be evaluated for OBE loads with AISC allowables used, and SSE loads with a maximum of 0.9Fy used for material strength. It should be noted they had to use greater than Fy for the building material strength to get the interaction coefficients at or equal to 1.0. These calculations were performed without lifted load included to determine if any revisions to beams would be required. See Inspection Report 07200037/2001-001 to see how well the required mods were implemented.

Even then, all these issues were contrary to Branch Technical Position APCSB 9-1 (the fore-runner of NUREG-0554 and NUREG-0012), and we still approved the crane (temporarily) and let them carry casks over safety related equipment unanalyzed. At least the reactor was supposed to be shut down.

Here we are, 25 years later, with the same unresolved issues, along with additional critical issues, ~~letting them~~ go again because of ...

In 1981, the crane was not load tested following extensive needed repairs of the girder as required by ANSI B-30.2. The licensee's calculations indicated that without the repairs, the main girders would be over stressed by 20%. Thus, necessitating repairs to restore it to Operability. Portions of the web plates were cut out and replaced along with the addition of cover plates over the bottom flanges. The licensee classified this as a minor repair and we are agreeing with them?

In 2001, in an attempt to restore the crane to Operability (because of years of neglect), a "major" (licensee's word) crane modification was performed which replaced all the crane controls including over 700 new electrical - terminations. Again, without the B-30.2 required load test and we are agreeing with them again?

On May 13, 1996, in their response to us from Bulletin 96-02, they re-affirmed their commitment to us not to carry heavy loads over safety related equipment while the reactor was at power because it's prohibited by Technical Specifications. They further stated that if such movements would be done in the future, they would demonstrate that they can safely shutdown the reactor as a result of a load drop inside the building. Later on in 1996, we allowed the licensee to remove all requirements and restrictions from the Technical Specifications concerning the Reactor Building Crane, and implement them through administrative procedures, which the amendment reviewer never saw (see Amendment dated June 28, 1996.) However, the requirement to not move heavy loads over safety-related equipment at power was conveniently never heard from again. When the inspector informed the licensee that they had a commitment (which did make it into the procedure), to demonstrate that they could shut down the reactor if there was a load drop, the licensee subsequently also deleted that commitment.

The licensee deleted the commitment "to demonstrate the capability of performing the actions necessary for safe shutdown in the presence of the radiological source term that may result from a breach of the dry fuel storage cask, damage to the fuel, and damage to safety-related equipment as a result of a load drop inside the facility." Their rational was that the Reactor Building Crane is single-failure-proof and thus a load drop analysis is not required to be performed.

Even though HQ concluded that the deficiencies noted above exist, they do not create an imminent threat of adequate protection, and no NRC action to intervene is required; there still is the unanswered question of does the proposed activity increase the consequences of an accident. HQ conclusion was based upon the fact that we issued a paper 25 years ago that said it was single-failure-proof. They also indicated that since the crane has operated for many years without dropping a load, i.e., the rope or repaired girder haven't failed, nor has an earthquake occurred during prior fuel cask handling, it must be ok.

Prior commission approval is required if the proposed change, test or equipment involves a change to commitments incorporated in the license or an unreviewed safety question exists.

Both of these are in effect here. Commitments made to the NRC have been deleted and there is an unreviewed safety question in moving the cask over the torus and other safety-related equipment while the reactor is at power.

An unreviewed safety question exists (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The proposed cask movement activities represent an unreviewed safety question that should be submitted for NRC review and approval per 10 CFR 50.59 and 50.90. This is based on the movement of loads heavier than those previously analyzed in the FSAR. This is also based on the fact that the load drop had not been previously evaluated, and on the possibility that a drop in the reactor building while the reactor is at power could result in consequences that are greater than those previously postulated in the FSAR.

Therefore, although the licensee had reduced the probability of dropping the cask, a load drop could result in an increase in the potential consequences, accordingly, as defined in 50.59(c), if an activity is found to involve an unreviewed safety question, an application for a license amendment must be filed with the commission pursuant to 50.90.

In summary, allowing Dresden to use the reactor building crane with an unacceptable rope, an untested crane, a crane or building structure that wouldn't support the load in an earthquake, all while Unit 2 is at power, does not meet the intent of our regulations and should be stopped. Furthemore, we stopped Oyster Creek in 1996 from doing the same thing with an identical non-single-failure-proof crane. Why is this different?

From: Jim Dyer
To: Ross Landsman
Date: Wed, May 23, 2001 3:52 PM
Subject: DPV Memo Dated 5/23/01

Ross,

I received your memo to me identifying your disagreement with the NRC decisions concerning the startup of dry cask storage loading activities at the Dresden Station. This memo identified several issues concerning the historical use of the crane as well as its current configuration and readiness to safely conduct cask loading activities.

As we discussed this morning, it is premature to start the DPV process on this issue before the NRC has come to a decision on these issues. We are focusing our current efforts on identifying the activities necessary for cask loading and will disposition the historical issues via the enforcement process. A Region III public meeting is being held on the subject today with the licensee and further inspection activities are planned in the near future. Therefore, as we agreed to earlier, I will hold your DPV until a NRC decision is made and then review with you the alternatives and whether you wish to proceed with establishing a DPV panel in accordance with MD 10.159. I discussed this approach with Jim McDermott, OHR, and he agreed with this deviation from the MD 10.159 timeline for establishing the DPV panel.

Additionally, I want to thank you for allowing me to provide copies of your memo to the NRC staff to better prepare for the public meeting today and facilitate further deliberations on the subject. I hope that you fully participated in today's public meeting and raised your concerns to the licensee and attending NRC staff.

Jim.

CC: Bruce Berson; James Caldwell; James McDermott; ...

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

April 2, 2002

Rec'd 4/3 605PM JEDyer

MEMORANDUM TO: J. E. Dyer, Regional Administrator

FROM: J. A. Grobe, Director, Division of Reactor Safety

SUBJECT: RECOMMENDATION OF AD HOC REVIEW PANEL FOR DIFFERING PROFESSIONAL VIEW: STARTUP OF CASK STORAGE LOADING CAMPAIGN AT DRESDEN UNITS 2 AND 3

In accordance with your memo dated July 20, 2001, to me (Reference 1), an Ad Hoc Differing Professional View (DPV) Review Panel (Panel) was formed in accordance with Management Directive 10.159 with myself as Chairman and John Jacobson and Patrick Hiland as members. The Panel reviewed several issues related to the loading and handling of spent fuel dry storage casks at the Dresden facility. The purpose of this memorandum is to provide you with the Panel's review, conclusions, and recommendations for this DPV. The schedule for resolution of this DPV was protracted due to the NRC's response to the September 11, 2001, event, the need for input on several complex technical and licensing basis issues from the Office of Nuclear Reactor Regulation (NRR) and the Spent Fuel Project Office, and the nexus between the DPV issues and a backfit analysis Task Interface Agreement on Dresden dry cask transfer issues under review by NRR.

The DPV addressed three main issues related to the Reactor Building and Cask Transfer Facility (CTF). The first issue concerned the integrity of the Reactor Building structure with respect to design basis loading conditions and loads associated with a cask lift. The second issue concerned the compliance of the Cask Transfer Facility to applicable codes and standards. The third main issue concerned the quality of some welds on the CTF. The DPV also addressed six issues related to the Reactor Building crane. These issues (Reference 2) were developed through review of various documents including the draft and final reports (References 1 and 3) and several meetings with the Submitter. The summary of the issues (Reference 2) was compiled by the Panel and provided to the Submitter. The Submitter acknowledged that the summary adequately captured his concerns.

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During the review of this DPV, the Panel met on several occasions, interviewed the Submitter, interviewed key Region III managers (Reference 10), and conducted several telecons with both NMSS and NRR staff and management. Written responses were requested (Reference 4) and received (References 5, 6, and 7) for portions of the three main issues.

The Panel did not identify any immediate safety concerns regarding dry cask movement activities at Dresden. The Panel did identify several regulatory and compliance issues warranting further staff consideration. The Panel's review, conclusions, and recommendations are discussed in the attachment.

PANEL RESULTS OF DPV REVIEW

SUBSTANTIVE ISSUES

1.a The reactor building design for Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) load cases did not include the 125 ton crane load (live load) as described in the Updated Final Safety Analysis Report (UFSAR).

REVIEW

The first issue raised by the Submitter was that while the Normal and Wind load analyses for the Reactor Building included the 125T crane load, the analyses for the OBE and SSE load cases did not include the crane load. The Submitter contended that the UFSAR requires that the crane load be included in the OBE and SSE analyses. The licensee's position, presented during a meeting in RIII on May 23, 2001(Reference 8), was that the Dresden design basis did not include consideration of the crane load for the OBE and SSE analyses. The licensee also presented the results of a "beyond design basis" analysis for the SSE load case which did include the crane load. The licensee indicated that results were acceptable. This is discussed in the DNMS inspection report (Reference 3). The Submitter was in attendance at that meeting.

Because it was licensed early, Dresden Unit 2 was included in the Systematic Evaluation Program (SEP). The SEP reviewed the seismic design of Dresden Unit 2 under SEP Topic III-6, "Seismic Design Considerations." The SEP reviewed load combinations under SEP Topic III-7.B, "Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria." The results of the SEP Topic III-6 review is reported in NUREG/CR-0891, "Seismic Review of Dresden Nuclear Power Station - Unit 2 for the Systematic Evaluation Program," dated April 1980 and in the SEP Topic III-6 Safety Evaluation for Dresden Unit 2 dated June 30, 1982. The SEP seismic review only evaluated the Safe Shutdown Earthquake (SSE) seismic design. SEP Topic III-6 identified no open items related to crane live loads and the reactor building structural design.

The load combinations used in the design of Dresden 2 for the reactor building and all other Class I structures are listed in Table 4-4 of NUREG/CR-0891 as D+R+E and D+R+E' where D = Dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads or operating pressures and **live loads expected to be present when the plant is operating** [emphasis added], E = Design earthquake load, and E' = Maximum earthquake load. The SEP Topic III-6 safety evaluation does not specifically state that the SEP considered that heavy loads on the reactor building crane were loads expected to be present when the plant is operating. The SEP review used the Standard Review Plan (SRP), NUREG-75-087, as the basis for its review. Section 3.8.4 of the 1975 SRP gives load combinations consistent with Table 4-4 of NUREG/CR-0891 although it breaks down D into D (dead loads) + L (live loads). SRP Section 3.8.4 defines L as "Live loads or their related internal moments and forces including any movable equipment loads and other loads which may vary with intensity and occurrence, such as soil pressure." SRP 3.8.4 allows deviations from the acceptance criteria for loads and load combinations if the deviations have been adequately justified. NRC did not identify any justifications in the Dresden licensing basis for excluding reactor building crane lifted loads.

NRC completed its review of SEP Topic III-7.B and issued an SE by letter dated August 23, 1990. With respect to the crane live load, NRC's contractor stated in TER-C5506-425 dated November 15, 1983, that the reviewers did not have access to actual design calculations. Also, we have not identified any lists of actual loads. Therefore, it does not appear that NRC or its contractor reviewed individual live loads in their review of Topic III-7.B. With respect to OBE seismic evaluations, the licensee identified in its letter to the NRC dated August 2, 1982, that Sargent & Lundy reactor building superstructure calculations did not include OBE loads but that it was Sargent & Lundy's judgement that the SSE evaluation would control the reactor building superstructure structural evaluation.

The Dresden Units 2 and 3 Reactor Building (including superstructure) licensing basis is described in the UFSAR as follows: UFSAR Section 3.2.1 classifies the Reactor Building as a Class 1 structure. UFSAR Section 3.8.4 defines the load combinations for Class 1 structures to include the dead load plus **live loads expected to be present when the plant is operating** [emphasis added] plus the OBE load (E) for the OBE case or the SSE load (E') for the SSE case.

In preparation for beginning a campaign of spent fuel transfers, Sargent and Lundy performed an extensive evaluation for the licensee (calculation DRE98-0020) (Reference 13) to analyze and evaluate the building superstructure during various loading conditions including OBE (without live load) and SSE (with live load). The licensee states that this calculation includes the loads from the SSE plus the effects of the maximum lifted load of 125T. The effects of the lifted load on the structure include the application of the load vertically as well as the pendulum effects of the lifted load during a SSE hanging from the crane during a seismic event. We note, however, that the licensee refers to SSE plus lifted load as "beyond design basis" although the NRC staff considers SSE plus lifted load to be within the licensing basis if the crane is being used to lift loads while the plant is operating.

CONCLUSION

The UFSAR correctly describes the licensing basis for the Reactor Building as dead loads, plus live loads expected to be present when the plant is operating, plus the seismic load, for both the OBE and SSE load cases. If the licensee intends to lift spent fuel casks when the plant is operating, the spent fuel cask is then a live load expected to be present on the Reactor Building crane when the plant is operating. Therefore, the licensing basis of the plant requires analysis of OBE plus lifted loads and SSE plus lifted loads for the Reactor Building structure.

RECOMMENDATIONS

Notify the licensee that the design basis of the plant requires that both the OBE and SSE load cases for the Reactor Building be analyzed with the 125T (or actual) crane load present if casks (or other heavy loads) are to lifted when the plant is operating. NRC should consider the potential enforcement aspects of this issue if spent fuel casks have been lifted in the past when the plant was operating prior to performing the required analysis.

1.b Calculations indicate that some Reactor Building structural components exceed both yield and ultimate tensile strength for the SSE load case.

REVIEW

There is a long history of calculations which show multiple Reactor Building structural members and connections to be outside design limits (several examples are described in Inspection Report 2001-002(DNMS). For example, Dresden Calculation No. DRE98-0013 is discussed as showing some crane support girders, interior building columns, and roof truss members exceed design allowable stress limits. The licensee concluded that the overstress was acceptable based on probabilistic considerations. Dresden Calculation No. DRE98-0020 (Reference 13) indicates some roof truss members exceed design allowable stress limits by 5%. The licensee accepted these results based on "normal practice to accept overstress of up to 10%" (Reference 8). Unresolved Item 05 in Inspection Report 2001-002 (DNMS) which follows the discussion of the overstress conditions does not directly address the design compliance issue, rather "long term acceptability of this equipment for handling large numbers of dry fuel storage casks". Unresolved Item 06 addresses the licensee's practice of accepting a 10% overstress condition however, the unresolved item does not address the acceptability of the licensee's use of a probabilistic approach to resolution of design issues.

CONCLUSION

Apparently, the licensee has calculations which indicate that some Reactor Building structural members do not conform to the design allowable limits. All calculations of record showing loads beyond design limits must be reconciled and documented. For example, for the SSE load case, the licensee may elect to use the Limit-Design approach.

With respect to the acceptance of 10% overstress, the Panel is not aware of any recognized code or standard which supports this practice. If the licensee's design practices or methodology inherently includes greater than 10% margin with respect to design, it is up to the licensee to demonstrate and document this. Regarding the use of probabilistic considerations to resolve overstress conditions, the Panel is not aware of any Agency approvals supporting this approach to resolve overstress conditions. If the licensee uses this approach, they need to justify the basis. Typically, these issues are resolved by refining the calculations (removing demonstrated conservatism) or, if necessary, through modifications.

RECOMMENDATIONS

Further inspection should be conducted to verify satisfactory resolution of identified overstress conditions. Evaluation of licensee actions should also be conducted to assess compliance with the requirements of 10 CFR Part 50, Appendix B, Criteria III and XVI.

1.c A 1998 calculation indicates an overstress of reactor building structural components of five percent. The applicable code does not allow any overstress conditions. In addition, the inspection report documents only a three percent overstress.

REVIEW

The 1998 calculation DRE98-0020 shows Rx Bldg structural members to exceed allowable stress by 5% for the normal load case. This is a specific example of the problem stated in 1.b above. This overstress was incorrectly presented by the licensee as 3% (Reference 8) during the May 23, 2001 licensee presentation in RIII and subsequently documented incorrectly in an NRC inspection report (Reference 3).

CONCLUSION

The calculation documents a 5% overstress with respect to design allowable stress levels. All calculations of record showing loads beyond design limits must be reconciled and documented. The licensee's May 23, 2001 presentation slides indicate that it is normal practice to accept overstress of up to 10%. With respect to the acceptance of 10% overstress, the Panel is not aware of any recognized code or standard which supports this practice. If the licensee's design practices or methodology inherently includes greater than 10% margin with respect to design, it is up to the licensee to demonstrate and document this.

RECOMMENDATIONS

Further inspection should be conducted to verify satisfactory resolution of identified overstress conditions. Evaluation of licensee actions should also be conducted to assess compliance with the requirements of 10 CFR Part 50, Appendix B, Criteria III and XVI.

2.a The cask lifting yolks for both the CTF and the Unit 2/3 crane do not meet ANSI N14.6 standards as required by the Certificate of Conformance.

REVIEW

Since the cask lifting yoke did not include a latching device, the Submitter questioned the basis for concluding that the cask transfer yoke met the licensing requirements. The Certificate of Compliance (CoC) (Reference 9) for the Cask Transfer Facility (CTF) requires the device to be single failure proof, and the application states that no single failure will result in a dropped load. Further, the CoC states that the device must meet NUREG-0612 which requires that special lifting devices meet ANSI N14.6. The cask lifting yokes are special lifting devices. ANSI N14.6 indicates that, if it is possible for a load carrying component to become disengaged, it shall be lifted with a latching device with an actuating mechanism that securely engages and disengages. The licensee's purchase specification and the CoC require that the lifting yokes on the CTF and the Reactor Building crane meet ANSI N14.6. Inspection Report 07200037/2001-002(DNMS) (Reference 3) documented that the Spent Fuel Project Office did not attempt to determine how the yokes met the ANSI provisions, but instead, focused on whether any of the provisions were violated (pg. 21, Reference 3).

The panel requested the staff (Reference 4) to provide the basis for the conclusion that the cask lifting yokes meet the licensing basis requirements. The staff response, documented in Reference 6, states that the ANSI N14.6 (1978) contains two provisions that allow the CTF design not to utilize a latching mechanism. As stated in the ANSI N14.6, Section 3.3.5 and 3.3.6, a latching mechanism is required if the "Load-carrying components that **may become** [emphasis added] inadvertently disengaged" or "An actuating mechanism shall be used, **if needed**, [emphasis added]...." The staff responded that for normal lifting operation, the cask is not subject to any lateral load, thus it is not possible for the yokes to become disengaged from the cask trunnions. Additionally, the staff concluded that for seismic events, the cask is pin-supported in a pendulum like configuration, suggesting that the cask will not be subject to any meaningful lateral force.

CONCLUSION

The Panel concurs with the staff's conclusion that the cask lifting yokes appear to meet the licensing basis without a latching device.

RECOMMENDATIONS

None.

2.b The CTF lift platform beam does not meet the single failure proof criteria of NUREG-0554.

REVIEW

The Submitter questioned whether an adequate basis was provided by the licensee to conclude that the CTF lift platform beam satisfied single failure proof requirements. The staff's overall safety evaluation for the design and testing of the Cask Transfer Facility, including the lift platform is referenced in Inspection Report 07200037/2001-002(DNMS), dated August 13, 2001. As part of the staff's safety evaluation (Reference 12), a detailed assessment of the single failure proof design of the lift platform was performed. The staff concluded that "...the lift platform is conservatively designed and is, therefore, acceptable for the design service load of 280,000 lbs."

The panel reviewed the staff's safety evaluation with particular emphasis on the lift platform analysis. For completeness, the following excerpts from Reference 12 were reviewed by the panel:

> 3.2.1.1 Lift Platform Evaluation
>
> The lift platform is bolted at two ends to the screw jack nuts, which, in turn, are raised or lowered by turning the screw jacks against the nuts through a motor/shaft/gear assembly mounted on the CTF top bridge girder. Holtec reports the nut thread bending safety factors of 19 and 48 against F_y and F_u, respectively. The reported nut thread shear safety factors are 50 and 194. These safety factors are more than adequate to satisfy the intent of NUREG-

0612 guidelines to improve the reliability of the handling system through increased factors of safety in certain active components. The lift platform serves a structural support function equivalent to that of a crane bridge girder. CMAA 70 states, "The crane girders shall be welded structural steel box sections, wide flange beams, standard I-beams, reinforced beams, or box sections fabricated from structural shapes." The staff notes that the bridge girder should be conservatively designed but need not be considered single failure proof, in accordance with NUREG-0554. In the following, the staff compares safety factors inherent to the Subsection NF, Level A stress allowables to those of crane industry standards. By considering the stress "design margins" presented in the Holtec report, the staff then computed the overall safety factor to demonstrate that the lift platform is conservatively designed.

Inherent Safety Factors. Using the common structural steel A-36 (F_y = 36 ksi) as a basis, the stress allowable, specified as a fraction of the yield strength, and the inherent safety factor (ISF), defined as the inverse of this fraction, are computed and listed below for the basic tension/compression and bending stress categories considered by three industry standards.

Standard (Bridge Girders)	Basic Tension/Comp.		Bending Stress	
	Allowable	ISF	Allowable	ISF
CMAA 70	0.6 F_y	1.67	0.6 F_y[1]	1.67
Subsection NF, Level A	14.5 ksi[2]	2.48	21.75 ksi[3]	1.66
ASME NOG-1[4]	0.5 F_y	2.0	0.49 F_y[5]	2.04

Notes:

1. Not specified explicitly for bending, but used the basic tension/compression allowable
2. ASME Section II, Part D, Table 1A; 14.5 ksi = 0.40 F_y, approximately
3. Bending allowable = tension/compression allowable x 1.5 (21.75 ksi = 14.5 x 1.5)
4. "Rules for Construction of Overhead and Gantry Cranes," which includes cranes with single-failure-proof features
5. Section NOG-4313: AISC stress allowable (0.66 F_y) divided by 1.12N, where N=1.2 for operating loads

For bending stresses, which usually govern a design, the comparison table above shows that ISFs are essentially identical for the CMAA 70 and the ASME, Subsection NF, criteria. The staff notes that, for the A-36 steel, compared to the CMAA 70 or Subsection NF standard, the ISF, per NOG-1, is about 23% larger for bending stresses.

The staff notes further that all structural steel design ISFs are smaller than the basic safety factor of 3 against the yield strength associated with the mechanical design of the HI-TRAC and MPC Lifter components. This crane industry practice of adopting relatively smaller ISFs for bridge girders is consistent with the common structural steel design philosophy. It is risk informed and acceptable, recognizing that steel bridge girders undergo bounded deformation when overloaded, thereby providing sufficient advanced warning for necessary remedial actions.

Lift Platform Stress Design Margin. The Holtec report defines safety factor as the ratio of the allowable stress and the calculated stress; a safety factor greater than one is considered acceptable. For this evaluation, however, the staff considers Holtec stress safety factors as stress "design margins."

The Holtec lift platform is fabricated with the A-516 Grade 70 carbon steel with a yield strength of 38 ksi and bending stress allowable of 26.25 ksi in accordance with Subsection NF. For a service load of 280,000 lbs plus a 15% dynamic load effect, Holtec reports a minimum design margin of 1.45, which is greater than one. This design margin is above and beyond the ISF of 1.45 (38/26.25 = 1.45) for the A-516 Grade 70 steel although it is slightly smaller than the ISF of 1.66 for the A-36 steel discussed above.

Overall Safety Factor. The staff considers an overall safety factor (OSF), defined as the product of design margin and ISF, for comparing stress design adequacy associated with different design standards for the lift platform. The design margin of 1.45 and the ISF of 1.45 result in an OSF of 2.10 (1.45 x 1.45 = 2.10), on the basis of Subsection NF. As indicated in the ISF comparison table above, a stress design margin of greater than one, which is acceptable on the basis of the more conservative NOG-1 stress allowables, amounts to an OSF of greater than 2.04 (1.0 x 2.04 = 2.04). Thus, the lift platform based on the Subsection NF stress allowables and a design margin of 1.45 achieves an OSF of 2.10, which is greater than the minimum acceptable crane girder OSF standard of 2.04, per NOG-1, for a design margin of one. On this basis, the staff concludes that the lift platform is conservatively designed and is, therefore, acceptable for the design service load of 280,000 lbs."

CONCLUSION

The panel concurs with the staff's June 15, 2001, safety evaluation and determination that Dresden Cask Transfer Facility lift platform design is acceptable.

RECOMMENDATIONS

None.

3. Existing records are inadequate to establish weld structural quality for welds on the Cask Transfer Facility.

REVIEW

The issue raised by the Submitter was that the adequacy of individual CTF welds could not be verified based on a review of quality records. The CTF fabricator's Quality Assurance (QA) manager consolidated the weld inspection records into weld groups according to size. All welds for the entire CTF were signed off by the QA manager on the same day. Since original weld documentation is no longer available, welder identity and fabrication sequence could not be established. A specific example identified by the Submitter was a fabricator's non-conformance report (NCR-46), dated September 12, 2000, that documented an incorrect weld made on a box beam. While that particular weld was repaired, there are no records to indicate that the specific welder didn't make the same mistake on other box beams. As documented in NRC Inspection Report 2001-002(DNMS) (Reference 3) the fabrication welds were determined to be "proper" based on the licensee's assertion that all welds were inspected and identified discrepancies corrected; the documented results of Quality Control inspector activities (weekly Holtec Users Group reports); and the fabricator's QA manager's certification of the cumulative welding data.

The Panel believed that the documented evidence of welding and inspection activities would likely be insufficient for similar nuclear power plant welding for which 10CFR 50, Appendix B applied, and it requested the staff to provide the Panel with the NRC's expectations and quality standards for this issue. The staff responded to the Panel in Reference 6 and also provided additional email correspondence (Reference 7) on February 12, 2002.

The staff's response detailed that metal weldment of the CTF structure, including the lift platform, should comply with the material, fabrication, inspection, and testing requirements of ASME Section III, Subsection NF, Class 3 for linear structures. For weld quality verification, the staff relies on Dresden's quality assurance programs for controlling CTF fabrication activities, including weld quality inspection, to provide adequate confidence that the CTF will perform satisfactorily.

As for weld quality verification, the staff noted that the CTF weld fabrication standards were not submitted for staff review and approval. That is, the staff relies on Dresden's quality assurance programs, per 10 CFR Part 72, Subpart G, for controlling CTF fabrication activities, including weld quality inspection, to provide adequate confidence that the CTF will perform satisfactorily. Thus, upon staff's site inspection and audit, all applicable CTF welds are expected to be in compliance with their quality standards.

The staff's February 12, 2002, correspondence provided a specific record quality trail required by the CoC. As outlined by the staff, ASME Code Article NCA 4000, Quality Assurance, includes NCA 4234.10, Inspection. The applicable requirements include the preparation of process sheets, travelers, or checklists, with space provided for recording results of examinations or tests. The requirements state the document shall include space for: a signature, initials, or stamp: the date that the activity was performed by the Certificate Holders representative, and the date on which those activities were witnessed. The staff noted that the Code requirements for the CTF weld inspection records did not agree with the description of available records documented in Reference 3.

The staff also noted that the CoC, Section 3.3.2, allows for exceptions to the ASME Code requirements when authorized by the Director of the Office of Nuclear Materials Safety and Safeguards when the Certificate holder demonstrates that the proposed alternates provide an acceptable level of quality and safety or result in hardship without a compensating increase in the level of quality and safety. The current CoC, Table 3-1 of Appendix B, does not include a Code exception for CTF weld records.

CONCLUSION

The Panel agrees with the staff's observation that the current weld quality records are not in agreement with the Code requirements. The NRC determination documented in Reference 3 that the CTF welds were "proper," based on licensee assertions and alternate quality verification methods, appears to grant a Code exemption without authorization from the Director of the Office of Nuclear Materials Safety and Safeguards.

RECOMMENDATIONS

The Panel recommends that the licensee be asked to demonstrate how the existing quality records meet Code requirements. If this cannot be demonstrated, the licensee should request an exemption from the requirements of the ASME Code in accordance with the CoC. The Panel also notes that the alternate quality verification methods for CTF weld fabrication documented in Reference 3, by themselves, may not support a Code exemption.

ADDITIONAL ISSUES

1. The crane wire rope does not meet the required safety factor of eight as specified in the UFSAR.

REVIEW

The wire rope is required to have a safety factor of 7.5 as stated in Dresden Amendments 19 and 22. The licensee committed to an inspection and replacement program, however, they did not commit to upgrade the wire rope. The inspection report 2001-002(DNMS) issued an NCV for failure to update the UFSAR which incorrectly reflected a safety factor of 8.

CONCLUSION

The licensing basis for Dresden does not require the wire rope to meet a safety factor of 8, rather, 7.5. Therefore the existing wire rope with a safety factor of 7.798 is acceptable.

RECOMMENDATIONS

None.

2. The current inappropriate operation and testing of the overload protection device (load cell) is dispositioned in the inspection report (Reference 3) as an unresolved item, however, the inspection report does not address the identified deficiencies in competency and training of the staff and technicians who operate and calibrate the load cell.

REVIEW

The inappropriate operation and testing of the overload protection device (load cell) is dispositioned in report 2001-002(DNMS) as an unresolved item. The report does mention equipment and personnel performance challenges, but concludes that actions to correct the problems were successfully implemented.

CONCLUSIONS

The report as issued does not discuss competency and training issues. The Submitter's draft report (Reference 1) does discuss training deficiencies.

RECOMMENDATIONS

The unresolved item should be followed up with further inspection. It is recommended that the identified deficiencies in competency and training of the staff who operate and calibrate the load cell be included in the follow up inspection activities.

3. The inspection report states that the load cell on the Unit 2 and 3 crane hoist was routinely bypassed for 20 years when the crane was in the restricted mode, which was outside the licensing basis. This is a violation of requirements, but is not characterized as a violation in the inspection report.

REVIEW

The issued report does state that the use of the crane for cask handling with the load cell bypassed was outside the licensing basis.

CONCLUSIONS

It appears that a violation occurred, however, no violation was issued.

RECOMMENDATIONS

It is recommended that the licensee be issued a violation, if in fact this occurred, or the report should be clarified.

4. The 1981 repairs to the crane bridge girders were incorrectly classified as a minor repair.

REVIEW

The Panel reviewed the design report for the repairs prepared by Nutech (Reference 14) and the Staff Review of Crane issues (Reference 11). The Nutech report concluded that the repairs were not considered “extensive” as defined by the 1976 ANSI B30.2 code. The Staff review concluded that there was no regulatory or technical basis to challenge this conclusion.

CONCLUSIONS

ANSI/ASME B30.2 - 1967 to which the licensee was committed, specified a 125% load test for “extensively repaired” cranes. While it can be debated whether or not the crane repairs were “extensive” there is no regulatory basis or accepted criterion defining the term “extensively repaired” when referring to crane repairs. The licensee performed the repairs to restore margin of safety for the OBE load case. Additionally, the licensing basis classifies the crane as non-seismic. For the NRC to make a determination of what was intended by the ANSI code would require a backfit analysis. The Panel has no basis to challenge the Nutech conclusion.

RECOMMENDATIONS

None.

5. The 1974 analysis of the bridge girders indicates a two percent overstress condition during an OBE considering only static loads. This over-stress condition is documented in the inspection report, but there is no documentation of the basis for the acceptability of this over-stress condition. In addition, there is no analysis of stresses in the trolley for the OBE or SSE load cases.

REVIEW

Since the Dresden crane is classified as non-seismic, the licensee committed (from Reference 5) to analyze the bridge and trolley in a manner consistent with applicable design codes. Allowable stresses were limited to 90% of yield with only static loads considered.

CONCLUSIONS

While the licensee committed to analyze the crane for the new trolley with static lifted loads, it was stated that the crane licensing basis classified the crane as non-seismic. Therefore there is no apparent regulatory basis to compel the licensee to fully meet the OBE load case. No analysis was located for the trolley.

RECOMMENDATIONS

Request the licensee to produce the trolley analysis per the commitment (from Reference 5).

6. During a crane inspection conducted by licensee representatives, five deficiencies in the crane were identified as needing correction. The licensee initiated a corrective action document, but only corrected one of the deficiencies and closed the corrective action document as acceptable.

REVIEW

The crane inspection performed by the vendor was not a safety related or QA type audit. The inspection was focused on crane reliability and none of the deficiencies related to conditions adverse to quality as defined in 10 CFR 72.172. Therefore the recommendations were up to the discretion of the licensee. The vendor inspection was not done to qualify the crane for cask lifting, rather economics (reliability) for general use during outages.

CONCLUSIONS

Correction of the deficiencies noted by the vendor was up to the discretion of the licensee.

RECOMMENDATIONS

None.

REFERENCES:

1. Memorandum Dyer to Grobe: AD HOC REVIEW PANEL FOR DIFFERING PROFESSIONAL VIEW CONCERNING STARTUP OF THE CASK STORAGE LOADING CAMPAIGN AT DRESDEN UNITS 2 AND 3, dated July 20, 2001(includes attachments).

2. E-mail Grobe to Dyer: DPV Update, dated September 21, 2001.

3. NRC Inspection Report 07200037/2001-002(DNMS), dated August 13, 2001.

4. Memorandum Grobe to Zwolinski, et al., dated December 28, 2001.

5. Memorandum Zwolinski to Grobe: RESPONSE TO REQUEST FOR HQ INPUT ON DPV CONCERNING SEISMIC/STRUCTURAL ANALYSIS FOR DRESDEN UNITS 2 AND 3 SPENT FUEL CASK HANDLING, dated February 22, 2002.

6. Memorandum Brach to Grobe: RESPONSE TO DPV STRUCTURAL ISSUES REGARDING THE DRESDEN SPENT FUEL CASK TRANSFER FACILITY, dated February 4, 2002.

7. E-mail Narbut to Grobe: DRESDEN CTF WELD DOCUMENTATION REQUIREMENTS, dated February 12, 2002.

8. Memorandum Jorgenson to File: MEETING WITH EXELON [May 23, 2001] REGARDING DRESDEN UNIT 2/3 Reactor Building CRANE ISSUES, dated June 1, 2001.

9. Certificate of Compliance (No. 1014) issued to Holtec International, dated May 31, 2000.

10. Memorandum Grobe to Dapas and Jorgenson: DPV REGARDING STRUCTURAL ISSUES ON THE DRESDEN Reactor Building AND 125 TON CRANE, dated November 2, 2001.

11. Memorandum Carpenter to Pederson: STAFF REVIEW OF DRESDEN Reactor Building CRANE ISSUES, dated June 15, 2001.

12. Memorandum Brach to Pederson: SAFETY EVALUATION OF DRESDEN CASK TRANSFER FACILITY, dated June 15, 2001.

13. Commonwealth Edison Calculation NO. DRE98-0020 :"Dresden Reactor Building Steel Superstructure Interaction Summary", dated March 16, 1998.

14. Design Report for Reactor Building Crane Bridge Girder Evaluation and Repairs (Nutech), dated September 29, 1981.

From: John Grobe , R3
To: Jim Dyer , R3
Date: 9/21/01 11:44AM
Subject: Dresden DPV - Update

Jim,

Attached is a document that contains the issues that Ross has agreed represent his concerns. These have been developed through review of various documents including the draft and final reports and meetings with Ross.

Pat, John and I are mapping out our strategy to address these issues and will begin early next week.

I would be glad to answer any questions you have.

Jack

CC: James Caldwell; John Jacobson; Patrick Hiland; Ross Landsman

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DRESDEN STRUCTURAL ISSUES

SUBSTANTIVE ISSUES NOT ADEQUATELY ADDRESSED BY THE NRC

1. Reactor building structural issues.

   A. The reactor building design for Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) load cases did not include the 125 ton crane load (live load) as described in the Updated Final Safety Analysis Report (UFSAR), while the live load considerations were included in the analysis of non-earthquake load cases.

