September 21, 2001

Mr. Gary Van Middlesworth
Site Vice President
Duane Arnold Energy Center
Nuclear Management Company, LLC
3277 DAEC Road
Palo, IA 52324-0351

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT FOR REVISED THERMAL-HYDRAULIC ANALYSIS FOR SPENT FUEL POOL (TAC NO. MB0596)

Dear Mr. Van Middlesworth:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 242 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Operating License in response to your application dated November 17, 2000, as supplemented on February 16, and April 9, 2001.

The amendment changes the license to allow refueling activities in accordance with a revised thermal-hydraulic analysis based upon use of advanced core designs employing advanced fuel, increased fuel burnup, increased cycle length, and increased reload batch size. The revised analysis also corrects several input parameter discrepancies in the existing analysis.

A copy of the Safety Evaluation and *Federal Register* Notice of Issuance is also enclosed.

Sincerely,

Brenda L. Mozafari, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures:
1. Amendment No. 242 to License No. DPR-49
2. Safety Evaluation
3. Notice

cc w/encls: See next page

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Site Vice President
Duane Arnold Energy Center
Nuclear Management Company, LLC
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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.242
License No. DPR-49

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee) dated November 17, 2000, as supplemented February 16 and April 9, 2001, complies with the standards and requirements of the Atomic Energy Act of l954, 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;

   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 l0 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 242, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: September 21, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 242

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised areas are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License Page 3	License Page 3

2.B.(2) NMC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended as of June 1992 and as supplemented by letters dated March 26, 1993, and November 17, 2000.

2.B.(3) NMC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

2.B.(4) NMC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated radioactive apparatus components;

2.B.(5) NMC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

Maximum Power Level

2.C.(1)* NMC is authorized to operate the Duane Arnold Energy Center at steady state reactor core power levels not in excess of 1658 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 242, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 242 TO FACILITY OPERATING LICENSE NO. DPR-49

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By application dated November 17, 2000, as supplemented February 16 and April 9, 2001, Nuclear Management Company, LLC (NMC, or the licensee) requested a license amendment for the Duane Arnold Energy Center (DAEC). The licensee proposes to allow refueling activities in accordance with a revised spent fuel pool thermal-hydraulic analysis based upon use of advanced core designs employing advanced fuel, increased fuel burnup, increased cycle length, and increased reload batch size. The revised analysis would also correct several input parameter discrepancies in the existing analysis. The revised spent fuel pool thermal-hydraulic analysis supercedes the analysis previously provided by the licensee in a letter dated October 3, 1997. The revised analysis was performed by the licensee in anticipation of an increase in the maximum power level of DAEC as requested by the licensee in a letter dated November 16, 2000. The NRC's final action regarding the licensee's request to increase the maximum power level is pending.

The February 16, 2001, supplement was included in the original Federal Register notice (66 FR 13793). The April 9, 2001, supplement provided clarifying information and did not expand the scope of the original Federal Register notice.

2.0 EVALUATION

The proposed action would change the license to allow refueling activities in accordance with a revised thermal-hydraulic analysis needed to support DAEC plans to pursue advanced core designs beginning with Cycle 18, including the use of General Electric (GE)-14 fuel, increased fuel burnup, increased cycle length, and increased reload batch size. The proposed action revises the thermal-hydraulic analysis for the spent fuel pool (SFP) submitted to the NRC by letter dated October 3, 1997. The proposed action also corrects discrepancies made in the existing thermal-hydraulic analysis. The proposed action was reviewed by the NRC staff and its contractor, Brookhaven National Laboratory (BNL). The detailed evaluation is contained in the attached Technical Evaluation Report and is summarized below.

NUREG-0800, "Standard Review Plan," provides criteria related to the design and performance of the spent fuel pool. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," provides methods acceptable for the licensee to implement General Design Criteria 61 of

Appendix A to 10 CFR Part 50 which requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. NRC memorandum, “Office Technical Position for Review and Acceptance of Spent Fuel Storage and Handling Applications,” dated April 14, 1978, and modified by Addendum dated January 18, 1979, provides key design criteria and regulatory guidance for new spent fuel storage racks.

