October 10, 2003

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS - HEAVY LOADS HANDLING (TAC NOS. MB7840 AND MB7841)

Dear Mr. Skolds:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 204 to Facility Operating License No. DPR-19 and Amendment No. 196 to Facility Operating License No. DPR-25 for Dresden Nuclear Power Station, Units 2 and 3. The amendments are in response to your application dated February 26, 2003, as supplemented by letters dated June 12, July 25, September 11, and October 9, 2003.

The amendments approve changes to the Dresden Nuclear Power Station, Units 2 and 3, as described in the Updated Final Safety Analysis Report (UFSAR). The amendments would revise the UFSAR to include a description of a load drop analysis performed for handling reactor cavity shield blocks weighing greater than 110 tons with the Unit 2/3 reactor building crane during power operation.

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Maitri Banerjee, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos.: 50-237 and 50-249

Enclosures:
1. Amendment No. 204 to DPR-19
2. Amendment No. 196 to DPR-25
3. Safety Evaluation

cc w/encls: See next page

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Dresden Nuclear Power Units 2 and 3

cc:

Site Vice President - Dresden Nuclear Power Station
Exelon Generation Company, LLC
6500 N. Dresden Road
Morris, IL 60450-9765

Dresden Nuclear Power Station Plant Manager
Exelon Generation Company, LLC
6500 N. Dresden Road
Morris, IL 60450-9765

Regulatory Assurance Manager - Dresden
Exelon Generation Company, LLC
6500 N. Dresden Road
Morris, IL 60450-9765

U.S. Nuclear Regulatory Commission
Dresden Resident Inspectors Office
6500 N. Dresden Road
Morris, IL 60450-9766

Chairman
Grundy County Board
Administration Building
1320 Union Street
Morris, IL 60450

Regional Administrator
U.S. NRC, Region III
801 Warrenville Road
Lisle, IL 60532-4351

Illinois Emergency Management
Agency
Division of Disaster Assistance &
Preparedness
110 East Adams Street
Springfield, IL 62701-1109

Document Control Desk - Licensing
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Senior Vice President, Nuclear Services
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Vice President - Operations Support
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Chief Operating Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Vice President - Licensing and Regulatory
Affairs
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Director - Licensing
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Senior Counsel, Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Manager Licensing - Dresden and
Quad Cities
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 204
License No. DPR-19

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated February 26, 2003, as supplemented by letters dated June 12, July 25, September 11, and October 9, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended to authorize revision of the Updated Final Safety Analysis Report (UFSAR) as set forth in the application for amendment by the licensee, dated February 26, 2003, and as supplemented by letters dated June 12, July 25, September 11, and October 9, 2003. The licensee shall update the UFSAR to include a description of a load drop analysis performed for handling reactor cavity shield blocks weighing greater than 110 tons with the Unit 2/3 reactor building crane during power operation, as authorized by this amendment and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to refueling outage D2R18 heavy loads handling operation at Dresden Nuclear Power Station.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: October 10, 2003

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 196
License No. DPR-25

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated February 26, 2003, as supplemented by letters dated June 12, July 25, September 11, and October 9, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended to authorize revision of the Updated Final Safety Analysis Report (UFSAR) as set forth in the application for amendment by the licensee, dated February 26, 2003, and as supplemented by letters dated June 12, July 25, September 11, and October 9, 2003. The licensee shall update the UFSAR to include a description of a load drop analysis performed for handling reactor cavity shield blocks weighing greater than 110 tons with the Unit 2/3 reactor building crane during power operation, as authorized by this amendment and in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to refueling outage D2R18 heavy loads handling operation at Dresden Nuclear Power Station.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: October 10, 2003

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 204 TO FACILITY OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 196 TO FACILITY OPERATING LICENSE NO. DPR-25