   B. Calculations indicate that some building structural components exceed both yield and ultimate tensile strength for the SSE load case.

   C. A 1998 calculation indicates an overstress of reactor building structural components of five percent. The applicable code does not allow any overstress conditions. In addition, the inspection report documents only a three percent overstress.

2. Cask Transfer Facility (CTF) structural design issues.

   A. The cask lifting yolks for both the CTF and the Unit 2/3 crane do not meet ANSI N14.6 standards as required by the Certificate of Conformance.

   B. The CTF lift platform beam does not meet the single failure proof criteria of NUREG-0554.

3. Cask Transfer Facility weld quality.

   Existing records are inadequate to establish weld structural quality for welds on the CTF.

ADDITIONAL ISSUES NOT ADEQUATELY ADDRESSED BY THE NRC

1. The crane wire rope does not meet the required safety factor of eight as specified in the UFSAR. This condition was acknowledged by the NRC and accepted in a 1976 license amendment based on a commitment to implement compensatory actions and to upgrade of the wire rope to one that has a safety factor of at least eight at the time of its next replacement. The wire rope has been replaced a number of times with like-for-like deficient (e.g., safety factor less than eight) wire rope. The inspection report characterizes this issue as a non-cited violation for failure to update the UFSAR; however, the report does not address the failure to upgrade the wire rope with rope that meets the original design or the failure to continue implementing the compensatory measures following the 1976 amendment.

2. The current inappropriate operation and testing of the overload protection device (load cell) is dispositioned in the inspection report as an unresolved item, however, the inspection report does not address the identified deficiencies in competency and training of the staff and technicians who operate and calibrate the load cell.

3. The inspection report states that the load cell on the Unit 2 and 3 crane hoist was routinely bypassed for 20 years when the crane was in the restricted mode, which was outside the licensing basis. This is a violation of requirements, but is not characterized as a violation in the inspection report.

4. The 1981 repairs to the crane bridge girders were incorrectly classified as a minor repair. One outcome of classifying the repair as minor was that there was no requirement to perform a full load test of the crane following the repair as would have been required had the repair been classified as extensive. Part of the licensee's basis for justification of the classification of the repair as minor was predicated on the Nutech analysis which demonstrated that had the repairs not been made the girders would only have been over-stressed 22 percent. That Nutech analysis was based on allowable stresses specified in a design code that was different than the code specified in the licensing basis for the crane. The correct design code for the crane is CMMA-1975. Utilizing that design code allowable stress (17.6 ksi) and re-performing the calculations demonstrated that the girders would have been approximately 50 percent over-stressed. It does not appear that this was considered in the NRC evaluation of the acceptability of the minor repair classification.

5. The 1974 analysis of the bridge girders indicates a two percent overstress condition during an OBE considering only static loads. This over-stress condition is documented in the inspection report, but there is no documentation of the basis for the acceptability of this over-stress condition. In addition, there is no analysis of stresses in the trolly for the OBE or SSE load cases

6. During a crane inspection conducted by licensee representatives, five deficiencies in the crane were identified as needing correction. The licensee initiated a corrective action document, but only corrected one of the deficiencies and closed the corrective action document as acceptable. There is no documented justification for not correcting the remaining four deficiencies. This was a violation of the licensee's corrective action program, but was not documented as a violation in the inspection report.

OTHER ISSUES INVOLVING CRANE ADEQUATELY ADDRESSED BY THE NRC

1. There are no safety lugs on the crane bridge rails as described in the UFSAR. This issue was acceptably dispositioned in the inspection report as a non-cited violation.

2. The crane design does not include an inching motor as described in the UFSAR. This issue was acceptably dispositioned in the inspection report as a non-cited violation.

3. A 1996 safety evaluation regarding the revision of the UFSAR addressing provisions for preventing overloading the crane concluded that an unreviewed safety question did not exist. That evaluation was inadequate in that it was based on inaccurate information regarding the crane. This issue was acceptably dispositioned in the inspection report as a non-cited violation.

August 13, 2001

EA-01-209

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company, LLC
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: NRC INSPECTION REPORT 07200037/2001-002(DNMS); DRESDEN - PREPARATIONS FOR SPENT FUEL LOADING INTO DRY STORAGE CASKS

Dear Mr. Kingsley:

On June 22, 2001, the NRC completed a special inspection of preparations for fuel loading from the Unit 2 spent fuel pool at the Dresden Station into the Holtec dry fuel cask system. The enclosed report presents the results of the inspection. These results were discussed regularly with members of your staff as the inspection progressed, and a special exit meeting was conducted at the site on June 7. Additional inspection activities relating to Cask Transfer Facility issues were conducted via a meeting in the Region III office on June 18, and a final exit discussion was held on June 22.

This inspection involved a number of complex evaluations of design, engineering, and operability issues. The information which we examined spanned more than 25 years, from 1974 to the present day. This was necessitated, in part, by the licensee's decision to re-establish a 1976 licensing basis for the Unit 2/3 reactor building crane as a single-failure-proof crane, so that heavy loads could be handled with the Unit 2 reactor in power operations. Multiple elements of your organization, as well as multiple contractors, were involved in various aspects of the process. Two meetings were conducted with representatives from your staff during the inspection for the purpose of ensuring that the NRC received clear statements of the licensee's positions on various issues. These meetings are addressed in the enclosed report. In addition, the NRC Office of Nuclear Reactor Regulation (NRR) and the Spent Fuel Project Office (SFPO) in the Office of Nuclear Material Safety and Safeguards were consulted regarding some issues. The assessments provided by these offices were considered in arriving at an agency conclusion regarding the acceptability of the approaches taken by your staff at the Dresden Station in resolving the various issues. The conclusions of NRR and the SFPO are referenced in this inspection report.

During our inspection, certain of your activities were determined to be in noncompliance with NRC requirements. These involved design aspects of heavy load handling equipment, as described in the enclosed report. Because of their low direct safety significance, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. While of low safety significance, the violations are of concern because they reflect failure over a long period of time to recognize and resolve discrepancies between heavy load handling equipment in the facility, and associated descriptions in the Updated Final

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Safety Analysis Report, and failure to identify the discrepancies as part of a specific effort during 2001 to "restore" the subject equipment to an NRC-approved licensing basis as single-failure-proof. If you deny these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Dresden Station.

A number of other problems or concerns were identified in various areas, and not all of the issues or concerns were resolved at the conclusion of the inspection; however, the NRC ultimately concluded that your staff's preparations for fuel loading were acceptable. The inspection activity itself appeared to prompt certain reviews and technical evaluations by your staff and/or contractors, as well as result in *post facto* analyses, which should have been accomplished without NRC prompting as part of your staff's comprehensive assessment of the readiness, both from an equipment and human performance perspective, to safely load spent fuel into dry storage casks and transport those casks to the storage pad. Some problems emerged which your organization and/or other licensees had previously encountered and which, therefore, should have been avoided. These problems indicated that your dry cask loading and associated heavy load handling activities were not subject to an appropriate level of planning and focused attention, and also reflected that an appropriate degree of coordination among the several responsible parties did not occur.

Regarding the overall issue of seismic qualification, the NRC has not determined that the current heavy load handling facilities are acceptable for the long term. Deviations from generally applicable standards exist with regard to the Unit 2/3 reactor building crane and reactor building superstructure. Detailed analyses of record have not addressed the capability to prevent a full spent fuel cask from dropping during an earthquake. Some non-record analyses indicate that some building structural elements could potentially fail due to overstress under certain conditions. The potential consequences of dropping a full cask have not been analyzed. While the NRC determined that there are not any issues that would preclude you from safely using the Holtec dry fuel cask system to load, handle, or transport spent fuel in the near term, the NRC staff is in the process of determining whether to impose new or different requirements to address the seismic qualification of the reactor building crane and superstructure. In evaluating this issue, we will be using the "backfit" process. We will communicate the results of the NRC review of this issue separately.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available **electronically** for public inspection in the NRC Public Document Room **or** from the *Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from* the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

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

Sincerely,

/RA by M. Dapas acting for/

Cynthia D. Pederson, Director
Division of Nuclear Materials Safety

Docket Nos. 05000010; 05000037;
05000249; 07200037

Enclosures: Inspection Report 0720037/2001-002(DNMS)
Safety Evaluation Report by SFPO - June 15, 2001
Report of Meeting - May 23, 200

cc w/encls: R. P. Tuetken, General Manager,
Decommissioning Projects and Services
K. A. Ainger, Director of Dresden 1 and
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000010; 05000237; 05000249; 07200037
Report No:	07200037/2001-002(DNMS)
Licensee:	Exelon Nuclear
Facility:	Dresden Station
Location:	6500 N. Dresden Road Morris, IL 60450
Dates:	April 4 - June 22, 2001
Inspectors:	Ross B. Landsman, Project Engineer Roy J. Leemon, Reactor Decommissioning Inspector
Approved by:	Bruce L. Jorgensen, Chief Decommissioning Branch Division of Nuclear Materials Safety

EXECUTIVE SUMMARY

Dresden Station
NRC Inspection Report 07200037/2001-002(DNMS)

This was a special inspection to review licensee preparedness to load spent fuel from the Unit 2 spent fuel pool at the Dresden Station into the Holtec dry fuel cask system. Unresolved items from the previous NRC inspection were examined and the licensee's approach to resolving these items was determined to be acceptable.

During this inspection, a number of new issues were identified primarily relating to the design, maintenance, and operation of equipment for the handling of heavy loads. The NRC Office of Nuclear Reactor Regulation (NRR) and the Spent Fuel Project Office (SFPO) in the Office of Nuclear Material Safety and Safeguards (NMSS) were both consulted and each provided interpretations and analyses to support the inspection.

A wide spectrum of licensee records were examined, spanning a period of more than 25 years. These included licensing submittals, technical evaluations and reports, maintenance and repair records, engineering evaluations and supporting calculations, quality assurance documents, and regulatory bases including the license, Updated Final Safety Analysis Report (UFSAR), and applicable codes and standards. Numerous phone conferences between or among Region III, NRR, SFPO, and the licensee were conducted. On two occasions, inspection meetings were conducted in the NRC regional office to obtain clear information on licensee activities and positions.

The NRC staff identified examples of the licensee's failure to update the UFSAR, as required by 10 CFR 50.71(e), to completely and accurately describe elements of the Unit 2/3 reactor building crane, and failure to provide an adequate basis for a determination that a change to the facility did not constitute an Unreviewed Safety Question as required by 10 CFR 50.59. The examples were of concern because of their long duration and the fact that opportunities were missed to identify and correct them. If reviews had been performed under 10 CFR 50.59 for the changes which should have been made to the UFSAR, they would have provided an opportunity to identify issues which degraded safety features of the crane. In addition, the issues were not identified by the licensee as part of a focused restoration of the crane in 2001 to an NRC-approved licensing basis as a single-failure-proof crane.

Ultimately, licensee preparations to begin loading spent fuel from the Unit 2 and Unit 3 spent fuel pools into dry cask storage proved to be adequate. Some reviews, evaluations, and quality verifications were apparently conducted by the licensee or its contractors after the NRC staff asked questions regarding the acceptability of issues. Overall, the licensee did not focus effectively on ensuring that all requirements were met. Further, problems occurred which had been previously encountered across the dry cask storage industry. Consequently, these problems were avoidable. These findings indicated that the licensee had not focused appropriately on preparations for the cask loading campaign, and the several parties with responsibilities for aspects of the project did not appropriately coordinate their activities. In addition, the licensee's process for capturing and acting upon vendor recommendations resulting from the annual crane inspection, was not effective.

Regarding the overall issue of seismic qualification, the NRC has not determined that the current heavy load handling facilities are acceptable for the long term. Deviations from generally applicable standards exist with regard to the Unit 2/3 crane and the reactor building susperstructure. Detailed analyses of record have not addressed the capability to prevent a full spent fuel cask from dropping during an earthquake. Some non-record analyses indicate that some building structural elements could potentially fail due to overstress under certain conditions. The potential consequences of dropping a full cask have not been analyzed. While the NRC determined that there are not any issues that would preclude the licensee from safely using the Holtec dry fuel cask system to load, handle, or transport spent fuel in the near term, the NRC staff is in the process of determining whether to impose new or different requirements to address the seismic qualification of the reactor building crane and superstructure. In evaluating this issue, the NRC will be using the "backfit" process, and will communicate the results of the NRC review of this issue separately.

Report Details[1]

1.0 General

This special inspection examined design, fabrication, and testing of equipment for use in removing spent fuel from the Unit 2 fuel pool into components of the Holtec dry fuel cask system. The Unit 2/3 reactor building crane and the Cask Transfer Facility were examined in detail. Previously identified unresolved items were examined and the licensee's resolution of the associated issues was determined to be acceptable.

2.0 Handling of Heavy Loads

2.1 Unit 2/3 Reactor Building Crane

2.1.1 Licensing Basis for the Dresden Crane

a. Inspection Scope

The Unit 2/3 reactor building crane was being relied upon by the licensee as a single-failure-proof crane, which the licensee stated was supported by the plant licensing basis. The Office of Nuclear Reactor Regulation (NRR) was requested to determine the licensing basis for the Dresden Unit 2/3 reactor building crane.

b. Observations and Findings

The Office of Nuclear Reactor Regulation reviewed the current licensing basis documents for the Dresden Reactor Building Crane, noting that Dresden Units 2 and 3 are Systematic Evaluation Plants which pre-dated the General Design Criteria, and pre-dated NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."

By letter to the licensee dated June 3, 1976, the NRC staff issued Technical Specification License Amendments Nos. 19 and 22 approving changes governing the operation and surveillance of the modified crane handling system for Dresden Units 2 and 3. In the safety evaluation report (SER), the NRC staff concluded that the Reactor Building Crane met the intent of the NRC requirements and was acceptable for handling spent fuel casks weighing up to 100 tons. In its application, the licensee (Commonwealth Edison, now Exelon) committed to perform load tests in accordance with American National Standards Institute (ANSI) B30.2, "Overhead and Gantry Cranes," at 125 percent of design-rated load in the event the hoist system should be extensively repaired or altered.

By letter to the licensee dated July 11, 1983, the NRC staff approved the licensee's Phase I heavy loads program in accordance with NUREG-0612, Phase I. In the safety evaluation, the staff concluded that the licensee had provided for the crane to be tested and operated in accordance with ANSI B30.2.

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[1]NOTE: A list of acronyms used in this report is included at the end of these Report Details.

By letter to the licensee dated June 28, 1984, the NRC staff approved, via draft-technical evaluation report, the licensee's heavy loads program under NUREG-0612, Phase II, endorsing the crane as a single-failure-proof crane provided that the licensee assured that no single failure would occur in the crane electric power and control systems, and provided that the license evaluated all load attachment points to single-failure criteria. However, in Generic Letter 85-11, "Completion of Phase II of Heavy loads at Nuclear Power Plants, NUREG-0612," the NRC closed out the need for licensees to satisfy NUREG-0612, Phase II. In the generic letter, the licensee was noted as taking credit for its single-failure-proof crane.

Because the "crane industry," as of 1976, had not yet developed codes or standards that adequately covered the design, operation, and testing for a single-failure-proof crane, the NRC staff developed a position statement to provide a consistent basis for reviewing overhead handling systems. This statement was Auxiliary and Power Conversion Systems Branch (APCSB) Technical Position 9-1, also referred to as Branch Technical Position (BTP) 9-1. In NUREG-0554 and BTP 9-1, the NRC states that a single-failure-proof crane is designed not to drop a load up to the maximum critical load in the event of the failure of any single component of the crane.

c. Conclusions

The licensing basis for the Dresden Unit 2/3 reactor building crane was established by the 1976 licensing action for moving loads up to 100 tons. The additional NRC correspondence, i.e., letters to the licensee dated July 11, 1983, and June 28, 1984, as well as Generic Letter 85-11, has resulted in the licensee taking credit for a single-failure-proof crane.

2.1.2 Review of Previously Identified Items

a. Inspection Scope

Licensee actions, responses, and clarifications regarding an Unresolved Item (URI 07200037/2001-001-001(DNMS)) were examined. The Unresolved Item (URI) consisted of a number of individual elements. These elements are addressed in the following section in the same sequence as originally documented in NRC Inspection Report 07200037/2001-001.

b. Observations and Findings

(1) UFSAR Commitments

(a) Safety Lugs

During the previous inspection, the inspectors raised a question regarding whether safety lugs were installed on the Unit 2/3 crane trolley and bridge rails. The Updated Final Safety Analysis Report (UFSAR), in Sections 6.2.3.2.1 and 9.1.4.2.2, describes provisions made (safety lugs) to ensure the Unit 2/3 reactor building crane trolley and bridge do not become dislodged during an earthquake. During this inspection, the inspectors were able to verify that safety lugs were installed on the trolley. However, the inspectors could not determine whether safety lugs were installed on the bridge rails. The licensee subsequently determined that the bridge did not have the specified safety lugs. They had apparently never been installed. The licensee provided the inspectors with a 1998 analysis of reactor building superstructure performance, which indicated that the walls supporting the crane rails would not experience any significant displacement during a seismic event. From this analysis, the licensee concluded that safety lugs are not required on the bridge rails.

Title 10 CFR 50.71(e) requires, in part, that the licensee periodically update the Final Safety Report (FSAR) to assure that the information in it contains the latest material developed. Paragraph (3)(i) of this regulation specifies that the first revision of the original FSAR was to be submitted within 24 months of July 22, 1980, or the date of license issuance, whichever is later. Title 10 CFR 50.71(e)(4) requires that subsequent revisions be filed annually or 6 months after each refueling outage, provided that the interval between refueling outages does not exceed 24 months. The failure to update the UFSAR from 1982 until the date of this inspection to reflect that there are no safety lugs on the crane bridge rails is a violation of 10 CFR 50.71(e). This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 07200037/2001-002-01). This violation is in the licensee's corrective action program as Condition Report (CR) No. D2001-03331. Additional violations of 10 CFR 50.71(e) are described in subsequent sections of this report.

(b) Wire-Rope Safety Factor

During the previous inspection, the inspectors raised a question regarding the safety factor for the crane wire-rope. The NRC's current guidance for crane cables is contained in NUREG-0554 and NUREG-0612, which were issued after Dresden License Amendment No. 22 for Unit 2 and Amendment No. 19 for Unit 3, and recommend a safety factor of 10 to 1. Based on a review of the licensing basis, the NRC staff determined that a safety factor of 10 to 1 is not a requirement for Dresden Units 2 and 3. It appeared the wire-rope on the Dresden Unit 2/3 reactor building crane had a safety factor of 8 to 1, per the UFSAR, Table 9.1-3, but the inspectors learned that it actually had a factor of 7.798 to 1.

The NRC staff determined that the NRC wrote to the licensee on January 30, 1976, stating the wire rope safety factors were not acceptable, and requesting a proposed inspection/replacement program for the wire rope. The licensee responded in a letter dated March 2, 1976, that the ropes would be inspected and if required, replaced to assure compliance.

On June 3, 1976, in License Amendment No. 22 for Unit 2 and License Amendment No. 19 for Unit 3, the NRC staff accepted the wire-rope static safety factor of 7.798 to 1 and the lead line safety factor of 6.564 to 1. To compensate for the reduced factor, the staff incorporated surveillance and action requirements in the Technical Specifications. Specifically, inspection requirements were imposed in accordance with ANSI B30.2.

Through the years, the licensee has been replacing the rope with like-for-like. The licensee is currently considering providing an additional safety enhancement by replacing the rope the next time with a 10 to 1 margin rope. The failure to update the UFSAR from 1982 until the date of this inspection to state the actual safety factor of 7.798 to 1 is a violation of 10 CFR 50.71(e). This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 07200037/2001-002-02). This violation is in the licensee's corrective action program as CR Nos. D2001-00834 and D2001-00909.

(c) Overload Protection

An issue relating to the apparent lack of overload protection on the Unit 2/3 reactor building crane hoist was identified during the previous NRC inspection. The initial licensee submittal in support of Amendment No. 22 for Unit 2 and Amendment No. 19 for Unit 3, Dresden Special Report No. 41, stated that a load sensing readout with high and low limit cut-offs would be provided as an overload protection feature. Section 9.1.4.2.2 of the UFSAR states a digital-type weight indicator for the main hoist is provided and that when the weight to be lifted is above the setpoint on the weight indicator, the control circuit for the slow speed motor will prevent its operation and the main hoist brakes will set.

Based on a review of the history of the system since initial installation of the load cell in 1976, the inspectors noted that the load cell was routinely out-of-service because of repeated problems with locking up the hoist. The licensee routinely installed electrical jumpers, thereby bypassing the "restricted mode" limitation in Technical Specification 3.10(F)1. Use of the crane for cask handling with the load cell bypassed was outside the licensing basis.

Trouble with the digital load limit setpoint was so common that the licensee, in Dresden Fuel-Handling Procedure (DFP) 0800-20, "Operation of 2/3 Reactor Building 125/9 Ton Crane," described how to jumper out the load cell signal in order to use the crane. The procedure also directed use of Dresden Administrative Procedure (DAP) 07-04, "Control of Temporary System Alteration Procedure," which specified a specific time frame that the temporary alteration may remain installed. The inspectors determined that the load cell was out-of-service for an undocumented number of years. The licensee provided the inspectors with an operability determination dated December 13, 1991, which addressed operability of the load cell and indicated that it had been inoperable for many years.

The licensee lifted at least 68 fuel casks between 1976 and 1984. The inspectors did not identify, nor was the licensee able to provide, plant records to indicate the status of the load cell during that time frame. Consequently, it appears likely that the load cell and its associated overload protective feature were out-of-service, and therefore inoperable, during these heavy load lifts. This lack of digital load indication and associated overload protection reduced the effectiveness of the defense-in-depth design approach to ensuring the crane was not overloaded. This condition may have resulted in overloading the crane when the weight of a lifted load was determined via calculation instead of through direct measurement.

During 1996, while re-baselining the UFSAR to the current design requirements for the reactor building crane, the December 13, 1991 operability determination was re-examined by the licensee. As documented in this operability determination, the licensee justified crane operability based upon the crane's having another load limiting device, specifically, the "over-torque limit." However, the licensee could not explain to the inspectors what part of the crane actually constituted this "over-torque limit" protective feature. To meet the original intent of the digital weight indicator function, the licensee recommended in the 1991 operability determination document that the indicator be restored to operation. As a temporary alternative, engineering staff recommended revising station procedures to add the requirement of stationing additional supervisory personnel during crane operation to ensure load hang ups did not occur. The inspectors noted that Dresden Maintenance Procedure (DMP) 5800-18, "Load Handling of Heavy Loads and Lifting Devices," Revision 07, still contains this statement ten years later. In 2001, the licensee installed modifications and conducted post-modification testing to re-establish operability of the load cell and its associated overload protective feature.

A 10 CFR 50.59 safety evaluation was conducted in 1996 to revise UFSAR Section 9.1.4.2.2 to include the statement, "As an alternative to the digital load limiter, station procedures require supervising personnel to ensure load hang ups do not occur during reactor building crane operation." The 1991 operability determination was used by the licensee in the 1996 safety evaluation as a basis for concluding that no unreviewed safety question existed. This meant that the licensee was relying on the unverified "over-torque limit" configuration.

Title 10 CFR 50.59, "Changes, Tests and Experiments," permits, in part, licensees to make changes to the facility as described in the UFSAR without Commission approval provided the changes meet certain conditions. The revision of this regulation in effect in 1996 included in those provisions that the change not involve an unreviewed safety question. Further, it required that licensees maintain records of changes to the facility and that these records include a written safety evaluation which provides the bases for the determination that the changes did not involve an unreviewed safety question. Reliance on the "over-torque limit," which does not exist on the crane, and on a procedure which was intended to be temporary, resulted in the licensee's 10 CFR 50.59 basis for accepting the "overload protective feature" configuration, being inadequate. The licensee's complete replacement of the load cell and overload protective feature via the 2001 modification using the design change process, obviated the need to rely on the 1996 50.59 safety evaluation. However, the inadequate 1996 safety evaluation was a violation of 10 CFR 50.59. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy

(NCV 07200027/2001-002-03). As described, no further corrective action is needed for this violation.

(d) Restricted Load Handling

During the previous inspection, the inspectors raised a question about use of a slow-speed or "inching" motor when operating the crane in the "restricted mode." Section 9.1.4.2.2 of the UFSAR described fuel cask handling above the 545-foot level of the reactor building as a restricted load evolution requiring raising and lowering of the load in slow-speed with the fast-speed circuitry disabled. The slow-speed motor malfunctioned in 1976, and since that time it has not been used despite operation of the crane in the "restricted mode" on many occasions. In the letter transmitting the NRC's safety evaluation report (SER) for License Amendments Nos. 19 and 22, dated June 3, 1976, the NRC described the reliance on the slow-speed motor as an item for which the licensee had requested a temporary waiver, until the end of August, 1976. Upon further inspection and based on discussion with the licensee, the NRC staff determined that this statement in the NRC's SER was in error.

During the NRC's amendment review process, the licensee requested an alternate approach, relying on a speed control circuit which could limit hoisting speed to five feet per minute, consistent with APCSB Technical Position 9-1 (BTP 9-1). As documented in the SER, the NRC approved the licensee's alternate approach. In the NRC's SER for the 1976 crane modification, the NRC noted that because of problems with the slow speed motor installation, the licensee could not complete the work to correct these problems before planned fuel shipments in June 1976. The licensee's proposal to modify the electrical circuit to limit the maximum attainable hoisting speed of the crane main hoist to five feet per minute was considered consistent with BTP 9-1. The NRC staff further concluded that the licensee's proposed electrical modifications did not degrade the capabilities of the crane from the standpoint of reliability or ability to withstand single failures. Thus, the "inching" motor was never part of the single-failure-proof licensing basis for the crane.

The inspectors noted that as of June 22, 2001, Section 9.1.4.2.2 of the UFSAR contained eight references to the crane having a slow speed (inching) motor. The failure to revise the UFSAR, from 1982 until the date of this inspection, to accurately describe the hoist speed limiting feature, including deletion of the slow-speed motor descriptions, is considered a violation of 10 CFR 50.71. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 07200037/2001-002-04). This violation is in the licensee's corrective action program as CR No. D2001-01376.

(2) Modifications to the Crane

A concern was identified during the previous NRC inspection regarding whether the crane had been "extensively repaired" after it was damaged in 1981 such that a load test would be required per ANSI B30.2. No load test had been conducted by the licensee.

In 1981, the crane bridge box girders were damaged by impact and compression during an over-hoisting event involving the strongback for the reactor vessel head. The crane girders are built-up box beams approximately 8 ½ feet deep, 2 feet wide, and 113 feet long. The damaged surfaces were confined to the lower portion of the inside webs (buckling) and the inside portion of the bottom flanges (bending) on both of the girders approximately 35 feet from one end. The Office of Nuclear Reactor Regulation (NRR) evaluated the licensee's actions to determine whether the crane had been extensively repaired.

The licensee's repairs consisted of cutting out the deformed section of each girder web and welding a new plate over the area, and welding a plate over the deformed section of each flange. The licensee hired a contractor, Nutech, to evaluate the extent of necessary repairs. As documented in their report, Nutech concluded that the extent of damage to the west crane bridge girder was such that it would be 22 percent overstressed at rated capacity (125 tons) if the girder was not repaired. Based on a 22 percent reduction in load carrying capacity with the damaged section unaccounted for, the NRR staff concluded that the crane would be capable of lifting 100 tons without overstress.

Nutech concluded that the repairs to the crane, if implemented per the specifications described in their report, would restore the capacity of the crane girders to their original design value; thus, the crane would be able to support a lift of 125 tons without appreciable overstress in all conditions, including seismic. The NRR staff considered this a conservative assessment since the damaged areas of the girder's bottom flanges were not removed, but instead covered, and the crane girders and other critical structural components were designed to carry 125 tons with acceptable safety factors. Additionally, based on their analysis, Nutech concurred with the licensee's conclusion that the repair work was not in the extensive repair or major alteration category as defined by the 1976 ANSI B30.2 code.

The NRC staff evaluated the damage to the crane, and the repairs performed, based on the information in the Nutech repair report and the information provided by the licensee in a meeting with NRC staff on May 23, 2001 (refer to enclosed meeting minutes). During the meeting with the licensee, the implications of the 1981 incident were discussed. The licensee stated that the Unit 2/3 reactor building crane was damaged in 1981, during a lift involving the reactor vessel head lift "strongback" (approximately a 10-ton load), when the strongback struck the underside of the crane bridge girders. The inside web and the bottom flange of each girder were deformed. The licensee presented information to support its view that the damage incurred and the repairs to restore the crane girders to "design" were not "extensive." This was an issue because ANSI/ASME B30.2 - 1967, to which the licensee was committed, specified a 125 percent rated load test be performed on "new, extensively repaired or altered" cranes.

Calculations were presented by the licensee to define the "overstress" experienced by the girders. In that the licensing basis for the crane was Safety Class II, "non-seismically qualified," the licensee maintained that restoration of the crane was not mandatory by regulation. Rather, the restoration was to regain margin of safety which was established in beyond-licensing-basis calculations. Specifically, one girder would experience 22 percent overstress in the event of an Operational Basis Earthquake (OBE) with full rated crane load. Thus, the margin represented in meeting a 0.6 Fy (minimum yield strength of the material) criterion was reduced, but all stresses were less than yield. An expert consultant's assessment of the damage and the repair was presented in support of the licensee's position that the repairs were not "extensive."

Information was also provided by the licensee to address the performance of the crane since the 1981 repair, which indicated that loads up to 125 tons had been lifted at least 42 times. No indication of stress or distortion has been observed in the repaired or adjacent sections, although the licensee acknowledged that no focused inspection has been performed with the specific intent of looking for such stress or distortion.

In addition, the NRC staff noted that the licensee has conducted a 50-ton load "dry run" performance test at power using an empty fuel cask. This dry run test was intended to verify that the crane, including its hoist and single-failure proof controls, were operating properly. This dry run also provided additional assurance to the NRC staff on the capabilities of the crane. Based on licensee activities and staff evaluations to date, the NRC staff concluded that additional load testing of the crane is not required to provide assurance of safety.

(3) Reactor Building Superstructure

Inspection report 07200037/2001-001 references a series of calculations from the 1960's forward which disclosed examples of various building structural members (roof trusses or columns) being overstressed, either with or without rated load assumed to be on the crane, and including or not including design-basis earthquake conditions. During this inspection, substantial additional reviews of the licensing basis for the reactor building, and of the expected performance of the building superstructure in seismic conditions, were conducted.

The Unit 2/3 reactor building is classified as a Seismic Category I structure in Section 3.2.1 of the UFSAR, and is designed to accommodate various load conditions and satisfy specific stress criteria described in Section 3.8.4. Section 3.8.4.1.3 states that the following load combinations will be used for Class I structures, with OBE and SSE defined as operational basis earthquake and safe shutdown earthquake, respectively.

- D (dead load + live load) + E (OBE load)

- D + E' (SSE load)

This section of the UFSAR also specifies the included live load as, "live loads expected to be present when the plant is operating." The licensee's definition of live load included the crane rated load for the "normal" load condition; however, it did not include the crane rated load for the OBE and SSE conditions. Consequently, the licensee's seismic analysis of record for the reactor building included the rated crane load for "normal" loads (winds, snow, etc.), but the analyses for design basis earthquake conditions (OBE and SSE) assumed no lifted load on the crane. The licensee addressed this at the meeting with the NRC on May 23.

The licensee identified the crane as Safety Class II (non-seismic), which the licensee indicated had been made known to the NRC and accepted by the NRC as part of the licensing basis. The licensee further stated that a commitment in Special Report 41, Supplement A, to perform analyses of the crane bridge girders for OBE and SSE conditions, had been met, with the result that the stress condition for the bridge girders with a suspended load of 125 tons was acceptable. The licensee also maintained that the crane trolley was not seismically qualified (refer to Section 2.1.3.b).

The licensee claimed that the reactor building superstructure had to be capable of accommodating the following specified loading combinations: normal loads, which included rated load of the crane; OBE without rated crane load; and SSE without rated crane load. The licensee stated that based on its analyses, the reactor building superstructure met all design allowables for these load combinations. The licensee indicated the NRC understood and approved this design approach within the licensing basis, even though BTP 9-1 included an expectation that the SSE with load would be analyzed. Separately, the licensee informed the NRC staff of the results of "beyond-design-basis" calculations for the building, considering SSE plus full suspended crane load (125 tons), as per BTP 9-1. According to the licensee, the results of these calculations were that all structural members met design allowables.

A long history of various seismic analyses was reviewed by the inspectors. Calculations in the early 1960's for the reactor building, without lifted load included, indicated that the stresses in the support girders, support columns, and members of the roof truss were above the yield stress for the SSE loading. In some of the roof truss connections, the loading exceeded the ultimate capacity of the connections. Nothing was documented outside of the calculations to indicate how these issues were resolved by the licensee.

In 1973, the licensee performed new calculations to evaluate the effects of the new, heavier single-failure-proof trolley without lifted load. These calculations showed that the columns were overstressed by up to 30 percent for the SSE loading using 0.9 Fy as the allowable stress. The roof trusses and vertical bracing were not evaluated. Nothing was documented outside of the calculations to indicate how the licensee resolved this issue.

In 1975, the licensee performed additional calculations for the columns and the vertical bracing to address the effects of the new trolley without lifted load. Modifications were designed to bring all elements within code allowable stresses. The modifications were subsequently not implemented. The Dresden calculation book carries the notation, "Project Canceled, calculation not approved," without further explanation.

In 1998, the reactor building superstructure framing was examined once again (calculation DRE98-0013) by the licensee as part of the dry cask project. The calculations showed that crane support girders, interior building columns, and roof truss members, had interaction coefficients above 1.0. The calculation results were accepted by the licensee based on probabilistic considerations that the earthquake would not occur during crane use, i.e., the crane is used only a small fraction of the time. This probabilistic rationale was not addressed in the UFSAR.

The licensee performed additional calculations in 1998 (DRE98-0020), which again indicated that some structural members would be overstressed. Given the small magnitude of overstress (3 percent), the licensee characterized the results as acceptable. The NRC has not determined that this characterization is acceptable (refer to Section 2.1.2.b(5).

Section 3.8.4.1.4 of the UFSAR states that stresses are to be maintained below the minimum yield point as a general case. Stresses may exceed the yield point in some elements, if the energy absorption capacity is shown to exceed the energy input, using the Limit-Design approach. When requested, the licensee could not show that an energy balance had been performed to establish whether absorption capacity exceeds energy input.

In an earthquake, particularly one causing building deformation, the load could also be subjected to lateral forces. The crane yoke for holding the spent fuel cask load is not equipped with positive latching capable of retaining the cask against significant lateral force (refer to paragraph 2.2.b under "ANSI N14.6"). This suggests that a cask could slide off the yoke and fall, even if the crane and building do not fail catastrophically. This potential scenario also has not been analyzed.

Ultimately, the NRC staff could not conclusively determine from the calculations reviewed whether the reactor building, the crane, and/or the load (a spent fuel cask containing 68 spent fuel assemblies) would fall during an earthquake. Considering the potential vulnerability of these facilities for heavy load handling, long-term acceptability of this equipment for handling large numbers of dry fuel storage casks is considered an Unresolved Item (URI 07200037/2001-002-05).

(4) Crane Inspections

During the previous NRC inspection, the inspectors noted that the crane manufacturer (Whiting Crane) had been conducting annual inspections of the Unit 2/3 reactor building crane, and had identified a number of discrepancies during these inspections. The inspectors also noted that these discrepancies were not being addressed by the licensee in a timely manner. In addition, there were a number of crane equipment failures/problems during the last Unit 3 outage. Affected equipment included the main hoist brake, the trolley brake, the equalizing bar circuitry, an inverter on the trolley, a motor exciter, and a trolley conductor shoe.