The licensee submitted a revised thermal-hydraulic analysis, which included maximum SFP temperatures, minimum time-to-boil after loss of forced cooling, and local water and fuel cladding temperatures. The licensee calculated the maximum bulk SFP temperatures for the following three cases: (a) planned full core offload scenario with full core discharge beginning at 60 hours after reactor shutdown, with one train of the fuel pool cooling and cleanup (FPCCU) system in operation; (b) planned full core offload scenario, the same scenario as case (A) except that two trains of FPCCU are in operation; and (c) unplanned full core offload scenario consisting of a normal refueling outage of 36 days, followed by 45 days of full power operation and a subsequent unplanned discharge of the full core to the SFP beginning 60 hours after reactor shutdown, with two trains of FPCCU in operation. Based on its review, the NRC staff concluded that the methodology and assumptions used by the licensee to calculate the decay heat loads and to calculate the SFP bulk temperatures met the intent of the applicable NRC guidelines. The maximum SFP bulk temperatures of the revised hydraulic analysis are below the onset of boiling and are below the SFP temperatures approved by the NRC staff for the current thermal-hydraulic analysis.

The licensee also evaluated the effect of a complete loss of forced cooling to the SFP, which was assumed to occur when the SFP was at the maximum SFP bulk temperature. The calculated minimum time from the loss of pool cooling at peak pool water temperature until the pool boils for the worst case was 3.8 hours for the revised analysis, which was a slight decrease from the 4.5 hours of the current analysis, but still substantially longer than the 2 hours required to align the emergency service water system to provide makeup water to the SFP. In addition, various other sources of emergency makeup water would be available in less than 2 hours. Based on its review, the NRC staff concluded that in the unlikely event that there is a complete loss of cooling, the licensee is capable of aligning the makeup water from various sources to the pool before boiling begins and that makeup water will be supplied at a rate which exceeds the boil-off rate, and that cooling the SFP by adding makeup water in the unlikely event that there is a complete loss of cooling to the SFP conforms to NRC guidance.

The NRC staff has completed its evaluation of the proposed action and concludes that the proposed revision to the thermal-hydraulic analysis complies with the applicable regulatory documents and will allow for the continued safe storage of spent fuel.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATIONS

Pursuant to 10 CFR 51.21, 10 CFR 51.32, and 10 CFR 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on July 24, 2001 (66 FR 38442), for this amendment. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The staff 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: Technical Evaluation Report

Principal Contributors: B. Thomas
D. Shum

Date: September 21, 1001

TECHNICAL EVALUATION REPORT

Review of
Duane Arnold Energy Center Operating License Change Request
(TSCR-040)
Regarding A Revised Thermal-Hydraulic Analysis For The Spent Fuel Pool

Docket Number: 50-331

TAC Number: MB0596

Prepared by:
Jae Jo and Edward Grove
Department of Energy Sciences and Technology
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June 25, 2001

Prepared for:
B.E. Thomas and D.H. Shum, NRC Technical Monitors
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

JCN J-2427, Task Order No. 13

ABSTRACT

The Nuclear Management Company, LLC is proposing a change to Operating License Number DPR-49. This report provides the results of Brookhaven National Laboratory's evaluation related to the thermal hydraulic aspects of this proposed operating license change. This proposed change allows for the use of advanced core designs including GE-14 fuel, increased fuel burnup, increased cycle length, and increased reload batch size. In addition, this proposed change also revises some discrepancies in the previously submitted thermal hydraulic analysis dated October 3, 1997, in support of the current operating license.

