NMC
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DAEC Plant Support Center
Operated by Nuclear Management Company, LLC

December 4, 2002
NG-02-1106

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Single-Failure-Proof Status of Reactor Building Crane
Reference: NG-01-1428, dated December 21, 2001, from G. Van Middlesworth to NRC; Single-Failure-Proof Status of Reactor Building Crane
File: A-101a, T-31, SPF-164

By the referenced letter, Nuclear Management Company, LLC submitted information regarding the Duane Arnold Energy Center's (DAEC's) reactor building crane. The information was submitted in an effort to resolve open issues regarding the single-failure-proof status of the crane. During conference calls held with the Staff to discuss electronic mail regarding the submittal, additional information was requested to aid in their review. This information is provided in the attachments.

Should you have any questions regarding this matter, please contact this office.

Sincerely,

Kenneth S. Putnam
Manager, Nuclear Licensing

Attachments:
1. Additional Information Concerning the DAEC Reactor Building Crane
2. Portions of Calculations
3. DAEC Response to NRC Bulletin 96-02

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Additional Information Concerning the DAEC Reactor Building Crane

Attachment 2 provides applicable portions of calculations performed to demonstrate the adequacy of the DAEC reactor building crane. Attachment 3 contains a copy of the DAEC's response to NRC Bulletin 96-02.

As shown in the calculations provided in Attachment 2, the wheel loads, diaphragm spacing, and diaphragm thickness are acceptable. The allowable spacing between diaphragms is calculated based on the equation

$$\frac{108000\ S}{W}$$

where: S = Section modulus of rail in inches cubed and
W = Maximum trolley wheel load in pounds with rated load but without impact.

This results in a required spacing of approximately 23.8″. The maximum existing spacing is 24″ (center-to-center) (23.625″ (edge-to-edge)). As discussed in Attachment 2, since the edge-to-edge spacing is less than the required spacing of 23.8″, the existing spacing of the crane girder diaphragm plates is acceptable.

The diaphragm thickness that is required in order to resist the trolley wheel load is calculated using the vertical wheel load without impact, as discussed in Attachment 2. This results in a value of 0.343″. The actual thickness of 0.375″ is greater than the required value. These thickness values are calculated based on allowable stress values determined from material strengths shown in the certified material test reports (CMTRs).

Attachment 3 provides the DAEC response to NRC Bulletin 96-02. That response provided the chronology of the DAEC's actions in response to the Generic Letter that transmitted NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." As stated in the DAEC Bulletin response, the DAEC provided submittals on December 15, 1981, December 2, 1982, August 22, 1983, September 22, 1983, and May 16, 1984. The NRC provided a Safety Evaluation (SE) and Technical Evaluation Report (TER) by letter dated June 12, 1984 that concluded that the guidelines of NUREG-0612, Sections 5.1.1 and 5.3 had been satisfied and that Phase I of this issue for the DAEC was acceptable.

In 1985, the DAEC Reactor Building Crane was modified to meet the requirements of NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants." The design of the Ederer hoist and trolley system was evaluated in a Staff SER of the Generic Licensing Topical Report EDR-1, Rev. 3, for Ederer's Nuclear Safety-Related Extra Safety and Monitoring (X-SAM) Cranes, dated August 3, 1983.

The handling of heavy loads over safety-related equipment while the reactor is at power is conducted in accordance with our generic heavy loads procedure which is in accordance with the methodology described in our response to Phase I of NUREG-0612.

Attachment 2
to NG-02-1106

Portions of Calculations

CALCULATION SHEET | STONE & WEBSTER, INC.