EXELON GENERATION COMPANY, LLC

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated February 26, 2003 (Reference 1), as supplemented by letters dated June 12, July 25, September 11, and October 9, 2003 (References 3, 4, 5, and 6), Exelon Generation Company (EGC, the licensee) requested amendment to Facility Operating Licence Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station, Units 2 and 3 (DNPS), pursuant to Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.90), “Application for amendment of license or construction permit,” and 10 CFR 50.59, “Changes, tests, and experiments.” The license amendment request proposes to revise Section 9.1.4.3.2, “Reactor Building Overhead Crane,” and Section 9.15, “References,” of the DNPS Updated Final Safety Analysis Report (UFSAR). The revision would allow using the 125-ton design-rated reactor building overhead crane, which is approved as single-failure-proof for up to 110-ton loads, to lift the 116-ton reactor cavity shield blocks. The licensee performed a load drop analysis, in accordance with NUREG-0612, Appendix A guidance, to show the damage to equipment in redundant or dual safe shutdown paths as a consequence of a postulated load drop would be limited so as not to result in loss of required safe shutdown functions.

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* Notice.

2.0 REGULATORY EVALUATION

2.1 NRC Guidance Documents

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.

In NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” dated July 1980, the staff provided regulatory guidelines for heavy load lifts. These guidelines were divided into two phases (Phase I and II) for implementation by 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 might 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 to either further reduce the probability of a load handling accident or mitigate the consequences of heavy load drops. These alternatives included using a single-failure-proof crane for increased handling system reliability, employing electrical interlocks and mechanical stops for restricting crane travel to safe areas, or performing load drops and consequence analyses for assessing the impact of dropped loads on plant safety and operations. NUREG-0612, Appendix A provides guidelines for an analysis of postulated load drops and evaluation of potential consequences.

The basis for the guidelines in NUREG-0612 was to minimize the occurrence of the principal causes of load handling accidents and to provide an adequate level of defense-in-depth for handling of heavy loads near spent fuel and safe shutdown systems. Defense-in-depth is generally defined as a set of successive measures that reduce the probability of accidents and the consequences of such accidents. In the area of control of heavy loads, the emphasis is on measures that prevent load drops or other load handling accidents. These measures include: use of rigorous crane design standards with substantial safety margins; implementation of prudent maintenance, testing, and inspection guidance; selection and use of appropriate lifting devices; and establishment of crane operator training programs and heavy load handling procedures. Measures to reduce the consequences of potential load handling accidents include: restricting, by procedure or interlock, the travel of heavy loads to reduce the potential that a dropped load would damage spent fuel or safe shutdown equipment; verifying by analysis that intervening structures would prevent a dropped load from damaging spent fuel or safe shutdown equipment; and verifying by analysis that the damage to critical structures, systems, and components from a dropped load would remain within acceptable limits. In addition to measures that reduce the probability of a load drop, the operational restrictions that maintain heavy loads within selected safe load paths provide some defense-in-depth by reducing the likelihood that a dropped load could damage critical structures, systems, and components (SSCs).

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 on the basis of improvements obtained from the implementation of NUREG-0612 Phase I. Generic Letter 85-11, however, encouraged licensees to implement actions they perceived to be appropriate to provide adequate safety.

NRC Bulletin 96-02, dated April 11, 1996, clarified that “[a]lthough the generic letter stated that the NRC staff review of the Phase II submittals did not indicate the need to require further generic action at that time, it did not preclude the possible further need for the staff to review additional heavy load handling concerns and to require, as appropriate, further actions by licensees.”

2.2 Plant Specific Information

DNPS is a boiling water reactor (General Electric design) which commenced commercial operation in 1971, and its current operating license will expire in January 2011. By letter dated March 21, 2002, the licensee submitted to the NRC its 2001 Commitment Change Summary for DNPS. The revisions to NRC commitments were processed using Nuclear Energy Institute 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes," dated July 1999. The commitment summary provided the licensee's revision to previous commitments it made in response to NRC Bulletin 96-02. The licensee (then Commonwealth Edison Co.) provided its previous commitments to the NRC in a letter, dated May 13, 1996. The original commitment stated the following:

> Current plans at [EGC] do not include the implementation of activities involving the handling of heavy loads over the spent fuel pool, fuel in the reactor core, or safety-related equipment which result in the potential for an unreviewed safety question per the provisions of 10 CFR 50.59, prior to April 11, 1998 (two years from the date of the NRC Bulletin 96-02) . . . [EGC] currently has no plans for any movement of 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). However, should such movements be planned in the future, [EGC] will 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 storage cask, damage to the fuel, and damage to safety-related equipment as a result of a load drop inside the facility.