During the annual inspection by the crane vendor conducted before the Unit 3 outage, five equipment issues were identified. The licensee initiated a work action request (AR990093876) to address these issues. Only one item was addressed; however, the action request was closed. Correction of these items most likely would have prevented some of the crane equipment failures that occurred during the Unit 3 outage. Based on a historical review of work action requests associated with the crane, the inspectors noted that over the years numerous action requests have been canceled, with the justification that the repairs were not required. The inspectors concluded that the licensee's process for capturing and acting upon vendor recommendations, specific to the crane, was not effective. The licensee informed the inspectors that it is addressing this issue.

During this NRC inspection, repairs were performed for each of the four vendor-identified discrepancies from the latest vendor inspection, or the affected equipment was replaced as part of the digital control system modification. No previously-identified material discrepancies remained when the crane was used by fuel storage project personnel for heavy load handling.

(5) Meeting with Licensee

On May 23, 2001, licensee representatives met in the NRC Region III Office with NRC staff having inspection and oversight responsibilities for Independent Spent Fuel Storage Installation (ISFSI) activities at the Dresden site. The purpose of the meeting was to provide the licensee the opportunity to present clear position statements and supporting information regarding the examples previously classified as parts of the Unresolved Item - as detailed above. The meeting minutes, list of attendees, and the information package presented by the licensee are enclosed with this inspection report. No major inconsistencies were identified during review and verification of this information throughout the inspection. However, the licensee's presentation included a characterization that for seismic analyses, overstress conditions up to about ten percent overstress, as determined by calculation, are considered generally acceptable. This has not been determined by the NRC to be an acceptable position. Further NRC review is required to determine whether the licensee's characterization and its actual application in practice are acceptable. Pending the required additional NRC review, this matter is being classified as an Unresolved Item (URI 07200037/2001-002-06).

c. Conclusions

The issues previously identified in NRC Inspection Report 07200037/2001-001 are resolved to the extent that the NRC determined that the Unit 2/3 reactor building crane meets applicable regulatory and industry standards, and the crane is considered acceptable in the near-term for use in handling heavy loads (up to 100 tons) in support of the spent fuel dry cask storage project. Unresolved Item URI 07200037/2001-001-001(DNMS) is considered closed.

During this inspection, violations of 10 CFR 50.71(e) were identified. These relate to features or attributes of the Unit 2/3 reactor building crane which are not accurately described in updates to the UFSAR since 1982. Specifically, UFSAR Sections 6.2.3.2.1 and 9.1.4.2.2 describe safety lugs on the reactor building crane bridge rails which do not exist **(NCV 07200037/2001-002-01)**; Table 9.1-3 displays the safety factor for the crane wire rope as 8 to 1 when it is actually 7.798 to 1 **(NCV 07200037/2001-002-02)**; and Section 9.1.4.2.2 contains multiple references to a slow speed (inching) motor which does not exist **(NCV 07200037/2001-002-04)**. In addition, the licensee, in its Safety

Evaluation for a 1996 revision to UFSAR Section 9.1.4.2.2, regarding the crane digital load limiter, used a 1991 operability determination as a basis for concluding no unreviewed safety question existed. The 1991 operability determination relied on an "over-torque limit" which apparently did not exist, and on a procedure which was intended to be temporary. Consequently, the licensee's bases for concluding the UFSAR change did not constitute an unreviewed safety question was inadequate, contrary to the requirements of 10 CFR 50.59 **(NCV 07200037/2001-002-03)**.

While the NRC has concluded that the Unit 2/3 reactor building crane satisfies its licensing basis, the NRC has not made a determination regarding the long-term acceptability of the reactor building, the crane, and their interface, considering their apparent potential vulnerability to failure under seismic loads. If the NRC should determine that some licensee action is required to ensure long-term acceptability, this would constitute imposition of a new regulatory position on the licensee. The NRC uses a disciplined "backfit" process to assess the safety benefits of any proposed new requirements, and that process will be applied in this case. Pending a final determination regarding the long-term acceptability of the current facilities considering seismic vulnerability, this matter is being considered an Unresolved Item **(URI 07200037/2001-002-05)**.

In addition, further NRC review is required to evaluate the acceptability of the licensee's practice of accepting results in seismic stress analyses up to about 10 percent overstress, without further analyses or corrective action. This is considered an Unresolved Item **(URI 07200037/2001-002-06).**

2.1.3 Additional Crane Evaluations

a. Inspection Scope

During this inspection, the seismic qualification of Unit 2/3 reactor building crane itself was evaluated, and "dry runs" of the crane were observed.

b. Observations and Findings

Seismic Qualification

Dresden Special Report No. 41, dated November 8, 1974, stated that the entire crane trolley and existing bridge girders would be reviewed for the revised weight of a proposed new trolley, in conjunction with the lifted load requirements, to establish compliance with Crane Manufacturers Association of America (CMAA) permissible stress ranges. Design values for the Operational Basis Earthquake (OBE) were to be based on the American Institute of Steel Construction (AISC) Code requirements of 0.60 times the minimum yield strength of the material (Fy) and 0.90 Fy for the Design Basis Earthquake, which is the Safe Shutdown Earthquake (SSE). Before all the analyses were submitted, the NRC issued Branch Technical Position (BTP) APCSB 9-1 in April 1975. The BTP specified that a "single-failure-proof" crane used to handle heavy loads be classified as Seismic Category I and that it should be capable of retaining the maximum design load during an SSE. This meant that the design rated load plus operational and seismically-induced pendulum (swinging load) effects, should be factored into the design of the trolley, and they should be added with trolley weight, for the design of the bridge.

Dresden Special Report No. 41, Supplement A, dated June 3, 1975, stated that the Dresden Unit 2/3 reactor building crane was identified as Safety Class II in the plant operating license and it was not practical to consider reclassifying the crane as Seismic Class I. This would have required a new bridge and extensive modifications to the bridge track way (the reactor building superstructure). Supplement A further stated that the bridge and trolley would be analyzed with only static lifted loads considered. This meant that the licensee was not going to include seismically induced pendulum effects. While this approach was not consistent with BTP 9-1, it was ultimately accepted by the NRC.

The crane bridge girders were evaluated in 1974 for static lifted loads without any pendulum or swinging loads. Calculations indicated the girders were acceptable for the non-seismic loading conditions for the new trolley additional weight of 25,000 pounds. The calculations also indicated that the girders were acceptable for SSE static loading. However, the calculations indicated that the girders were two percent overstressed for OBE static loading which the licensee considered acceptable. While the licensee did not commit to analyze the crane bridge for pendulum loads, the licensee did perform calculations which indicated the bridge girders would be overstressed by six percent relative to AISC allowables (0.6 Fy for an OBE) with the added pendulum loads. The licensee apparently did not fulfill a commitment to analyze the trolley, considering static lifted loads. As of June 22, 2001, the licensee was not able to locate any evaluations for the seismic stresses on the trolley as committed to in Supplement A. However, as noted above, the trolley has no seismic qualification and was accepted by the NRC knowing that was the case.

Dry Runs

When the crane was upgraded in early 2001, a new digital load indication and overload protection system was installed. During initial testing, the licensee used a test load of 8000 pounds. The upper load limit trip was initially set according to drawing 12E-6510 to 110 percent of the 125 ton rated load. The inspectors questioned the licensee regarding whether the trip setpoint should be 100 percent of the rated load value. The licensee subsequently determined that the setpoint was not correct and adjusted it to 100 percent of the crane rated load.

During various dry runs, the inspectors noted that the new load cell was indicating a negative load with a weight on the main hook. The licensee had informed the inspectors that the load cell was accurate to within 50 pounds, or to within one percent of full scale. The inspectors determined, based on discussions with various crane operators, that the operators were not "zeroing out" the crane's load cell correctly. The licensee provided the operators with additional training.

The inspectors also noticed that the load cell indicator was reading 5200 pounds with only the Hi-Trac lifting yoke on the hook. This yoke has a calibrated weight of 3400 pounds. The total lifted weight of a full cask had been calculated as 199,394 pounds, using a 3400 pound yoke weight. The inspectors noted that if the yoke weight is actually 5200 pounds, the total lifted weight of a full cask will exceed 100 tons. The licensee stopped work to calibrate the new load cell system and decided that no cask lifts would be conducted until this issue was resolved. After the licensee re-calibrated the load cell and considered the indication problem resolved, the actual loaded cask (with fuel) indicated 215,000 pounds on the digital readout. This exceeded the estimated calculated weight of 192,000 pounds. Subsequently, on June 20, 2001, the licensee conducted tests with a calibrated dynamometer that indicated the load cell readings were up to 40 percent greater than actual load. The load cell was calibrated again. The inspectors noted that the licensee did not follow standard industry practice in calibrating the load cell at increments of 0, 50, 100, 50, and 0 percent of full scale to account for hysteresis losses, creep, and non-linearity.

The inspectors were informed that the signal which drives the load cell indicator also feeds into the overload protection feature. The problems encountered with the indicator caused the inspectors to question whether the overload protection feature had been properly installed and tested to demonstrate functionality, and therefore, could be relied on to perform its function. Regulations in 10 CFR 72.162 require that the licensee establish a test program to ensure that all testing, required to demonstrate that components will perform satisfactorily in service, is performed. The inspectors questioned the acceptability of the licensee's load cell calibration and testing methodology in demonstrating that the load cell and associated overload protection feature are functioning properly. The licensing basis for the crane as a single-failure-proof crane incorporates reliance on the overload protection feature. The licensee has concluded that based on testing conducted to date, the overload protective feature is operable and the single-failure-proof qualification of the crane is not in question. However, additional NRC inspection is necessary to verify that the overload protective feature will actuate at the proper setpoint. Pending the additional NRC evaluation, this is considered an Unresolved Item (URI 07200027/2001-002-07).

Other problems were encountered during the dry runs. For example, cable-stretch was not properly accounted for during the initial handling of an empty (approximately 50-ton) Hi-Trac cask. As a result, the maximum lift height limit was reached before the cask was actually six inches above the floor at the refueling elevation. This prevented placing the empty cask on the decontamination pad because it would not clear the surrounding curb. The same cable-stretch phenomenon had been previously encountered during handling of a Hi-Star cask in Dresden Unit 1 during 2000. Thus, this was an avoidable problem.

NRC Bulletin 96-02

When NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," was developed, the NRC policy in the area of handling heavy loads was set forth in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and the Standard Review Plan (SRP). Branch Technical Position 9-1 was not part of the policy at the time, having been replaced previously in 1979 and 1980 by the two NUREGs. The NRC staff has determined that the provisions of the BTP were subsumed into the NUREGs.

In its response to NRC Bulletin 96-02, dated May 13, 1996, the licensee stated that it had no plans for any movement of dry storage casks over safety-related equipment with the reactor at power. However, if such movements were planned in the future, the licensee committed to demonstrate the capability of safe plant shut-down in the presence of the radiological source term that might result from a breach of the dry storage cask, damage to the fuel, and damage to safety-related equipment as a result of a cask load drop inside the reactor facility.

This commitment was incorporated into site procedure DMP 5800-18, "Load Handling of Heavy Loads and Lifting Devices," Revision 7. Upon learning this, the inspectors requested to see the licensee's evaluation of the ability to safely shut down the plant. After being informed that the evaluation was forthcoming, the inspectors were instead given "Commitment Change Evaluation Form, No. 2001-002," dated subsequent to the inspector's request, that revised the original licensee commitment. The rationale provided by the licensee for modifying its original commitment was that the reactor building crane is single-failure-proof; dropping a cask is a non-credible event, so an evaluation of safe shut down is not required.

c. Conclusions

Despite the crane being at variance from some of the established criteria, the Dresden Unit 2/3 reactor building crane was accepted as a single-failure-proof crane in 1976 by the NRC as documented in its June 3, 1976 SER for License Amendments Nos. 19 and 22. Subsequently, as the "heavy loads" criteria evolved, the licensee committed, in its response to NRC Bulletin 96-02, to forego specified activities, or to perform evaluations showing the plant could be safely shut down. The licensee subsequently canceled these commitments, relying on the "single-failure-proof" qualification of the crane as an acceptable basis. The NRC staff reviewed the basis for the licensee's commitment change and discussed the licensee's approach during an April 27, 2001 conference call and the May 23 meeting. The NRC concluded the licensee's actions were acceptable.

The licensee encountered some problems during dry runs conducted to test and demonstrate equipment readiness for loading fuel from the Unit 2 or Unit 3 spent fuel pools. These included both equipment and personnel performance challenges. Actions to correct the problems were successfully implemented. The NRC staff had not resolved their concerns regarding demonstration of the proper performance of the new load cell and associated overload protection feature; this will be reviewed further and is considered an Unresolved Item **(URI 07200037/2001-002-07)**.

2.2 Cask Transfer Facility (CTF)

a. Inspection Scope

Removing fuel from the Unit 2/3 reactor building required the use of another heavy load device, the Cask-Transfer Facility (CTF), designed and supplied by Holtec. The inspection included an evaluation of various engineering and design attributes, including whether the design of the CTF satisfied all requirements in accordance with the Certificate of Compliance (CoC). In addition, fabrication of the CTF was examined.

b. Observations and Findings

Single-Failure-Proof Design

The concept of a CTF initially appeared in the Hi-Storm Topical Safety Analysis Report (TSAR). As the NRC considered the concept, additional detail regarding the design of a CTF was needed if it was to be included in the CoC for the Hi-Storm system. In response, Holtec expanded Section 2.3.3.1 of the Hi-Storm TSAR to include design features that any CTF would have to meet. However, the CoC does not address the CTF.

Section 2.3.3.1 of the TSAR addresses single-failure-proof design in several places: Section A.iii defines the CTF structure as the "stationary, anchored" portion of the CTF; Section A.iii also defines Single-Failure-Proof as a device wherein all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria in NUREG-0612. This definition was refined by Section 203.12 of the licensee's CTF purchase specification, which required the CTF to be able to withstand a failure of any one part or component without resulting in an uncontrolled lowering of the load.

Section 2.3.3.1.C.ii of the TSAR states that three main portions of the CTF shall comply with NUREG-0612 guidance: the connector bracket, Hi-Trac lifter, and MPC lifter. These three portions are defined rather broadly, but clearly enough to permit distinction between these portions and the stationary CTF structure itself. The licensee, in its purchase specification (Section 203.6), stated that the CTF structure includes the fixed-position vertical structural members that support the cask.

Section 5.1.6 of NUREG-0612 invokes NUREG-0554 for the design of cranes. It defines single-failure-proof as a system that is designed so that a single failure will not result in the loss of the system to safely retain the load. By issuing the CoC, the NRC approved TSAR Section 2.3.3.1; however, Section 3.5 of Appendix B to the CoC specified that the Hi-Trac lifter and the MPC lifting device must be designed, fabricated, operated, tested, inspected, and maintained in accordance with NUREG-0612. The NUREG-0612 requirements include the requirements of NUREG-0554, which it invokes. These appear to apply to the Hi-Trac lift system, including all load bearing components and parts, which includes the lift platform itself, and not just the cask lifting yoke pieces and the screw jacks (the Hi-Trac lifter).

The licensee and the CTF vendor did not consider the CTF to be a crane. Neither did they classify the CTF as a "special lifting device." Rather, the CTF was designed as a composite assembly, referred to as a "stationary jacking tower." Due to the complexity of the composite approach, the applicable regulatory requirements could not be readily determined. The NRC Spent Fuel Project Office (SFPO) was requested to analyze the design of the CTF to determine whether it satisfies the requirements of the cask CoC. Further, SFPO staff were requested to examine CTF testing to determine whether applicable requirements were met. By memorandum dated June 15, 2001, SFPO management reported the results of the staff's safety evaluation (enclosed with this report), concluding that the Dresden CTF satisfies all design bases in the CoC, and that CTF component and system testing met applicable requirements and was acceptable.

<u>Cask Transfer Facility Fabrication</u>

A joint audit of Holtec's implementation of its quality assurance (QA) program at its subcontractor (Omni Fabricators) was performed on May 22-26, 2000. The audit team identified the following issues:

- Omni was not on the approved supplier list.

- Omni did not have a quality assurance program that conformed to the requirements of Parts 71 or 72, nor to the requirements of 10 CFR 50, Appendix B.

- Fabrication process controls were less than effectively implemented. Weaknesses were identified with special processes (i.e., welding), work process control documents, and documented instructions and procedures for controlling activities that have an impact on quality.

- Holtec, consistent with its quality program, had extended its quality program to Omni, but Holtec's oversight and implementation of its quality program at Omni's facility were considered to be ineffective at the time of the audit.

- Weld procedures still needed to be qualified, as did welders.

Based on the audit findings, a follow up audit was conducted August 29-31, 2000. This "limited-scope" audit assessed the adequacy of corrective action implementation in response to the five findings documented in the May audit report. The corrective actions included: revising the incorrect procedures and records identified in the May audit; developing procedures that the audit team found missing; and providing training to personnel on the new procedures. Corrective actions were narrowly scoped, only targeted past work, and did not address programmatic issues. The finding that Holtec had not been effectively implementing its quality assurance program at Omni was not re-examined. The audit team did not observe any work activity at Omni. The audit report stated the number of qualified welders was very limited; weld procedures still needed to be qualified, as did welders; and employee experience at Omni with demands of a quality program mandated by the NRC, was essentially non-existent. The audit findings did not lead to a work stoppage. Instead, both the licensee and Holtec assigned full-time quality verification staff to the Omni shop. At times, five quality control personnel were auditing or observing the work of four welders.

Subsequently, condition reports were written at the Dresden site for items manufactured at Omni, which included the transfer cask (Hi-Trac) with its various ancillary equipment, and the CTF. Condition reports identified: incomplete and/or inaccurate CoC information, missing documentation, fit up problems with various pieces of equipment, and corrective actions that demonstrated lack of compliance with the quality program. However, the licensee apparently did not examine the actual fabrication records for compliance.

The inspectors requested various welding and inspection records for specific welds on the CTF. The available records consisted of weld data and inspection data assembled cumulatively by the Omni QA Manager, in weld groups, according to the size of welds. The date recorded for the cumulative data was the date that the last welding activity had been conducted for a particular group of welds. This resulted in the formal record documenting that all of the welds for the CTF had been conducted properly and visually inspected by a certified quality control (QC) inspector being signed off by the QA Manager on the same day. Holtec's procedure for recording weld inspection results HSP-211, "Visual Weld Examination," provides that the QC inspector may compile a single Examination Report Form, but does not address (neither provides for, nor prohibits) consolidation of data by the QA Manager. The reason given for the data consolidation practice was the poor condition of the shop traveler paperwork and drawings after being exposed to the shop environment (dirt, grease, etc.). When the original data was requested by the inspectors, it could not be provided by the licensee because it had been discarded.

The inspectors raised a concern that because individual weld data is not available, a condition adverse to quality involving a specific welder cannot be fully evaluated. For example, nonconformance Report No. 46, dated September 12, 2000, identified that Welder K made 5/16 inch stitch welds incorrectly. The faulty weld was repaired; however, the inspectors noted that other similar welds may have been made which are now inaccessible. Specific information regarding welder identity and the fabrication sequence does not exist; thus, specific documentary evidence regarding the quality of other specific welds is not available for the inspectors' review. This issue was discussed extensively with the licensee during the June 18 meeting.

As noted above, licensee and vendor oversight of fabrication activities at Omni involved multiple, full-time QC inspectors being assigned to examine every aspect of the job. Weekly reports by the Holtec Users Group inspector constitute a generalized record of activities inspected, and indicate all the work was being performed by qualified and certified welders and Non-Destructive Examination (NDE) technicians, using qualified procedures, with no specific details given. As a result of this approach, the licensee was confident of fabrication quality. Licensee representatives at the June 18 meeting indicated specifically that all the welds similar to the one which Welder K had completed incorrectly had been inspected and all deficiencies corrected. Based on the information provided by the licensee, the NRC concluded that the final CTF welds are proper. More specifically, this is based on the licensee's assertion that all welds were inspected and identified discrepancies corrected; the documented results of QC inspector activities (weekly Holtec Users Group reports); and the QA managers' certification of the cumulative welding data. However, the inspectors and involved NRC staff did note that the adequacy of individual CTF welds cannot be verified based on records of quality verification specific to each individual weld, e.g., certified weld specification drawings.

ANSI N14.6

The CoC requires that the CTF be designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," which requires that "special lifting devices" satisfy the guidelines of ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More." This ANSI standard requires that (1) special lifting devices that require remote engagement with the shipping container be provided with lead-in guides and sufficient clearance between the container attachment points and the lifting hook to allow simple motion engagement; (2) the means of attaching the special lifting device to the shipping container be addressed during the design to ensure the security of the attachment method under load; (3) the actuating mechanism used securely engage or disengage; (4) load-carrying components that may become inadvertently disengaged be fitted with a retaining latch; and (5) engagement indication be provided whenever it is difficult to observe the attachment points between the special lifting device and the shipping container.

The licensee's purchase specification required that the lifting yokes (on the CTF and the Unit 2/3 crane) be designed in accordance with ANSI N14.6. Holtec's lift yoke design criteria also specified design in accordance with ANSI N14.6. However, during a review of the criteria, the inspectors could not determine how the lifting yokes met the intent of the ANSI N14.6 requirements stated above. The lifting yoke employs two air-operated swing arms with circular cut-outs which fit over the cask lift trunnions. No physical locks or latches are provided, nor is any type of flag provided as an indication of engagement.

This matter was also discussed with the Spent Fuel Project Office (SFPO) in determining if the CTF and Unit 2/3 reactor building crane design satisfied ANSI N14.6 requirements. The SFPO staff did not attempt to determine how the yokes met the ANSI N14.6 provisions. Instead, they focused on whether any of these provisions were violated. The conclusion of the SFPO assessment was that the design of the lifting yokes for the CTF and crane does not violate any provisions of ANSI N14.6. Subsequent to the inspectors raising questions on this point, Holtec revised the design criteria to include statements that "each cask lifting trunnion is equipped with an end cap, which allows for visual verification that the lift yoke arms are properly engaged." The criteria also included the statements, "because the design of the lift yoke shall ensure that the lift yoke arms hang plumb to engage the lifting trunnions, there is no need to provide an additional security device to maintain attachment. There are no credible loads that would apply a side load between the cask lifting trunnion and lift yoke arm. Therefore, the security of the attachment method under load in all handling positions is assured." The NRC reviewed the revised criteria against the provisions of ANSI N14.6 and concluded that each of these provisions was satisfied.

During "dry run" testing, the licensee ensured that the lift yoke arms were plumb through engineering inspection and adjustment. The licensee is considering a procedure step requiring verification that the yoke arms are plumb. As noted in Section 2.1.2 of this report, a seismic event could be a source of side loads between the cask trunnion and the lift yoke arm. Generally, seismic events are considered credible; however, as detailed in several sections above, various heavy-load-handling components at Dresden have been accepted without rigorous seismic qualification and analyses.

c. Conclusions

The NRC staff determined that the Cask Transfer Facility complies with applicable CoC requirements specific to design and testing. The NRC staff also concluded that the lifting yoke design satisfies the provisions of ANSI N14.6; a set of design criteria, produced by the contractor after the NRC questioned the design, formed parts of the basis for this conclusion. Seismic event effects have not been analyzed in detail and may have the potential to apply a side load on lift yoke arms not designed to withstand such loads. Seismic issues relating to the CTF and Unit 2/3 reactor building crane lifting yokes will be examined further as part of the review of the Unresolved Item concerning overall seismic issues. However, the NRC did not identify any concerns that would prohibit the licensee from safely proceeding with cask loading in accordance with the licensee's planned schedule.

2.3 Quality Verification/Receipt Inspection

a. Inspection Scope

The inspection included a review of the receipt inspection documentation, including nonconformance reports, for the Cask Transfer Facility (CTF), Hi-Trac (fuel transfer cask), and Multi-Purpose Canister (MPC 006). The inspectors also reviewed selected 10 CFR 72.48 packages addressing manufacturer identified nonconformances for the CTF, Hi-Trac, and MPC 006.

b. Observations and Findings

The inspection focused on the documentation for the receipt inspection of the Hi-Trac fuel transfer cask, since it was to be used for many years. The documentation pertaining to MPC-006 was examined because it was the first to be loaded. The CTF was fabricated and supplied under what was essentially a "Turn Key" contract. The inspectors reviewed the documentation to determine if the manufacturer had provided the licensee with Suppliers Manufacturing Deviation Reports (SMDRS), if the licensee had conducted engineering evaluations to address the issues identified by the manufacturer in the SMDRS, and if the licensee effected proper repairs to return the component deviations to specifications, or identified the need to perform a 10 CFR 72.48 evaluation for the deviations from the specification. The inspectors also reviewed the licensee's engineering evaluation for any 10 CFR 72.48 change to the components.

Numerous SMDRS were reported concerning problems encountered during work by Omni, the manufacturer of the Hi-Trac cask. Some were repetitive (cut hole in wrong place in plate, re-welded hole, cut new hole too large, then used too much weld build up.) Several 10 CFR 72.48 evaluations had to be performed because the finished component was not made exactly to specifications. For example, Omni welded on two different size lifting trunnions. The manufacturer of the MPCs (US Tool & Die) necessitated a few evaluations because the finished component was not exactly to specifications. Based on their review, the inspectors concluded that the evaluations performed under 10 CFR 72.48 were acceptable.

c. Conclusions

Receipt documentation for the Cask Transfer Facility, Hi-Trac fuel transfer cask, and Multi-Purpose Canister (MPC 006) disclosed numerous problems that occurred during fabrication, but did not contain information indicating there were any significant quality deficiencies in the final, delivered components. In addition, the inspectors did not identify any concerns with the licensee's 10 CFR 72.48 evaluations that addressed permanent changes to the components.

3.0 Management Meetings

The inspectors presented the inspection results to licensee management daily during the ongoing inspection and at a special exit meeting on June 7, 2001, and during a subsequent phone call on June 22. The licensee acknowledged the findings presented. The licensee did not identify any of the documents or processes reviewed as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Dale Ambler, Regulatory Assurance Manager
Robert Fisher, Unit 2-3 Station Manager
Paul Planing, Unit 1 Manager
Nate Leech, Dry Cask Storage Project Manager
Bob Rybak, Regulatory Assurance
Ken Ainger, Regulatory Services
Joe Sipek, Nuclear Oversight Manager
Dave Schupp, Operations
Preston Swafford, Site Vice President
Tom Luke, Site Engineering Director
Chip Cerovac, Training
Dave Williams, Electrical Maintenance Superintendent
Ken Bowman, Operations Manager
Joe Kotowski, Operations Supervisor
Pete Scardigno, Site Project Manager

Illinois Department of Nuclear Safety

Rick Zuffa, Dresden Resident Inspector

NRC - Region III

Marc L. Dapas, Deputy Director, DNMS
Bruce L. Jorgensen, Chief, Decommissioning Branch, DNMS

The inspector also interviewed other licensee personnel in the course of the inspection.

INSPECTION PROCEDURES USED

IP 60853: Construction of an ISFSI

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

07200037/2001-002-01	NCV	Facilities not as described in the UFSAR - no safety lugs on reactor building crane bridge rails.
07200037/2001-002-02	NCV	Facilities not as described in the UFSAR - wire rope safety factor is not 8 to 1.
07200037/2001-002-03	NCV	Documented bases for 50.59 safety evaluation inadequate - relied on temporary procedure and non-existent protective device.
07200037/2001-002-04	NCV	Facilities not as described in the UFSAR - no slow-speed (inching) motor on the reactor building crane.
07200037/2001-002-05	URI	Long-term acceptability of the Unit 2/3 reactor building, crane, and ancillary equipment for handling large numbers of dry fuel storage casks.
07200037/2001-002-06	URI	Acceptability of licensee characterization of overstress conditions greater than 1.0, but less than about 1.1 as "generally acceptable."
07200037/2001-002-07	URI	Adequacy of licensee verification of operability of crane overload protection associated with "new" load indication system.

Closed

07200037/2001-001-01	URI	Issues potentially indicating the Unit 2/3 reactor building crane is not acceptable for handling dry fuel storage casks.

Discussed

None

LIST OF ACRONYMS USED

ANSI	American National Standards Institute
BTP	Branch Technical Position
CoC	Certificate of Compliance
CTF	Cask Transfer Facility
ISFSI	Independent Spent Fuel Storage Installation
MPC	Multi Purpose Canister
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NUPIC	Nuclear Utilities
OBE	Operating Basis Earthquake
QA	Quality Assurance
SER	Safety Evaluation Report
SSE	Safety Shutdown Earthquake
SFP	Spent Fuel Pool
SMDRS	Suppliers Manufacturing Deviation Reports
SRP	Standard Review Plan
TER	Technical Evaluation Report
TS	Technical Specification
TSAR	Topical Safety Analysis Report
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item

DOCUMENTS REVIEWED

As stated in the Report Details.

December 28, 2001

MEMORANDUM TO: John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Gary M. Holahan, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

E. William Brach, Director
Spent Fuel Program Office
Office of Nuclear Materials Safety and Safeguards

FROM: John A. Grobe, Director
Division of Reactor Safety

SUBJECT: DIFFERING PROFESSIONAL VIEW CONCERNING STRUCTURAL ISSUES REGARDING THE DRESDEN REACTOR BUILDING/125 TON CRANE AND THE SPENT FUEL CASK TRANSFER FACILITY

I have been serving as Chair of the Review Panel evaluating the subject Differing Professional View (DPV). The specific issues captured in the DPV are included in the attachment to this memorandum. The Review Panel has met several times to evaluate information associated with the concerns and has interviewed Region III staff and management to begin to develop a sound understanding of the agency's positions on these issues.

Three questions have emerged regarding the licensing basis involved in these issues that need clarification from program office representatives.

1. Regarding Substantive Issue No. 1.A., the NRC apparently accepted the position of the licensee expressed during a public meeting in 2001 that the licensing basis for the Dresden reactor building did not require consideration of live or lifted loads on the crane when analyzing the structure for the Operating Basis and Safe Shutdown Earthquake (OBE and SSE) load cases. Where is this described in the licensing basis for the Dresden facility, e.g., application, letters responding to questions, safety evaluations, etc.? What was the licensing guidance, e.g., Standard Review Plan, Branch Technical Position, Regulatory Guide, etc., at the time of this licensing review regarding consideration of crane live loads in OBE and SSE load case structural analyses of Seismic Category I structures?

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2. Regarding Substantive Issue No. 2.A., the Certificate of Conformance (C of C) for the Cask Transfer Facility requires the device to be single failure proof and the application states that no single failure will result in a dropped load. Further the C of C states that the device must meet NUREG-0612 which requires that special lifting devices meet ANSI N14.6. The cask lifting yolks are special lifting devices. ANSI N14.6 indicates that, if it is possible for a load carrying component to become disengaged, it shall be fitted with a latching device with an actuating mechanism that securely engages and disengages. The cask lifting yolk design does not include a latching device. What is the basis for the conclusion that the cask lifting yolks meet the licensing basis requirements for the device?

3. Regarding Substantive Issue No. 3., what were the NRC expectations and approved fabrication standards for weld quality verification for the Cask Transfer Facility?

The Review Panel would like to review records and interview program office staff and management regarding these three questions. Please contact me at your earliest convenience (E-mail: JAG; Telephone: 630-829-9700) if you have any questions and to arrange a time for the Review Panel to complete this work.

Attachment: Dresden Structural Issues

cc w/o attach: J. Dyer
J. Caldwell
R. Landsman
P. Hiland
J. Jacobson

DRESDEN STRUCTURAL ISSUES

SUBSTANTIVE ISSUES NOT ADEQUATELY ADDRESSED BY THE NRC

1. Reactor building structural issues.

   A. The reactor building design for Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) load cases did not include the 125 ton crane load (live load) as described in the Updated Final Safety Analysis Report (UFSAR), while the live load considerations were included in the analysis of non-earthquake load cases.

   B. Calculations indicate that some building structural components exceed both yield and ultimate tensile strength for the SSE load case.

   C. A 1998 calculation indicates an overstress of reactor building structural components of five percent. The applicable code does not allow any overstress conditions. In addition, the inspection report documents only a three percent overstress.

2. Cask Transfer Facility (CTF) structural design issues.

   A. The cask lifting yolks for both the CTF and the Unit 2/3 crane do not meet ANSI N14.6 standards as required by the Certificate of Conformance.

   B. The CTF lift platform beam does not meet the single failure proof criteria of NUREG-0554.

3. Cask Transfer Facility weld quality.

   Existing records are inadequate to establish weld structural quality for welds on the CTF.

ADDITIONAL ISSUES NOT ADEQUATELY ADDRESSED BY THE NRC

1. The crane wire rope does not meet the required safety factor of eight as specified in the UFSAR. This condition was acknowledged by the NRC and accepted in a 1976 license amendment based on a commitment to implement compensatory actions and to upgrade of the wire rope to one that has a safety factor of at least eight at the time of its next replacement. The wire rope has been replaced a number of times with like-for-like deficient (e.g., safety factor less than eight) wire rope. The inspection report characterizes this issue as a non-cited violation for failure to update the UFSAR; however, the report does not address the failure to upgrade the wire rope with rope that meets the original design or the failure to continue implementing the compensatory measures following the 1976 amendment.

2. The current inappropriate operation and testing of the overload protection device (load cell) is dispositioned in the inspection report as an unresolved item, however, the inspection report does not address the identified deficiencies in competency and training of the staff and technicians who operate and calibrate the load cell.

3. The inspection report states that the load cell on the Unit 2 and 3 crane hoist was routinely bypassed for 20 years when the crane was in the restricted mode, which was outside the licensing basis. This is a violation of requirements, but is not characterized as a violation in the inspection report.

4. The 1981 repairs to the crane bridge girders were incorrectly classified as a minor repair. One outcome of classifying the repair as minor was that there was no requirement to perform a full load test of the crane following the repair as would have been required had the repair been classified as extensive. Part of the licensee's basis for justification of the classification of the repair as minor was predicated on the Nutech analysis which demonstrated that had the repairs not been made the girders would only have been over-stressed 22 percent. That Nutech analysis was based on allowable stresses specified in a design code that was different than the code specified in the licensing basis for the crane. The correct design code for the crane is CMMA-1975. Utilizing that design code allowable stress (17.6 ksi) and re-performing the calculations demonstrated that the girders would have been approximately 50 percent over-stressed. It does not appear that this was considered in the NRC evaluation of the acceptability of the minor repair classification.

5. The 1974 analysis of the bridge girders indicates a two percent overstress condition during an OBE considering only static loads. This over-stress condition is documented in the inspection report, but there is no documentation of the basis for the acceptability of this over-stress condition. In addition, there is no analysis of stresses in the trolly for the OBE or SSE load cases

6. During a crane inspection conducted by licensee representatives, five deficiencies in the crane were identified as needing correction. The licensee initiated a corrective action document, but only corrected one of the deficiencies and closed the corrective action document as acceptable. There is no documented justification for not correcting the remaining four deficiencies. This was a violation of the licensee's corrective action program, but was not documented as a violation in the inspection report.

<u>OTHER ISSUES INVOLVING CRANE ADEQUATELY ADDRESSED BY THE NRC</u>

1. There are no safety lugs on the crane bridge rails as described in the UFSAR. This issue was acceptably dispositioned in the inspection report as a non-cited violation.

2. The crane design does not include an inching motor as described in the UFSAR. This issue was acceptably dispositioned in the inspection report as a non-cited violation.