A5010 61

J.O./W.O./CALCULATION NO.	REVISION	PAGE
12133- SS-3.	Ø1	16

PREPARER/DATE	REVIEWER/CHECKER/DATE	INDEPENDENT REVIEWER/DATE
DWH 9/18/2001	[illegible] 09/27/01	[illegible] 10-02-01

SUBJECT/TITLE	QA CATEGORY/CODE CLASS
DAEC CRANE GIRDER CHECK	I

DIAPHRAGM & VERTICAL STIFFENERS   REF: #12 Sect. 3.3.3.1.4 & #17 Pg 89

SPACING BTW DIA. $\leq \frac{108{,}000\ S}{W}$

S = SECTION MODULUS of RAIL (in^3)

W = wheel LOAD, (LBS.) [No IMPACT] (SEE PG 16A ⚠)

135^{LB} RAIL, $S = 17.3\ in^3$

$W = 78{,}400$ LB.

SPACING $\leq 23.8''$ ACTUAL SPACING APPROX. 24"

SEE PG 16A ⚠ — OK, SINCE APPROX EQUAL BY ENGINEERING JUDGEMENT

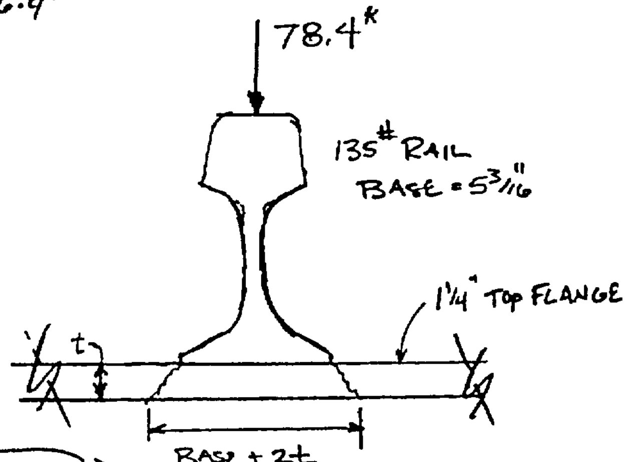
NOTE: DIA. SPACING IS 24" MAX, MOST IS LESS.

THICKNESS of DIA. - SUFFICIENT TO RESIST WHEEL IN BEARING @ 26.4^{ksi}

29.7^{ksi} SEE PG 16B ⚠

$26.4^{ksi} = \frac{78.4^K}{t \times (5\frac{3}{16}'' + 2 \times 1\frac{1}{4}'')}$

$t \geq \frac{78.4}{26.4 \times 7\frac{11}{16}} = 0.386''$ (29.7 ⚠; .343" ⚠)



ACTUAL $t = \frac{3}{8}''$

~~THICKNESS UNDER = (.386 - .375)/.375 = 2.9%~~ DELETE ⚠

SAY OK, SEE BELOW.

ACTUAL BRG STRESS $= \frac{78.4}{.375\,(5\frac{3}{16} + 2 \times 1.25'')} = 27.19$ KSI   ~~3% OVER~~ < 29.7^{ksi} ⚠

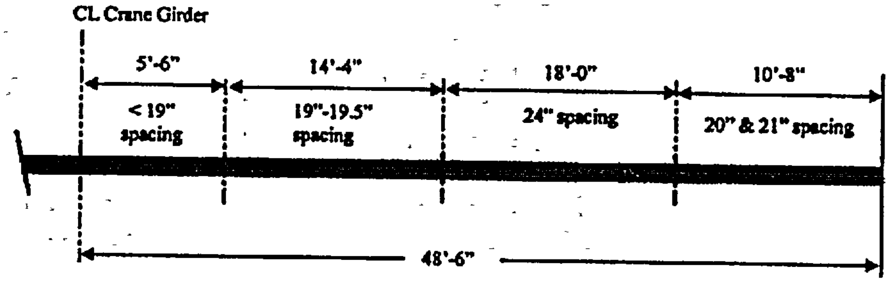
SAY OK SINCE BASED ON MAX. WHEEL LOADS & BEARING STRESS IS $\frac{27.2}{36}$ = 76% of YIELD, JUDGED TO BE OK.