In the March 21, 2002, letter revising its commitments to the NRC, the licensee stated that the existing commitment was too restrictive and that it had performed all spent fuel cask lifts in the DNPS reactor building using a single-failure-proof crane equipped with a special lifting device. The licensee further stated that the NRC had granted single-failure-proof status to the reactor building crane in a letter to the licensee (then Commonwealth Edison Co.), dated June 3, 1976. Thus, the licensee concluded that it did not have to analyze the consequences of a load drop. However, the NRC staff noted that this commitment change argument was valid only for loads up to the 110-ton single-failure rating of the crane.

During an inspection of preparations at DNPS for loading spent fuel into dry storage casks, the NRC staff identified concerns regarding the long term acceptability of heavy load handling facilities at DNPS. These concerns included apparent deviations from generally applicable standards for seismic qualification and single-failure-proof criteria for handling heavy loads with regard to the DNPS reactor building superstructure and reactor building crane. The inspection findings are documented in NRC Inspection Report 07200037/2001-002(DNMS), dated August 13, 2002.

As stated in the Inspection Report, the NRC staff had issued License Amendment Nos. 22 and 19, dated June 3, 1976, for Dresden Nuclear Power Station, Units 2 and 3, respectively, approving changes to the DNPS Technical Specifications governing the operation and surveillance of the upgraded crane with "single-failure-proof" capability. In the associated safety evaluation, the NRC staff stated that the reactor building crane met the intent of the guidance in Branch Technical Position, Auxiliary and Power Conversion Systems Branch 9-1, “Overhead Handling Systems for Nuclear Power Plants,” for handling heavy loads weighing up to 110 tons, with the following exceptions: (1) the redundant mechanical limit switch in the main hoist power circuit (for two blocking), (2) an electrical interlock system to prevent crane travel outside its safe load path, and (3) a slow speed drive motor to limit the hoisting speed. The staff also stated that it expected that completion of the crane modifications would satisfy the intent of Branch Technical Position, Auxiliary and Power Conversion Systems Branch 9-1.

As stated above, the NRC staff cited in the June 3, 1976, safety evaluation the specific modifications that were needed to support the single-failure-proof capability of the crane. Such modifications would enable the licensee to conform to its licensing basis. Following modification of the reactor building crane to conform to its original licensing basis, the licensee is prohibited from lifting loads exceeding 110 tons as a “single-failure-proof” crane. The guidelines of NUREG-0612, Section 5.1.4, "Reactor Building - BWR," apply to any loads above 110 tons to show that the evaluation criteria of Section 5.1, "Recommended Guidelines," are satisfied. The guidelines of NUREG-0612, Section 5.1.5, also apply to any load above 110 tons to demonstrate that the largest postulated load handled by the handling system could not penetrate the ceiling or cause spalling that could cause failure of the safe shutdown equipment where safe shutdown equipment has a ceiling separating it from the overhead handling system.

The DNPS reactor building crane is needed to lift reactor shield blocks, weighing up to 116 tons with its associated rigging. However, evaluation of the potential drop of a shield block during power operations was not previously addressed in the UFSAR and the shield blocks exceed the 110-ton single-failure proof rating of the reactor building crane. Thus, 10 CFR Part 50.59 requires that the licensee obtain a license amendment prior to using the crane for moving the reactor shield blocks.

The proposed amendment would revise the DNPS Units 2 and 3 reactor building overhead crane licensing basis to support movement of heavy loads exceeding the capacity of the single failure proof crane over units operating at power. The licensee has provided its assessment of the reactor building crane’s capability and the basis for the changes to the UFSAR to support the requested license amendment. The staff reviewed the procedural controls described in the changes to the UFSAR and assumptions used in the load drop analysis submitted for the top layer of reactor cavity shield blocks weighing up to 116 tons. The staff also used section 9.1.5, “Overhead Heavy Load Handling Systems,” of NUREG-0800, “NRC Standard Review Plan,” that references the guidelines of NUREG-0612, and NUREG-0554, for implementation of these criteria in the design of overhead heavy load handling systems. The NRC staff’s evaluation of the licensee’s assessment and basis for the changes is described in Section 3.0 below.

