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Duane Arnold Energy Center
Operated by Nuclear Management Company, LLC

March 11, 2003
NG-03-0168

10CFR50.90
10CFR50.59

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

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Request for Operating License Change (TSCR-062): Single-Failure-Proof Reactor Building Crane

References:
1. Amendment No. 195 to Facility Operating License No. DPR-49, dated February 2, 1994 (TAC No. M86284)
2. NG-96-1637, Letter dated September 13, 1996, from K. Peveler (IES) to NRC, Resolution of Single Failure Proof Status of Reactor Building Crane
3. Letter dated August 3, 2001, from B. Mozafari (NRC) to G. Van Middlesworth (NMC), Duane Arnold Energy Center - Single-Failure-Proof Status of Reactor Building Crane (TAC No. M97242)
4. NG-01-1029, dated August 31, 2001, from G. Van Middlesworth to NRC; Single-Failure-Proof Status of Reactor Building Crane
5. NG-01-1428, dated December 21, 2001, from G. Van Middlesworth to NRC; Single-Failure-Proof Status of Reactor Building Crane
6. December 4, 2002, NG-02-1106, Single-Failure-Proof Status of Reactor Building Crane, Kenneth S. Putnam to NRC

File: A-117, T-31, SpF-164

Amendment 195 to the Duane Arnold Energy Center (DAEC) Operating License (OL) was issued to allow re-racking the DAEC spent fuel pool (Reference 1). The Safety Evaluation (SE) of the Amendment contained a statement that the reactor building crane could not be considered single-failure-proof because the NRC had not reviewed the seismic analysis. By Reference 2, the DAEC provided information regarding the crane modification to the NRC, and requested that the Staff concur with the conclusion that the crane was single-failure-proof.

In Reference 3, the Staff stated that they could not concur on the status of the crane without reviewing the seismic calculations. The letter also requested that Nuclear Management Company, LLC (NMC) revise the DAEC Updated Final Safety Analysis Report (UFSAR) to clarify that the NRC has not endorsed the crane as single-failure-proof. A footnote was added to the UFSAR to denote that the NRC had not endorsed the crane as single-failure-proof (Reference 4). References 5 and 6 provided portions of the seismic analysis and additional information for the Staff's review.

During a teleconference between the Staff and NMC personnel held on January 22, 2003, it was decided that the most expeditious way to resolve the single-failure-proof status of the DAEC reactor building crane would be to submit a license amendment request.

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A001

In accordance with the Code of Federal Regulations, Title 10, Sections 50.59 and 50.90, NMC hereby requests revision to the Operating License (OL) for the DAEC. Specifically, NMC requests approval to revise the DAEC UFSAR to delete the notation that the NRC does not endorse the reactor building crane as single-failure-proof.

This application has been reviewed by the DAEC Operations Committee. A copy of this submittal, along with the evaluation of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to 10 CFR Section 50.91.

As discussed in the conference call, NMC requests approval of this application by May 7, 2003 in order to support preparations for cask movements currently scheduled for August of 2003.

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

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

Nuclear Management Company, LLC

By ______________________
Mark Peifer
DAEC Site Vice President

State of Iowa
(County) of Linn

Signed and sworn to before me on this 11th day of March, 2003,

by Mark Peifer.

KATHLEEN M. FURMAN
Commission Number 199226
MY COMMISSION EXPIRES
11-3-2004

Kathleen M. Furman
Notary Public in and for the State of Iowa

November 3, 2004
Commission Expires

Attachments: 1. Evaluation Of Change Pursuant To 10 CFR Section 50.92
2. Safety Assessment
3. Environmental Consideration

cc: C. Rushworth (w/a)
D. Hood (NRC-NRR) (w/a)
J. Dyer (Region III) (w/a)
D. McGhee (State of Iowa) (w/a)
NRC Resident Office (w/a)
IRMS (w/a)

EVALUATION OF CHANGE PURSUANT TO 10 CFR SECTION 50.92

Background:

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.

In 1994, the NRC issued Amendment 195 to the Duane Arnold Energy Center (DAEC) Operating License which allowed re-racking the DAEC spent fuel pool. The safety evaluation of the amendment contained a statement that the reactor building crane could not be considered single-failure-proof because the NRC had not reviewed the seismic analysis. By letter dated August 3, 2001, the NRC requested that Nuclear Management Company, LLC (NMC) revise the DAEC Updated Final Safety Analysis Report (UFSAR) to clarify that the NRC has not endorsed the crane as single-failure proof. A footnote was added to the UFSAR to that effect.

Nuclear Management Company, LLC, Docket No. 50-331
Duane Arnold Energy Center, Linn County, Iowa
Date of Amendment Request: March 11, 2003

Description of Amendment Request:

The DAEC UFSAR would be revised to delete the notation that the NRC does not endorse the reactor building crane as single-failure-proof.

Basis for proposed No Significant Hazards Consideration:

The Commission has provided standards (10 CFR Section 50.92(c)) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

After reviewing this proposed amendment, NMC has concluded:

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

For heavy load handling associated with the spent fuel pool, Section 5.1.4(2) of NUREG-0612 states "The effects of heavy load drops in the reactor building should be analyzed to show that the evaluation criteria of Section 5.1 are satisfied."