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TECHNICAL EVALUATION REPORT
RELATED TO THE
DUANE ARNOLD ENERGY CENTER OPERATING LICENSE
CHANGE REQUEST (TSCR-040)
REVISED THERMAL-HYDRAULIC ANALYSIS
FOR THE SPENT FUEL POOL

1.0 INTRODUCTION

By letter dated November 17, 2000, the Nuclear Management Company, LLC (NMC, or the licensee) requested a revision to Operating License Number DPR-49 for the Duane Arnold Energy Center (DAEC) (Reference 1). DAEC is a boiling-water reactor (BWR) which commenced commercial operation in February, 1975. The purpose of this Operating License (OL) Amendment request is to revise and supercede the previously submitted, and approved, thermal-hydraulic analysis for the spent fuel pool. Specifically, this request proposes advanced core designs including GE-14 fuel, increased fuel burnup, increased cycle length, and increased reload batch size. The licensee included a licensing report entitled "Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie (Revision 4)" prepared by Holtec International, which contained the revised thermal-hydraulic analysis to justify the proposed changes. Note that the revised thermal-hydraulic analysis was performed in anticipation of increasing the thermal power output by 15.3% (from 1658 MWt to 1912 MWt) as requested by the licensee in letter dated November 16, 2000.

Brookhaven National Laboratory (BNL), in support of the NRC Office of Nuclear Reactor Regulation (NRR), reviewed the proposed revisions, and requested clarifying and additional information, which was transmitted by the NRC to the licensee electronically on March 8, 2001. A conference call was held between representatives of the licensee, BNL, and the NRC on March 27, 2001, to discuss the licensee's proposed responses. The licensee submitted responses to these requests by letter dated April 9, 2001 (Reference 2).

2.0 BACKGROUND

In the current submittal, the licensee referenced previously submitted requests for OL Amendments, in which IES Utilities Inc., on March 26, 1993, requested an Operating License/Technical Specification Amendment to allow rerack of the DAEC spent fuel pool (Reference 3). In support of that request, a thermal-hydraulic analysis was performed for four refueling scenarios, the most limiting which represented a full core offload, starting 120 hours after shutdown, following an 18-month cycle, using two trains of the Fuel Pool Cooling and Clean Up (FPCCU) system to provide spent fuel pool (SFP) cooling. This Operating License/Technical Specification Amendment request was granted by the NRC as Amendment No. 195 (Reference 4).

IES Utilities Inc., on October 3, 1997, requested an OL Amendment to allow the start of core offload as soon as 60 hours after shutdown (Reference 5). In support of this request, the licensee submitted Revision 2 of the "Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie." This OL Amendment request was granted as Amendment No. 222 (Reference 6).

In addition to the proposed changes in fuel design, the Revision 4 submittal includes corrections of recently discovered discrepancies which existed in the licensee's previous submittals. These discrepancies are summarized in Table 1 and reviewed as a part of this submittal.

Table 1 Design Value Discrepancies

Calculation Input Parameter	Value Used in 1993 and 1997 Submittals	Revised Design Value
Reactor Cavity Volume	157,000 gallons	136,000 gallons
RHR Service Water Flow Rate	4800 gpm	4080 gpm
Emergency Service Water Makeup Flow	75 gpm	59.5 gpm
Height of spent fuel racks above bottom of SFP	14.67 ft	15.146 ft
Height of active fuel above bottom of SFP	13.52 ft	13.96 ft
RHR Flow Rate to SFP	1300 gpm	2000 gpm

This Technical Evaluation Report (TER) presents the results of the review of this proposed OL Amendment request by BNL in support of the NRR. BNL's evaluation is focused on the thermal-hydraulic analysis of the licensee's submittal. The licensee's revised thermal-hydraulic analysis included maximum SFP temperatures, minimum time-to-boil after a loss of forced cooling, and local water and fuel cladding temperatures. This TER reviews the licensee's analysis of the maximum SFP temperatures, and the minimum time-to-boil after a loss of forced cooling. Previous submittals and NRC approvals are referenced wherever necessary.

NUREG-0800, "Standard Review Plan" provides criteria related to the design and performance of the spent fuel pool. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis" provides methods acceptable for the licensee to implement General Design Criteria 61 of Appendix A to 10 CFR Part 50 which requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. The NRC memorandum entitled "Office Technical Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978, and modified by Addendum dated January 18, 1979, provides key design criteria and regulatory guidance for new spent fuel storage racks.