Separately, in a letter dated October 4, 2002, the staff had approved exigent license amendments 196/189 consisting of a one-time change to the DNPS UFSAR to state that lifting heavy loads up to and including 116 tons was allowed prior to and during the Dresden Nuclear Power Station, Unit 3 refueling outage number 17. The staff approved these exigent amendments in accordance with 10 CFR 50.91(a)(6).

2.3 UFSAR Changes

The proposed changes to the DNPS UFSAR Section 9.1.4.3.2 include the following:

- Add text after the 11th paragraph, including the following:

    - A load drop analysis has been performed for handling the top layer of the Units 2 and 3 reactor cavity shield blocks weighing up to 116 tons, for the designated safe load path. This analysis demonstrated that the postulated load drop will not affect any safety related equipment because the analysis predicted no scabbing or perforation of the refueling floor and the overall response of the floor system was acceptable.

    - This load drop analysis was performed in accordance with the guidance of NUREG-0612, Appendix A.

    - Extension of the load drop analysis methodology is not approved for application to other heavy loads or load paths.

    - Key assumptions used in the load drop analysis, include the following:

        - "The weight of the dropped top half-layer reactor cavity shield block is considered to be 116 tons, including the slings and other rigging used to lift the cavity shield block, and excluding the weight of the crane load block. The maximum drop height for the 116-ton shield block is assumed to be 1'-0" above the floor."

        - "The overall adequacy of the impacted structural elements is determined by calculating the total strain energy in the impacted elements corresponding to an allowable ductility limit, and comparing this energy to the impact energy imparted to the impacted elements."

        - "The kinetic energy of the reactor cavity shield block at impact is conservatively assumed to be transferred entirely to the impacted structural elements."

    - "The designated safe load path, hoisting height restrictions, and maximum weight of the reactor cavity shield block and rigging are described in applicable procedures. When handling the top layer of shield blocks weighing more than 110 tons, crane controls incorporate travel limits and hoisting height restrictions."

The changes to the DNPS UFSAR Section 9.1.5 include adding references used for load drop analysis, described in Section 3.0 below, that was submitted as the basis for changes to the UFSAR Section 9.1.4.3.2.

3.0 TECHNICAL EVALUATION

The reactor building crane is designed to handle loads up to 125 tons. In the response to GL 80-113, "Control of Heavy Loads," which was issued on December 22, 1980, the licensee committed to the NUREG-0612 Phase I guidelines on measures for reducing the likelihood of dropping heavy loads. During its review, the NRC staff did not identify any changes or modifications needed to satisfy the NUREG-0612, Phase I guidelines. Subsequently, the NRC staff issued a safety evaluation, dated July 11, 1983, accepting the licensee's NUREG-0612 Phase I heavy-loads program. Heavy-load lifts are subject to the licensee's commitments to NUREG-0612 Phase I guidelines. The current reactor building crane is single-failure-proof for handling loads up to 110 tons, as discussed in Section 3.1 of this report. That is, the current reactor building crane configuration meets the intent of NUREG-0612 Phase II guidelines for increased handling system reliability. Therefore, the reactor building crane is capable of lifting reactor cavity blocks, weighing up to 116 tons, without significant probability of a load drop.

In the submittal, dated February 26, 2003, the licensee stated that it estimated the weight of the top layer shield blocks of Dresden Nuclear Station, Unit 2 as 116 tons, based on review of dimensional drawings. The licensee stated that it would verify the weight of each piece of top layer of the Unit 2 shield blocks. In a request for additional information, the staff asked the licensee to describe how and when the licensee would verify the weights of these shield blocks. In response, in a letter dated, July 25, 2003, the licensee stated that it would weigh the shield blocks using a load cell before starting the DNPS Unit 2 refueling outage number 18 (D2R18) in early October 2003. The methodology would be the same as the one the licensee used for weighing the top layer of the Unit 3 shield blocks on September 20, 2002. The load cell will be similar to the one used for weighing the top layer of the Unit 3 shield blocks. The staff finds the licensee response acceptable.