An alternative to this is Section 5.1.4(1): "The reactor building crane, and associated lifting devices used for handling of ... heavy loads, should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report."

The upgraded crane and handling systems satisfy the guidelines of Section 5.1.6. The evaluation criteria of NUREG-0612, Section 5.1 are met with a single-failure-proof crane that satisfies the guidelines of Section 5.1.6, or consequence analysis that satisfies Section 5.1.4(2).

Section 5.2 of NUREG-0612 states that an evaluation of fault trees shows that:
"(1) The likelihood for unacceptable consequences in terms of excessive releases of gap activity or potential for criticality due to accidental dropping of postulated heavy loads after implementation of the guidelines of Section 5.1 is very low; and
(2) The potential for unacceptable consequences is comparable for any of the alternatives evaluated by fault trees, indicating the relative equivalency between alternatives."

Since the NRC fault tree evaluation shows that the potential for unacceptable consequences is comparable for the two alternatives in Section 5.1.4 of NUREG-0612, the proposed request does not significantly change the potential for unacceptable consequences to the plant in conducting heavy load handling above the spent fuel pool. The probability of a load drop accident caused by use of the reactor building crane has been reduced to where it is so small to be considered not credible within regulatory accepted standards. The reason for this is attributed to the following:
(a) The reactor building crane is single-failure-proof. 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.
(b) The rigging used with the crane will be single-failure-proof per Section 5.1.6 of NUREG-0612.
(c) The requirements of NUREG-0612 Phase I have been implemented. 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.

Therefore, this proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The crane has been upgraded to meet single-failure-proof requirements in accordance with the applicable provisions of NUREG-0612 and NUREG-0554. The use of a single-failure-proof crane with rigging and procedures that implement the requirements of NUREG-0612 assures that a cask drop is not credible. The loading on the single-failure-proof crane will not exceed the design rated load of the crane.

Rigging for critical loads will meet NUREG-0612 requirements for single-failure-proof handling systems whenever a critical load is to be lifted over safety related equipment, or over the spent fuel pool, or over the cask when it is in the reactor building and loaded with fuel. When a cask is loaded on the crane hook, the crane trolley and bridge movements will be maintained within well defined limits of operation.

The loading conditions, load combinations, allowable stress limits, and methods of analysis used in the evaluations are consistent with the current licensing basis for the DAEC and NRC approved methods.

Therefore this proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in a margin of safety.

In 1985, the reactor building crane was upgraded to single-failure-proof in compliance with NUREG-0554. The upgraded crane and handling system is in compliance with NUREG-0612, Sections 5.1.1 and 5.1.6. The NRC in NUREG-0612, Section 5.2 documented their review of the potential consequences of a load drop when handled by a single-failure-proof crane using single-failure-proof rigging compared with other alternatives and concluded as follows:
"The likelihood for unacceptable consequences in terms of excessive releases of gap activity or potential for criticality due to accidental dropping of postulated heavy loads after implementation of the guidelines of Section 5.1 is very low."

This means that a load drop is considered to be unlikely within regulatory accepted standards when the load is handled by a single-failure-proof crane and handling system, and performed in accordance with Section 5.1 of NUREG-0612. A single-failure-proof crane design incorporates the applicable design basis event that in this case is a seismic event. A load drop is of such low probability that it is considered unlikely when it is handled with the reactor building crane since the crane and its handling systems satisfy the NUREG-0612 criteria for a single-failure-proof crane. Therefore, any load lifted over the spent fuel pool using the reactor building crane has a very low probability of falling into the spent fuel pool accidentally or as a result of a design basis event.

Therefore, this proposed amendment will not involve a significant reduction in a margin of safety.

Based upon the above, we have determined that the proposed amendment will not involve a significant hazards consideration.

Attorney for Licensee: Jonathan Rogoff, Esquire, General Counsel, NMC, LLC, 700 First St., Hudson, WI, 54016.

SAFETY ASSESSMENT

Description of Change

As an alternative to the restrictions required by a non-single-failure-proof crane, the DAEC upgraded its crane to single-failure-proof using the guidelines in NUREG-0612, Appendix C, and NUREG-0554. The proposed UFSAR change will delete the notation that the NRC does not endorse the reactor building crane as single-failure-proof.

Background

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. The NUREG superseded Draft Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants," dated 1976.

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides regulatory guidelines in two phases (Phase 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. Phase I guidelines address measures for reducing the likelihood of dropping heavy loads and provide criteria for establishing safe load paths, procedures for load handling operations, training of crane operators, design, testing, inspection, and maintenance of cranes and lifting devices, and analyses of the impact of heavy load drops. Phase II guidelines address alternatives for mitigating the consequences of heavy load drops, including using either (1) a single-failure-proof crane for increased handling system reliability, or (2) electrical interlocks and mechanical stops for restricting crane travel, or (3) load drops and consequence analyses for assessing the impact of dropped loads on plant safety and operations. NUREG-0612, Appendix C provides alternative means of upgrading the reliability of the crane to satisfy the guidelines of NUREG-0554.