(40.5 ksi ⚠; .67% ⚠)

~~01767~~

Spacing of Crane Girder Diaphragm Plates

The calculation on page 16 shows that the required diaphragm stiffener spacing should be less than or equal to 23.83". Portions of the girder have a stiffener spacing of 24" as diagramed below, which is approximately 3/16" greater than allowed.



Crane Girder Stiffener Spacing (Ref. 8)

The stiffener spacing formula in the code is a simplification of a concentrated load on a simple span beam. The formula limits the bending stress in the rail to 18,000 psi assuming that the wheel load is applied at the center of a 3 span continuous beam as demonstrated below:

$f_b = M / S \Rightarrow 18{,}000 = P\,L / 6\,(S) \Rightarrow L = 108{,}000\ S / P$

$L = 108{,}000\ S / W$ where: L = Stiffener Spacing; W = wheel load (P)

Since the stiffener spacing (center to center) can be equated with the "Beam Span" one can rationally substitute the "clear span" between stiffeners in the above formula. Deducting ½ of the stiffener plate thickness from each end of the center to center diaphragm spacing reduces the span to:

$$24" - 2(3/16") = 23.625" < 23.83" \quad \textbf{O.K.}$$

Therefore the 24" diaphragm spacing is judged adequate.

Crane Impact Loads

After Revision 1 of this calculation was issued, a question raised as to whether the design wheel load for the crane girder diaphragm stiffeners (see page 16) should include a 15% increase due to impact.

The relevant section of the code (CMAA 70-75, Section 3.3.3.1.5.5) does not explicitly mention whether or not wheel loads should include impact. It should be noted however, that the original crane design calculation performed in 1971, using the referenced code, does not include the 15% impact factor. The relevant portions of the original calculation are included as Attachment 5 herein.

Also Section 3.3.3.1.5.6 of CMAA 70-75, which provides criteria for checking bending stress in the trolley rail, clearly states that impact loads need not consider impact. Since both of these sections (3.3.3.1.5.5 & 3.3.3.1.5.6) relate to the design and spacing of the box girder diaphragms, it is reasonable to assume that the two criteria would use the same loads and that loads without impact should also be used in Section 3.3.3.1.5.5.

To reinforce this argument, one can look at the allowable stresses in the AISC code (Reference 20) which was in use at the same time as CMAA 70-75. Referring to Section 1.5.1.5.1, the allowable bearing stress for stiffeners is 0.9 Fy (or 32.4 ksi for A36 steel). Saying this another way, if the box girder had been designed to AISC requirements, the allowable bearing stress with impact would be 0.9 (36) = 32.4 ksi. The CMAA 26.4 ksi allowable is too low if impact is to be considered.

Therefore, it is reasonable to assume that the 15% crane impact load is not to be included in the design wheel load when checking the adequacy of the crane girder diaphragm plates.

Allowable Yield Stress of Crane Girder Diaphragm Plates

Due to the 3% overstress calculated for the crane girder diaphragm plates on page 16, a record search was performed to locate the material certifications for the 3/8" plate material used in the crane. The results of the search, which are included as Attachment 5 herein, show that the yield strength of the 3/8" material used range in value from a high of 49,000 ksi to a low of 40,500 ksi. Using the lowest value increases the allowable bearing stress for the stiffeners to:

$$\frac{26.4\text{ ksi }(40.5\text{ksi})}{36\text{ ksi}} = 29.7\text{ksi}$$

Attachment 3
to NG-02-1106

DAEC Response to
NRC Bulletin 96-02

IES Utilities Inc
200 First Street S E
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Telephone 319 398 8162
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UTILITIES

John F. Franz, Jr.
Vice President. Nuclear

May 10, 1996
NG-96-1035

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
IES Response to NRC Bulletin 96-02: Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment, dated April 11, 1996

Reference: NRC Bulletin 96-02: Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment, dated April 11, 1996

File: A-101a, T-31

On April 11, 1996, the NRC issued the referenced bulletin requesting that licensees review plans and capabilities for handling heavy loads while the reactor is at power in accordance with existing regulatory guidelines and licensing basis. The bulletin requested that licensees submit a report within 30 days addressing this review. We have performed this review; our report is attached.