3. A 1996 safety evaluation regarding the revision of the UFSAR addressing provisions for preventing overloading the crane concluded that an unreviewed safety question did not exist. That evaluation was inadequate in that it was based on inaccurate information regarding the crane. This issue was acceptably dispositioned in the inspection report as a non-cited violation.

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

June 1, 2001

MEMORANDUM TO: License File Nos. DPR-19, DPR-25
Docket Nos. 050-00237, 050-00249

FROM: Bruce L. Jorgensen, Chief, Decommissioning Branch

SUBJECT: MEETING WITH EXELON REGARDING DRESDEN UNIT 2/3 REACTOR BUILDING CRANE ISSUES

Exelon Generating Company representatives, in conversations with NRC representatives, including a discussion between Marc Dapas of RIII and Rod Krich of Exelon on April 27, 2001, offered to attend a meeting to discuss issues and questions about the Dresden reactor building crane.

A meeting was held in RIII on May 23, 2001, for the purpose of clarifying the issues and to provide the licensee the opportunity to present verbal and written information to support their views regarding crane history, licensing basis and current qualification status. Minutes of the meeting are attached. These minutes reflect licensee positions; NRC conclusions regarding some of the issues discussed remain under consideration. A copy of the Agenda, a copy of the licensee's handout, and a list of attendees are also attached.

The licensee addressed each issue and question presented. Their position may be summarized as follows: As of May, 2001, the Dresden Unit 2/3 crane is a single-failure-proof crane. This is by virtue of its having been so designated and approved by the NRC in 1976 (i.e. the licensing basis for this crane is as a single-failure-proof crane) and by virtue of the licensee's having maintained or restored each and every attribute relied upon in 1976 for the designation.

Selected portions of the information presented will be verified as part of Region III's inspection program for the Dresden ISFSI project. For example, the Region plans to verify the licensee's interpretation that Technical Specifications permitted handling heavy loads while the reactor was at power, and to re-examine the meaning of the term "over-stressed," in the consultant's report on the 1981 impact event. In addition, the Region specifically requested a copy of the 50.59 evaluation for bypassing of the "load cell" and its integral overload protection function, and the Region plans to identify and review any other 50.59 evaluations done to support other modifications to the crane compared to the licensee's description in their licensing basis.

Selected information contained in the attachment may be incorporated, as appropriate, in the Region's inspection report(s).

Attachments: 1. Meeting Minutes
2. Meeting Agenda
3. Meeting Handout
4. Meeting Attendees

cc: J. Zwolinski, NRR
S. Bajwa, NRR
C. Carpenter, NRR
J. Hannon, NRR
J. E. Dyer, RIII
J. L. Caldwell, RIII
C. D. Pederson, RIII
M. L. Dapas, RIII
M. A. Ring, RIII

MEETING MINUTES

May 23, 2001 Meeting Between NRC RIII and Exelon Generating Company Dresden Unit 2/3 Reactor Building Crane

INTRODUCTION

A meeting was held in the NRC RIII offices on May 23, 2001, for the purpose of clarifying the licensee's position regarding qualification status of the Dresden Unit 2/3 reactor building crane. Inspection activities conducted by NRC as part of the oversight of the Dresden Independent Spent Fuel Storage Installation (ISFSI) had generated questions and concerns about the subject crane. Some of these issues were documented as Unresolved Items in NRC Inspection Report No. 07200037/2001-001(DNMS).

An Agenda was prepared for the meeting which is attached to these minutes. All of the items on the Agenda were discussed, in a sequence which followed a handout prepared by the licensee. The handout package is also attached to these minutes, as is a list of meeting attendees.

LICENSING BASIS

The licensing basis for the Dresden Unit 2/3 reactor building crane was established in a 1976 licensing action.

The licensee discussed the regulatory history leading up to the licensing action, indicating that the original crane design (per CMAA-70, a manufacturer's standard) was identified as needing to be improved when, in about 1973-4, plans were being developed to use the crane to load spent fuel into transport casks for shipment to a fuel reprocessing facility. Special Report 41 was submitted to NRC in about early 1975 to define the crane improvements deemed necessary. Subsequently, NRC issued Branch Technical Position BTP APCSB 9-1 to address Agency expectations for cranes to be deemed adequate to ensure against load drop in the event of any single failure. The licensee's Supplement A to Special Report 41 addressed additional items derived from BTP 9-1, in mid- to late-1975, and several exchanges of correspondence followed in the form of NRC Requests for Additional Information (RAIs) and licensee responses.

The licensee's Special Report 41 and Supplement, along with subsequently submitted information, were discussed. These became the licensing basis when the NRC amended the license and issued the supporting Safety Evaluation Report (SER) on June 3, 1976. The SER considered the crane rating as 125 tons, evaluated for casks weighing up to 100 tons. The design redundancies included hoist and trolley brakes, the cask lifting device and crane control components. Single element components were all specified to have safety factors of at least 7.5.

The licensee indicated that NRC had concluded the intent of BTP 9-1 was met, except for the reeving system (wire rope safety factor and fleet angles) and the protection provided against "two-blocking" (upper limit switch on travel.) Temporary waivers were approved at the licensee's request to allow for alternative means (operator at main electrical breaker) to accomplish the function of a mechanical upper travel limit switch and load travel restrictions only over a designated "safe load path." The rope safety factor issue was discussed in some detail separately - see below. Original stipulations were that an "inching motor" was to be in operation when the crane was being used in its single-failure-proof mode (i.e. in "restricted mode"); this was replaced by a speed-limiting control circuit which ensured hoisting speed would not exceed 5 feet per minute. The hoisting speed limit was described as being in compliance with BTP 9-1.

A "load cell," which was provided to indicate load and which contained an integral overload protection, was discussed. The load cell did not function properly and was subsequently bypassed, including the overload protection feature. A design change review under 10CFR50.59 supported the bypassing of the load cell, with administrative controls to serve in lieu of the overload protection. NRC representatives requested the 50.59 review package be made available for inspection, which the licensee agreed to do. As detailed below ("DIGITAL CONTROL UPGRADE") the overload protection function has been restored.

SEISMIC DESIGN

The licensing basis was also discussed from the perspective of seismic design and NRC review of the crane and the reactor building superstructure.

The licensee identified the crane as Safety Class II (non-seismic), which they indicated had been made known to NRC and recognized as part of the licensing basis. A commitment to perform analyses of the crane bridge girders in Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) conditions was met, with the result that the girders were found to be "acceptable" with a suspended load of 125 tons. The crane trolley was said to have no seismic qualification.

The building superstructure was specified to be capable for loading combinations which included normal loads with a full crane lifted load, OBE without lifted load, and SSE without lifted load. The analyses were said to have found the superstructure "acceptable" for these load combinations. The licensee indicated the NRC understood and approved this design within the licensing basis, even though BTP 9-1 included an expectation the SSE with load would be analysed and found acceptable.

Separately, beyond-design-basis calculations were reported for the building, considering SSE plus full suspended load (125 tons), as per BTP 9-1. The result of this calculation was reported to be that all structural members met design allowables.

BULLETIN 96-02

The Bulletin and the licensee's response to the Bulletin were discussed. The licensee talked through the Technical Specifications applicable to the Dresden plant, indicating that those specifications never restricted use of the crane in "restricted mode" to refueling or outage conditions. Data on actual historic crane use for loading casks was provided. The response to the Bulletin was "generic" to address all the (then) ComEd operating reactors, and stated that there were no plans to lift heavy loads over the fuel pool or safety-related equipment with the reactor operating at power. The response indicated changes to the technical specifications would be required should such lifting be planned, and that safe shutdown capability would be demonstrated should cask movements be required.

The licensee expressed the view that, because Dresden specifically had a licensing basis as a single-failure-proof crane, the "generic" response contained an over-commitment. Dresden was considered to be within its licensing basis to perform heavy load lifts with the reactor in power operation. Safe shutdown capability demonstration was considered to apply to cranes which could experience drop of a load, which was "not credible" for a crane classified as single-failure-proof. Thus, the commitments contained in the Bulletin response, as they applied to Dresden, were withdrawn earlier this year.

The 1976 T/S which pertained to inspection and surveillance requirements for the wire rope were deleted in 1996 in an action unrelated to the Bulletin, and the equivalent requirements were placed into station procedures, where they remain.

The question of "safe load paths" was discussed, and the licensee went over the diagram contained in their handout which illustrates the paths so designated.

CRANE DAMAGE IN 1981

The implications of a 1981 incident were discussed. The Unit 2/3 reactor building crane was damaged in 1981, during a lift involving the reactor vessel head lift "strongback" (approximately a 10-ton load), when the strongback struck the under side of the crane bridge girders. The inside web and the bottom flange of each girder were deformed. The licensee presented information to support their view that the damage incurred and the repairs to restore the crane girders to "design" were not "extensive." This was an issue because ANSI/ASME B30.2.0 - 1967, to which the licensee was committed, specified a 125% rated load test be performed on "new, extensively repaired or altered" cranes.

Calculations were presented to define the "overstress" experienced by the girders. In that the licensing basis for the crane was Safety Class II, the licensee maintained that restoration of the crane was not mandatory by regulation. Rather, the restoration was to regain margin of safety which was established in beyond-design-basis calculations. Specifically, one girder was found to have experienced 22% overstress for OBE with full load. Thus, the margin represented in meeting a 0.6 Fy criterion was reduced, but all stresses were less than yield.

The repair consisted of cutting out the deformed section of each girder web and welding a new plate over the area, and welding a plate over the deformed section of each flange. An expert consultant's assessment of the damage and the repair were presented in support of the licensee's position that the repairs were not "extensive."

Information was also provided to address the performance of the crane since the repair, which indicated that loads up to 125 tons had been lifted at least 42 times. No indication of stress or distortion has been observed in the repaired or adjacent sections, although no focused inspection has been performed with the specific intent of looking for such stress or distortion.

DIGITAL CONTROL UPGRADE

Modifications to the crane controls in 2001, to replace the control, indication and protection systems with a new digital system were discussed. The licensee indicated the design, installation and testing of the new system were specifically aimed at restoring each and every feature of the original system captured in the 1976 licensing basis. Specific examples of features provided included a variable speed drive controller for the hoist function, to limit the hoist speed to 5 feet/min, an overload protection feature, and limits to control upper lift limit and safe load path. The licensee maintained that, as installed and tested, the new digital system conforms with the licensing basis.

WIRE ROPE

The licensing basis, performance, inspection and replacement of the wire rope were discussed. The rope was last replaced in 2000 in accordance with the inspection program, and was described as in good condition, not requiring replacement.

The licensee indicated that in 1976, the NRC was aware of the fact the wire rope had a safety factor of 7.798 (for a 125 ton load) and that, while BTP 9-1 specified a safety factor of 8 (by limiting load to 12.5 % of rope yield), the rope was reviewed and approved. As part of the approval, technical specifications were put in place for limiting conditions for operation and for surveillance and inspection of the wire rope, to ensure the rope did not degrade from the "original design" condition.

The licensee concluded the wire rope has been and remains as approved by the NRC in the licensing basis. Future replacements, as necessary, will continue to be like-for-like, as a minimum.

CONCLUSION

The licensee indicated the crane was certified and licensed as a single-failure-proof crane in 1976. The NRC recognized and accepted certain identified conditions which did not literally meet all of the expectations set forth in BTP 9-1. In some cases, the conditions noted were allowed only on a temporary basis, and permanent upgrades were put into place as required. In other instances, exceptions were made on a permanent basis. The licensee's objective, in proceeding with planned use of the crane for the upcoming ISFSI cask loading activities, was to restore the crane to the 1976 licensing basis, restoring it to single-failure-proof classification. They indicated that they have achieved this objective.

NRC representatives expressed appreciation for the licensee's efforts in researching dated records and in organizing them for the presentation. The licensee was informed that NRC will consider the information and we will inform them of our conclusions in the near future.

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

AGENDA

NRC/Exelon Meeting - Dresden ISFSI
(Unit 2/3 Crane - History/Status)

Opening/Purpose

Licensing Basis
- Design Features
- (licensing basis; modifications; SE & OBE plus load)
- Operating MODE Restraints (TS restrictions)
- 2001 Equivalence

Bulletin 96-02
- Defined Safe Load Paths
- Safe Shutdown Capability/Plans/Procedures
- T/S history & current Admin Procedures without MODE

ASME/ANSI B30.2.0
- "Extensive repair"
- "Equivalent degree of protection " for exception(s)
  - crane performance history (#/size of lifts)
  - inspection history

Alterations to controls/limits/indications/alarms
- Scope
- Testing

Wire Rope
- Inspection/performance history (decision processes)
- Replacement plans/schedules

Open Forum

Closing

Meeting with NRC
Unit 2/3 Reactor Building Crane

Exelon Generation Company, LLC

Dresden Nuclear Power Station

May 23, 2001

Meeting Objective

- Review licensing basis, history, and current status of the Reactor Building Crane to show that:
  - Crane is single failure proof and rated to 125 tons
  - Original licensing basis is maintained
  - Repairs and modifications have been sufficiently tested

Topics

- Licensing Basis
- NRC Bulletin 96-02
- 1981 Crane Repair
- 2001 Modifications
- Wire Rope

Licensing Basis

Licensing Basis

- Licensing basis is described in two primary documents
  - ComEd Special Report 41 and Supplement A
  - NRC Safety Evaluation, June 1976
- Single failure proof features of Reactor Building Crane
  - Dual load path through gear train, reeving system, and load block

Licensing Basis

- Single failure proof features (cont'd)
  - Redundancy
    - Hoist and trolley brakes
    - Cask lifting device
    - Crane control components
  - Dual element stresses comply with CMAA-70
  - Single element components minimum safety factor of 7.5

Licensing Basis

- Restricted load path
  - Limit switches
  - Administrative controls on operation with failed control area limit switches (DFP 800-45)
- NRC Safety Evaluation states that crane meets intent of BTP APCSB 9-1, except for
  - Reeving system
  - Protection against "two blocking"

Licensing Basis

- NRC Safety Evaluation accepts compensating features
  - Wire rope inspection and replacement program compensates for reeving system (DMS 5800-01)
  - Mechanical limit switch provides for "two blocking" protection

Modifications and Waivers

- Electrical interlocks for safe load path temporarily waived until testing could be completed
- Mechanical limit switch for "two blocking" waived until it could be installed
- Installation of a slow speed drive motor

Modifications Status

- Electrical interlocks were installed and tested; have remained in place
- Mechanical limit switch was installed and has remained in place
- In lieu of slow speed drive motor, modified electrical circuit of main hoist to limit maximum lift speed to 5 feet per minute in restricted mode operation
  - Consistent with BTP APCSB 9-1
  - Accepted by NRC in June 3, 1976 Safety Evaluation

Spent Fuel Cask Handling Technical Specifications

- Covered restricted mode operation and wire rope inspection
  - Original Technical Specifications (TS) issued June 3, 1976
  - TS upgrade relocated restricted mode description to UFSAR
  - Surveillance requirements implemented by DMS 5800-01 and DOS 0800-06

Licensing Basis Conclusi

- The licensing basis approved for the Reactor Building Crane and cask han in the June 3, 1976, NRC Safety Eva has been maintained

Licensing Basis - Seismic

- Reactor Building Crane is Safety Class II (non-seismic) (UFSAR Sections 9.1.4.2.2 and 3.8.5)
- Crane bridge girders have been evaluated for both OBE and SSE conditions with 125 ton load and found acceptable

Licensing Basis - Seismic

- Reactor building superstructure
  - Licensing basis (UFSAR Section 3.8)
    - Loading combinations
      - Normal loads with full crane lifted load
      - OBE without crane lifted load
      - SSE without crane lifted load

Licensing Basis - Seismic

- Reactor building superstructure (cont'd)
  - Calculation of record (DRE98-0020) documents design basis
    - Normal case: all except one member meet design allowables; roof girder had 3% overstress
      - Small overstress acceptable considering conservative support assumptions
      - Normal practice to accept overstress of up to 10%
    - OBE case: all members meet design allowables
    - SSE case: all members meet design allowables
    - Performed "beyond design basis" case of SSE plus full crane load: all members meet design allowables

NRC Bulletin 96-02

NRC Bulletin 96-02

- Bulletin requests
  - Review of plans and capabilities for handling heavy loads with reactor at power
  - Determine whether activities are within licensing basis
  - If outside licensing basis, get NRC approval prior to handling heavy loads

NRC Bulletin 96-02

- ComEd May 13, 1996 response
  - No plans for movement of dry storage cask over spent fuel, fuel in the core, or safety related equipment at power
  - Would demonstrate safe shutdown capability should cask movements be required

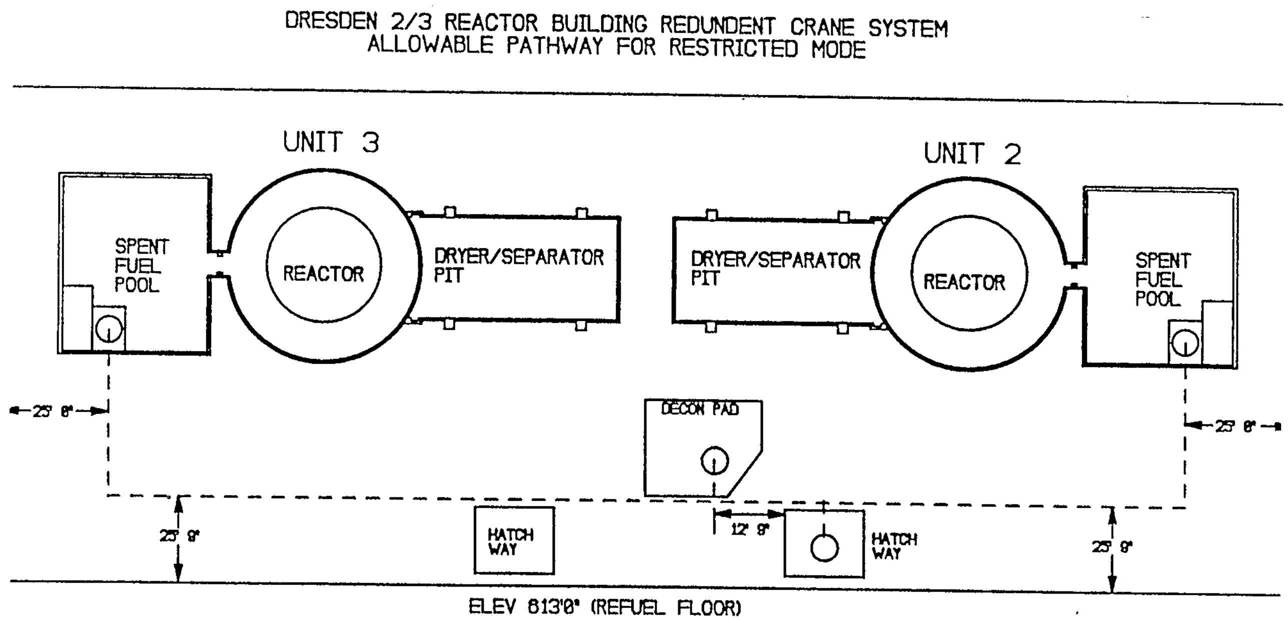
NRC Bulletin 96-02

- Commitment changed in 2001
  - Original response was an over-commitment and was not required if cask moves were within licensing basis
  - Single failure proof crane means that load drop accident is not credible

Safe Load Paths

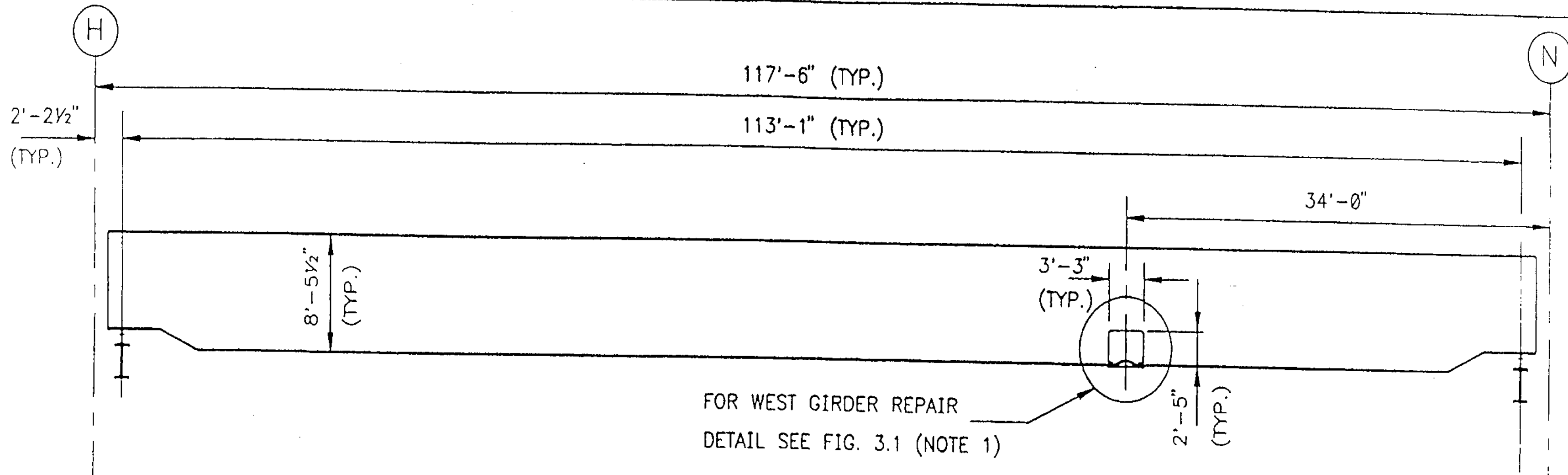
- Crane bridge and trolley movement is restricted to ensure the crane remains within a predefined pathway
- Governed by station procedures (DFP 0800-20 and DFP 0800-45)
  - Fuel cask handling above 545-foot level is "Restricted Mode"
  - Only permitted outside "Restricted Mode" in emergency or due to equipment failure to place the load in safe condition
- Reinforced by crane design
- Visual aid for crane operator

Restricted Mode Path

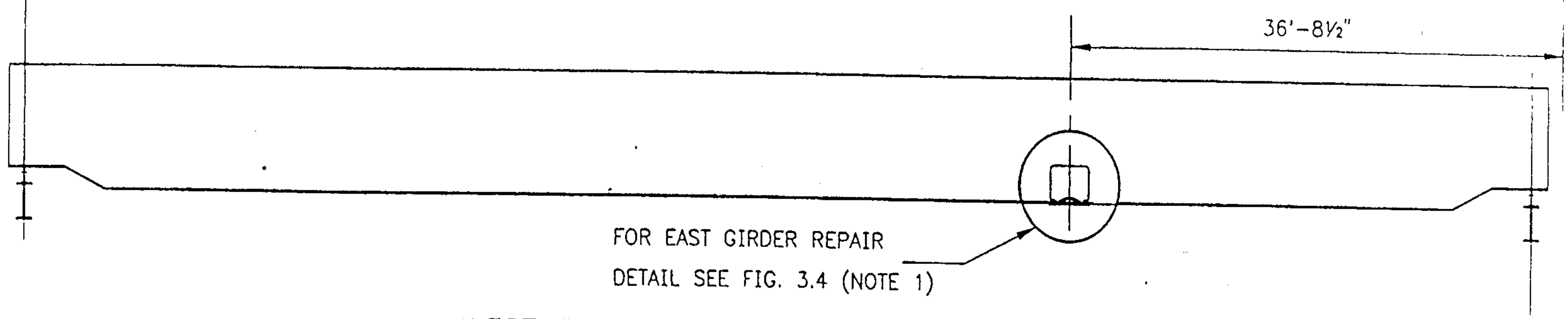


1981 Crane Girder Damage

- Bottom of girders damaged by a 10 ton lifted load
- Damage affected 2' x 2'2" lower section of inside web and bent bottom flange on each girder (0.3% of total bridge girder surface area)



EAST SIDE OF WEST CRANE BRIDGE GIRDER
ELEVATION VIEW (LOOKING EAST)



WEST SIDE OF EAST CRANE BRIDGE GIRDER
ELEVATION VIEW (LOOKING EAST)

FOR REFERENCE

NOTE (1): INFORMATION TAKEN FROM "NUTECH" DESIGN REPORT NO. COM-21-011 REV. 0 DATED SEPTEMBER 1981, AND FIELD WALKDOWN.

CRANE BRIDGE GIRDERS
(LOCATION OF GIRDER REPAIRS)

Exelon. Nuclear
Dresden Station 12 Unit: 2

SCALE : 1/8"=1'-0"
DATE : ——
DRAWN BY : ——
ORG. BY :

SKETCH A

SHEET NUMBER.

SIZE B

Damage Assessment

- Conservative analysis of damage without repair
  - Assumes damaged material removed and not replaced
  - All stresses less than yield; no permanent deformation
    - 22% overstress for OBE with full load (.6 Fy)
    - Within allowable for SSE with full load (.9 Fy)

Evaluation of Repaired Section

- Repair was to cut out web plates and replace with cover plates along web and bottom flange in two girders
- Repaired girders are stronger than original
  - Welds visually inspected by QC inspector
- Function of crane unchanged

Conclusion - Extent of Repair

- Nutech report reviewed repairs against ANSI B30.2 and concluded that "the suggested repair work to the crane bridge girders is not in the extensive repair or major alterations category. The repair is a minor repair."
- This certified design report was approved by two qualified Professional Engineers and reviewed and certified by an independent Registered Professional Engineer from State of Illinois

Conclusion - Extent of Repair

- Conclusion confirmed by independent expert
  - Stephen N. Parkhurst
    - Crane and Equipment Handling Specialist
    - Chairman and Member of ASME Committee on Cranes for Nuclear Facilities (CNF)
  - Summary of Findings
    - Concurs with the original findings of NUTECH engineers that the repairs performed were not extensive
    - Does not recommend re-load testing the crane based upon the localized girder repairs
    - Re-load testing only required if crane modified or re-rated, where a modification included an item such as girder extension

History Since Repair

- Reactor Building Crane has lifted up to 125 tons at least 42 times since 1982. No distress or distortion observed at the repaired section or adjacent sections

2001 Modifications

- Purpose - improve reliability and replace obsolete equipment
- Added variable speed drive controller
- Installed digital crane controls
  - Not an extensive alteration
    - Does not affect single failure proof capability
  - Functional testing completed
    - No-load and load testing of controllers
- Crane design remains consistent with licensing basis

Wire Rope

- Last replaced in 2000 in accordance with inspection program
- No current need to replace
- When replacement is required, will be like-for-like as a minimum

Conclusion

- The licensing basis, history, and current status of the Reactor Building Crane has shown that:
  - Crane is single failure proof and rated to 125 tons
  - Original licensing basis is maintained
  - Repairs and modifications have been sufficiently tested

NRC/Exelon Meeting - Dresden ISFSI
Attendees 5/23/01

Exelon Attendees

K. A. Ainger
Dale F. Ambler
Tom Luke
David Schupp
Pat Simpson
M. Molaei
C. Chhablani
P. F. Scardigno
Timothy P. Heisterman
John Zappia

NRC Attendees

Marc Dapas
Mark Ring
Bruce Jorgensen
Ross Landsman
Paul Pelke

NRC FORM 651
(3-1999)
10 CFR 72

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket Number	Amendment No.	Amendment Date	Package Identification No.
1014	05/31/00	06/01/20	72-1014	0		USA/72-1014

Issued To: (Name/Address)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

Safety Analysis Report Title

Holtec International Inc., Final Safety Analysis Report for the HI-STORM 100 Cask System
Docket No. 72-1014

CONDITIONS

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications) and Appendix B – (Approved Contents and Design Features), and the conditions specified below:

1. CASK

a. Model No.: HI-STORM 100 Cask System

The HI-STORM 100 Cask System (the cask) consists of the following components: (1) interchangeable multi-purpose canisters (MPCs), which contain the fuel; (2) a storage overpack (HI-STORM 100), which contains the MPC during storage; and (3) a transfer cask (HI-TRAC), which contains the MPC during loading, unloading and transfer operations. The cask stores up to 24 pressurized water reactor (PWR), fuel assemblies or 68 boiling water reactor (BWR) fuel assemblies.

b. Description

The HI-STORM 100 Cask System is certified as described in the Topical Safety Analysis Report (SAR) and in NRC's Safety Evaluation Report (SER) accompanying the Certificate of Compliance. The cask comprises three discrete components: the MPCs, the HI-TRAC transfer cask, and the HI-STORM 100 storage overpack.

12

1. b. Description (continued)

The MPC is the confinement system for the stored fuel. It is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. It is made entirely of stainless steel except for the neutron absorbers and aluminum heat conduction elements. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. The honeycombed basket, which is equipped with Boral neutron absorbers, provides criticality control.

There are three types of MPCs: the MPC-24, the MPC-68, and the MPC-68F. The MPC-24 holds up to 24 PWR fuel assemblies that must be intact. The MPC-68 holds up to 68 BWR fuel assemblies that may be intact or damaged (i.e., with known or suspected cladding defects greater than hairline cracks or pinholes). The MPC-68F holds up to 68 BWR fuel assemblies that may be intact, damaged, or in the form of fuel debris (i.e., with known or suspected defects such as ruptured fuel rods, severed fuel rods, and loose fuel pellets). All three MPCs have the same external dimensions.

The HI-TRAC transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the spent fuel pool to the storage overpack. The transfer cask is a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a water jacket attached to the exterior. Two types of HI-TRAC transfer casks are available: the 125 ton-HI-TRAC and the 100 ton HI-TRAC. The weight designation is the maximum weight of a loaded transfer cask during any loading, unloading or transfer operation. Both transfer cask types have identical cavity diameters. The 125 ton HI-TRAC transfer cask has thicker lead and water shielding and larger outer dimensions that the 100 ton HI-TRAC transfer cask.

The HI-STORM 100 storage overpack provides shielding and structural protection of the MPC during storage. The overpack is a heavy-walled steel and concrete, cylindrical vessel. Its side wall consists of plain concrete that is enclosed between inner and outer carbon steel shells. The overpack has four air inlets at the bottom and four air outlets at the top to allow air to circulate naturally through the cavity to cool the MPC inside. The inner shell has channels attached to its interior surface to guide the MPC during insertion and removal, provide a flexible medium to absorb impact loads, and allow cooling air to circulate through the overpack. A loaded MPC is stored within the HI-STORM 100 storage overpack in a vertical orientation.

2. OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 8 of the SAR.

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

3. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 9 of the SAR.

4. QUALITY ASSURANCE

Activities in the areas of design, purchase, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important to safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the cask system.

5. HEAVY LOADS REQUIREMENTS

Each lift of an MPC, a HI-TRAC transfer cask, or a HI-STORM 100 overpack must be made in accordance to the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant-specific safety review (under 10 CFR 50.59 or 10 CFR 72.48, if applicable) is required to show operational compliance with existing plant specific heavy loads requirements. Lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Section 3.5 of Appendix B to this certificate.

6. APPROVED CONTENTS

Contents of the HI-STORM 100 Cask System must meet the fuel specifications given in Appendix B to this certificate.

7. DESIGN FEATURES

Features or characteristics for the site, cask, or ancillary equipment must be in accordance with Appendix B to this certificate.

8. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to the certificate, which includes Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

NRC FORM 651A
(3-1999)
10 CFR 72

U.S. NUCLEAR REGULATORY COMMISSION

Certificate No. 1014

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

9. AUTHORIZATION

The HI-STORM 100 Cask System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, and the attached Appendix A and Appendix B.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Materials Safety
and Safeguards

Attachments:

1. Appendix A
2. Appendix B

DESIGN FEATURES

3.5 Cask Transfer Facility (CTF)

3.5.1 TRANSFER CASK and MPC Lifters

Lifting of a loaded TRANSFER CASK and MPC outside of structures governed by 10 CFR Part 50 shall be performed with a CTF that is designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and the below clarifications. The CTF Structure requirements below do not apply to heavy loads bounded by the regulations of 10 CFR Part 50.

3.5.2 CTF Structure Requirements

3.5.2.1 Cask Transfer Station and Stationary Liftining Devices

1. The metal weldment structure of the CTF structure shall be designed to comply with the stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. The applicable loads, load combinations, and associated service condition definitions are provided in Table 3-3. All compression loaded members shall satisfy the buckling criteria of ASME Section III, Subsection NF.

2. If a portion of the CTF structure is constructed of reinforced concrete, then the factored load combinations set forth in ACI-318 (89) for the loads defined in Table 3-3 shall apply.

3. The TRANSFER CASK and MPC lifting device used with the CTF shall be designed, fabricated, operated, tested, inspected and maintained in accordance with NUREG-0612, Section 5.1.

0612 says or

4. The CTF shall be designed, constructed, and evaluated to ensure that if the MPC is dropped during inter-cask transfer operations, its confinement boundary would not be breached. This requirements applies to CTFs with either stationary or mobile lifting devices.

(continued)

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

November 2, 2001

MEMORANDUM TO: Marc L. Dapas, Deputy Director
Division of Nuclear Material Safety

Bruce L. Jorgensen, Chief, Decommissioning Branch
Division of Nuclear Material Safety

FROM: John A. Grobe, Director
Division of Reactor Safety

SUBJECT: DIFFERING PROFESSIONAL VIEW REGARDING STRUCTURAL ISSUES ON THE DRESDEN REACTOR BUILDING AND 125 TON CRANE

The Review Panel has met on several occasions to clarify the structural concerns at Dresden and evaluate various agency records related to each concern. The Panel has determined that it is necessary to interview you to further understand the regulatory framework and the technical basis used to resolve the various issues.

We are enclosing a listing of the issues to allow you to prepare for a focused and succinct discussion. We have scheduled the following meeting times:

Bruce Jorgensen: November 9, 2001, from 9:00 to 11:00 a.m.

Marc Dapas: November 9, 2001, from 2:00 to 4:00 p.m.

To facilitate comprehensive review of the issues, please be prepared to address the following questions, as appropriate:

1. The accuracy of the stated concern.
2. The agency's position on the concern. Where is the basis for that position documented (internal memorandum, inspection report, meeting minutes, etc.)?
3. What staff and managers in Region III and the program offices were involved in formulating and approving that position?
4. If the agency has not yet taken a position on the concern and the issue remains unresolved, who has action to resolve the issue and where is the issue documented internally, e.g., Task Interface Agreement?
5. For "Substantive Issues" 1, 2 and 3 and "Additional Issues" 1, 2, 4, 5 and 6, what was the basis for changing the characterization of the issue in the draft report to the final report?
6. For "Additional Issue" 3, what was the basis for not concluding that a violation occurred?

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Copies of the draft and final inspection reports are attached for your reference.

Attachments: 1. Dresden Structural Issues
2. Draft Inspection Report
3. Final Inspection Report 0720037/2001-002 (DNMS) dated 8/13/01

cc w/o encl: J. E. Dyer
J. Caldwell
R. Landsman
P. Hiland
J. Jacobson

COM-21-011
Revision 0
September, 1981
64.2100.0036

DESIGN REPORT
FOR
REACTOR BUILDING
CRANE BRIDGE GIRDER
EVALUATION AND REPAIRS

DRESDEN NUCLEAR POWER STATION
UNITS 2 AND 3

Prepared for:
Commonwealth Edison Company

Prepared by:
NUTECH
San Jose, California

Approved by:

B. P. Tripathi
B. P. Tripathi, P.E.
Project Engineer

T. J. Wenner
T. J. Wenner, P.E.
Engineering Manager

Issued by:

H. W. Massie
H. W. Massie, Jr.
Project Manager

Date: 9/29/81

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CERTIFICATION BY REGISTERED PROFESSIONAL ENGINEER

I hereby certify that this report was reviewed by me and that I am a duly Registered Professional Engineer under the laws of the State of Illinois and that I am competent to review this document. I also certify that this report is in accordance with the requirements of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings", February 1969, unless otherwise noted.