3.0 EVALUATION

3.1 Spent Fuel Pool Cooling System

The DAEC Fuel Pool Cooling and Clean Up (FPCCU) system removes the decay heat and radioactivity released from the spent fuel elements stored in the spent fuel pool (SFP). The FPCCU system consists of the SFP, two full capacity pumps, two heat exchangers, two filter demineralizers, two skimmer surge tanks and associated piping, valves and instrumentation. The FPCCU heat exchangers are cooled by the Reactor Building Closed Cooling Water system (RBCCW). As stated in the licensee's submittal, either loop of the Residual Heat Removal (RHR) system can be operated in the Supplemental Fuel Pool Cooling (RHR-SFPC) mode to provide additional cooling to the SFP. The RHR heat exchangers are shell-and-tube heat exchangers with reactor water on the shell side and river water on the tube side. If the RHR system is to be used to provide cooling to the SFP, the RHR system must be capable of rejecting the decay heat of the core plus the residual decay heat in the previously discharged fuel assemblies in the SFP.

In the RHR-SFPC mode of operation, a portion of the flow discharged from the heat exchanger of one loop of the RHR system is diverted to the SFP to cool the SFP. The water discharged from the RHR system to the SFP must be returned to the reactor cavity to prevent the SFP or skimmer surge tanks from overflowing. This can be achieved either by opening the transfer canal gates and allowing water to flow from SFP to the flooded reactor cavity, or by operating the FPCCU system with suction from skimmer surge tanks and discharge to the reactor cavity. In the calculation of the maximum SFP bulk temperatures, the licensee assumed that the transfer canal gates were open and the FPCCU system was off. This assumption is conservative, since this configuration would provide less cooling capacity for the SFP by not taking a credit for the additional heat removal capacity of the FPCCU system.

Both the RHR and FPCCU heat exchanger heat removal rates were modeled using constant temperature effectiveness values, which were based on the heat exchanger performance with the hot water inlet temperature of 120° F. As temperature effectiveness values would rise with increasing hot water temperature, the use of constant temperature effectiveness values was conservative.

3.2 Maximum Bulk SFP Temperatures

In Revision 4, the licensee calculated the maximum bulk SFP temperatures for the following three cases.

Case A: Planned full core offload scenario. Full core discharge begins at 60 hours after reactor shutdown, with one train of FPCCU initially in operation.

Case B: Planned full core offload scenario. This scenario is identical to Case A (i.e., 60 hours after reactor shutdown) except that two trains of FPCCU are in operation at the beginning of the fuel discharge operation.

Case C: Unplanned full core offload scenario. This offload scenario consists of a normal refueling outage with a length of 36 days followed by 45 days of full power operation and a subsequent unplanned discharge of the full core to the SFP. Full core discharge begins at 60 hours after reactor shutdown, with two trains of FPCCU in operation.

These were the same three cases presented in Revision 2 except for Case C, the unplanned full core offload scenario. The current submittal (Revision 4) assumed for Case C that recently loaded fresh fuel had undergone 45 days of exposure, while Revision 2 assumed that recently loaded fresh fuel had undergone 18 months of exposure. The assumption in Revision 4 was consistent with the actual scenario, while the assumption in Revision 2 was conservative. The scenarios and analyses in Revision 2 were reviewed and approved for Amendment No. 222 by the NRC.

In all three cases, it was assumed that the core offload proceeded at a rate of six fuel assemblies per hour. It was also assumed that the FPCCU would be used to cool the SFP until the bulk pool temperature reached 120° F, and that, once the pool temperature reached 120° F, the RHR-SFPC would be initiated and the FPCCU operation would be discontinued. In this mode of RHR operation (RHR-SFPC), a portion of the flow discharged from the RHR system heat exchangers was diverted to the SFP. The water discharged from the RHR system to the SFP was returned to the reactor cavity through the open transfer canal gates. Discontinuing the FPCCU system was a conservative assumption since it would result in a smaller total cooling capacity for the combined SFP-reactor cavity system and higher bulk temperatures.