As discussed in the attachment, our review confirmed that we continue to meet our commitments to existing regulatory guidelines and our licensing basis. Changes to the Duane Arnold Energy Center (DAEC) Technical Specifications are not required. However, the review identified several minor inconsistencies between plant documents and the Updated Final Safety Analysis Report (UFSAR). These issues will be resolved via the following new commitments:

(1) Revise the UFSAR to clarify which Special Lifting Devices are currently in use at the DAEC and correct the reference to Figure 9.1-29. This revision will be made in the next cyclic UFSAR update currently scheduled for May, 1997.

(2) Resolve single failure proof status of the Reactor Building Crane, with respect to seismic analysis review by the NRC.

Should you have any questions concerning this submittal, please contact this office.

This letter is true and accurate to the best of my knowledge and belief.

IES UTILITIES INC.

By ______________________
John F. Franz
Vice President, Nuclear

State of Iowa
(County) of Linn

Signed and sworn to before me on this 9th day of May, 1996,

by John F. Franz.

NOTARIAL SEAL IOWA NANCY S FRANCK

Nancy S. Franck
Notary Public in and for the State of Iowa

9-28-98
Commission Expires

Attachment

JFF/CJR/cjr

cc: C. Rushworth
L. Liu
G. Kelly (NRC-NRR)
H. Miller (Region III)
NRC Resident Office
Docu

IES RESPONSE TO NRC BULLETIN 96-02

NRC Request 1

For licensees planning to implement activities involving the handling of heavy loads over spent fuel, fuel in the reactor core, or safety-related equipment within the next 2 years from the date of this bulletin, provide the following·

A report, within 30 days of the date of this bulletin, that addresses the licensee's review of its plans and capabilities to handle heavy loads while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) in accordance with existing regulatory guidelines. The report should also indicate whether the activities are within the licensing basis and should include, if necessary, a schedule for submission of a license amendment request. Additionally, the report should indicate whether changes to Technical Specifications will be required.

IES Response 1

Background

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," was issued in July 1980 and provided guidelines to ensure the safe handling of heavy loads. An unnumbered generic letter (GL) dated December 22, 1980, Control of Heavy Loads , requested that licensees implement the heavy load control guidelines in NUREG-0612. This generic letter also requested immediate implementation of interim actions, as well as a 6-month follow-up response on the status of the implementation of Section 5.1.1 of NUREG-0612 (Phase I).

Our response to the December 22, 1980 generic letter was submitted on December 15, 1981. We provided additional information in supplemental responses on December 2, 1982, August 22, 1983, September 22, 1983, and May 16, 1984. The NRC provided a Safety Evaluation (SE) and Technical Evaluation Report (TER) by letter dated June 12, 1984 (D. Vassallo, NRC to L. Liu, IELP; Control of Heavy Loads (Phase I) for the Duane Arnold Energy Center (DAEC)). The SE concluded that the guidelines of NUREG-0612, Sections 5.1.1 and 5.3 had been satisfied and that Phase I of this issue for the DAEC was acceptable. On June 28, 1985, the NRC issued GL 85-11, Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612 which informed licensees that implementation of Phase II was not necessary.