THOMAS J. VICTORINE 62-32180 REGISTERED PROFESSIONAL ENGINEER OF ILLINOIS

Certified by:

T. J. Victorine

T. J. Victorine, P.E.

Professional Engineer

State of Illinois

Registration No. 62-32180

Date: 9/23/81

TABLE OF CONTENTS

LIST OF FIGURES AND TABLES

1.0 INTRODUCTION

The Reactor Building Crane Bridge Box Girders at the Commonwealth Edison Company's (CECo) Dresden Nuclear Power Station Units 2 and 3 were damaged due to impact and subsequent compression on the bottom of the girders by the strongback used for lifting the reactor head.

The two girders are part of the Reactor Building Crane and Hoist System. They are 113'-1" long and span in the north-south direction. The girders travel laterally in the east-west direction on rails located at the ends of the girders.

As a result of the strongback impact, the lower portion of the inside webs were buckled and bending of the inside portion of the bottom flanges on both the girders occurred. The effect on the girders due to this abnormal incident was very localized. Also, there were minor visible mars and scars along the length of the girders at various spots where the contact between the lifting apparatus and the inboard flanges occurred. The nature, extent and description of the damage was assumed to be as shown on Sargent and Lundy sketches (Reference 5.1)

At CECo's request NUTECH has performed a simplified but conservative structural analysis of the crane bridge girders. The analysis indicated that a repair is required to restore the crane bridge girders to their original design value.

NUTECH has also estimated an upperbound reaction load which could have existed between the bridge girder and the strongback during the incident in order to evaluate if stresses in any of the crane components exceeded material yield values.

The proposed repair consists of cutting out and replacing a portion of the web plate and adding cover plates over an approximate 3 ft. length on the bottom flanges of the existing crane bridge girders.

This report documents the results of the design and analysis of the crane bridge girder and calculation of the bounding reaction load between the crane bridge girder and the strongback.

The pertinent design criteria and loads are presented in Section 2.0. The analysis and design and design summary are provided in Section 3.0. Section 4.0 presents a list of documents required to be maintained by the station. Appropriate references are presented in Section 5.0. Reference 5.16 contains the detailed calculations of bridge girder analysis and modification. Reference 5.17 contains the calculations of the bounding reaction load between the crane bridge girder and the strongback.

2.0 DESIGN CRITERIA AND LOADS

The applicable design requirements and design loads are as follows:

2.1 The crane bridge girder modifications are designed to meet the requirements of the American Institute of Steel Construction (AISC) (Reference 5.2) unless otherwise noted.

2.2 The modified girder is designed for a seismic acceleration of 0.20g horizontal and 0.10g vertical for OBE condition and 0.40g horizontal and 0.20g vertical for DBE condition, in accordance with Sargent and Lundy Specification K-2177 (Reference 5.3).

2.3 The wheel loads, trolley loads and girder dead loads are taken from the original design values as reported in Whiting Corporation Design (Reference 5.4).

2.4 The load combinations and allowables used in the evaluation are as follows:

	Load Combination	Allowables
a.	Dead Load + Live Load + Impact (Normal Operation)	AISC (See Note 1 below)

b.	Normal Operation + OBE	AISC (See Note 1 below)
c.	Normal Operation + DBE	0.9 Fy (See Note 2 below)

NOTES:

(1) The Sargent and Lundy Specification K-2177 (Reference 5.3) states that the allowable stresses for crane bridge box girder design shall be taken from CMAA Specification (Reference 5.5) (Formerly known as EOCI), and the AISC Specifications (Reference 5.2), whichever governs. However at CECo's direction, to be consistent with the original design, the analysis and design presented herein is performed in accordance with AISC Specification (Reference 5.2).

(2) The material stresses are not to exceed 90 percent of the material yield strength as used in Reference 5.4.

3.0 **ANALYSIS AND DESIGN SUMMARY**

3.1 The detailed calculations for the evaluation, analysis and design of the crane bridge girder and its repair are documented in Reference 5.16.

The detailed calculations for the bounding reaction load between the crane bridge girder and the strongback and crane components failure evaluation are documented in Reference 5.17.

3.2 Summary of Crane Bridge Girder Structural Analysis

A structural analysis of the west crane bridge girder was performed in order to determine the stresses at rated capacity (i.e., 125 tons). This analysis was simplified in a conservative manner by excluding from the structural calculations the damaged areas of the web and flange plates (i.e., no credit was taken for any structural integrity available in areas of the damaged web and flange plates). Based on this conservative assumption, the west crane bridge girder is calculated to be approximately 20% overstressed at rated capacity. The east crane bridge girder would be overstressed in a similar manner but less than 20%.

Based on the structural analysis performed, repairs to the west and east girders as presented in this report in Figure 3-1 through Figure 3-6 are recommended in order to restore their capacity to the original design value. Repairs to the crane bridge girders allow the structural calculations to be modified so that credit can be taken for the structural steel in the vicinity of the repaired areas. Based on this modified structural calculation, the repairs will restore the capacity of the crane bridge girders to the original design value.

Table 3-1 provides a comparison of the bridge girder maximum stresses with the allowable stresses.

NUTECH Document Number COM-21-004 (Reference 5.6) presents the repair procedure required in order to accomplish the repairs specified in this report.

The repairs and modifications to the existing crane girder shall be treated as safety related and shall be in accordance with AISC Specification (Reference 5.2). All welding for the repair shall be in accordance with AWS D1.1-1981 (Reference 5.7) and the Dresden Station welding procedures (Reference 5.8).

The safety evaluation report (SER) is provided in NUTECH Document Number COM-21-002 (Reference 5.9).

At CECo's request we have reviewed the 1976 ANSI B30.2 code (Reference 5.10) requirements and we concur that the suggested repair work to the crane bridge girders is not in the extensive repair or major alteration category. The repair is a minor one.

All work will be performed in accordance with a QA program complying with ANSI N45.2 (Reference 5.11)

3.3 Summary of Crane Component Evaluation

A review of the component failure analysis provided by the Whiting Corporation (Reference 5.12), which presents the safety factors for each of the crane components, indicates that in addition to the bridge girders, the rod eye and the load cell brackets are the next two critical components. The rod eye and the load cell bracket have safety factors (to yielding) of 3.3 and 3.1, which represents a yield capacity of 412 and 387 tons, respectively.

3.4 Summary of Bounding Reaction Load Between the Crane Bridge Girder and the Strongback

The deformations of both the strongback and the box girder were analyzed to determine load on the crane. All results are based on an assumed rigid-perfectly-plastic material with a flow stress of 70 ksi and a plane strain deformation mode. The combination of these assumptions is conservative for the purpose of calculating an upperbound for the load. A rigid-perfectly-plastic solution always gives an upperbound to the actual load unless the geometry change is large. The conservatism of the plane strain assumption can be seen from the nature of the damage of the strongback which clearly indicates that the material plastically flowed in the thickness direction which would take a lower force to produce the depression than with the plane strain solution. In estimating the force required to produce the deformation, a lifted weight of ten tons was assumed.

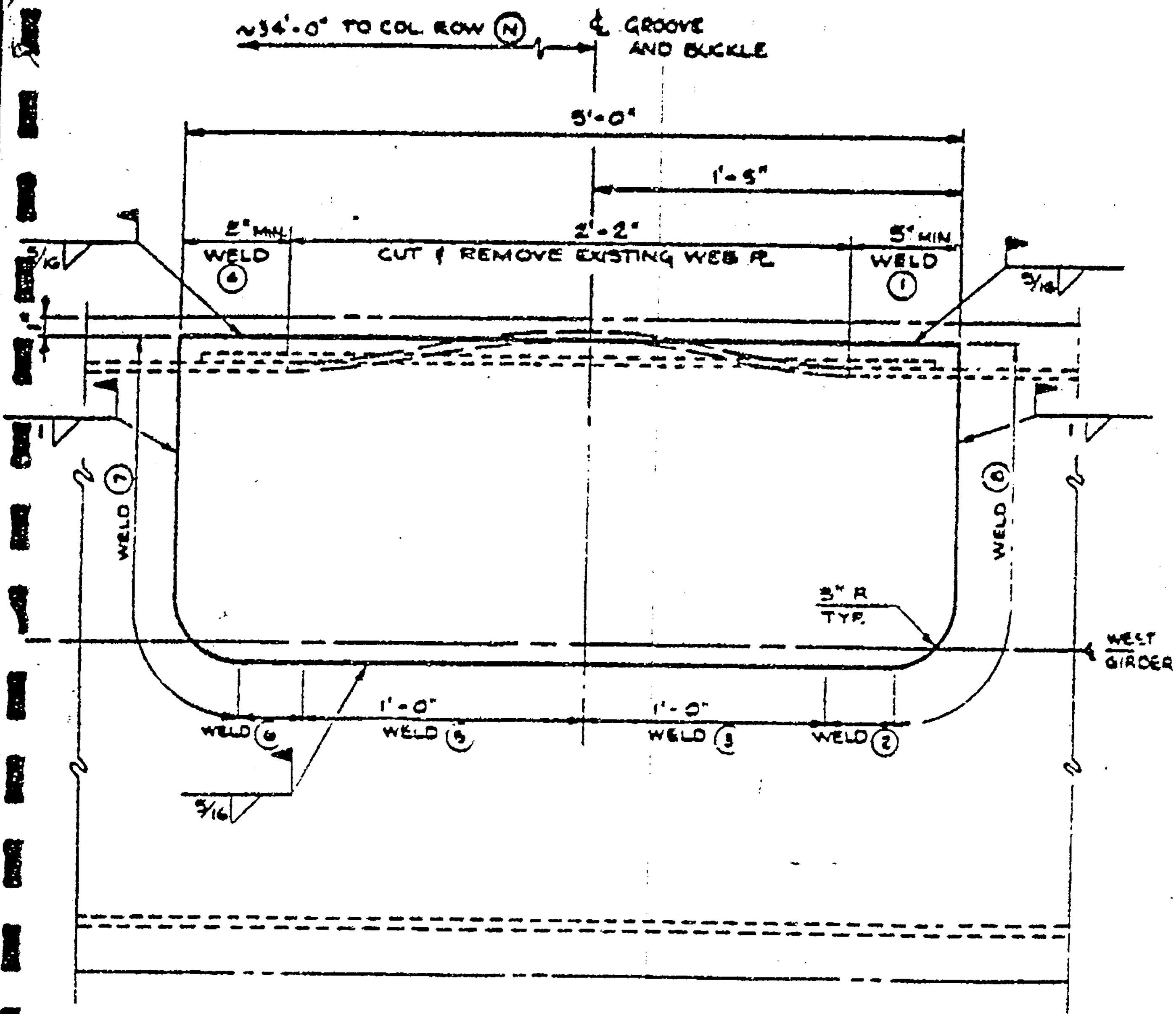
Two modes of deformation of the strongback were considered. The first mode assumed the box girder flange was rigid and cut into the strongback to the depths observed (Figure 3-7a). This would be somewhat unconservative since the small contact area would require a small load to produce permanent local deformation and in reality the flange bends increasing the contact area. The crane load for this case was calculated to be 130 tons. The second was that of a rigid plug, with a length equal to the entire length of the depression, being pushed into the strongback (Figure 3-7b). The crane load calculated was 500 tons. This would be an unrealistically conservative number, since it ignores the flexibility of the crane bridge girder.

The observed deformation of the box girder is too complex to determine the actual load applied. An upperbound load was calculated by assuming that the ultimate shear stress acted through the thickness all along the indentation in the flange of the box girder (Figure 3-7c). The calculated crane load for this case was 360 tons.

The final conclusions of these calculations is that the crane experienced a load of greater than 130 tons and less than 360 tons. The latter value is about 2.9 times the design value of 125 tons. The minimum factor of safety (to elastic limit), as provided in the Whiting Corporation's Component Failure Analysis (Reference 5.12) indicate that there was enough safety margin (greater than three), to conclude that the crane components were not stressed beyond yield, and hence no permanent deformations have resulted in any of the crane components.

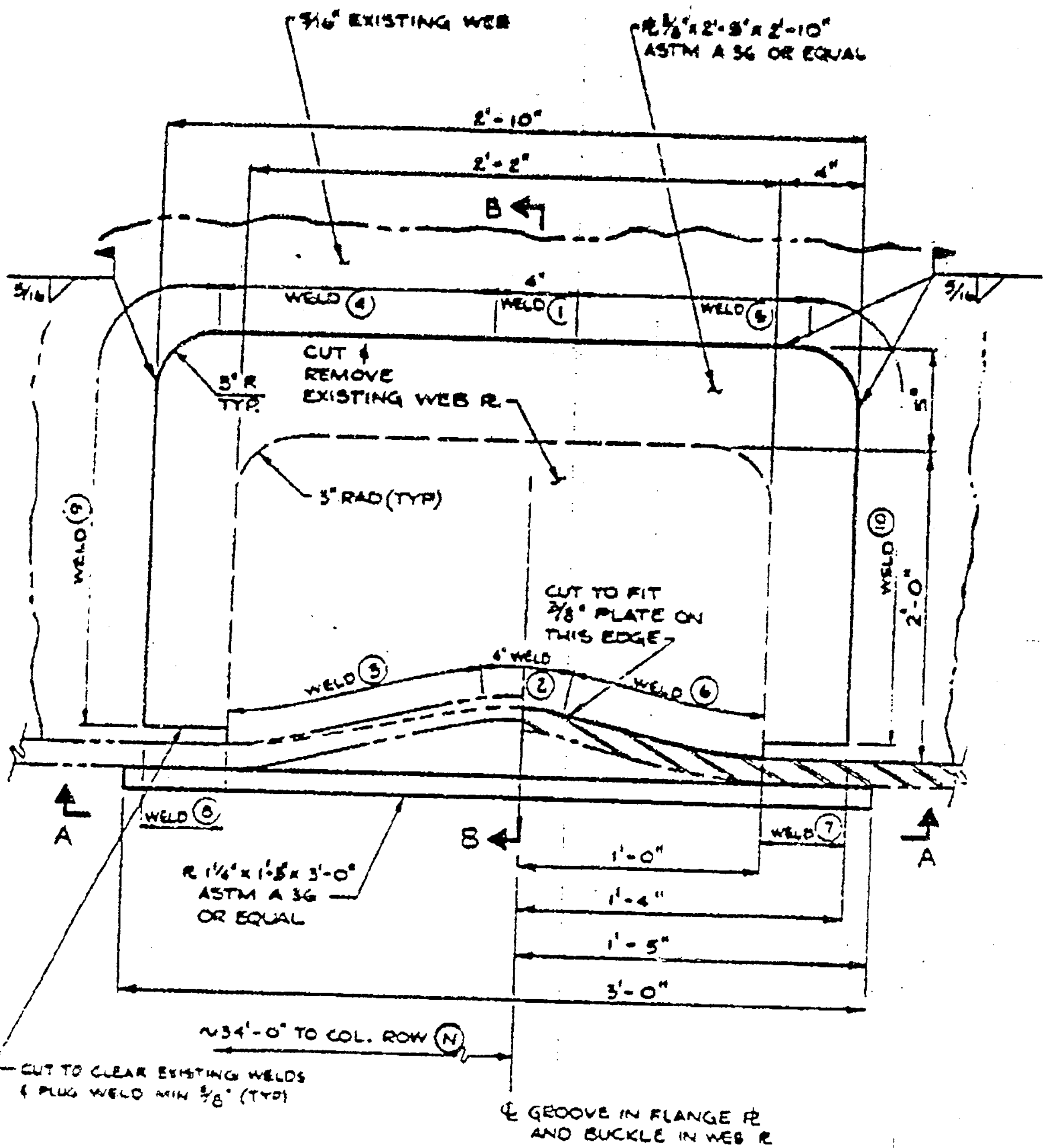
In order to further justify these conclusions, CECo must inspect the rod eye and load cell bracket to verify that no damage occurred on these components. Based on a successful examination of the rod eye and load cell brackets, CECo can be assured that the crane components were not overloaded so as to cause any permanent deformations.

NUTECH Document Number COM-21-014 (Reference 5.13) presents the inspection plan for these two crane components.



PLAN VIEW(A-A) WEST GIRDER

FIG. 3-2



WEST GIRDER REPAIR PLATES

FIG. 3.1

NOTES: 1) ALL WELDING MATERIAL SHALL BE AWS A5.1 E7018 ELECTRODE.

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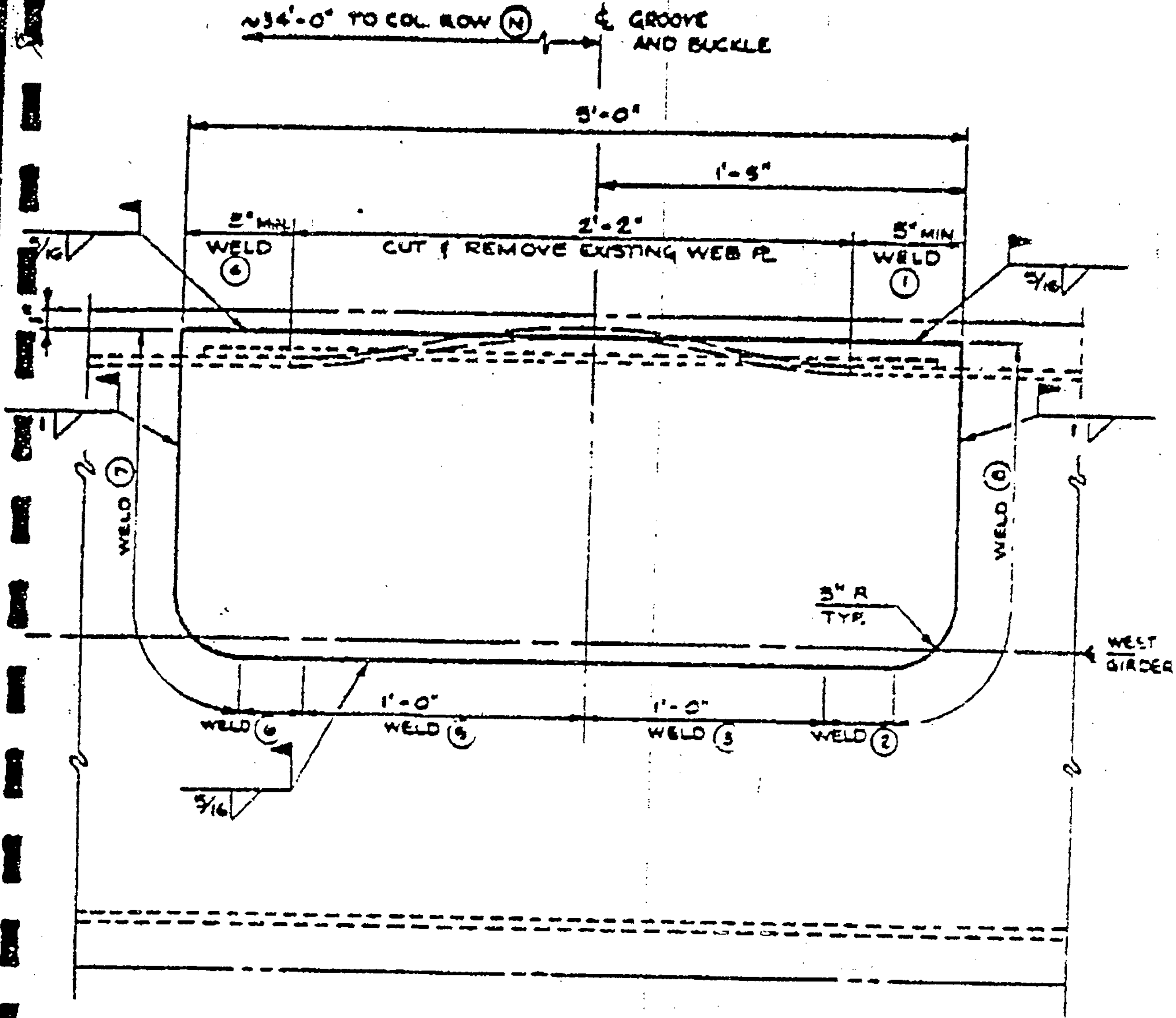
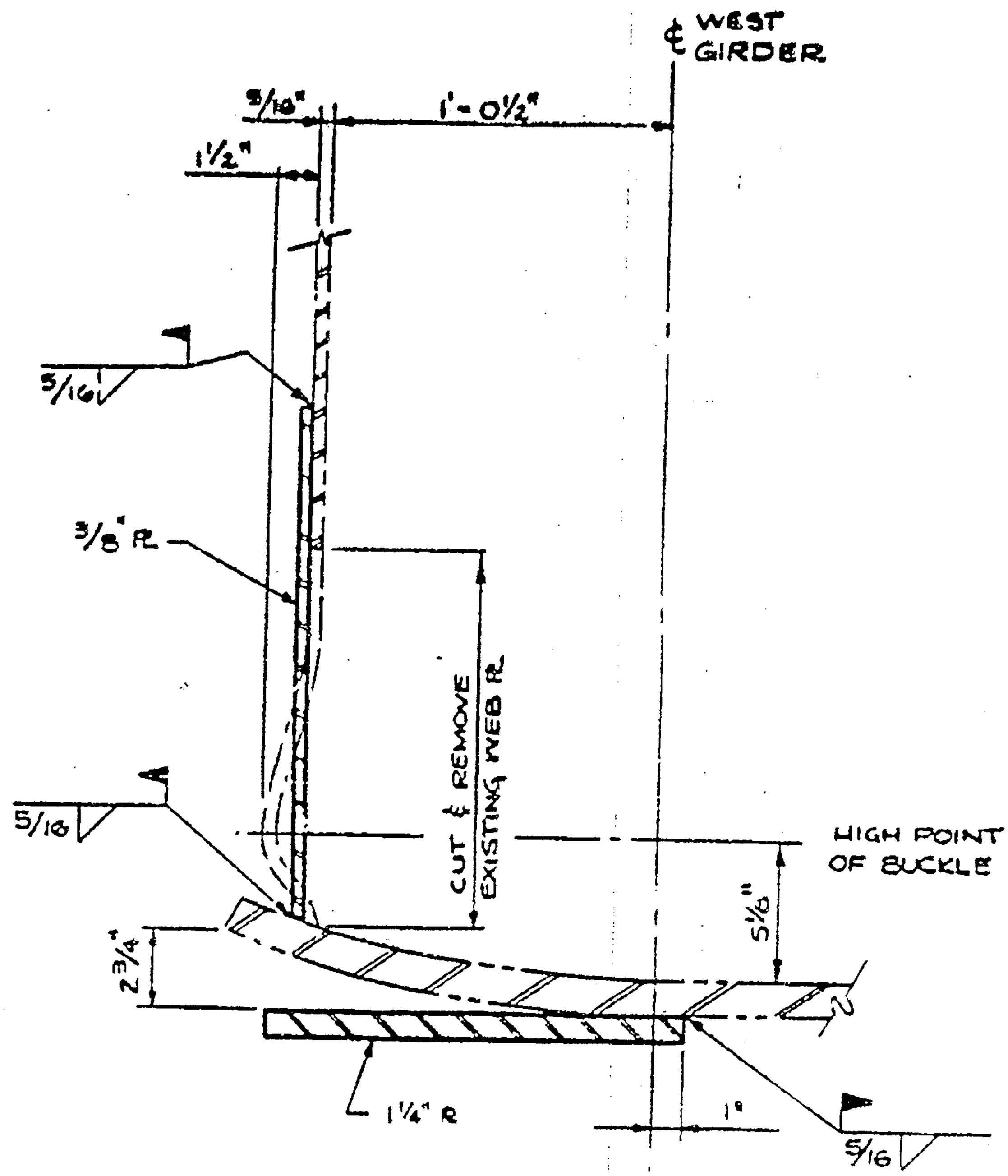


PLAN VIEW(A-A) WEST GIRDER

FIG. 3-2



SECTION VIEW (B-B) WEST GIRDER

FIG. 3-3

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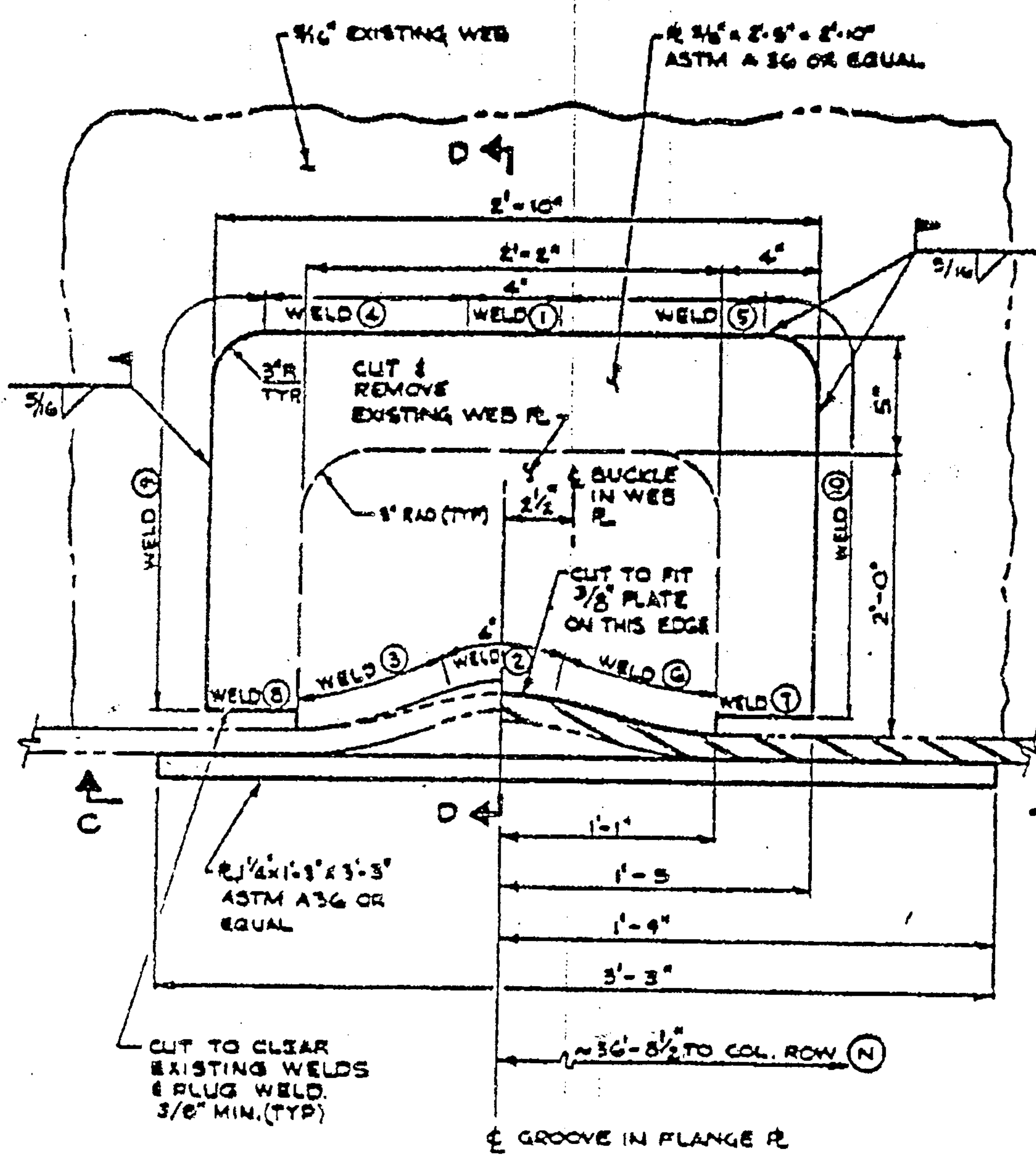

EAST GIRDER REPAIR PLATES
FIG. 3-4

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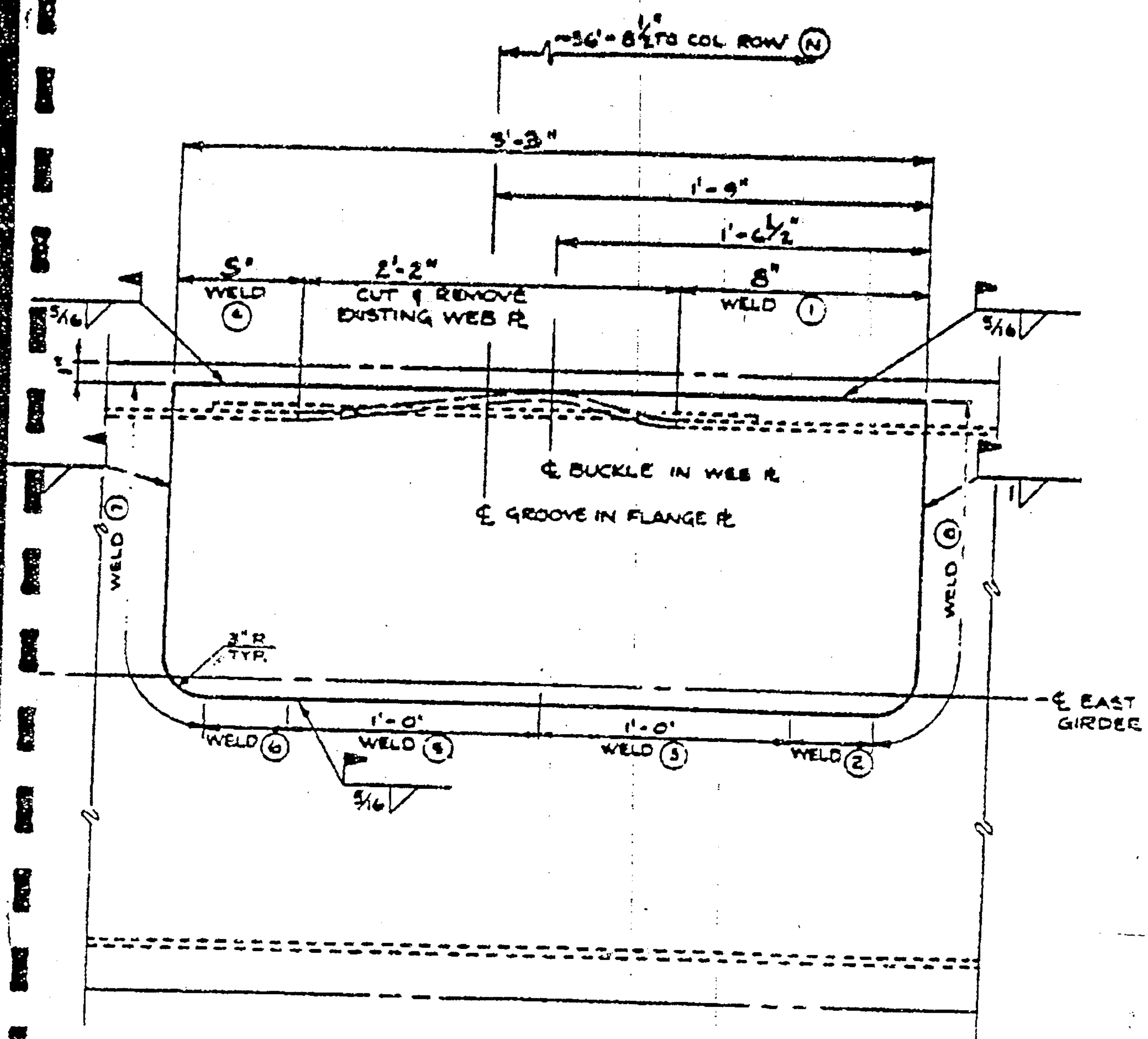
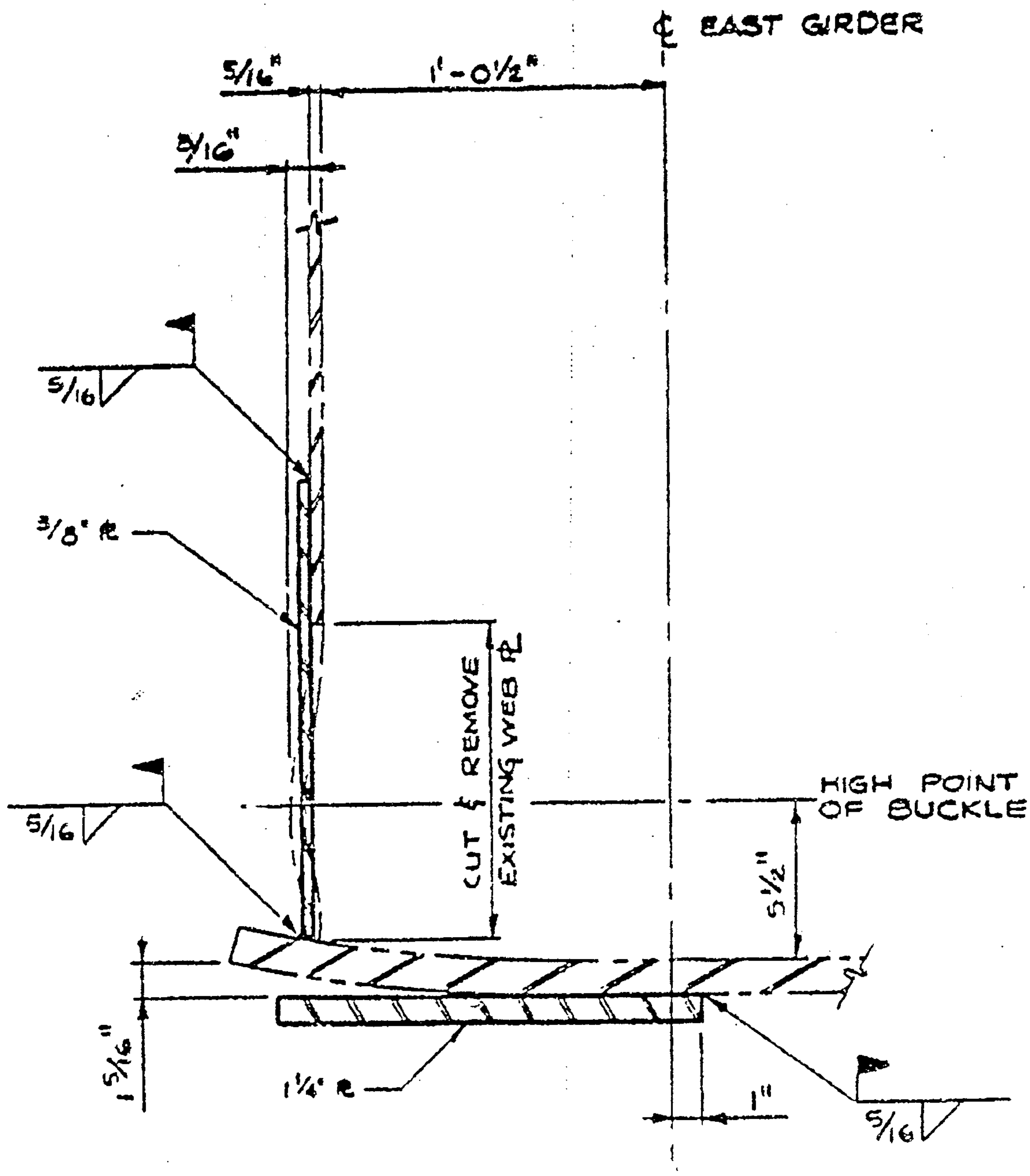


PLAN VIEW (C-C) EAST GIRDER
FIG. 3-5



SECTION VIEW (D-D) EAST GIRDER

FIG. 3-6

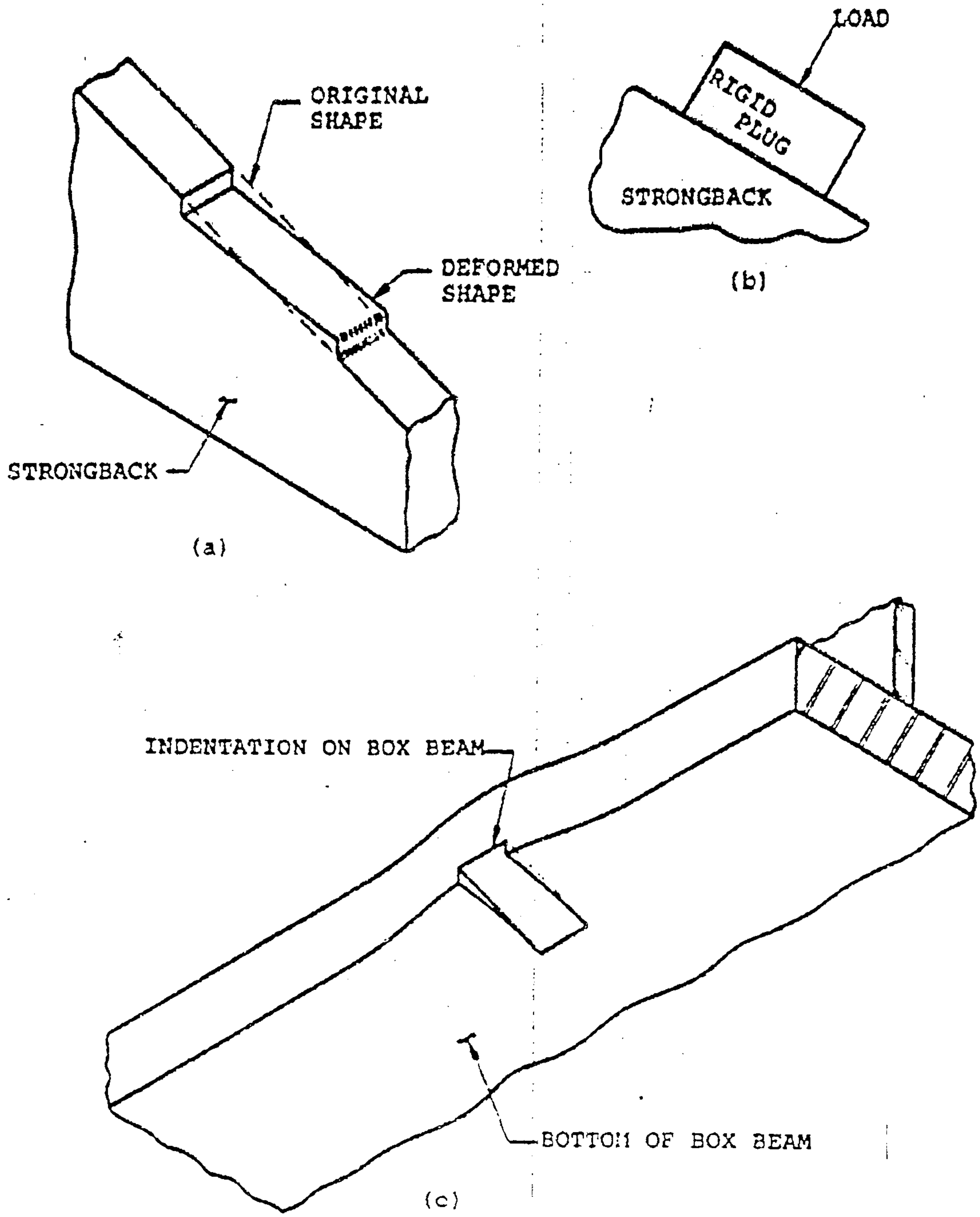


FIGURE 3-7 - MODELS USED TO CALCULATE LIMITING CRANE LOAD

TABLE 3-1

COMPARISON OF BRIDGE GIRDER MAXIMUM STRESSES TO ALLOWABLE STRESSES

Component and Stress		Maximum Stress (ksi)	Allowable Stress (ksi)	
West Girder	OBE	19.737	21.600	(0.6Fy)
(Combined Bending)	DBE	24.348	32.400	(0.9Fy)
East Girder	OBE	20.341	21.600	(0.6Fy)
(Combined Bending)	DBE	25.004	32.400	(0.9Fy)
Modified Web Plate	OBE	2.522	14.400	(0.4Fy)
(Weld Shear)	DBE	2.783	21.600	(0.6Fy)
Flange Cover Plate	OBE	1.605	14.400	(0.4Fy)
(Weld Shear)	DBE	0.666	21.600	(0.6Fy)

NOTES:

(1) Repair plates material shall be ASTM (Reference 5.14) A36.