The licensee designated safe load paths for the movement of the reactor cavity shield blocks to minimize the potential effect of a load drop while remaining within the practical limitations due to the size of the shield blocks and space available on the refueling floor. The load path for the top layer of the shield blocks is shown in the figure (unnumbered) in Attachment C to the licensee letter, dated September 11, 2003 (Reference 5). The licensee states that it considered the following in designating these load paths:

- General practices incorporated into DNPS procedures as a result of NUREG-0612 guidelines ensure that heavy load heights are maintained as low as practical and that the movement of heavy loads over the spent fuel pool and open reactor cavity is prohibited.

- The radius of the semi-circular top layer of reactor cavity shield blocks is approximately 21 feet 6 inches. The load path ensures that the shield blocks remain over reactor building structural members supporting the refueling floor during movement.

- A drop of the reactor cavity shield blocks in the designated safe load paths would not directly impact any safety-related equipment because the shield blocks are handled only on the refueling floor, which contains no such equipment.

The licensee analyzed consequences of dropping reactor cavity shield blocks on the structures in the designated safe load path. The staff reviewed only the cases involving a drop of the top layer of shield blocks. The staff did not review the cases involving drops of the middle and bottom layers of shield blocks because the reactor building crane can safely handle them as they weigh less than the single failure proof capacity of the crane (110 tons).

In the load drop analysis presented in the original submittal, dated February 26, 2003 (Reference 1), the licensee assumed up to half of the kinetic energy being dissipated during the impact leaving only a half of the kinetic energy for possible structural damage. The staff noticed that such a treatment was inconsistent with the NUREG-0612, Appendix A, Section 1(7) guidance, which recommends that the analysis should postulate the “maximum damage” that could result, i.e., the analysis should consider that all energy is absorbed by the structure and/or equipment that is impacted. In a request for additional information (RAI), the staff asked the licensee to justify assuming significant loss of kinetic energy during the impact, which was inconsistent with the NUREG-0612, Appendix A guidance. In the RAI, the staff also asked the licensee to explain in what pathways the energy is dissipated. The licensee’s response, submitted in a letter dated, July 25, 2003 (Reference 4), did not adequately explain the reasons for the inconsistency nor did it describe the pathways of energy dissipation. During a teleconference with the licensee and its contractors on September 10, 2003, the staff told the licensee that its assumed energy loss was unacceptable because the significant amount of kinetic energy that is lost could damage the structures, which the licensee had ignored in the analysis. In response, in a letter, dated September 11, 2003 (Reference 5), the licensee submitted an addendum to the load drop analysis, which did not assume any energy loss during the impact. Additionally, following a phone call between the staff and the licensee on October 8, 2003, the licensee submitted additional information regarding impact on safety-related equipment supported from the underside of the refuel floor covered by the safe load path if a load drop creates cracking in the area (Reference 6).

Based on the above discussion, the staff finds that the energy absorbed by the structure and equipment that is impacted in the resubmitted analysis is consistent with the NUREG-0612 guidance and, hence, acceptable.

3.1 Technical Evaluation of Structural Impacts

A load drop analysis has been performed in Reference 1 (as supplemented by References 5 and 6) for handling the top layer of the Units 2 and 3 reactor cavity shield blocks weighing up to 116 tons for the designated safe load path. The analytical results demonstrate that a postulated load drop will not affect any safety-related equipment since there will be no scabbing or perforation of the refueling floor, and the overall response of the floor system will be acceptable. Extension of the methodology is not approved for application to other heavy loads or load paths.

Standards and guides which have been used for determining allowable stress limits and other acceptance criteria are consistent with industry practice and have previously been accepted by the staff for similar applications. The licensee's load drop analysis used the following key assumptions and methods:

- The weight of the dropped top half-layer reactor cavity shield block is considered to be 116 tons, including the slings and other rigging used to lift the cavity shield block, and excluding the weight of the crane load block. The maximum drop height for the 116-ton shield block is assumed to be 1' - 0" above the floor.