In 1985, the DAEC Reactor Building Crane was modified to meet the requirements of NUREG-0554 "Single Failure Proof Cranes for Nuclear Power Plants". The single-failure proof status of the crane is discussed further in the NRC SE for Amendment No. 195 to the DAEC Technical Specifications (TS). This amendment revised the TS to allow reracking the DAEC spent fuel pool with high density fuel storage racks. The SE, dated February 2, 1994, states that the design of the Ederer hoist and trolley system was evaluated in a staff SER of the Generic Licensing Topical Report EDR-1, Rev. 3, for Ederer's Nuclear Safety-Related Extra Safety and Monitoring (X- SAM) Cranes, dated August 3, 1983. The SE documented that the design of the main hoist and trolley complies with the criteria for single-failure proof cranes presented in NUREG-0554. Since the trolley system was installed on an existing bridge, IES was required to perform a seismic analysis to determine whether the bridge and trolley system meet the seismic analysis guidance of Regulatory Guide 1.29. While we have performed an analysis concluding seismic requirements were met, the NRC has not reviewed the seismic analysis. The SE states that the crane system is not, therefore, considered single failure proof.

Bulletin 96-02 Review

The Heavy Loads program at DAEC is based on commitments originating from our response to Phase I of NUREG-0612. As discussed above, these commitments were reviewed and approved by the NRC via the SE and TER transmitted by letter dated June 12, 1984. The Bulletin 96-02 review was performed to verify compliance with these commitments.

Numerous documents were reviewed, including the Updated Final Safety Analysis Report (UFSAR); NUREG-0612; NRC SE and TER transmitted by letter dated June 12, 1984; various DAEC submittals on Control of Heavy Loads; unnumbered generic letter dated December 22, 1980; GL 81-07, Control of Heavy Loads, dated February 3, 1981; GL 83-42, December 19, 1983; and GL 85-11 dated June 28, 1985, Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612. Various DAEC procedures and safety evaluations were also reviewed. The specifics of these reviews are discussed below.

Operation and Maintenance Procedures Review

The procedures which govern operation and maintenance of the plant cranes and hoists have been reviewed for compliance with existing commitments. Numerous DAEC Procedures were reviewed, such as the Crane Operating Instruction; Maintenance Procedures for Inspection, Tagging and Testing of Special Lifting Devices, Slings, and General Use Hoists; Refueling Procedures for the Removal/Installation of the Reactor Vessel Head, Steam Dryer, Shroud Head and Separator, Reactor Vessel Head Insulation, Shipping Cask Storage Pool and Spent Fuel Pool Gates, Reactor Vessel Well Plugs and Cattle Chute Shield, Refuel Floor Hatches and Plugs, Drywell Head; and Fuel and Reactor Component Handling Procedures on Removal/Installation of Fuel Support Piece, Control Rods, Control Rod Guide Tubes, Jet Pump Seal Plugs. The procedures for Handling of Spent Fuel Shipping Casks and Radwaste Handling Casks and LPRM Replacement were also reviewed.

A review of these procedures identified no discrepancies with our commitments to NUREG-0612. All revisions to those procedures which affect crane operations have received Operations Committee review in accordance with DAEC procedures.

Equipment Database Review:

The equipment used to handle Heavy Loads which are entered into the DAEC computer equipment data base (CHAMPS) include permanently installed cranes and hoists, and special lifting devices. A review of these items was conducted to determine if the entered information is consistent with existing regulations and licensing commitments. Two discrepancies were identified as a result of this review. TOOL-E189, Reactor Vessel and Drywell Head Strongback, and TOOL-E190, Dryer/Separator Sling were identified as having a Quality Level of QL-2. This identifies these items as not being Nuclear Safety Related but as having Quality Assurance requirements relating to a regulatory commitment. This is in conflict with UFSAR Table 3.2-1 which identifies these items as Safety Class 2. A designation of Safety Class 2 indicates that these devices perform a Nuclear Safety function. The UFSAR safety class is based on the assumption that a drop of one of these loads (the Reactor Head, Drywell Head, Separator or Dryer) would cause damage to fuel in the reactor core. A load drop analysis performed as part of DAEC's response to NUREG-0612 concluded that a drop of any of these devices over the core would not result in damage to the fuel. Nevertheless, the CHAMPS database must agree with the UFSAR. A revision to the UFSAR to make it consistent with the current supporting analysis is being considered; however, at present, the position taken in the UFSAR is conservative. A Q-200 Code Data Sheet for each of these devices has been initiated and approved which re-classifies each as QL-1, Nuclear Safety Related. A maintenance and modification history search was conducted in accordance with the Evaluation of Items Which Have Increased in Quality Level. This search identified that no maintenance or modification has been performed on these devices which would have adversely affected their ability to perform their nuclear safety function during the time they were misclassified. The misclassification in the CHAMPS database was apparently made as a result of a mass update during conversion from a paper database to an electronic database. These items have now been restored to QL-1 in the CHAMPS equipment database, consistent with the UFSAR.