(2) Welding material shall be AWS (Reference 5.15) A5.1 E7018 Electrode.

DOCUMENTATION

The following documentation, shall be maintained by the station to become part of the permanent record:

a. Material Documentation (Certified Material Test Reports)

b. As-built Drawings or Sketches

c. Nondestructive Examination Reports

d. Nondestructive Examination Procedures and Personnel Qualification

e. Welder Qualifications

f. Welding Procedures and Welding Procedure Qualifications

g. Repair Program

h. Crane Components Inspection Plan

i. Completed Station Traveler

5.0 REFERENCES

5.1 Sargent and Lundy's E.R. Weaver's letter to E. R. Zebus of CECo, dated July 20, 1981 (Project No. 6415.00) including July 13, 1981 Dresden Station visit notes.

5.2 American Institute of Steel Construction, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings", February 1969.

5.3 Sargent and Lundy specification K-2177 dated 2-21-66, "Specification for Travelling Bridge Crane".

5.4 Stress Analysis for Crane #9492 - an enclosure with letter dated October 29, 1974 from Whiting's W. M. Weaver to Sargent and Lundy's Frank Spakoski.

5.5 Specification for Electric Overhead Travelling Cranes - CMAA Specification #70 revised 1975 (supercedes EOCI Specification #61).

5.6 "Dresden Units 2 and 3 Reactor Building Crane Bridge Girder Repair Program", NUTECH Document COM-21-004 (Revision 1) dated August 1981.

5.7 American Welding Society (AWS), "Structural Welding Code", AWS D1.1-1981.

5.8 Commonwealth Edison Company Welding Procedure Number GS-AWS-1 (Revision 1) dated 6-17-77.

5.9 Safety Evaluation for Dresden Units 2 and 3 Reactor Building Crane and Hoist System (System #5800), NUTECH Document COM-21-002 (Revision 0) dated August 1981.

5.10 American National Standard, "Overhead and Gantry Cranes", ANSI B30.2.0-1976.

5.11 American National Standard, "Quality Assurance Program Requirements for Nuclear Facilities", ANSI/ASME N45.2-1977.

5.12 Whiting Corporation's Component Failure Analysis, 125/5 Ton Capacity Reactor Room Crane for Commenwealth Edison Company, 63688 - Dresden Nulear Power Station, dated 11-4-74.

5.13 "Dresden Units 2 and 3 Reactor Building Crane Components Inspection Plan", NUTECH Document Number COM-21-014 (Revision 0) dated September 1981.

COMMONWEALTH EDISON COMPANY

CALCULATION NO. DRE98-0020	PROJECT NO. 10128-099	PAGE NO. 880
REVISION NO. 0		
PREPARED BY: Hani alnakib H. Al-Nakib DATE: 3/16/98	REVIEWED BY: A-AlDabbagh A. AL-Dabbagh	DATE: 3/16/98

DRESDEN REACTOR BUILDING STEEL SUPERSTRUCTURE INTERACTION SUMMARY

No.	Element	Normal Snow+ Lift	Wind Snow+Lift+ Wind	OBE (Design Basis)	SSE (Design Basis)	SSE + Lift (Beyond Design Basis)
1	Interior Crane Column Members (W14x119/ W24x145) (H/N / 39 - 49)	0.992 (Pg. 778)	0.90 (Pg. 808)	0.71 (Pg. 467)	0.65 (Pg. 589)	0.80 (Pg. 715)
2	Interior Building Column Members (W24x145) (H/N / 39 - 49)	0.996 (Pg. 776)	1.00 (Pg. 807)	0.83 (Pg. 463)	0.86 (Pg. 585)	1.00 (Pg. 711.1)
3	Interior Crane/Building Column Base Connections (H/N / 39 - 49)	0.79 (Pg. 785)	Normal controls	0.67 (Pg. 482)	0.80 (Pg. 596)	0.95 (Pg. 730)
4	Exterior Column Members (W24x76) (Rows 38 & 50, except Rows H & N)	0.14 (Pg. 787)	OBE Controls	0.27 (Pg. 494)	0.18 (Pg. 618)	0.18 (Pg. 618)
5	Exterior Column Base Connections (Rows 38 & 50, except Rows H & N)	0.22 (Pg. 786)	OBE Controls	0.83 (Pg. 490)	0.97 (Pg. 612)	0.96 (Pg. 738)
6	Corner Crane Column Members (W14x119/ W24x145) (H-38, H-50, N-38, N-50)	OBE Controls	OBE Controls	0.40 (Pg. 437)	0.37 (Pg. 557)	0.50 (Pg. 683)
7	Corner Building Column Members (W24x145) (H-38, H-50, N-38, N-50)	OBE Controls	OBE Controls	0.40 (Pg. 436)	0.42 (Pg. 556)	0.55 (Pg. 682)
8	Corner Crane/Building Column Base Connections (H-38, H-50, N-38, N-50)	OBE Controls	OBE Controls	0.88 (Pg. 443)	0.88 (Pg. 564)	0.86 (Pg. 690)
9	Vertical Bracing Members Column Rows 38/50	N/A	OBE Controls	0.41 (Pg. 324)	0.51 (Pg. 629)	0.51 (Pg. 755)
10	Vertical Bracing Connections Column Rows 38/50	N/A	OBE Controls	0.61 (Pg. 510)	0.87 (Pg. 635)	0.88 (Pg. 761)
11	Vertical Bracing Members Column Rows H / N	N/A	OBE Controls	0.49 (Pg. 383)	0.61 (Pg. 622)	0.61 (Pg. 748)
12	Vertical Bracing Connections Column Rows H / N	N/A	OBE Controls	0.50 (Pg. 503)	0.73 (Pg. 628)	0.73 (Pg. 754)
13	Roof Truss Members (Double Angles)	~~N/A~~	OBE Controls	0.75 (Pg. 513)	0.90 (Pg. 638)	Same as SSE
14	Roof Truss Members (W14x48)	0.33 (Pg. 792)	OBE Controls	0.72 (Pg. 530)	0.74 (Pg. 656)	Same as SSE
15	Roof Truss Members (Plate Girders)	1.05 (Pg. 797)	0.83 (Pg. 811)	0.95 (Pg. 537)	0.64 (Pg. 663)	Same as SSE
16	Roof Truss Members (W24x68)	0.11 (Pg. 793)	OBE Controls	0.50 (Pg. 533)	0.55 (Pg. 659)	Same as SSE
17	Roof Truss Members (W14x30)	0.56 (Pg. 790)	OBE Controls	0.71 (Pg. 528)	0.64 (Pg. 654)	Same as SSE
18	Roof Truss Connections (Double Angles)	N/A	OBE Controls	0.88 (Pg. 514)	0.90 (Pg. 646)	Same as SSE
19	Roof Truss Connections (W14x48)	0.11 (Pg. 792)	OBE Controls	0.45 (Pg. 531)	0.56 (Pg. 657)	Same as SSE
20	Roof Truss Connections (Plate Girders)	0.48 (Pg. 799)	OBE Controls	0.33 (Pg. 541)	0.22 (Pg. 668)	Same as SSE
21	Roof Truss Connections (W24x68)	0.013 (Pg. 794)	OBE Controls	0.61 (Pg. 535)	0.76 (Pg. 661)	Same as SSE
22	Roof Truss Connections (W14x30)	0.113 (Pg. 791)	OBE Controls	0.31 (Pg. 529)	0.39 (Pg. 655)	Same as SSE
23	Crane Girder Member	0.93 (Pg. 857)	0.93 (Pg. 857)	0.63 (Pg. 831)	0.68 (Pg. 839)	0.78 (Pg. 847)
24	Crane Girder Connections	0.98 (Pg. 875)	0.98 (Pg. 875)	0.39 (Pg. 875)	0.48 (Pg. 876)	0.62 (Pg. 876)

NOTES:

1. This table provides Interaction Coefficients for Critical Members of the Reactor Building Superstructure and the page number of Calculation DRE98-0020 where the Interaction Coefficient can be found. Interaction Coefficients are listed for each applicable Loading Combination.

2. Interaction Coefficients (IC) = (Actual Stress) / (Allowable Stress)

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

April 30, 2002

MEMORANDUM TO: Ross Landsman, Project Engineer
Division of Nuclear Materials Safety

FROM: J. E. Dyer for James P. Caldwell
Regional Administrator

SUBJECT: RESOLUTION OF DIFFERING PROFESSIONAL VIEW ON STARTUP OF CASK STORAGE LOADING CAMPAIGN AT DRESDEN UNITS 2 AND 3

I have reviewed the report of the Differing Professional View (DPV) panel concerning the cask loading campaign at Dresden Units 2 and 3 which you filed on May 23, 2001, but which was held in abeyance with your concurrence until July 11, 2001. A copy of the panel's April 2, 2002, memorandum to me and report are attached. I agree with the panel's conclusions on the issues addressed and am implementing the panel's recommendations with the modifications discussed below.

The panel recommended further action to develop information on six issues. Specifically, the panel recommended inspection for issue 1.b (reactor building structural components exceeding yield under SSE loads; issue 1.c (overstress of reactor building structural components); and additional issue 2 (operation and testing of the load cell). The panel recommended obtaining additional written information from the licensee on the other three issues: issue 1.a (reactor building design); issue 3 (weld quality of the Cask Transfer Facility (CTF)); and additional issue 5 (trolley analysis).

The licensee is responsible for addressing all six of these issues. In this regard, NRR issued a Request for Additional Information (RAI) dated February 26, 2002, to Exelon which requested the licensee to specifically address concerns which included the substance of issues 1.a and additional issue 5. The licensee submitted its response to NRR in a letter dated April 12, 2002. In addition, issues 1.b and 1c. were discussed on April 18, 2002, in a conference call among NRR, Region III, and the licensee pertaining to the licensee's response to the RAI as it relates to the seismic analysis of the reactor building super-structure (issue 1.a). The NRC is reviewing the licensee's RAI response. Additional issue 2 will be addressed as part of the inspection follow-up for the unresolved issue associated with the licensee's load cell calibration and testing methodology described in Inspection Report 07200037/2001-002 (DNMS). I am modifying the recommendation for inspection of issues 1.b. and 1.c, as recommended by the panel. Specifically, upon completion of the RAI response review, the NRC will determine what follow-up action, including possible additional inspection, is warranted.

With respect to issue 3 (weld quality of the CTF), by copy of this memorandum, I am directing DNMS to coordinate the preparation of a letter to the licensee requesting a written response. \16

The letter should be issued by May 15, 2002, following appropriate coordination with NRR and/or NMSS and discussions with the licensee about the need for the requested information. Subsequent actions, including the inspection, will be considered following evaluation of the licensee's response. Additionally, I am directing DNMS to review and initiate, if determined appropriate, enforcement action on those issues identified in the panel report as potentially warranting such action.

I am further requesting DNMS to provide you with a copy of the letter to the licensee, an explanation if issuance of the letter is delayed beyond the above date, and copies of the licensee's response and any additional enforcement action resulting from our review of these issues.

I appreciate and commend your willingness to utilize the DPV process. I am aware that we did not meet the timeliness goals for resolution of your DPV specified in Management Directive (MD) 10.159, but I understand that you were advised of the reasons for the delay, i.e., NRC's response to September 11th, the need for input from NRR and the Spent Fuel Project Office, and the relationship of the DPV to the ongoing backfit analysis on Dresden dry cask transfers issues under review by NRR. In accordance with the MD, a summary of the issue and its disposition will be included in the Weekly Information Report to advise interested employees of the outcome. DPVs are not normally made available to the public. However, if you would like to have your DPV case file made public, with or without the release of your name, please contact Bruce Berson.

Our review of your DPV is now considered complete. Should you wish, you may now initiate the Differing Professional Opinion process as described in Management Directive 10.159.

Attachment: As stated

cc w/o att: C. Pederson, DNMS

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

April 2, 2002

Rec'd 4/3 605PM PLDyer

MEMORANDUM TO: J. E. Dyer, Regional Administrator

FROM: J. A. Grobe, Director, Division of Reactor Safety

SUBJECT: RECOMMENDATION OF AD HOC REVIEW PANEL FOR DIFFERING PROFESSIONAL VIEW: STARTUP OF CASK STORAGE LOADING CAMPAIGN AT DRESDEN UNITS 2 AND 3

In accordance with your memo dated July 20, 2001, to me (Reference 1), an Ad Hoc Differing Professional View (DPV) Review Panel (Panel) was formed in accordance with Management Directive 10.159 with myself as Chairman and John Jacobson and Patrick Hiland as members. The Panel reviewed several issues related to the loading and handling of spent fuel dry storage casks at the Dresden facility. The purpose of this memorandum is to provide you with the Panel's review, conclusions, and recommendations for this DPV. The schedule for resolution of this DPV was protracted due to the NRC's response to the September 11, 2001, event, the need for input on several complex technical and licensing basis issues from the Office of Nuclear Reactor Regulation (NRR) and the Spent Fuel Project Office, and the nexus between the DPV issues and a backfit analysis Task Interface Agreement on Dresden dry cask transfer issues under review by NRR.

The DPV addressed three main issues related to the Reactor Building and Cask Transfer Facility (CTF). The first issue concerned the integrity of the Reactor Building structure with respect to design basis loading conditions and loads associated with a cask lift. The second issue concerned the compliance of the Cask Transfer Facility to applicable codes and standards. The third main issue concerned the quality of some welds on the CTF. The DPV also addressed six issues related to the Reactor Building crane. These issues (Reference 2) were developed through review of various documents including the draft and final reports (References 1 and 3) and several meetings with the Submitter. The summary of the issues (Reference 2) was compiled by the Panel and provided to the Submitter. The Submitter acknowledged that the summary adequately captured his concerns.

During the review of this DPV, the Panel met on several occasions, interviewed the Submitter, interviewed key Region III managers (Reference 10), and conducted several telecons with both NMSS and NRR staff and management. Written responses were requested (Reference 4) and received (References 5, 6, and 7) for portions of the three main issues.

The Panel did not identify any immediate safety concerns regarding dry cask movement activities at Dresden. The Panel did identify several regulatory and compliance issues warranting further staff consideration. The Panel's review, conclusions, and recommendations are discussed in the attachment.

PANEL RESULTS OF DPV REVIEW

SUBSTANTIVE ISSUES

1.a The reactor building design for Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) load cases did not include the 125 ton crane load (live load) as described in the Updated Final Safety Analysis Report (UFSAR).

REVIEW

The first issue raised by the Submitter was that while the Normal and Wind load analyses for the Reactor Building included the 125T crane load, the analyses for the OBE and SSE load cases did not include the crane load. The Submitter contended that the UFSAR requires that the crane load be included in the OBE and SSE analyses. The licensee's position, presented during a meeting in RIII on May 23, 2001(Reference 8), was that the Dresden design basis did not include consideration of the crane load for the OBE and SSE analyses. The licensee also presented the results of a "beyond design basis" analysis for the SSE load case which did include the crane load. The licensee indicated that results were acceptable. This is discussed in the DNMS inspection report (Reference 3). The Submitter was in attendance at that meeting.

Because it was licensed early, Dresden Unit 2 was included in the Systematic Evaluation Program (SEP). The SEP reviewed the seismic design of Dresden Unit 2 under SEP Topic III-6, "Seismic Design Considerations." The SEP reviewed load combinations under SEP Topic III-7.B, "Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria." The results of the SEP Topic III-6 review is reported in NUREG/CR-0891, "Seismic Review of Dresden Nuclear Power Station - Unit 2 for the Systematic Evaluation Program," dated April 1980 and in the SEP Topic III-6 Safety Evaluation for Dresden Unit 2 dated June 30, 1982. The SEP seismic review only evaluated the Safe Shutdown Earthquake (SSE) seismic design. SEP Topic III-6 identified no open items related to crane live loads and the reactor building structural design.

The load combinations used in the design of Dresden 2 for the reactor building and all other Class I structures are listed in Table 4-4 of NUREG/CR-0891 as D+R+E and D+R+E' where D = Dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads or operating pressures and **live loads expected to be present when the plant is operating** [emphasis added], E = Design earthquake load, and E' = Maximum earthquake load. The SEP Topic III-6 safety evaluation does not specifically state that the SEP considered that heavy loads on the reactor building crane were loads expected to be present when the plant is operating. The SEP review used the Standard Review Plan (SRP), NUREG-75-087, as the basis for its review. Section 3.8.4 of the 1975 SRP gives load combinations consistent with Table 4-4 of NUREG/CR-0891 although it breaks down D into D (dead loads) + L (live loads). SRP Section 3.8.4 defines L as "Live loads or their related internal moments and forces including any movable equipment loads and other loads which may vary with intensity and occurrence, such as soil pressure." SRP 3.8.4 allows deviations from the acceptance criteria for loads and load combinations if the deviations have been adequately justified. NRC did not identify any justifications in the Dresden licensing basis for excluding reactor building crane lifted loads.

NRC completed its review of SEP Topic III-7.B and issued an SE by letter dated August 23, 1990. With respect to the crane live load, NRC's contractor stated in TER-C5506-425 dated November 15, 1983, that the reviewers did not have access to actual design calculations. Also, we have not identified any lists of actual loads. Therefore, it does not appear that NRC or its contractor reviewed individual live loads in their review of Topic III-7.B. With respect to OBE seismic evaluations, the licensee identified in its letter to the NRC dated August 2, 1982, that Sargent & Lundy reactor building superstructure calculations did not include OBE loads but that it was Sargent & Lundy's judgement that the SSE evaluation would control the reactor building superstructure structural evaluation.

The Dresden Units 2 and 3 Reactor Building (including superstructure) licensing basis is described in the UFSAR as follows: UFSAR Section 3.2.1 classifies the Reactor Building as a Class 1 structure. UFSAR Section 3.8.4 defines the load combinations for Class 1 structures to include the dead load plus **live loads expected to be present when the plant is operating** [emphasis added] plus the OBE load (E) for the OBE case or the SSE load (E') for the SSE case.

In preparation for beginning a campaign of spent fuel transfers, Sargent and Lundy performed an extensive evaluation for the licensee (calculation DRE98-0020) (Reference 13) to analyze and evaluate the building superstructure during various loading conditions including OBE (without live load) and SSE (with live load). The licensee states that this calculation includes the loads from the SSE plus the effects of the maximum lifted load of 125T. The effects of the lifted load on the structure include the application of the load vertically as well as the pendulum effects of the lifted load during a SSE hanging from the crane during a seismic event. We note, however, that the licensee refers to SSE plus lifted load as "beyond design basis" although the NRC staff considers SSE plus lifted load to be within the licensing basis if the crane is being used to lift loads while the plant is operating.

CONCLUSION

The UFSAR correctly describes the licensing basis for the Reactor Building as dead loads, plus live loads expected to be present when the plant is operating, plus the seismic load, for both the OBE and SSE load cases. If the licensee intends to lift spent fuel casks when the plant is operating, the spent fuel cask is then a live load expected to be present on the Reactor Building crane when the plant is operating. Therefore, the licensing basis of the plant requires analysis of OBE plus lifted loads and SSE plus lifted loads for the Reactor Building structure.

RECOMMENDATIONS

Notify the licensee that the design basis of the plant requires that both the OBE and SSE load cases for the Reactor Building be analyzed with the 125T (or actual) crane load present if casks (or other heavy loads) are to lifted when the plant is operating. NRC should consider the potential enforcement aspects of this issue if spent fuel casks have been lifted in the past when the plant was operating prior to performing the required analysis.

1.b Calculations indicate that some Reactor Building structural components exceed both yield and ultimate tensile strength for the SSE load case.

REVIEW

There is a long history of calculations which show multiple Reactor Building structural members and connections to be outside design limits (several examples are described in Inspection Report 2001-002(DNMS). For example, Dresden Calculation No. DRE98-0013 is discussed as showing some crane support girders, interior building columns, and roof truss members exceed design allowable stress limits. The licensee concluded that the overstress was acceptable based on probabilistic considerations. Dresden Calculation No. DRE98-0020 (Reference 13) indicates some roof truss members exceed design allowable stress limits by 5%. The licensee accepted these results based on "normal practice to accept overstress of up to 10%" (Reference 8). Unresolved Item 05 in Inspection Report 2001-002 (DNMS) which follows the discussion of the overstress conditions does not directly address the design compliance issue, rather "long term acceptability of this equipment for handling large numbers of dry fuel storage casks". Unresolved Item 06 addresses the licensee's practice of accepting a 10% overstress condition however, the unresolved item does not address the acceptability of the licensee's use of a probabilistic approach to resolution of design issues.

CONCLUSION

Apparently, the licensee has calculations which indicate that some Reactor Building structural members do not conform to the design allowable limits. All calculations of record showing loads beyond design limits must be reconciled and documented. For example, for the SSE load case, the licensee may elect to use the Limit-Design approach.

With respect to the acceptance of 10% overstress, the Panel is not aware of any recognized code or standard which supports this practice. If the licensee's design practices or methodology inherently includes greater than 10% margin with respect to design, it is up to the licensee to demonstrate and document this. Regarding the use of probabilistic considerations to resolve overstress conditions, the Panel is not aware of any Agency approvals supporting this approach to resolve overstress conditions. If the licensee uses this approach, they need to justify the basis. Typically, these issues are resolved by refining the calculations (removing demonstrated conservatism) or, if necessary, through modifications.

RECOMMENDATIONS

Further inspection should be conducted to verify satisfactory resolution of identified overstress conditions. Evaluation of licensee actions should also be conducted to assess compliance with the requirements of 10 CFR Part 50, Appendix B, Criteria III and XVI.

1.c A 1998 calculation indicates an overstress of reactor building structural components of five percent. The applicable code does not allow any overstress conditions. In addition, the inspection report documents only a three percent overstress.

REVIEW

The 1998 calculation DRE98-0020 shows Rx Bldg structural members to exceed allowable stress by 5% for the normal load case. This is a specific example of the problem stated in 1.b above. This overstress was incorrectly presented by the licensee as 3% (Reference 8) during the May 23, 2001 licensee presentation in RIII and subsequently documented incorrectly in an NRC inspection report (Reference 3).

CONCLUSION

The calculation documents a 5% overstress with respect to design allowable stress levels. All calculations of record showing loads beyond design limits must be reconciled and documented. The licensee's May 23, 2001 presentation slides indicate that it is normal practice to accept overstress of up to 10%. With respect to the acceptance of 10% overstress, the Panel is not aware of any recognized code or standard which supports this practice. If the licensee's design practices or methodology inherently includes greater than 10% margin with respect to design, it is up to the licensee to demonstrate and document this.

RECOMMENDATIONS

Further inspection should be conducted to verify satisfactory resolution of identified overstress conditions. Evaluation of licensee actions should also be conducted to assess compliance with the requirements of 10 CFR Part 50, Appendix B, Criteria III and XVI.

2.a The cask lifting yolks for both the CTF and the Unit 2/3 crane do not meet ANSI N14.6 standards as required by the Certificate of Conformance.

REVIEW

Since the cask lifting yoke did not include a latching device, the Submitter questioned the basis for concluding that the cask transfer yoke met the licensing requirements. The Certificate of Compliance (CoC) (Reference 9) for the Cask Transfer Facility (CTF) requires the device to be single failure proof, and the application states that no single failure will result in a dropped load. Further, the CoC states that the device must meet NUREG-0612 which requires that special lifting devices meet ANSI N14.6. The cask lifting yokes are special lifting devices. ANSI N14.6 indicates that, if it is possible for a load carrying component to become disengaged, it shall be lifted with a latching device with an actuating mechanism that securely engages and disengages. The licensee's purchase specification and the CoC require that the lifting yokes on the CTF and the Reactor Building crane meet ANSI N14.6. Inspection Report 07200037/2001-002(DNMS) (Reference 3) documented that the Spent Fuel Project Office did not attempt to determine how the yokes met the ANSI provisions, but instead, focused on whether any of the provisions were violated (pg. 21, Reference 3).

The panel requested the staff (Reference 4) to provide the basis for the conclusion that the cask lifting yokes meet the licensing basis requirements. The staff response, documented in Reference 6, states that the ANSI N14.6 (1978) contains two provisions that allow the CTF design not to utilize a latching mechanism. As stated in the ANSI N14.6, Section 3.3.5 and 3.3.6, a latching mechanism is required if the "Load-carrying components that **may become** [emphasis added] inadvertently disengaged" or "An actuating mechanism shall be used, **if needed**, [emphasis added]...." The staff responded that for normal lifting operation, the cask is not subject to any lateral load, thus it is not possible for the yokes to become disengaged from the cask trunnions. Additionally, the staff concluded that for seismic events, the cask is pin-supported in a pendulum like configuration, suggesting that the cask will not be subject to any meaningful lateral force.

CONCLUSION

The Panel concurs with the staff's conclusion that the cask lifting yokes appear to meet the licensing basis without a latching device.

RECOMMENDATIONS

None.

2.b The CTF lift platform beam does not meet the single failure proof criteria of NUREG-0554.

REVIEW

The Submitter questioned whether an adequate basis was provided by the licensee to conclude that the CTF lift platform beam satisfied single failure proof requirements. The staff's overall safety evaluation for the design and testing of the Cask Transfer Facility, including the lift platform is referenced in Inspection Report 07200037/2001-002(DNMS), dated August 13, 2001. As part of the staff's safety evaluation (Reference 12), a detailed assessment of the single failure proof design of the lift platform was performed. The staff concluded that "...the lift platform is conservatively designed and is, therefore, acceptable for the design service load of 280,000 lbs."

The panel reviewed the staff's safety evaluation with particular emphasis on the lift platform analysis. For completeness, the following excerpts from Reference 12 were reviewed by the panel:

> 3.2.1.1 Lift Platform Evaluation
>
> The lift platform is bolted at two ends to the screw jack nuts, which, in turn, are raised or lowered by turning the screw jacks against the nuts through a motor/shaft/gear assembly mounted on the CTF top bridge girder. Holtec reports the nut thread bending safety factors of 19 and 48 against F_y and F_u, respectively. The reported nut thread shear safety factors are 50 and 194. These safety factors are more than adequate to satisfy the intent of NUREG-

0612 guidelines to improve the reliability of the handling system through increased factors of safety in certain active components. The lift platform serves a structural support function equivalent to that of a crane bridge girder. CMAA 70 states, "The crane girders shall be welded structural steel box sections, wide flange beams, standard I-beams, reinforced beams, or box sections fabricated from structural shapes." The staff notes that the bridge girder should be conservatively designed but need not be considered single failure proof, in accordance with NUREG-0554. In the following, the staff compares safety factors inherent to the Subsection NF, Level A stress allowables to those of crane industry standards. By considering the stress "design margins" presented in the Holtec report, the staff then computed the overall safety factor to demonstrate that the lift platform is conservatively designed.

Inherent Safety Factors. Using the common structural steel A-36 (F_y = 36 ksi) as a basis, the stress allowable, specified as a fraction of the yield strength, and the inherent safety factor (ISF), defined as the inverse of this fraction, are computed and listed below for the basic tension/compression and bending stress categories considered by three industry standards.

Standard (Bridge Girders)	Basic Tension/Comp		Bending Stress	
	Allowable	ISF	Allowable	ISF
CMAA 70	0.6 F_y	1.67	0.6 F_y(1)	1.67
Subsection NF, Level A	14.5 ksi(2)	2.48	21.75 ksi(3)	1.66
ASME NOG-1(4)	0.5 F_y	2.0	0.49 F_y(5)	2.04

Notes:

1. Not specified explicitly for bending, but used the basic tension/compression allowable
2. ASME Section II, Part D, Table 1A; 14.5 ksi = 0.40 F_y, approximately
3. Bending allowable = tension/compression allowable x 1.5 (21.75 ksi = 14.5 x 1.5)
4. "Rules for Construction of Overhead and Gantry Cranes," which includes cranes with single-failure-proof features
5. Section NOG-4313: AISC stress allowable (0.66 F_y) divided by 1.12N, where N=1.2 for operating loads

For bending stresses, which usually govern a design, the comparison table above shows that ISFs are essentially identical for the CMAA 70 and the ASME, Subsection NF, criteria. The staff notes that, for the A-36 steel, compared to the CMAA 70 or Subsection NF standard, the ISF, per NOG-1, is about 23% larger for bending stresses.

The staff notes further that all structural steel design ISFs are smaller than the basic safety factor of 3 against the yield strength associated with the mechanical design of the HI-TRAC and MPC Lifter components. This crane industry practice of adopting relatively smaller ISFs for bridge girders is consistent with the common structural steel design philosophy. It is risk informed and acceptable, recognizing that steel bridge girders undergo bounded deformation when overloaded, thereby providing sufficient advanced warning for necessary remedial actions.

Lift Platform Stress Design Margin. The Holtec report defines safety factor as the ratio of the allowable stress and the calculated stress; a safety factor greater than one is considered acceptable. For this evaluation, however, the staff considers Holtec stress safety factors as stress "design margins."

The Holtec lift platform is fabricated with the A-516 Grade 70 carbon steel with a yield strength of 38 ksi and bending stress allowable of 26.25 ksi in accordance with Subsection NF. For a service load of 280,000 lbs plus a 15% dynamic load effect, Holtec reports a minimum design margin of 1.45, which is greater than one. This design margin is above and beyond the ISF of 1.45 (38/26.25 = 1.45) for the A-516 Grade 70 steel although it is slightly smaller than the ISF of 1.66 for the A-36 steel discussed above.

Overall Safety Factor. The staff considers an overall safety factor (OSF), defined as the product of design margin and ISF, for comparing stress design adequacy associated with different design standards for the lift platform. The design margin of 1.45 and the ISF of 1.45 result in an OSF of 2.10 (1.45 x 1.45 = 2.10), on the basis of Subsection NF. As indicated in the ISF comparison table above, a stress design margin of greater than one, which is acceptable on the basis of the more conservative NOG-1 stress allowables, amounts to an OSF of greater than 2.04 (1.0 x 2.04 = 2.04). Thus, the lift platform based on the Subsection NF stress allowables and a design margin of 1.45 achieves an OSF of 2.10, which is greater than the minimum acceptable crane girder OSF standard of 2.04, per NOG-1, for a design margin of one. On this basis, the staff concludes that the lift platform is conservatively designed and is, therefore, acceptable for the design service load of 280,000 lbs."

CONCLUSION

The panel concurs with the staff's June 15, 2001, safety evaluation and determination that Dresden Cask Transfer Facility lift platform design is acceptable.

RECOMMENDATIONS

None.

3. Existing records are inadequate to establish weld structural quality for welds on the Cask Transfer Facility.

REVIEW

The issue raised by the Submitter was that the adequacy of individual CTF welds could not be verified based on a review of quality records. The CTF fabricator's Quality Assurance (QA) manager consolidated the weld inspection records into weld groups according to size. All welds for the entire CTF were signed off by the QA manager on the same day. Since original weld documentation is no longer available, welder identity and fabrication sequence could not be established. A specific example identified by the Submitter was a fabricator's non-conformance report (NCR-46), dated September 12, 2000, that documented an incorrect weld made on a box beam. While that particular weld was repaired, there are no records to indicate that the specific welder didn't make the same mistake on other box beams. As documented in NRC Inspection Report 2001-002(DNMS) (Reference 3) the fabrication welds were determined to be "proper" based on the licensee's assertion that all welds were inspected and identified discrepancies corrected; the documented results of Quality Control inspector activities (weekly Holtec Users Group reports); and the fabricator's QA manager's certification of the cumulative welding data.