- The overall adequacy of the impacted structural elements is determined by calculating the total strain energy in the impacted elements corresponding to an allowable ductility limit, and comparing this energy to the impact energy imparted to the impacted elements.

- The kinetic energy of the reactor cavity shield block at impact is conservatively assumed to be transferred entirely to the impacted structural elements.

- The energy absorption of the impacted elements is calculated using constructed elasto-plastic load-deflection diagrams of the structural elements. The ductility limit is determined using [Reference 9], Appendix C, Section C.3, and the area under the load-deflection diagram up to the applicable ductility limit is used as the measure of the energy absorption capacity of the elements.

- The shear failure load is estimated using [Reference 10]. The shear failure load is at least 1.20 times the flexural resistance load in order to use the flexural mode of failure to calculate the strain energy. Otherwise, the ductility ratios given in [Reference 9], Section C.3.7 or C.3.9 are used.

- The calculation uses the actual concrete compressive strength, as noted in station documents.

- The potential for scabbing of the underside of the refueling floor is investigated based on drop of the cavity shield block when one of the three lift points of the lifted cavity shield block fail. The calculation is based on the local damage equations given in [Reference 7].

Various scenarios are addressed in Reference 1 (as supplemented by References 5 and 6) and are evaluated in detail below. The maximum weight of the block that can be safely lifted to clear a height of 1'-0" above the floor is determined by equating the “Total Drop Energy Before Impact” to the “Target Energy Capacity.” The least value of the lifted weight from all scenarios was determined to be the governing weight of the reactor shield block.

The licensee used an energy method to demonstrate the adequacy of the impacted structural elements (beams, slabs, columns, and walls) under the drop of a shield block. The licensee calculated an impact energy of a block using the equation, $E_{IE} = P \times H$, where P and H are defined as block weight and drop height, respectively. The licensee did not consider any energy dissipation during the impact in the calculation of the impact energy. To determine the target energy absorbing capacity of a structural element, the licensee calculated the strain energy, which is the area under the load-deflection curve of a structural element, using the equation, $E_{SE} = R \times [0.5 \times \Delta_e + (\mu \times \Delta_e - \Delta_e)]$, where R, μ and Δ_e are defined as yield resistence, allowable ductility ratio and elastic deformation of a structural element in Reference 1. This energy approach is acceptable to the staff because: (1) this indicates a capacity of the structural element to absorb the impact energy without sustaining structural damage, and (2) energy dissipation was not considered during the impact in order to calculate a conservative value of the impact energy.

The designated safe load path, hoisting height restrictions, and the maximum weight of the reactor cavity shield block and rigging are described in applicable procedures. When handling the top layer of shield blocks weighing more than 110 tons, crane controls incorporate travel limits and hoisting height restrictions.

According to the licensee, the following load drop scenarios envelope all potential load drops of the reactor shield blocks on the reactor cavity and on relevant floor areas. The load movements are limited to the areas shown in the application (Reference 1 as revised per Reference 5). The licensee analyzed the following five different load drop scenarios (Reference 1 as supplemented by References 5 and 6):

(1) shield block drop during initial lift from Unit 2 (3) cavity and during laydown on top of Unit 3 (2) shield blocks;
(2) full drop of a shield block on a single column;
(3) full drop of a shield block on a system of two adjacent slabs with a beam between the slabs;
(4) full drop of a shield block on two adjacent columns; and
(5) full drop of a shield block on a wall at column row 44.

The licensee provided the results of the impact analyses in Reference 1 (as supplemented by References 5 and 6). The results show that all calculated impact energies (except one load drop condition in Scenario 2, which is discussed below), are smaller than the corresponding strain energies for drop of the top layer shield block from 1 foot above the refuel floor. This demonstrates that the structural elements are capable of handling a shield block weight of 116 tons dropping from a height of up to 1 foot above the floor without sustaining structural damage.