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. GL 85-11, however, encouraged licensees to implement actions they perceived to be appropriate to provide adequate safety.

In Nuclear Regulatory Commission Bulletin (NRCB) 96-02, "Movement of Heavy Loads over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," dated April 1996, the NRC staff addressed specific instances of heavy load handling concerns and requested licensees to provide specific information detailing their extent of compliance with the guidelines and their licensing basis.

Basis for Change

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. The DAEC heavy load handling program continues to meet NUREG-0612 Phase I guidelines as approved by the NRC staff. The combination of the crane upgrade to single-failure-proof and meeting NUREG-0612 Phase I guidelines satisfies the defense-in-depth philosophy of NUREG-0612 and assures a consistent level of protection in handling heavy loads at the DAEC.

As stated in NUREG-0554, when reliance for the safe handling of heavy loads is placed upon the crane system, the crane should be designed such that a single failure will not result in the loss of the capability of the system to safely retain the load. NUREG-0554 identifies the features of the design, fabrication, installation, inspection, testing and operation of the single-failure-proof overhead crane systems that are used for handling of heavy loads. These features are limited to the hoisting system and the braking systems for the trolley and the bridge. Other load bearing components (e.g., girders) should be conservatively designed, but are not required to be considered single-failure-proof. Also, NUREG-0612, Appendix C, provides alternative means to satisfy certain guidelines of NUREG-0554 when an existing crane system is upgraded to single-failure-proof.

In 1985, the DAEC reactor building crane was modified. A new Ederer single-failure-proof crane trolley and main hoist system meeting the guidelines of Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants," and NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," was installed. The design of the Ederer hoist and trolley system was evaluated in a Staff Safety Evaluation Report (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 main hoist has a 100-ton load capacity and has undergone a 125-ton load test.

The trolley system was installed on the existing bridge; the bridge itself was not replaced. A seismic analysis was performed to verify that the reactor building overhead crane would be capable of safely supporting its rated load during a seismic event after installation of the new, heavier, single-failure-proof trolley. This calculation concluded that the combined vertical and horizontal stresses developed in an operational basis earthquake (OBE) and a design basis earthquake (DBE) are within the allowable stress limits defined in Crane Manufacturers Association of America (CMAA) 70 - 1975 (Specification for Electric Overhead Traveling Cranes).

Since that time, new analyses have been performed as part of the DAEC's dry cask storage project, and provided to the NRC by letters dated December 21, 2001 (G. Van Middlesworth to NRC) and December 4, 2002 (K. Putnam to NRC).

Calculations were performed to check the design of the reactor building crane girders for the increased loadings imposed by the trolley upgrade. Specific assumptions included:

- CMAA #70 1975 edition was used.
- UFSAR earthquake accelerations were used for the reactor building and runway.
- Vertical accelerations of 0.09g OBE and 0.18g DBE were used. The vertical seismic load combinations included the weight of the crane and the lifted load (100 tons).
- The horizontal accelerations used were 0.60g OBE and 1.20g DBE at the crane level. The lifted load and lower load block were assumed to be decoupled from the bridge and trolley with respect to horizontal earthquake accelerations.

The new analysis concluded that the existing crane girders are adequate. The girder combined stresses were shown to be less than allowable stresses. Deflections are less than allowable. The wheel loads, diaphragm spacing, and diaphragm thickness are acceptable.

Calculations were performed to check the design of the reactor building structure for the loadings imposed by the 100-ton capacity single-failure-proof reactor building crane. The calculations also evaluated the design condition of maximum lifted loads during a seismic event. The calculations utilized standard engineering practice and followed analysis methods described in previous DAEC calculations. Conservative inputs and assumptions were used throughout this calculation. Specific assumptions included:

- Vertical accelerations of 0.04g OBE and 0.08g DBE are used. The vertical seismic load combinations include the weight of the crane and the lifted load (100 tons).
- The horizontal accelerations used were 0.35g OBE and 0.70g DBE at the crane level and 0.52g OBE and 1.04g DBE at the roof level. The lifted load and lower load block are assumed to be decoupled from the bridge and trolley with respect to horizontal earthquake accelerations.

The analysis concluded that the existing crane runway girders and rigid frame are adequate. The reactor building crane support structure is adequately designed for the increased weight of the replacement trolley and for all appropriate load combinations, including maximum lifted load plus seismic.

ENVIRONMENTAL CONSIDERATION

10 CFR Section 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (3) result in a significant increase in individual or cumulative occupational radiation exposure. The Nuclear Management Company, LLC (NMC) has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9) for the following reasons:

1. As demonstrated in Attachment 1 to this letter, the proposed amendment does not involve a significant hazards consideration.

2. The proposed changes do not result in an increase in power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes will not affect the types or increase the amounts of any effluents released offsite.

3. The proposed changes will not result in changes in the configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste nor will the proposal result in any change in the normal radiation levels within the plant. Therefore there will be no increase in individual or cumulative occupational radiation exposure resulting from these changes.