The Panel believed that the documented evidence of welding and inspection activities would likely be insufficient for similar nuclear power plant welding for which 10CFR 50, Appendix B applied, and it requested the staff to provide the Panel with the NRC's expectations and quality standards for this issue. The staff responded to the Panel in Reference 6 and also provided additional email correspondence (Reference 7) on February 12, 2002.

The staff's response detailed that metal weldment of the CTF structure, including the lift platform, should comply with the material, fabrication, inspection, and testing requirements of ASME Section III, Subsection NF, Class 3 for linear structures. For weld quality verification, the staff relies on Dresden's quality assurance programs for controlling CTF fabrication activities, including weld quality inspection, to provide adequate confidence that the CTF will perform satisfactorily.

As for weld quality verification, the staff noted that the CTF weld fabrication standards were not submitted for staff review and approval. That is, the staff relies on Dresden's quality assurance programs, per 10 CFR Part 72, Subpart G, for controlling CTF fabrication activities, including weld quality inspection, to provide adequate confidence that the CTF will perform satisfactorily. Thus, upon staff's site inspection and audit, all applicable CTF welds are expected to be in compliance with their quality standards.

The staff's February 12, 2002, correspondence provided a specific record quality trail required by the CoC. As outlined by the staff, ASME Code Article NCA 4000, Quality Assurance, includes NCA 4234.10, Inspection. The applicable requirements include the preparation of process sheets, travelers, or checklists, with space provided for recording results of examinations or tests. The requirements state the document shall include space for: a signature, initials, or stamp: the date that the activity was performed by the Certificate Holders representative, and the date on which those activities were witnessed. The staff noted that the Code requirements for the CTF weld inspection records did not agree with the description of available records documented in Reference 3.

The staff also noted that the CoC, Section 3.3.2, allows for exceptions to the ASME Code requirements when authorized by the Director of the Office of Nuclear Materials Safety and Safeguards when the Certificate holder demonstrates that the proposed alternates provide an acceptable level of quality and safety or result in hardship without a compensating increase in the level of quality and safety. The current CoC, Table 3-1 of Appendix B, does not include a Code exception for CTF weld records.

CONCLUSION

The Panel agrees with the staff's observation that the current weld quality records are not in agreement with the Code requirements. The NRC determination documented in Reference 3 that the CTF welds were "proper," based on licensee assertions and alternate quality verification methods, appears to grant a Code exemption without authorization from the Director of the Office of Nuclear Materials Safety and Safeguards.

RECOMMENDATIONS

The Panel recommends that the licensee be asked to demonstrate how the existing quality records meet Code requirements. If this cannot be demonstrated, the licensee should request an exemption from the requirements of the ASME Code in accordance with the CoC. The Panel also notes that the alternate quality verification methods for CTF weld fabrication documented in Reference 3, by themselves, may not support a Code exemption.

ADDITIONAL ISSUES

1. The crane wire rope does not meet the required safety factor of eight as specified in the UFSAR.

REVIEW

The wire rope is required to have a safety factor of 7.5 as stated in Dresden Amendments 19 and 22. The licensee committed to an inspection and replacement program, however, they did not commit to upgrade the wire rope. The inspection report 2001-002(DNMS) issued an NCV for failure to update the UFSAR which incorrectly reflected a safety factor of 8.

CONCLUSION

The licensing basis for Dresden does not require the wire rope to meet a safety factor of 8, rather, 7.5. Therefore the existing wire rope with a safety factor of 7.798 is acceptable.

RECOMMENDATIONS

None.

2. The current inappropriate operation and testing of the overload protection device (load cell) is dispositioned in the inspection report (Reference 3) as an unresolved item, however, the inspection report does not address the identified deficiencies in competency and training of the staff and technicians who operate and calibrate the load cell.

REVIEW

The inappropriate operation and testing of the overload protection device (load cell) is dispositioned in report 2001-002(DNMS) as an unresolved item. The report does mention equipment and personnel performance challenges, but concludes that actions to correct the problems were successfully implemented.

CONCLUSIONS

The report as issued does not discuss competency and training issues. The Submitter's draft report (Reference 1) does discuss training deficiencies.

RECOMMENDATIONS

The unresolved item should be followed up with further inspection. It is recommended that the identified deficiencies in competency and training of the staff who operate and calibrate the load cell be included in the follow up inspection activities.

3. The inspection report states that the load cell on the Unit 2 and 3 crane hoist was routinely bypassed for 20 years when the crane was in the restricted mode, which was outside the licensing basis. This is a violation of requirements, but is not characterized as a violation in the inspection report.

REVIEW

The issued report does state that the use of the crane for cask handling with the load cell bypassed was outside the licensing basis.

CONCLUSIONS

It appears that a violation occurred, however, no violation was issued.

RECOMMENDATIONS

It is recommended that the licensee be issued a violation, if in fact this occurred, or the report should be clarified.

4. The 1981 repairs to the crane bridge girders were incorrectly classified as a minor repair.

REVIEW

The Panel reviewed the design report for the repairs prepared by Nutech (Reference 14) and the Staff Review of Crane issues (Reference 11). The Nutech report concluded that the repairs were not considered "extensive" as defined by the 1976 ANSI B30.2 code. The Staff review concluded that there was no regulatory or technical basis to challenge this conclusion.

CONCLUSIONS

ANSI/ASME B30.2 - 1967 to which the licensee was committed, specified a 125% load test for "extensively repaired" cranes. While it can be debated whether or not the crane repairs were "extensive" there is no regulatory basis or accepted criterion defining the term "extensively repaired" when referring to crane repairs. The licensee performed the repairs to restore margin of safety for the OBE load case. Additionally, the licensing basis classifies the crane as non-seismic. For the NRC to make a determination of what was intended by the ANSI code would require a backfit analysis. The Panel has no basis to challenge the Nutech conclusion.

RECOMMENDATIONS

None.

5. The 1974 analysis of the bridge girders indicates a two percent overstress condition during an OBE considering only static loads. This over-stress condition is documented in the inspection report, but there is no documentation of the basis for the acceptability of this over-stress condition. In addition, there is no analysis of stresses in the trolley for the OBE or SSE load cases.

REVIEW

Since the Dresden crane is classified as non-seismic, the licensee committed (from Reference 5) to analyze the bridge and trolley in a manner consistent with applicable design codes. Allowable stresses were limited to 90% of yield with only static loads considered.

CONCLUSIONS

While the licensee committed to analyze the crane for the new trolley with static lifted loads, it was stated that the crane licensing basis classified the crane as non-seismic. Therefore there is no apparent regulatory basis to compel the licensee to fully meet the OBE load case. No analysis was located for the trolley.

RECOMMENDATIONS

Request the licensee to produce the trolley analysis per the commitment (from Reference 5).

6. During a crane inspection conducted by licensee representatives, five deficiencies in the crane were identified as needing correction. The licensee initiated a corrective action document, but only corrected one of the deficiencies and closed the corrective action document as acceptable.

REVIEW

The crane inspection performed by the vendor was not a safety related or QA type audit. The inspection was focused on crane reliability and none of the deficiencies related to conditions adverse to quality as defined in 10 CFR 72.172. Therefore the recommendations were up to the discretion of the licensee. The vendor inspection was not done to qualify the crane for cask lifting, rather economics (reliability) for general use during outages.

CONCLUSIONS

Correction of the deficiencies noted by the vendor was up to the discretion of the licensee.

RECOMMENDATIONS

None.

REFERENCES:

1. Memorandum Dyer to Grobe: AD HOC REVIEW PANEL FOR DIFFERING PROFESSIONAL VIEW CONCERNING STARTUP OF THE CASK STORAGE LOADING CAMPAIGN AT DRESDEN UNITS 2 AND 3, dated July 20, 2001(includes attachments).

2. E-mail Grobe to Dyer: DPV Update, dated September 21, 2001.

3. NRC Inspection Report 07200037/2001-002(DNMS), dated August 13, 2001.

4. Memorandum Grobe to Zwolinski, et al., dated December 28, 2001.

5. Memorandum Zwolinski to Grobe: RESPONSE TO REQUEST FOR HQ INPUT ON DPV CONCERNING SEISMIC/STRUCTURAL ANALYSIS FOR DRESDEN UNITS 2 AND 3 SPENT FUEL CASK HANDLING, dated February 22, 2002.

6. Memorandum Brach to Grobe: RESPONSE TO DPV STRUCTURAL ISSUES REGARDING THE DRESDEN SPENT FUEL CASK TRANSFER FACILITY, dated February 4, 2002.

7. E-mail Narbut to Grobe: DRESDEN CTF WELD DOCUMENTATION REQUIREMENTS, dated February 12, 2002.

8. Memorandum Jorgenson to File: MEETING WITH EXELON [May 23, 2001] REGARDING DRESDEN UNIT 2/3 Reactor Building CRANE ISSUES, dated June 1, 2001.

9. Certificate of Compliance (No. 1014) issued to Holtec International, dated May 31, 2000.

10. Memorandum Grobe to Dapas and Jorgenson: DPV REGARDING STRUCTURAL ISSUES ON THE DRESDEN Reactor Building AND 125 TON CRANE, dated November 2, 2001.

11. Memorandum Carpenter to Pederson: STAFF REVIEW OF DRESDEN Reactor Building CRANE ISSUES, dated June 15, 2001.

12. Memorandum Brach to Pederson: SAFETY EVALUATION OF DRESDEN CASK TRANSFER FACILITY, dated June 15, 2001.

13. Commonwealth Edison Calculation NO. DRE98-0020 "Dresden Reactor Building Steel Superstructure Interaction Summary", dated March 16, 1998.

14. Design Report for Reactor Building Crane Bridge Girder Evaluation and Repairs (Nutech), dated September 29, 1981.

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

May 3, 2002

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: NOTIFICATION OF A POTENTIAL NON-COMPLIANCE ISSUE

Dear Mr. Kingsley:

On April 29 and 30, 2002, Mr. Marc Dapas, Deputy Director of the Division of Nuclear Materials Safety in the Region III Office, and other members of the NRC staff, including staff from the Spent Fuel Project Office, discussed weld inspection record requirements related to the fabrication of the Cask Transfer Facility (CTF) used at the Dresden Station, with Messrs. K. Jury, D. Bost, and other members of your staff during two telephone conference calls. As described by your staff in a meeting on June 18, 2001, with the NRC, and subsequently documented in NRC Inspection Report 07200037/2001-002(DNMS), dated August 13, 2001, the available CTF weld records consist of weld and inspection data signed by the Quality Assurance (QA) Manager for the CTF fabrication vendor (OMNI). The weld and inspection data were assembled cumulatively in weld groups, according to the size of welds, and dated with the date that the last welding activity had been conducted for each particular group of welds.

During the April 29th and 30th conference calls with your staff, the NRC provided verbal notification that the weld records associated with the CTF described in NRC Inspection Report 07200037/2001-002(DNMS) are not acceptable in terms of providing the specific record quality trail required by the Certificate of Compliance (CoC). Specifically, Section 3.5.1 of Appendix B to the CoC requires the CTF to be fabricated in accordance with the standards in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980. In addition, Section 3.5.2.1 of Appendix B to the CoC requires that the CTF be designed to the stress limits of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Division 1, Section III, Subsection NF. The fabrication record standards in NUREG-0612 start with Section 5.1.6, "Single-Failure-Proof Handling Systems." Section 5.1.6 refers to NUREG-0554, "Single-Failure Proof Cranes for Nuclear Power Plants", for design, fabrication and installation of new cranes. Section 10, "Quality Assurance", of NUREG-0554, dated May 1979, states that a quality assurance program should be established to the extent necessary to include the recommendations of the NUREG and that the program should be consistent with Regulatory Guide (RG) 1.28, "Quality Assurance Program Requirements (Design and Construction)." Regulatory Guide 1.28 endorses American National Standards Institute (ANSI) Standard N45.2, "Quality Assurance Program Requirements for Nuclear Facilities." Section 18, "Quality Assurance Records", of ANSI Standard N45.2, states that the quality assurance records should ("should" changed to "shall" by RG 1.28) include reports of inspections, examinations, and tests, and as a minimum identify the date of the inspection, the

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inspector, the type of observation, the results, and that records shall be identifiable and retrievable. American National Standards Institute Standard N45.2 further states that the records should be maintained in a suitable environment to minimize deterioration.

The weld inspection records described in NRC Inspection Report 07200037/2001-002(DNMS) do not satisfy the inspection record requirements defined in ANSI Standard N45.2. We understand that you invoked ASME Section III, Subsection NF for fabrication of the welds on the CTF structure in your purchase specification. The ASME Code records requirement in NCA 4234.10, "Inspection", are as rigorous as those in ANSI N45.2, and are an acceptable alternative.

Please note that the NRC did not provide a conclusion regarding the adequacy of the CTF weld records in Inspection Report 07200037/2001-002(DNMS). Rather, the conclusions documented in the report focused on the structural integrity of the CTF. For weld quality verification, the NRC relies on licensee QA programs, per 10 CFR Part 72, Subpart G, for controlling CTF fabrication activities, including weld quality inspection, to provide adequate confidence that the CTF will perform satisfactorily. As documented in Inspection Report 07200037/2001-002(DNMS), the NRC determined that the CTF fabrication welds were "proper" in the context of assuring adequate structural integrity of the CTF. This conclusion was based on the licensee's assertion that all CTF welds were inspected, and identified discrepancies were corrected following licensee and vendor oversight of fabrication activities at OMNI involving multiple, full-time Quality Control (QC) inspectors being assigned to examine every aspect of the fabrication work; the documented results of QC inspector activities (weekly Holtec Users Group reports); and the OMNI QA Manger's certification of the cumulative weld data.

With respect to the CTF weld records, 10 CFR Part 72.242(b) states that records required by the CoC shall be maintained. As such, please provide copies of the weld records for the CTF which demonstrate compliance with ANSI Standard N45.2 as described above, or in the alternative, if it was your intent to meet the ASME Code requirements invoked via the CTF purchase order, please provide copies of the weld records for the CTF which demonstrate compliance with the Code requirements. If records satisfying ANSI Standard N45.2 or the ASME Code requirements are not available, please inform the NRC of your plans to resolve the apparent non-compliance issue associated with the CTF weld records by May 10, 2002, followed by submssion of a letter on the docket describing the basis for your planned actions. Actions discussed with your staff to address the apparent non-compliance issue include inspection and testing, the Certificate holder requesting an exception to the Code, or the licensee requesting an exemption to the regulations with a supporting basis for why the existing CTF welds are acceptable.

We understand that your staff has documented its basis for concluding that the CTF is structurally adequate to perform its design functions, (i.e., transferring a loaded multi-purpose canister (MPC)-68 from the 100-ton HI-TRAC transfer cask into a HI-STORM storage overpack) in Condition Report (CR) 00106133, "Potential Nonconformance on CTF Weld Documentation". We have reviewed this CR. While this CR addresses the issue of CTF operability for transferring a presently loaded MPC-68 to a HI-STORM overpack, as discussed with your staff during the April 29th and 30th conference calls, it is our expectation that you resolve the apparent non-compliance issue regarding the CTF weld documentation before you use the CTF for subsequent cask transfer evolutions.

Should you or your staff have any further questions on this matter, please contact the Region III Decommissioning Branch Chief, Chris Miller, at 630-829-9633.

In accordance with 10 CFR Part 2.790 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Cynthia D. Pederson

Cynthia D. Pederson, Director
Division of Nuclear Materials Safety

Docket No. 07200037
License No. DPR-2

cc: Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Dresden Nuclear Power Station Decommissioning Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President, Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
Director Project Management
Senior Counsel, Nuclear Mid-West Regional Operating Group
Document Control Desk - Licensing
M. Aguilar, Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission
A. C. Settles, Illinois Department of Nuclear Safety

non-compliance issue regarding the CTF weld documentation before you use the CTF for subsequent cask transfer evolutions.

Should you or your staff have any further questions on this matter, please contact the Region III Decommissioning Branch Chief, Chris Miller, at 630-829-9633.

In accordance with 10 CFR Part 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Cynthia D. Pederson, Director
Division of Nuclear Materials Safety

Docket No. 07200037
License No. DPR-2

cc: Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Dresden Nuclear Power Station Decommissioning Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President, Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
Director Project Management
Senior Counsel, Nuclear Mid-West Regional Operating Group
Document Control Desk - Licensing
M. Aguilar, Assistant Attorney General
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A. C. Settles, Illinois Department of Nuclear Safety

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

June 15, 2001

MEMORANDUM TO: Cynthia D. Pederson, Director
Division of Nuclear Materials Safety
Region III

FROM: E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

SUBJECT: SAFETY EVALUATION OF DRESDEN CASK TRANSFER FACILITY

Attached is the staff's Safety Evaluation of the Dresden Cask Transfer Facility (CTF). On January 23, 2001, David Tang of my staff received a telephone call from Ross Landsman of your staff regarding the design basis of the Dresden CTF. Over the next several months, the cognizant Spent Fuel Project Office (SFPO) Project Manager (Stephen O'Connor), SFPO Senior Structural Engineer (David Tang), and Office of Nuclear Reactor Regulation Senior Reactor Engineer and heavy loads specialist (Brian Thomas) communicated with Ross Landsman on the design basis and testing requirements of the Dresden CTF. On April 11, 2001, Bruce Jorgensen of your staff requested a meeting with SFPO to establish the CTF design basis and test requirements.

On May 9, 2001, after several teleconferences between SFPO, Region-III and the licensee, SFPO provided a summary of the staff's review of the CTF to Bruce Jorgensen. In that teleconference, the staff stated that they had completed their review and concluded the following:

1. The design of the Dresden CTF satisfies all requirements in accordance with the design basis in the cask Certificate of Compliance.

2. The Dresden CTF component tests, and the CTF system functional, static and performance tests are in accordance with the applicable requirements and determined to be acceptable.

Please contact me if you need any additional information on the staff's review of the Dresden CTF design basis and test requirements.

Docket Nos: 72-37, 72-1014

Attachment: Safety Evaluation of Dresden
Cask Transfer Facility

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

SAFETY EVALUATION REPORT
Dresden Cask Transfer Facility

SUMMARY

The U.S. Nuclear Regulatory Commission's (NRC's) Region III conducted an inspection of the cask transfer facility (CTF) to be used at the Dresden plant to lift the Holtec International Corporation's HI-TRAC transfer cask and multiple-purpose canister (MPC) for the HI-STORM 100 dry spent fuel storage system. Following the inspection and ensuing discussions with the licensee (Exelon Generation Company), Holtec, and the NRC headquarters staff, concerns were raised on the adequacy of the design and testing of the CTF with respect to its licensing basis. By letter dated April 30, 2001, as supplemented, Holtec provided a report on the design and testing criteria for the CTF, including delineation of load bearing components and corresponding safety factors and procedures for conducting functional and load testing. The staff of NRC's Offices of Nuclear Reactor Regulation (NRR) and Nuclear Material Safety and Safeguards (NMSS) has completed review of the report and concludes that the design and testing of the CTF are in accordance with the licensing basis and are acceptable.

1. GENERAL DESCRIPTION

1.1 System Configuration

The CTF steel weldment, which consists of two towers and a top bridge, provides structural support for two heavy load handling systems: the HI-TRAC and MPC Lifters. For Dresden on-site storage application, a 100-ton transfer cask is to be lifted by the HI-TRAC Lifter to a height of approximately 18 ft so that it can be placed on top of a HI-STORM overpack, which is brought in later with air pallets. By harnessing the cask trunnions with the lifting arms of the connector bracket attached to the lift platform, the cask lifting is accomplished by raising the platform from both ends by a pair of screw jacks each installed along the inside face of the tower and turned by a drive motor/shaft/gear assembly. After the transfer cask is secured on top of the overpack, the loaded MPC is raised up slightly by the MPC Lifter to allow the retractable bottom lid of the transfer cask to withdraw so that the MPC can be lowered down to the bottom of the overpack.

1.2 MPC Lifter

The MPC Lifter is a dual-load-path system which provides component redundancy in lifting the Dresden design service load of 90,000 lbs. The load bearing components of interest for each of the two load paths are listed below, in the order the load is transferred from the MPC to the CTF steel weldment:

- one MPC lift cleat
- one basket hitch synthetic sling
- one inboard pulley/shaft assembly
- one outboard pulley/shaft assembly
- two sling adjuster plates
- one cylinder traveler plate
- one hydraulic cylinder
- one cylinder guide assembly
- CTF structure (towers, top bridge)

1.3 HI-TRAC Lifter

The HI-TRAC Lifter is a single-load-path system with a design capacity to lift 280,000 lbs. Dresden will use the HI-TRAC Lifter to lift a design service load of 200,000 lbs. The load bearing components of interest in the load path are listed below, in the order the load is transferred from the transfer cask to the CTF steel weldment:

- two cask trunnions (HI-STORM 100)
- one connector bracket
  - two lifting arms(lugs)/pins
  - one strongback plate assembly
- one connector bracket pin (interfacing lift point)
- one lift platform, simply supported
- two screw jacks
- CTF structure (towers, top bridge)

2. LICENSING BASIS

Section 3.5, Appendix B to Certificate of Compliance (CoC) No. 1014 for the HI-STORM 100 requires that a CTF be designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines in NUREG-0612 and ASME Section III, Subsection NF as follows:

- The HI-TRAC and MPC Lifters shall be in accordance with NUREG-0612, Section 5.1
- If dropped during the transfer operation, the MPC confinement boundary shall not be breached
- The CTF steel weldment shall be designed to comply with Level A stress limits of ASME Section III, Subsection NF

Section 2.3.3.1 of the HI-STORM 100 Final Safety Analysis Report (FSAR) identifies the major CTF components and their design basis. However, because the CTF is a site specific application it was not reviewed in detail during the HI-STORM CoC review.

3. EVALUATION

The staff notes that the CTF is neither a single-failure-proof crane per NUREG-0554 nor entirely a special lifting device per ANSI N14.6. The staff recognizes that the CTF is a heavy load hoisting/jacking system allowed by the flexibility built into Section 3.5, Appendix B to the CoC, and, as such, shall be designed and tested in accordance with the intent of NUREG-0612 to achieve improvement in the reliability of handling systems. In the following, through consideration of NUREG-0612 guidelines and applicable industry codes and standards, the staff summarizes its review findings based on the requirements for compliance for the design and testing for all CTF load bearing components.

3.1 Evaluation of MPC Lifter

3.1.1. MPC Lifter Design Requirements

Component	Requirement Source	Applicable Requirements	Acceptable
MPC lift cleat	0612, Sec. 5.1.6(3)(a)[1]	safety factor > 5 (F_u)	Yes
sling	0612, Sec. 5.1.6(1)(b)(i)[2]	rated load: 100,000 lbs > 50,000 lbs (load rating safety factor = 5)	Yes
pulley shaft	0612, Sec. 5.1.6(1)(a)[3]	safety factors > 3 and 5 (F_y and F_u)	Yes
adjuster plate	0612, Sec. 5.1.6(1)(a)	safety factors > 3 and 5 (F_y and F_u)	Yes
traveler plate	0612, Sec. 5.1.6(1)(a)	safety factors > 3 and 5 (F_y and F_u)	Yes
hydraulic cylinder	0612, Sec. 5.1.6(1)(b)(i)[4]	100,000 lbs rated load > 90,000 lbs	Yes
cylinder guide	Subsection NF[5], Level A	allowable/calculated stress >1.0	Yes
CTF structure	Subsection NF[5], Level A	allowable/calculated stress >1.0	Yes

Notes:
1. Interfacing Lift Points

3.1.2 MPC Lifter Testing Requirements

Component	Requirement Source	Applicable Requirements	Acceptable
MPC lift cleat	0612, Sec. 5.1.6(3)(a)[1]	150% service load	Yes
sling	0612, Sec. 5.1.6(1)(b)(i)[2]	200% rated load	Yes
pulley shaft	0612, Sec. 5.1.6(1)(a)[3]	150% service load	Yes
adjuster plate	0612, Sec. 5.1.6(1)(a)	150% service load	Yes
traveler plate	0612, Sec. 5.1.6(1)(a)	150% service load	Yes
hydraulic cylinder	0612, Sec. 5.1.6(1)(b)(i)[4]	manufacture tested	Yes
cylinder guide	part of system test	see System Test below	see below
CTF structure	part of system test	see System Test below	see below

Notes:
1. Interfacing Lift Points; no testing guideline, but follows ANSI-N14.6 practices
2. Lifting devices that are not specially designed, ASME B30.9, Section 9-6.4, Proof Test
3. Special lifting device, ANSI N14.6, dual load path
4. Lifting devices that are not specially designed

System Test	Requirement Source	Applicable Requirements	Acceptable
functional test[1]	CMAA 70	verify power supply, control, etc.	Yes
125% static test[2]	0554, Sec. 8.2	112,500 lbs (90,000 x 1.25)	Yes
100% performance[3, 4]	0554, Sec. 8.2	45 tons test weight	Yes

Notes:
1. Factory simulated no-load
2. HI-STAR cask plus weights = 112,500 lbs
3. Empty HI-TRAC, pool lid, rigging, and spreader beam at approximately 45 tons
4. The test was not run to the full lift height due to the use of the rigging and spreader beam. A total of about 100 inches of the 240-inch range of the two-stage hydraulic lift system was exercised, across both stages. However, to the extent practicable for the limited travel allowed and because the full rated load was used, the performance test meets the intent of demonstrating system operation and control, and is acceptable.

3.2. Evaluation of HI-TRAC Lifter

3.2.1 HI-TRAC Lifter Design Requirements

Component	Requirement Source	Applicable Requirements	Acceptable
cask trunnion	SER[1], HI-STORM 100	SER[1], HI-STORM 100	Yes
connector bracket - lifting arm/pin	0612, Sec. 5.1.6(1)(a)[2]	safety factors > 6 and 10 (F_y and F_u)	Yes
- strong back	0612, Sec. 5.1.6(1)(a)	safety factors > 6 and 10 (F_y and F_u)	Yes
connector bracket pin	0612, Sec. 5.1.6(1)(a)	safety factors > 6 and 10 (F_y and F_u)	Yes
lift platform	Subsection NF[3], Level A	see Section 3.2.1.1 below	see below
screw jack	0612, Sec. 5.1.6(1)(b)(ii)[4]	safety factors of 14 and 17 (F_y and F_u), Section 3.2.1.2 below	see below
CTF structure	Subsection NF[5], Level A	allowable/calculated stress >1.0	Yes

Notes:
1. NRC Safety Evaluation Report
2. Special lifting device, ANSI N14.6, single load path
3. AISC Steel Construction equivalent; for ASTM- A516 Grade 70 steel with F_y = 38 ksi, the bending stress allowable is 26.25 ksi (17.5 ksi x 1.5 = 26.25), which is about 0.69 F_y and corresponds to a safety factor of about 1.45 against yield strength
4. Lifting devices, as applicable, that are not specially designed
5. AISC Steel Construction equivalent; for ASTM-A36 steel with F_y = 36 ksi, the bending stress allowable is 21.75 ksi (14.5 x 1.5 = 21.75), which is about 0.6 F_y and corresponds to a safety factor of about 1.67 against yield strength

3.2.1.1 Lift Platform Evaluation

The lift platform is bolted at two ends to the screw jack nuts, which, in turn, are raised or lowered by turning the screw jacks against the nuts through a motor/shaft/gear assembly mounted on the CTF top bridge girder. Holtec reports the nut thread bending safety factors of 19 and 48 against F_y and F_u, respectively. The reported nut thread shear safety factors are 50 and 194. These safety factors are more than adequate to satisfy the intent of NUREG-0612 guidelines to improve the reliability of the handling system through increased factors of safety in certain active components. The lift platform serves a structural support function equivalent to that of a crane bridge girder. CMAA 70 states, "The crane girders shall be welded structural steel box sections, wide flange beams, standard I-beams, reinforced beams, or box sections fabricated from structural shapes." The staff notes that the bridge girder should be conservatively designed but need not be considered single failure proof, in accordance with NUREG-0554. In the following, the staff compares first safety factors inherent to the Subsection NF, Level A stress allowables to those of crane industry standards. By considering the stress "design margins" presented in the Holtec report, the staff then computed the overall safety factor to demonstrate that the lift platform is conservatively designed.

Inherent Safety Factors. Using the common structural steel A-36 (F_y = 36 ksi) as a basis, the stress allowable, specified as a fraction of the yield strength, and the inherent safety factor (ISF), defined as the inverse of this fraction, are computed and listed below for the basic tension/compression and bending stress categories considered by three industry standards.

Standard (Bridge Girders)	Basic Tension/Comp.		Bending Stress	
	Allowable	ISF	Allowable	ISF
CMAA 70	0.6 F_y	1.67	0.6 F_y[1]	1.67
Subsection NF, Level A	14.5 ksi[2]	2.48	21.75 ksi[3]	1.66
ASME NOG-1[4]	0.5 F_y	2.0	0.49 F_y[5]	2.04

Notes:
1. Not specified explicitly for bending, but used the basic tension/compression allowable
2. ASME Section II, Part D, Table 1A; 14.5 ksi = 0.40 F_y, approximately
3. Bending allowable = tension/compression allowable x 1.5 (21.75 ksi = 14.5 x 1.5)
4. "Rules for Construction of Overhead and Gantry Cranes," which includes cranes with single-failure-proof features
5. Section NOG-4313: AISC stress allowable (0.66 F_y) divided by 1.12N, where N=1.2 for operating loads

For bending stresses, which usually govern a design, the comparison table above shows that ISFs are essentially identical for the CMAA 70 and the ASME, Subsection NF, criteria. The staff notes that, for the A-36 steel, compared to the CMAA 70 or Subsection NF standard, the ISF, per NOG-1, is about 23% larger for bending stresses.

The staff notes further that all structural steel design ISFs are smaller than the basic safety factor of 3 against the yield strength associated with the mechanical design of the HI-TRAC and MPC Lifter components. This crane industry practice of adopting relatively smaller ISFs for bridge girders is consistent with the common structural steel design philosophy. It is risk informed and acceptable, recognizing that steel bridge girders undergo bounded deformation when overloaded, thereby providing sufficient advanced warning for necessary remedial actions.

Lift Platform Stress Design Margin. The Holtec report defines safety factor as the ratio of the allowable stress and the calculated stress; a safety factor greater than one is considered acceptable. For this evaluation, however, the staff considers Holtec stress safety factors as stress "design margins."

The Holtec lift platform is fabricated with the A-516 Grade 70 carbon steel with a yield strength of 38 ksi and bending stress allowable of 26.25 ksi in accordance with Subsection NF. For a service load of 280,000 lbs plus a 15% dynamic load effect, Holtec reports a minimum design margin of 1.45, which is greater than one. This design margin is above and beyond the ISF of 1.45 (38/26.25 = 1.45) for the A-516 Grade 70 steel although it is slightly smaller than the ISF of 1.66 for the A-36 steel discussed above.

Overall Safety Factor. The staff considers an overall safety factor (OSF), defined as the product of design margin and ISF, for comparing stress design adequacy associated with different design standards for the lift platform. The design margin of 1.45 and the ISF of 1.45 result in an OSF of 2.10 (1.45 x 1.45 = 2.10), on the basis of Subsection NF. As indicated in the ISF comparison table above, a stress design margin of greater than one, which is acceptable on the basis of the more conservative NOG-1 stress allowables, amounts to an OSF of greater than 2.04 (1.0 x 2.04 = 2.04). Thus, the lift platform based on the Subsection NF stress allowables and a design margin of 1.45 achieves an OSF of 2.10, which is greater than the minimum acceptable crane girder OSF standard of 2.04, per NOG-1, for a design margin of one. On this basis, the staff concludes that the lift platform is conservatively designed and is, therefore, acceptable for the design service load of 280,000 lbs.

3.2.1.2 Screw Jack Evaluation

Since there is no explicit NUREG-0612 guidelines for screw jacks, the staff considers NUREG-0612, Section 5.1.6(1)(b)(ii), as applicable, for evaluating the safety factor. The staff notes that ANSI N14.6 imposes basic safety factors of 3 and 5 (F_y and F_u). For improvement of reliability, twice the usual safety factors of 6 and 10 should be considered for the non-redundant lifting devices that are not specially designed. Holtec reports a rated capacity of 300 tons, with an ISF of 8 against the ultimate strength, for the pair of screw jacks. The service load of 140 tons results in an OSF of 17.1 (300/140 x 8 = 17.1) which is larger than 10 and is acceptable.

3.2.2 HI-TRAC Lifter Testing Requirements and Compliance

Component	Requirement Source	Applicable Requirements	Acceptable
connector bracket - lifting arm/pin - one strong back	 0612, Sec. 5.1.6(1)(a)[1] 0612, Sec. 5.1.6(1)(a)	 300% service load 300% service load	 Yes Yes
conn. bracket pin	0612, Sec. 5.1.6(1)(a)	300% service load	Yes
lift platform	see system tests	see System Test below	see below
screw jack	see system tests	see System Test below	see below
CTF structure	see system tests	see System Test below	see below

Notes:
1. Special lifting device, ANSI N14.6, single load path

System Test	Requirement Source	Applicable Requirements	Acceptable
functional test[1]	CMAA 70	verify power supply, control, etc.	Yes
125% static test[2]	0554, Sec. 8.2	351,600 lbs > 280,000 x 1.25	Yes
100% performance[3]	0554, Sec. 8.2	280,000 lbs test weight	Yes

Notes:
1. Factory simulated no-load
2. HI-STAR cask plus weights = 351,600 lbs (175 tons)
3. HI-TRAC plus weights = 280,000 lbs (140 tons)

4. CONCLUSIONS

The NRR and NMSS staff have completed review of the Holtec report summarizing design and testing of the Dresden CTF to lift a HI-TRAC service load of 100 tons and MPC service load of 45 tons, in accordance with Section 3.5 of Appendix B to CoC No. 1014, including NUREG-0612, Subsection NF of ASME Section III, ANSI N14.6, and NUREG-0554, as appropriate. On the basis of design for a service load of 140 tons for the HI-TRAC Lifter and 45 tons for the MPC Lifter, the staff concludes: 1) the heavy load lifting designs of the HI-TRAC and MPC Lifters satisfy all requirements, 2) all HI-TRAC Lifter component tests and system functional, static, and performance tests are in accordance with applicable requirements and acceptable, 3) all MPC Lifter component tests, and system functional and static tests are in accordance with applicable requirements and acceptable, and 4) the MPC Lifter performance test was not run to the full lift height, but meets the intent of demonstrating system operation and control at full rated load and is acceptable.

5. REFERENCES

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Resolution of Generic Technical Activity A-36, July 1980.

NUREG-0544, "Single-Failure-Proof Cranes for Nuclear Power Plants," May 1979.

ANSI N14.6-1993, -for Radioactive Materials, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More."

ANSI B30.2-1996, "Overhead and Gantry Cranes," (Top Running Bridge, Multiple Girder)

ANSI B30.9-2000, "Slings."

Crane Manufacturers Association of America (CMAA) Specification No. 70-2000, "Specifications for Electric Overhead Traveling Cranes."

ASME NOG-1-2000, "Rules for Construction for Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder."

ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, "Rules for Construction of Nuclear Power Plant Components."

American Society of Steel Construction, "Specification for Structural Steel Buildings - Allowable Stress Design (ASD) and Plastic Design."