Scenario 1 consists of all cases of the drop of a shield block on the reactor cavity. In the governing condition, the top layer block is postulated to drop on the two middle and the two bottom layer blocks. The top layer block may drop over the middle layer blocks situated in the reactor cavity, while being lifted one foot above the floor or a drop height of 3'-0". The “Total Impact Energy” for a block weighing 116 tons (232 kips), dropping from 3'-0", without considering the dissipation of the energy due to the impact, is determined to be 696 kip-ft. and the maximum weight that can be safely dropped from a height of 3'-0" is calculated to be 194.82 tons. Since this is greater than the weight of the heaviest shield block, the results of this Scenario are acceptable.

The licensee stated (References 1, 5, and 6) that due to a large area of impact, scabbing on the underside of the floor is not expected to occur if the failure occurs above the crane hook. If one of the three lift points of the block fails, there may be an impact on a smaller area of the floor from a corner of the block, which may cause scabbing. The evaluation of the potential scabbing of the refueling floor slab is based on an assumption that a piece of broken block weighing 10,000 lbs. dropping from a height of 2'-6" (30") impacts the concrete floor slab area of 1 sq. ft. The evaluation shows that this condition will result in scabbing only if the thickness of the floor slab is 13.708" or less. Since the floor slab thickness is 18", no scabbing will result.

The staff considers scabbing and potential for penetration of the floor to be more of a concern for high velocity impact. The drop velocity is expected to be low for this scenario of a tilted drop. Also, the licensee's analysis included a number of assumptions regarding energy conservation and uncertainties regarding the impact geometry on the referral floor. Recognizing this, the staff requested the licensee to verify that if cracking occurs on the underside of the floor slab or beams as a result of a load drop, safety-related equipment will not be adversely affected. In response (Reference 6), the licensee performed a walk-down of both units in the path of potential impact, and determined that most of the safety-related equipment in the area of concern is anchored by deeply embedded supports. Only one safety-related support each, in Unit 2 and Unit 3, used a combination of embedded plates and concrete expansion anchors (CEAs). Another Unit 3 safety-related cable tray support system used CEAs in addition to embedded plates and struts. This cable tray is very lightly loaded compared to its design rating. Given this information, the staff concluded that cracking of the refuel floor slab will not impact safety-related equipment supported from the underside of the slab in an adverse manner.

With regard to top layer shield blocks, the movement path for the top layer blocks has been restricted by the design of crane limit switches, which will limit the movement of the crane hook (center of gravity of the block) as shown in References 3 and 5. The path shows that the center of gravity of the shield blocks will not pass near the center of the column along Row M. The center of gravity of the shield blocks will be a minimum of 12'-5" from columns along Row M. Accordingly, Scenario 2 does not apply.

In Scenario 3, 4, and 5, the maximum weights that can be safely dropped from a height of 1'-0" to have the "Total Impact Energy" value equal to the "Net Available Strain Energy" value are determined to be 237.5, 139.44 and 317.5 tons respectively. Since all these weights are greater than the heaviest shield block, the results of Scenarios 3, 4, and 5 are acceptable.

The Unit 3 middle and bottom layers weighed less than 110 tons and the Unit 2 layers are expected to weigh at or below 110 tons. Since the single failure proof capacity of the crane is not exceeded when lifting these blocks, a load drop analysis is not needed for these blocks.

Based on its review as discussed above, the staff concludes that handling the reactor cavity shield blocks with the reactor building 2/3 crane during reactor power and refueling operations is acceptable as the consequence of the top layer shield block dropping from a height of up to 1'-0" above the floor is expected to cause no structural damage to the building elements with no scabbing of the floor slab and is not expected to impact adversely the operation of safety-related equipment supported from underside of the floor slab in the area of concern.

3.2 Controls While Handling Heavy Loads

The licensee will implement the following controls, in accordance with the NUREG-0612, Appendix A guidelines, when handling loads greater than 110 tons:

- Mechanical stops, electrical interlocks, or similar automatic controls to restrict travel outside the designated load path.

- Mechanical stops, electrical interlocks, or similar automatic controls to prohibit lifting the reactor cavity shield blocks over one foot above the refueling floor.