UFSAR Review:

The DAEC UFSAR addresses Heavy Loads as a general topic on an overview basis in Section 9.1.4.4. The specific commitments regarding the handling of Heavy Loads are addressed by reference. A specific discussion regarding Spent Fuel Cask Movement is included in UFSAR Section 9.1.4.4.5. A Spent Fuel Cask Drop accident is identified in the Nuclear Safety Operational Analysis (NSOA) as an "Other Event". Section 9.1.4.4.5 describes how DAEC meets our commitments to NUREG 0612 and ANSI-N14.6-1978. A review of this section showed that the UFSAR is consistent with current practices for spent fuel cask movements at the DAEC.

A review of the UFSAR identified a discrepancy in the list of Special Lifting Devices located in Section 9.1.4.4.2 of the UFSAR. This list does not include all Special Lifting Devices now in use at DAEC. However, they are correctly identified in the CHAMPS equipment database and are included in the appropriate maintenance procedures. Commitments to ANSI-N14.6-1978 are correctly identified and implemented in accordance with the DAEC's commitments to NUREG

0612. The UFSAR will be revised to eliminate this inconsistency. This revision will be made in the next cyclic update currently scheduled for May, 1997.

Another minor discrepancy was identified in Section 9.1.4.4.5. The text refers to a safe load path shown in Figure 9.1-29. This figure was removed during a later UFSAR revision, while the text was not revised to reflect the removal of that figure. This discrepancy will be corrected during the next UFSAR update.

An additional issue requiring clarification was identified during a review of the SE for Amendment 195 to the DAEC TS. This amendment revised the TS to allow reracking the spent fuel pool. SE Section 2.1.2 "Evaluation of Heavy Loads" states that the DAEC Reactor Building Crane is not considered single failure proof. This is because the seismic analysis which was performed in accordance with the guidance of Reg Guide 1.29 has not been reviewed by the NRC. UFSAR Section 9.1.4.4.5 states that the Reactor Building Crane is single failure proof in accordance with the requirements of NUREG-0554. Since the implementation of Phase II of NUREG-0612 was suspended with the issuance of Generic Letter 85-11, it is unclear whether NRC review of the seismic analysis is required to support the current UFSAR evaluation. Resolution of this issue will be pursued via future discussions with the NRC.

Safe Load Path Review:

Safe load paths used at DAEC are determined in one of two ways. They are either included in standing procedures for repetitive evolutions such as those which support refueling activities (e.g., reactor head movement, dryer/separator movement) or those handled on a generic basis in accordance with DAEC procedure GPM-032 "Generic Heavy Loads".

All standing procedures which include safe load paths have been reviewed to verify that the defined safe load paths are in concert with the analyses which support existing Heavy Loads commitments. Changes to these procedures require Operations Committee approval. Revision to Safe Load Paths is permitted with Operations Committee approval. This is consistent with commitments to NUREG-0612 documented in the TER.

GPM-032 establishes generic criteria based on the location of safety related equipment, crane travel paths, and system redundancy and separation. A safe load path is then determined and documented on a GPM-032 Data Sheet along with applicable rigging and QC inspection hold point requirements. This methodology has been reviewed and meets our current commitments to NUREG-0612.