PRINCIPAL CONTRIBUTORS: David T. Tang
Brian E. Thomas

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

February 4, 2002

MEMORANDUM TO: John A. Grobe, Director
Division of Reactor Safety, Region III

FROM: E. William Brach, Director /RA/
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

SUBJECT: RESPONSE TO DIFFERING PROFESSIONAL VIEW STRUCTURAL ISSUES REGARDING THE DRESDEN SPENT FUEL CASK TRANSFER FACILITY

REFERENCE: (1). "Differing Professional View Concerning Structural Issues Regarding the Dresden Reactor Building/125 Ton Crane and the Spent Fuel Cask Transfer Facility," Memo, J. Grobe to J. Zwolinski, et al., Dated 12/28/01

Attached are my staff's responses to the two questions, Q2 and Q3, of Reference 1, to support clarification of the licensing basis for the structural issues raised regarding the Dresden cask transfer facility (CTF).

On fitting a latching device to the actuating mechanism of the lifting yoke, the staff notes that, for normal HI-TRAC lifting operation or seismic events, the cask will not be subject to any meaningful lateral force. Thus, a retaining latch, per ANSI N14.6, need not be a design consideration or a licensing basis for the yoke actuating mechanism.

On the fabrication standards, consistent with the technical specification of Section 3.5.2.1(1), Appendix B to the Certificate of Compliance, the staff expects that metal weldment of the CTF structure, including the lift platform, should comply with the material, fabrication, inspection, and testing requirements of ASME Section III, Subsection NF, Class 3 for linear structures. For weld quality verification, the staff relies on Dresden's quality assurance programs for controlling CTF fabrication activities, including weld quality inspection, to provide adequate confidence that the CTF will perform satisfactorily.

Please contact me, or David Tang of my staff, if you need additional information on the staff's review of the CTF structural issues. Dr. Tang may be contacted at either (301) 415-8535 or DTT@nrc.gov.

Attachment: Staff Response to Dresden CTF
Structural Issues, Q2 and Q3

19

February 4, 2002

MEMORANDUM TO: John A. Grobe, Director
Division of Reactor Safety, Region III

FROM: E. William Brach, Director *(Original Signed by*
Spent Fuel Project Office *M. Wayne Hodges for:)*
Office of Nuclear Material Safety
and Safeguards

SUBJECT: RESPONSE TO DIFFERING PROFESSIONAL VIEW
STRUCTURAL ISSUES REGARDING THE DRESDEN SPENT
FUEL CASK TRANSFER FACILITY

REFERENCE: (1). "Differing Professional View Concerning Structural Issues Regarding the Dresden Reactor Building/125 Ton Crane and the Spent Fuel Cask Transfer Facility," Memo, J. Grobe to J. Zwolinski, et al., Dated 12/28/01

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Please contact me, or David Tang of my staff, if you need additional information on the staff's review of the CTF structural issues. Dr. Tang may be contacted at either (301) 415-8535 or DTT@nrc.gov.

Attachment: Staff Response to Dresden CTF
Structural Issues, Q2 and Q3

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Response to Dresden Cask Transfer Facility Structural Issues, Q2 and Q3
(Memo, J. Grobe to J. Zwolinski, et al., dated 12/28/01)

Q2 *Regarding Substantive Issue No. 2.A., the Certificate of Compliance (CoC) for the Cask Transfer Facility requires the device to be single failure proof and the application states that no single failure will result in a dropped load. Further, the CoC states that the device must meet NUREG-0612 which requires that special lifting devices meet ANSI N14.6. The cask lifting yokes are special lifting devices. ANSI N14.6 indicates that, if it is possible for a load carrying component to become disengaged, it shall be fitted with a latching device with an actuating mechanism that securely engages and disengages. The cask lifting yoke design does not include a latching device. What is the basis for the conclusion that the cask lifting yokes meet the licensing basis requirements for the device?*

Response. Recognizing Region III's inquiries on design bases and testing requirements, the staff's safety evaluation addressed design safety factors and test procedures for various load path elements of the cask transfer facility (CTF). No attempt was made to evaluate the CTF for other environmental and design conditions, although the staff was aware of the availability of the related safety analysis reports. As a result, the staff's lifting arm/pin (yoke) safety evaluation focused on the stress design factors and testing requirements for the ANSI N14.6 non-redundant lift of the HI-TRAC transfer cask, per NUREG-0612, Paragraph 5.1.6.1(a), "Special Lifting Devices." Holtec International (Holtec) performed analysis and testing of the lifting yoke. The analysis results showed the design safety factors greater than or equal to six (6) and ten (10) against the material yield and ultimate strengths, respectively. The testing, performed at a 300 percent service load, demonstrated structural integrity of the lifting yoke. The results are acceptable.

On yoke actuating mechanisms and latching devices, the staff notes two ANSI N14.6 (1978) provisions: (1) Section 3.3.6, "An actuating mechanism shall be used, if needed, to securely engage or to disengage a special lifting device and a container," and (2) Section 3.3.5, "Load-carrying components that may become inadvertently disengaged shall be fitted with cotter pins or lock pins of a positive locking type, lock wired, or provided with a retaining latch."

For normal HI-TRAC lifting operation, the cask is not subject to any lateral load, thus not possible for the yokes to become disengaged from the cask trunnions. In this case, the yoke actuating mechanisms may serve to ease the handling of the heavy weights.

For seismic events, the HI-TRAC cask is pin-supported in a pendulum like configuration. This suggests that, because of the "long" period of vibration, the cask will not be subject to any meaningful lateral force. Thus, a retaining latch, per ANSI N14.6, need not be a design consideration or a licensing basis for the yoke actuating mechanism.

Q3 *Regarding Substantive Issue No. 3., what were the NRC expectations and approved fabrication standards for weld quality verification for the Cask Transfer Facility?*

Response. Section 3.5.2.1 (3), Appendix B to the CoC states, "...the CTF shall be designed, fabricated, operated, tested, inspected and maintained in accordance with NUREG-0612, Section 5.1." The staff expects the ANSI N14.6 standards or equivalent to be applied to the special lifting devices such as the MPC Lifter adjuster and traveler plates as well as the HI-TRAC connector bracket lifting yoke and strong back subassemblies. Specifically, the ANSI N14.6 (1978), Section 4, "Fabrication," provisions should be considered.

Consistent with the design stress limits criteria of Section 3.5.2.1(1), Appendix B to the CoC, the other metal weldment of the CTF structure, including the lift platform, should comply with the material, fabrication, inspection, and testing requirements of ASME Section III, Subsection NF, Class 3, for linear structures. The staff notes that the CTF lift platform serves a structural support function equivalent to that of a crane bridge girder. Therefore, as an alternative, the applicable criteria and guidelines of ANSI B30.2, "Overhead and Gantry Crane" and of CMMA-70, "Specifications for Electric overhead Traveling Cranes," can also be considered, per Section 5.1.1(6) and (7) of NUREG-0612.

As for weld quality verification, the staff noted that the CTF weld fabrication standards were not submitted for staff review and approval. That is, the staff relies on Dresden's quality assurance programs, per 10 CFR Part 72, Subpart G, for controlling CTF fabrication activities, including weld quality inspection, to provide adequate confidence that the CTF will perform satisfactorily. Thus, upon staff's site inspection and audit, all applicable CTF welds are expected to be in compliance with their quality standards.

February 12, 2002

Dresden CTF Weld Documentation Requirements

On February 11, 2002, John Grobe of Region III requested that SFPO provide a clarifying Email describing the requirements for weld inspection documentation for work done for Part 72. The request was made during a Region III/NRR/SFPO telephone discussion concerning a DPV filed in regards to the welding records for the Core Transfer Facility (CTF) fabricated for Dresden. The request asked that we send the Email by COB 2/12/02.

The Certificate of Compliance (CoC) No. 1014 dated 5/31/00 for the HI-STORM cask system describes the requirements for the CTF in Section 3.5 of Appendix B, "Approved Contents and Design Features." The requirements of Section 3.5 were clarified by E. William Brach memorandum to John A. Grobe dated February 4, 2002 "Response to Differing Professional View Structural Issues Regarding the Dresden Spent Fuel Cask Transfer Facility," which stated the metal weldment, including the lift platform should comply with ASME Section III, Subsection NF, Class 3. The CoC, Appendix B Section 3.3 invokes the 1995 Edition with Addenda through 1997as the governing Code. Article NCA 4000, Quality Assurance, includes NCA 4134.10, Inspection. The requirements include the preparation of process sheets, travelers, or checklists, with space provided for recording results of examinations or tests. The requirements state the document shall include space for: a signature, initials, or stamp: the date that the activity was performed by the Certificate Holders representative, and the date on which those activities were witnessed.

The Code description of inspection record requirements as applied to CTF welds does not comport with the description of the CTF weld inspection records described in NRC Inspection Report 07200037/2001-002, page20. The report describes records consisting of weld data and inspection data signed by the vendor's Quality Assurance manager, assembled cumulatively in weld groups, according to size and dated with the date the last welding activity was performed. The CoC Section 3.3.2, allows for exceptions to the ASME Code requirements when authorized by the Director of the Office of Nuclear Material and Safeguards when the Certificate holder demonstrates that the proposed alternates provide an acceptable level of quality and safety or result in hardship without a compensating increase in the level of quality and safety. The current CoC, Table 3-1 of Appendix B, does not include a Code exception for CTF weld records.

In addition to the CoC, NRC Part 72 requirements for inspections and records are provided in Part 72.160, Licensee and certificate holder inspection, and 72.174, Quality assurance records, The regulations are in accord with the ASME Code requirements for weld inspection records, but are not as specific as the Code regarding signatures and dates.

-Paul P. Narbut

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June 15, 2001

MEMORANDUM TO: Cynthia D. Pederson, Director
Division of Nuclear Materials Safety
Region III

FROM: Cynthia A. Carpenter, Acting Deputy Director /RA/
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: STAFF REVIEW OF DRESDEN REACTOR BUILDING CRANE ISSUES. (TAC NO. MB1762)

Region III has requested that the Nuclear Reactor Regulation (NRR) staff assist in reviewing a number of technical and policy issues associated with qualification and operation of the Dresden Units 2 and 3 Reactor Building Crane. The Region conducted a special inspection to review licensee activities relating to the Holtec dry fuel cask system to be used to remove spent fuel from the Dresden fuel pools. This inspection raised issues concerning whether the Dresden Reactor Building Crane complies with the standards set forth in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The Region could not determine if the Dresden crane system met this and other standards. The Region noted that these issues needed to be resolved before cask handling in the reactor building could commence and classified these issues as unresolved items.

In response, the NRR staff evaluated the information provided by the Region and the licensee. The attachment provides NRR's response to the Region's questions associated with the Dresden Reactor Building Crane.

cc: G. Pangburn, RI
D. Collins, RII
D. Chamberlain, RIV
A. R. Blough, RI
L. R. Plisco, RII
J. Grobe, RIII
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NUCLEAR REACTOR REGULATION (NRR) RESPONSE TO REGIONAL TECHNICAL AND POLICY ISSUES ASSOCIATED WITH THE DRESDEN UNITS 2 AND 3 REACTOR BUILDING CRANE

BACKGROUND

Region III conducted a special inspection to review licensee activities relating to the Holtec dry fuel cask system to be used to remove spent fuel from the Dresden Units 2 and 3 fuel pools. The inspection raised issues concerning whether the Dresden Reactor Building Crane complies with the standards set forth in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and its licensing basis. The Region could not determine if the Dresden crane system meets this and other standards. The Region noted that these issues needed to be resolved before cask handling in the reactor building could commence and classified these issues as unresolved items. Two categories of issues emerged; technical issues surrounding the qualifications of the crane and repairs performed in 1981 and policy issues surrounding implementation of Nuclear Regulatory Commission (NRC) Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment."

TECHNICAL EVALUATION

The Region asked whether the NRR staff had any regulatory requirements or safety basis that would prevent the licensee from lifting up to a 100-ton load.

Licensing Basis for the Dresden Crane

The staff reviewed the current licensing basis documents for the Dresden Reactor Building Crane, noting that Dresden Units 2 and 3 are Systematic Evaluation Plants (SEP), pre-GDC plants, and pre-NUREG-0612 and NUREG-0554 plants.

By letter to the licensee dated June 3, 1976, the NRC staff issued Technical Specification (TS) amendments approving changes governing the operation and surveillance of the modified crane handling system for Dresden Units 2 and 3. In our safety evaluation, the NRC staff concluded that the Reactor Building Crane met the intent of the NRC requirements and was found acceptable for handling spent fuel casks weighing up to 100 tons. In their application, ComEd (the licensee, now Exelon) committed to perform load tests in accordance with American National Standards Institute (ANSI) B30.2.0, "Overhead and Gantry Cranes," at 125 percent of design-rated load in the event the hoist system is extensively repaired or altered.

By letter to the licensee dated July 11, 1983, the NRC staff approved the licensee's Phase I Heavy Loads program in accordance with NUREG-0612, Phase I. In the safety evaluation (SE), the staff concluded that the licensee had provided for the crane to be tested and operated in accordance with ANSI B30.2.0.

By letter to the licensee dated June 28, 1984, the NRC staff approved via draft-technical evaluation report (TER), the licensee's Heavy Loads program under NUREG-0612, Phase II, endorsing the crane as a "single-failure" proof crane provided that the licensee assures that no single failure would occur in the crane electric power and control systems, and provided that the license evaluates all load attachment points to single-failure criteria. However, in Generic Letter 85-11, the NRC closed out the need for licensees to satisfy NUREG-0612, Phase II. In the generic letter, the licensee was noted as taking credit for their "single-failure" proof crane.

In NUREG-0554 and BTP 9-1, the NRC states that a "single-failure" proof crane is designed not to drop a load up to the maximum critical load in the event of the failure of any single component of the crane.

In summary, the licensing basis for the Dresden Reactor Building Crane was established by the 1976 licensing action for moving loads up to 100 tons. An additional certification has resulted in the licensee taking credit for a "single-failure" proof crane.

Staff Evaluation of Damage to Crane

The crane girders and other critical structural components were designed to carry 125 tons with a minimum safety factor of three to the elastic limit, which equals a factor of safety of about six against failure based on girder material ultimate stress.

In 1981, the crane bridge box girders were damaged by impact and compression during an over-hoisting event involving the strongback for the reactor vessel head. The crane girders are built-up box beams approximately 8 ½ feet deep, 2 feet wide, and 113 feet long. The damaged surfaces were confined to the lower portion of the inside webs (buckling) and the inside portion of the bottom flanges (bending) on both the girders approximately 35 feet from one end.

The licensee's repairs consisted of a splice plate welded over the cutout portion of the damaged girder web plate and welding cover plates to the girders bottom flanges. These repairs were made by the licensee and Nutech was contracted to perform the repair evaluation. On page 3.1 of their repair report, Nutech concluded that the west crane bridge girder would be 20 percent overstressed at rated capacity (125 tons) with the damaged area removed. Based on 20 percent reduction in load carrying capacity with the damaged section unaccounted for, the NRR staff concluded that the crane would be capable of lifting 100 tons without overstress. The repairs made to the crane were determined to restore the capacity of the crane girders to the original design value (Page 3.2 of the Nutech report), thus the crane would be able to support a lift of about 125 tons without appreciable overstress. The staff notes that this is a conservative assessment since the damaged areas of the girder's bottom flanges were not removed, but instead covered.

Additionally, based on their analysis, Nutech concurred on Page 3.3 of their report that the repair work was not in the extensive repair or major alteration category as defined by the 1976 ANSI B30.2 code. If the repairs were found to be extensive, then ANSI B30.2 would require that the licensee conduct a load test on the crane.

The NRC staff has evaluated the damage to the crane and the repairs performed based on the information in the Nutech repair report. Of concern was a determination, independent of the licensee, that confirms that repairs were considered "entensive" or not. Should the staff determine that the repairs made to the crane were "extensive," the crane would need to be load tested. The staff in evaluating this matter notes that it does not have well-defined regulatory criteria for what is an "extensive" repair. To make such a determination, the NRC would need to impose a new staff position, and that action would ultimately require a backfit analysis. NRR is not confident that the staff can adequately justify this backfit, because the evaluation of the box girder capacity (with the repair) would most likely indicate no appreciable reduction in its load carrying capacity. The staff believes that it has no technical basis to challenge the licensee's conclusion that the repairs were not "extensive," especially since the licensee has been lifting loads in excess of 100 tons during every refueling outage since the 1981 repairs. Based on licensee activities, staff evaluations and conclusions to date, additional load testing of the crane now will not add assurance of safety to the staff's evaluation of the crane.

Plant's Operating Experience since 1981

Considering the licensee's operating experience with the crane, the licensee has been moving heavy loads such as the reactor head (approximately 100 tons) for the last 20 years since the crane damage was repaired. The license has stated that loads up to 125 tons had been lifted at least 42 times since the repairs. No indication of stress or distortion has been observed in the repaired sections. These previous moves provide the NRR staff with sufficient basis to conclude that future loads could be moved without incident. In addition, the NRC has not challenged the operation of the crane during this time. The licensee has stated that they have a prescribed safe-load path for heavy load lifts on the 613-foot elevation of the Reactor Building which assures that they can safely shut down the reactor following a heavy load drop. This information supports the licensee's Bulletin 96-02 commitment and provides the staff additional assurance.

The licensee has conducted a 50-ton load dry run performance test at power using an empty fuel cask. This dry run test evaluated the adequacy and performance for the hoist and the crane, including its single-failure proof controls. This dry run provided additional assurance to the staff on the capabilities of the crane.

Wire Rope Concerns

The Region asked the NRR staff if the Dresden Reactor Building Crane cable safety factor of 7.5 is acceptable. The staff accepted a cable safety factor of 7.5 in our SE dated June 3, 1976, for Dresden Units 2 and 3, Amendment Numbers 22 and 19. The safety factor for the current crane cable has been determined to be 7.798, exceeding the requirements of the Dresden licensing basis. It should be noted that the wire rope conforms to the original licensing basis and therefore remains acceptable to the staff.

It should be noted that NRC's current guidance for crane cables is contained in NUREG-0612, which was issued after Dresden Amendment Numbers 22 and 19, and recommends a safety factor of 10. A safety factor of 10 is not a requirement for Dresden Units 2 and 3, but it should be noted that the licensee's use of a cable with a higher safety factor provides additional margin.

Technical Evaluation Summary

In summary, NRR concludes that there are no regulatory requirements or safety basis findings that would prevent the licensee from lifting up to 100 ton loads. This is based on the staff's assessment of the licensee's licensing basis, the review of the adequacy of the repairs following the 1981 damage event, and post-repair operating experience of the crane.

BULLETIN 96-02 IMPLEMENTATION EVALUATION

The Region asked the NRR staff five questions in the implementation of NRC Bulletin 96-02 at Dresden. The staff has the following responses.

1. **The Bulletin [96-02] references only documents written in 1980 or later (i.e., NUREG-0612 and related Generic Letters) - were the provisions of the pre-1980 Branch Technical Position (BTP) 9.1 considered as establishing NRC policy in the area of handling heavy loads when the Bulletin was developed?**

   NRR Response

   When the Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," was developed, the NRC policy in the area of handling heavy loads at the time was set forth in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," and the Standard Review Plan (SRP). The BTP was not part of the policy at the time, having been replaced previously in 1979 and 1980 by the two NUREGs. The staff has determined that the provisions of the BTP were subsumed into the NUREGs.

2. **Are licensees whose licensing basis classifies their crane as "single-failure-proof" allowed to consider the probability of load drop to be zero, so that potential consequences may be ignored?**

   NRR Response

   The staff has developed general criterion guidelines in the NUREG-0612 and NUREG-0554 to assure that either the potential for a load drop is extremely small, or for each area addressed, the following evaluation criteria are satisfied:

   1. Releases for radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body;

   2. Damage to fuel and fuel storage rack based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that k-eff is larger than 0.95;

   3. Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel; and

   4. Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load will be limited so as not to result in loss of required safe shutdown functions.

As a result, licensees meeting the criterion above (i.e., with a single-failure-proof crane), do not need to consider load drop consequence analysis.

3. **What does a "single-failure-proof" classification in 1976, under BTP 9.1, mean in terms of whether and how Bulletin 96-02 applied to Dresden?**

   NRR Response

   NRC Bulletin 96-02 applied to Dresden and required response from the licensees. The licensee's response was determined by the staff to be within their licensing basis and commitments. The licensee replied that their load-handling operations were in accordance with NUREG-0612, Phase I.

4. **If exceptions were taken to BTP 9.1 (and approved by NRC) involving reduced safety factor of the wire/rope (7.5 vs. 10) and limited seismic analyses (OBE without load vs. with load) does that affect whether and how Bulletin 96-02 applied?**

   NRR Response

   NRC Bulletin 96-02 applies to Dresden. The licensing basis for the Dresden crane does not have a bearing whether or not Bulletin 96-02 applies.

5. **If Dresden (Exelon) simply withdraws their "response" to the Bulletin, is there a compliance issue regarding the 50.54(f) information request which the Bulletin demanded?**

   NRR Response

   No, licensees are allowed to make changes to their commitments if they properly follow their commitment management program and are consistent with the NEI-99-04 guidance as endorsed by the NRC in RIS 2000-17.

February 22, 2002

MEMORANDUM TO: John A. Grobe, Director
Division of Reactor Safety, Region III

FROM: John A. Zwolinski, Director ***/RA by L B Marsh For/***
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: RESPONSE TO REQUEST FOR HEADQUARTERS INPUT ON DIFFERING PROFESSIONAL VIEW CONCERNING SEISMIC/STRUCTURAL ANALYSIS FOR DRESDEN UNITS 2 AND 3 SPENT FUEL CASK HANDLING

REFERENCE: 1) Differing Professional View Concerning Structural Issues Regarding the Dresden Reactor Building/125 Ton Crane and the Spent Fuel Cask Transfer Facility - Memo from J. Grobe to J. Zwolinski, et al., dated 12/28/01

Attached is our response to Question 1 of Reference 1 concerning whether reactor building crane live or lifted loads must be included in the seismic analysis of the reactor building.

The reactor building crane live or lifted loads must be included in the reactor building structural analysis in combination with seismic loads if the crane is to be used to lift loads during plant operation. More detail is provided in the attached response.

Please contact me if you need additional information on the staff's review of this issue.

Attachment: Response to Region III question

cc: G. Holahan
E. W. Brach

February 22, 2002

MEMORANDUM TO: John A. Grobe, Director
Division of Reactor Safety, Region III

FROM: John A. Zwolinski, Director ***/RA by L B Marsh For/***
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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The reactor building crane live or lifted loads must be included in the reactor building structural analysis in combination with seismic loads if the crane is to be used to lift loads during plant operation. More detail is provided in the attached response.

Please contact me if you need additional information on the staff's review of this issue.

Attachment: Response to Region III question

cc: G. Holahan
E. W. Brach

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Response to Region III question concerning structural analysis/seismic design basis for Dresden Units 2 and 3 reactor building when moving spent fuel casks

Question 1. (Reference 1) Regarding Substantive Issue No. 1.A., the NRC apparently accepted the position of the licensee expressed during a public meeting in 2001 that the licensing basis for the Dresden reactor building did not require consideration of live or lifted loads on the crane when analyzing the structure for the Operating Basis and Safe Shutdown Earthquake (OBE and SSE) load cases. Where is this described in the licensing basis for the Dresden facility, e.g., application, letters responding to questions, safety evaluations, etc.? What was the licensing guidance, e.g., Standard Review Plan, Branch Technical Position, Regulatory Guide, etc., at the time of this licensing review regarding consideration of crane live loads in OBE and SSE load case structural analyses of Seismic Category I structures?

Response:

Original licensing basis and Systematic Evaluation Program: Dresden Unit 2 received its construction permit on January 10, 1966, and its operating license on December 22, 1969. The construction permit and operating license reviews preceded the Standard Review Plan, Branch Technical positions, and Regulatory Guides. Because it was licensed early, Dresden Unit 2 was included in the Systematic Evaluation Program (SEP). The SEP reviewed the seismic design of Dresden Unit 2 under SEP Topic III-6, "Seismic Design Considerations." The SEP reviewed load combinations under SEP Topic III-7.B, "Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria." SEP Topic IX-2, "Overhead Handling System (Cranes)," was deleted from the SEP because overhead handling systems were being reviewed under USI A-36, "Control of heavy loads near spent fuel" (NUREG-0649).

The results of the SEP Topic III-6 review is reported in NUREG/CR-0891, "Seismic Review of Dresden Nuclear Power Station - Unit 2 for the Systematic Evaluation Program," dated April 1980 and in the SEP Topic III-6 Safety Evaluation for Dresden Unit 2 dated June 30, 1982. The SEP seismic review only evaluated the Safe Shutdown Earthquake (SSE) seismic design. SEP Topic III-6 identified no open items related to crane live loads and the reactor building structural design.

The load combinations used in the design of Dresden 2 for the reactor building and all other Class I structures are listed in Table 4-4 of NUREG/CR-0891 as D+R+E and D+R+E' where D = Dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads or operating pressures and **live loads expected to be present when the plant is operating** [emphasis added], E = Design earthquake load, and E' = Maximum earthquake load. The SEP Topic III-6 safety evaluation does not specifically state that the SEP considered that heavy loads on the reactor building crane were loads expected to be present when the plant is operating. The SEP review used the Standard Review Plan (SRP), NUREG-75-087, as the basis for its review. Section 3.8.4 of the 1975 SRP gives load combinations consistent with Table 4-4 of NUREG/CR-0891 although it breaks down D into D (dead loads) + L (live loads). SRP Section 3.8.4 defines L as "Live loads or their related internal moments and forces including any movable equipment loads and other loads which may vary with intensity and occurrence, such as soil pressure." SRP 3.8.4 allows deviations from the acceptance criteria for loads and load combinations if the deviations have been adequately justified. We have not identified any justifications in the Dresden licensing basis for excluding reactor building crane lifted loads.

The SEP Topic III-7.B review spanned a decade. A draft Safety Evaluation (SE) for Topic III-7.B and NRC contractors Technical Evaluation Report (TER) number TER-C5257-321 dated May 17, 1982, was sent to the licensee by letter dated May 20, 1982. The licensee commented on that SE and TER by letter dated August 2, 1982. NRC responded to the licensee's response by letters dated September 21, 1982 and March 9, 1984. NRC's March 9, 1984 letter also transmitted TER-C5506-425 dated November 15, 1983, which was a supplement to TER-C5257-321. The licensee responded to NRC's comments by letter dated July 11, 1984. NRC issued a request for additional information (RAI) by letter dated July 26, 1989, which enclosed revised TER-C5506-425 dated June 3, 1986. The licensee responded to this RAI by letter dated August 30, 1989. NRC completed its review of SEP Topic III-7.B and issued an SE by letter dated August 23, 1990. With respect to the crane live load, NRC's contractor stated in TER-C5506-425 dated November 15, 1983, that the reviewers did not have access to actual design calculations. Also, we have not identified any lists of actual loads. Therefore, it does not appear that NRC or its contractor reviewed individual live loads in their review of Topic III-7.B. With respect to OBE seismic evaluations, the licensee identified in it's letter dated August 2, 1982, that Sargent & Lundy reactor building superstructure calculations did not include OBE loads but that it was Sargent & Lundy's judgement that the SSE evaluation would control the reactor building superstructure structural evaluation.

In summary, the SEP review does not provide a basis for excluding the crane live load from the reactor building superstructure seismic evaluation.

Updated Final Safety Analysis Report (UFSAR) licensing basis: The Dresden Units 2 and 3 reactor building (including superstructure) licensing basis is described in the UFSAR as follows: UFSAR Section 3.2.1 classifies the reactor building as a Class 1 structure. UFSAR Section 3.8.4 defines the load combinations for Class 1 structures to include the dead load plus **live loads expected to be present when the plant is operating** [emphasis added] plus the OBE load (E) for the OBE case or the SSE load (E') for the SSE case. UFSAR Section 9.1.4 states that the reactor building overhead crane is classified as Safety Class II equipment and is not seismically qualified. We are not aware of any inaccuracies in these UFSAR descriptions.

License amendment Nos. 19/22: Dresden license amendment Nos. 19/22 (Reference 2) involved the licensing basis for using the reactor building crane as a single failure proof crane to lift spent fuel casks when the unit is operating at power. The NRC staff used the criteria in Branch Technical Position (BTP) 9-1 (Reference 3) and various licensee submittals in concluding that the licensee's amendment request was acceptable. Although BTP 9-1 would have the licensee evaluate load handling systems for OBE and SSE plus lifted load, we are not aware if this was done. This does not change our conclusion that the licensing basis for the reactor building superstructure includes OBE plus lifted load and SSE plus lifted load if the reactor building crane is being used to lift loads when the plant is operating.

BTP 9-1 states in part, "An overhead handling system includes all the structural, mechanical, and electrical components that are needed to lift and transfer a load from one location to another... The crane should be classified as seismic Category I and should be capable of retaining the maximum design load during a safe shutdown earthquake, although the crane may not be operable after the seismic event." The licensee stated on page 24 of Special Report 41 (Reference 4), which predated BTP 9-1, that "A component failure analysis is currently being prepared by the Whiting Corporation for submittal at a later date. The analysis which has been referred to in the text of this submittal will evaluate the final engineering design and present values for each item relative to the crane safety. Calculations will provide vertical impacts with loading values, as defined in Section 70-3 of CMAA #70, and stress levels of operating conditions under seismic considerations based on AISC code requirements for OBE and DBE. This analysis will also discuss the safety factors, redundancy, and failure protection provided, based on the final engineering design." The licensee submitted the Whiting Corporation component failure analysis and Supplement A to Special Report 41 by letter dated June 10, 1975 (Reference 5).

On pages 1 and 2 of Supplement A to Special Report 41 (Reference 5), the licensee stated that, "We have reviewed the Branch Position on overhead crane handling systems, dated January 10, 1975, in light of the system proposed for installation by Commonwealth Edison at Dresden... This system is judged to be in substantial conformance with the position taken in the draft guideline, as is demonstrated by the itemized summary that follows... The Dresden and Quad Cities cranes are identified as Safety Class II equipment in the plant operating license. It is not practicable to consider reclassifying the hoist system as Seismic Class I, because this would most probably require a new bridge and extensive modifications to the bridge trackway. The bridge and trolley will be analyzed in a manner consistent with the design codes applicable at the time of original installation, that is, the allowable stress will be limited to 90% of yield, with only static lifted loads considered..."

The NRC staff reviewed Supplement A to Special Report 41 against BTP 9-1 and by letter dated October 16, 1975, identified areas that were not acceptable or where additional information was needed. This NRC letter did not question the seismic design. A second letter from NRC dated January 30, 1976, did not question the seismic design either. The licensee provided additional information for this amendment request by letters dated 12/8/75, 1/23/76, 2/9/76, 3/2/76, 3/29/76, and 5/20/76. These letters do not change the statements the licensee made in Reference 5 concerning seismic design.

The staff concluded in the safety evaluation for amendments 19/22 that, "Based on our review of data provided by the licensee, we have concluded that the integrated design of crane, controls, and cask lifting devices meets the intent of BTP APCSB 9-1 as regards single failure criteria..."

<u>Single-Failure reviews</u>: NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," dated May 1979, identifies features of the design, fabrication, installation, inspection, testing, and operation of single-failure-proof overhead crane handling systems that are used for handling critical loads. NUREG-0554 superseded Draft Regulatory Guide (RG) 1.104, Overhead Crane Handling Systems for Nuclear Power Plants," dated 1976 and BTP 9-1. NUREG-0554 Section 2.5, "Seismic Design," states that, "...the crane bridge and trolley should be designed and constructed in accordance with Regulatory Guide 1.29, "Seismic Design Classification," such that the maximum critical load plus operational and seismically induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for design of the bridge." Accordingly, licensees are expected to design and construct the lifting system so that an SSE and OBE may not result in any failures that could reduce the functioning of the spent fuel pool storage structure to an unacceptable safety level.

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides regulatory guidelines in two phases (Phases I and II) for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 28, 1985, dismissed the need for licensees to implement the guidelines of NUREG-0612, Phase II, based on the improvements obtained from the implementation of NUREG-0612, Phase I.

The staff issued an SER dated July 11, 1983, that accepted the licensee's Phase I Heavy Loads program in accordance with NUREG-0612, Phase I. Both the licensee and the staff, as part of their Phase I review, did not identify any changes and/or modifications needed to satisfy the guidelines of NUREG-0612, Phase I. The staff issued a draft TER dated June 28, 1984, that addressed the staff's review of Dresden's implementation of guidelines in NUREG-0612 Phase II. The draft TER concluded that the licensee provided a detailed account of the modifications to the crane and cask yoke assembly to demonstrate compliance with the single-failure-proof criteria. However, the draft TER was not issued as a final product due to the issuance of GL 85-11.

Since NRC's single-failure guidance for overhead handling systems requires that the crane be capable of withstanding seismic events with rated load on the crane, the building structure that supports the crane would also have to be capable of withstanding seismic events with rated load on the crane. Although we are not aware that these single-failure reviews confirmed the adequacy of the reactor superstructure loadings, this does not change our conclusion that the crane live load is part of the licensing basis if the crane is being used as a single-failure proof crane.

<u>Licensee's 1998 calculation</u>: In preparation for beginning a campaign of spent fuel transfers, Sargent and Lundy performed an extensive evaluation for the licensee (calculation DRE98-0020) to analyze and evaluate the building superstructure during various loading conditions including OBE (without live load) and SSE (with live load). The licensee states that this calculation includes the loads from the SSE plus the effects of the maximum lifted load of 250 kips. The effects of the lifted load on the structure include the application of the load vertically as well as the pendulum effects of the lifted load during a SSE hanging from the crane during a seismic event. We note, however, that the licensee refers to SSE plus lifted load as beyond design basis although the NRC staff considers SSE plus lifted load to be within the licensing basis if the crane is being used to lift loads while the plant is operating.

<u>Conclusion</u>: The UFSAR correctly describes the licensing basis for the reactor building as dead load plus live loads expected to be present when the plant is operating, plus the seismic load, both OBE and SSE. If the licensee intends to move fuel when the plant is operating, the spent fuel cask is then a live load expected to be present on the reactor building crane when the plant is operating. The staff notes that calculation DRE 98-0020 describes the SSE load case as, "SSE + Lift (Beyond Design Basis)." As discussed above, the licensing basis of the plant requires analysis of OBE plus lifted loads and SSE plus lifted loads.

References:

1) "Differing Professional View Concerning Structural Issues Regarding the Dresden Reactor Building/125 Ton Crane and the Spent Fuel Cask Transfer Facility," Memo from J. Grobe to J. Zwolinski, et al, dated 12/28/01.

2) Dresden Units 2 and 3 Technical Specification amendments Nos. 19 and 22, dated June 3, 1976, approving changes governing the operation and surveillance of the modified crane handling system for Dresden Units 2 and 3.

3) Auxiliary and Power Conversion Systems Branch (APCSB) Branch Technical Position 9-1 (BTP 9-1), Overhead Handling Systems for Nuclear Power Plants (undated).

4) Letter from ComEd dated November 8, 1974, and attached report Dresden Special Report No. 41 Quad Cities Special Report No. 16 Reactor Building Crane and Cask Yoke Assembly Modifications.

5) Letters from ComEd dated June 3 and June 10, 1975, and attached report Dresden Station Special Report No. 41 Quad-Cities Station Special Report No. 16 Supplement A Reactor Building Crane and Cask Yoke Assembly Modifications with Attachment 1 Component Failure Analysis and Attachment 2 Whiting Corporation's Redundant Trolley Design.