The licensee used the Holtec quality assurance (QA)-validated computer program, DECOR, to determine the decay heat generation rates of both the previously and freshly discharged fuel assemblies, as was used in Revision 2. This program can perform decay heat calculations using either Branch Technical Position ASB 9-2, or the ORIGIN2 computer code. For both analyses (Revisions 2 and 4), the ORIGIN2 option was used. All fuel assemblies were assumed to have been irradiated to the appropriate maximum burnup level. Based on this review, BNL concurs that the methodology and assumptions the licensee used to calculate the decay heat loads meet the intent of the applicable NRC guidelines.

Once the transient decay heat load with respect to time was determined for each case, the licensee solved the differential equations representing the transient heat balances and the thermal responses of the SFP and reactor vessel cavity, using the Holtec QA-validated computer program, MULPOOLD, to obtain the bulk SFP water temperatures. This program utilized the constant temperature effectiveness values (discussed in Section 2.2) for the heat removal capability of the heat exchangers. The assumptions discussed above were also incorporated into the model. Based on this review, BNL concurs with the methodology and assumptions the licensee used to calculate the SFP bulk temperatures.

The major differences of Revision 4 from Revision 2 are:

1. Revision 4 allows for advanced core designs including GE-14 fuel, increased fuel burnup, increased cycle length and increased reload batch size. This change will result in higher decay heat.

2. The fuel exposure time for Case C, unplanned full core discharge scenario was revised. Revision 4 assumes that recently loaded fresh fuel has undergone 45 days of exposure, while Revision 2 assumed that recently loaded fresh fuel had undergone 18 months of exposure. The assumption in Revision 4 is consistent with the actual scenario, while the assumption in Revision 2 was conservative. This modification of assumption will result in lower decay heat for Case C for Revision 4.

3. Revision 4 corrects discrepancies in six input parameters, which existed in the calculations in the licensee's previous submittals (including Revision 2) as listed in Table 1. The licensee stated that corrections of two of the six parameters (the flow rates for RHR-SFPC mode and RHR Service water) have the most impact on the maximum SFP bulk temperature, and that corrections of other parameters produced results that were not significantly different from the results of Revision 2. Specifically, the correction of the six parameters would decrease the SFP bulk temperature by about 9° F, most of which was due to the increased flow rate of the RHR-SFPC mode. This decrease in the SFP temperature was verified by BNL's independent calculations.

Table 2 provides a comparison of the coincident decay heat and maximum bulk pool water temperature calculated by the licensee for each case for Revisions 2 and 4.

It is noted in the table that:

1. The decay heat for Case A and B in Revision 4 increased substantially from Revision 2 due to the advanced core design, while the increase was smaller for Case C. This was due to the fact that Revision 4 assumed that recently loaded fresh fuel for Case C had undergone 45 days of exposure, while Revision 2 assumed 18 months of exposure, resulting in reduction in the calculated increase in the decay heat under Case C.

Table 2 Coincident Decay Heat and Maximum Bulk Pool Water Temperature

Case Scenario		Revision 2		Revision 4	
		Coincident Net Decay Heat (10^6 Btu/hr)	**Maximum Bulk SFP Water Temperature (°F)**	**Coincident Net Decay Heat (10^6 Btu/hr)**	**Maximum Bulk SFP Water Temperature (°F)**
A	Planned full core offload	20.87	154.31	23.87	152.50
B	Planned full core offload	20.87	154.30	23.87	152.50
C	Unplanned full core offload	22.54	159.87	23.92	152.46

2. The maximum bulk pool water temperatures of Case A and B were almost identical for both Revision 2 and Revision 4. Cases A and B differ in the number of FPCCU trains used before aligning the RHR for supplemental fuel pool cooling: 1 train for Case A and 2 trains for Case B. The very close temperatures of these two cases indicated that the limited amount of FPCCU cooling was rapidly overcome by the large cooling capacity of the RHR system and became insignificant once RHR-SPFC was initiated.