Technical Specification Review:

The DAEC Technical Specifications have been reviewed; there are no specifications which specifically address the handling of Heavy Loads.

Lifting and Handling Equipment Review:

A review has been performed to verify that all lifting systems are being maintained consistent with commitments made in DAEC's response to NUREG-0612. All overhead cranes comply with CMAA-70 with the exceptions noted in the response, slings conform to ANSI B30.9 and hoists conform to ANSI B30.16 or ANSI B30.21 (ANSI B30.21 governs lever hoists and did not exist at the time DAEC's response to NUREG-0612 was submitted to the NRC and is therefore not included in the commitment). Special lifting devices conform to applicable sections of ANSI N14.6. Items purchased since our response to NUREG-0612 have been purchased under the control of the IES Utilities Quality Assurance program to ensure that these requirements were met.

In addition to these commitments, the Reactor Building Crane (1H001) has been modified to meet the requirements of NUREG -0554 "Single Failure Proof Cranes for Nuclear Power Plants". This modification was completed in 1985. While the majority of loads carried by the Reactor Building Crane have had load drop evaluations performed which conclude that the consequences of a load drop are acceptable, certain load movements including movement of a Spent Fuel Cask would yield unacceptable results should a load drop occur. It was for this reason that the Reactor Building Crane was upgraded to a single failure proof design. 10CFR50.59 safety evaluations that were performed for various load movements involving the Reactor Building Crane rely on the single failure proof status of the crane to demonstrate the acceptability of a load movement and that the margin of safety has not been reduced.

Safety Evaluation Review:

Evolutions involving heavy load movements which deviate from those specifically addressed in the UFSAR (such as spent fuel cask movements inside the reactor building) have been reviewed via safety evaluations performed in accordance with 10CFR50.59. A review of these safety evaluations indicated that the load movements evaluated were conducted in such a way that the margin of safety for any Technical Specification or Safety Evaluation has not been reduced and the probability of occurrence of an accident has not been increased. This conclusion was based on the activities either being bounded by existing evaluations or by the activity having been found to be acceptable in accordance with the guidance of NUREG-0612 Section 5.1 (either the consequences of a load drop were found to be acceptable or the probability of a load drop is sufficiently small that the accident is not considered to be a credible event based on single failure proof design).

NRC Request 2

For licensees planning to perform activities involving the handling of heavy loads over spent fuel, fuel in the reactor core, or safety-related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) and that involve a potential load drop accident that has not previously been evaluated in the FSAR, submit a license amendment request in advance (6-9 months) of the planned movement of the loads so as to afford the staff sufficient time to perform an appropriate review.

IES Response 2

IES does not plan to perform activities involving the handling of heavy loads over spent fuel or fuel in the reactor core, other than previously evaluated refueling activities. The handling of heavy loads over safety-related equipment while the reactor is at power is conducted in accordance with our generic heavy loads procedure which is in accordance with the methodology described in our response to Phase I of NUREG 0612.

NRC Request 3

For licensees planning to move dry storage casks over spent fuel, fuel in the reactor core, or safety-related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) include in item 2 above, a statement of the capability of performing the actions necessary for safe shutdown in the presence of 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.

IES Response 3

IES does not plan to move dry storage casks over spent fuel, fuel in the reactor core, or safety-related equipment while the reactor is at power within the next two years.

NRC Request 4

For licensees planning to perform activities involving the handling of heavy loads over spent fuel, fuel in the reactor core, or safety-related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled), determine whether changes to Technical Specifications will be required in order to allow the handling of heavy loads (e.g., the dry storage canister shield plug) over fuel assemblies in the spent fuel pool and submit the appropriate information in advance (6-9 months) of the planned movement of the loads for NRC review and approval.

IES Response 4

No changes to the DAEC TS are required; the DAEC TS do not specifically address the handling of Heavy Loads.