The licensee will modify the existing procedural controls to include the following to ensure that the assumptions used in the load drop analysis are maintained:

- The applicable procedures will describe the weight of the shield blocks assumed in the analysis, the safe load path, and the hoisting height restrictions for the reactor cavity shield blocks.

- The applicable procedures will ensure that the mechanical stops, electrical interlocks, or automatic controls are not bypassed during handling of the reactor cavity shield blocks, unless a particular piece is shown to weigh less than 110 tons.

The following defense-in-depth approach provides reasonable assurance that handling of the reactor cavity shield blocks, weighing up to 116 tons, using the reactor building crane at DNPS will be performed in a safe manner without affecting the operation of safety-related SSCs:

(1) the licensee's commitments to meet the NUREG-0612 Section 5.1.1 (Phase I) guidelines for safe load paths, procedures, crane operators, special lifting devices, general lifting devices, crane inspection, crane testing and maintenance, and crane design;

(2) the licensee's commitments to meet the NUREG-0612 Section 5.1.2 (Phase II) guidelines for increased handling system reliability ensuring single-failure-proof handling capability for loads up to 110 ton; and

(3) demonstration of acceptable consequences for postulated load drops provided by the load analyses.

Therefore, the staff finds the proposed changes in the DNPS UFSAR acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes an inspection or a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 37576). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

On the basis of preceding discussions, the staff concluded that the proposed revisions to the DNPS UFSAR are in accordance with NUREG-0612 guidelines. To ensure the continued safety of the fuel and safety-related equipment during movement of the reactor cavity shield blocks, the licensee has (1) committed to meet the NUREG-0612, Section 5.1.1 guidelines for safe load paths, procedures, crane operators, special lifting devices, general lifting devices, crane inspection, crane testing and maintenance, and crane design; (2) committed to meet the NUREG-0612 Section 5.1.2 (Phase II) guidelines for increased handling system reliability ensuring single-failure-proof handling capability for loads up to 110 tons; and (3) demonstrated acceptable consequences for postulated load drops using analyses in accordance with the NUREG-0612, Appendix A guidelines. The staff finds that the proposed revisions to the DNPS UFSAR are acceptable and will not represent a decrease in oversight and control of the movement of heavy loads in the vicinity of safety-related structures, systems and components.

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. Therefore, the licensee’s proposed changes are acceptable.

7.0 REFERENCES

1. Letter from K. R. Jury (Exelon Generation Company, LLC) to U. S. NRC, “Request for License Amendment Related to Heavy Loads Handling,” dated February 26, 2003.

2. Letter from U. S. NRC to J. L. Skolds (Exelon Generation Company, LLC), “Dresden Nuclear Power Station, Units 2 and 3 - Request for Additional Information Regarding Heavy Loads Handling Amendment Request,” dated May 23, 2003.

3. Letter from Exelon Generation Company to U. S. NRC, “Dresden Nuclear Power Station Units 2 and 3 - Additional information regarding request for license amendment related to load handling,” dated June 12, 2003.

4. Letter from Exelon Generation Company to U. S. NRC, “Dresden Nuclear Power Station Units 2 and 3 - Additional information regarding request for license amendment related to load handling,” dated July 25, 2003.

5. Letter from Exelon Generation Company to U. S. NRC, “Dresden Nuclear Power Station Units 2 and 3 - Additional information regarding request for license amendment related to load handling,” dated September 11, 2003.

6. Letter from Exelon Generation Company to U.S. NRC, “Dresden Nuclear Power Station Units 2 and 3 - Additional information regarding request for license amendment related to load handling,” dated October 9, 2003

7. Second ASCE Conference on “Civil Engineering and Nuclear Power, Volume V: Report of the ASCE Committee on Impactive and Impulsive Loads,” September 1980, Knoxville, Tennessee.

8. NUREG-0612, “Control of Heavy Loads at Power Plants,” July 1980.

9. ACI 349-97, “Code Requirements for Nuclear Safety Related Concrete Structures.”

10. ACI 318-99, “Building Code Requirements for Structural Concrete.”

Principal Contributors: H. Wagage
J. Rajan

Date: October 10, 2003