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.

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 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.

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.

<u>CONCLUSION</u>

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

<u>RECOMMENDATIONS</u>

None.

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

<u>REVIEW</u>

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.