3. The maximum bulk temperature for Revision 4 is lower than Revision 2 for each case, although the decay heat is higher. This is due to the corrections of discrepancies in some input parameters, which existed in the licensee's previous submitted calculations, as discussed above.

BNL notes that the maximum bulk SFP water temperatures for the planned offload cases (A and B) for Revision 4 are greater than the maximum pool temperature recommended in the SRP for planned refueling offloads. However, the Revision 4 temperatures are lower than those of Revision 2, which the NRC staff reviewed and accepted in Amendment No. 222. The staff accepted the results of the Revision 2 in Amendment No. 222, since the licensee's analysis previously documented the SFP structure's acceptability during planned operations for temperatures up to 165° F (Reference 6). Therefore, BNL finds the maximum SFP temperature of 152.5° F acceptable for the planned outages. BNL also notes that the maximum SFP water temperature for the unplanned offload case (Case C) for Revision 4 is lower than that of Revision 2, which the NRC staff also reviewed and accepted in Amendment No. 222. The maximum bulk SFP temperatures of both Revision 2 and Revision 4 are below the onset of boiling, as recommended by the SRP.

Based upon this review, BNL concludes that the maximum SFP bulk temperatures, as calculated by the licensee, are acceptable.

3.3 Effect of SFP Boiling

In the unlikely event that there is a complete loss of cooling, the SFP water temperature will begin to rise, and eventually reach boiling and start to lose the SFP water.

The licensee performed an analysis to demonstrate the minimum time-to-boil following a loss of forced cooling and the maximum boil-off rate once bulk SFP boiling begins. The licensee also calculated the time for the SFP water level to drop to the active fuel height following a loss of forced cooling. The makeup water should be provided before this time at a rate exceeding the boil-off rate. The loss of forced cooling was assumed to occur when the SFP was at the maximum SFP bulk temperature. The licensee solved an equation governing the SFP thermal response with the RHR-SFPC mass flow rate set at zero, using the Holtec QA-validated computer program, TBOIL, with and without a coincident cask pit gate seal failure. Table 3 compares the licensee's results for the minimum time-to-boil, maximum boil-off rate, and time available to provide makeup water (which is the time for the SFP water level to drop to the active fuel height following a loss of forced cooling) with a coincident cask pit gate seal failure for Revisions 2 and 4. (The results for the case without a coincident cask pit gate seal failure are not presented here since it is less conservative than the case with a coincident cask pit gate seal failure; the time-to-boil was longer and the boil-off rate was less).

Table 3 Comparison of Boil-Off Calculations

	Rev. 2		Rev. 4	
	Cases A & B	Case C	Cases A & B	Case C
Time-to-Boil	1.15 Hrs	0.96 Hrs	1.07 Hrs	1.09 Hrs
Boil-off Rate	48.82 gpm	52.78 gpm	53.05 gpm	52.23 gpm
Time to Provide Makeup Water*	5.0 Hrs	4.5 Hrs	3.8 Hrs	3.9 Hrs

* 'Time to Provide Makeup Water' is the time available before the SFP water level drops to the active fuel height following a loss of forced cooling.

In its response to the request for additional information (RAI) (Reference 2), the licensee referenced the results of a Special Test (SpTP-198), which was performed in September 1996 that demonstrated the capability of Emergency Service Water System (ESW) to provide water to the SFP in 2 hours at the rate of 59.5 gpm from the ESW system, a Seismic Category I system. The calculated maximum boil-off rate for Revision 4, 53.05 gpm, is less than the makeup capacity of 59.5 gpm available from the ESW system. It is not significantly different from that of Revision 2, 52.78gpm, which the NRC staff reviewed and accepted in Amendment No. 222. The licensee further stated in its response to the RAI that various other sources of emergency makeup water would be available in less than 2 hours such as the RHR Low Pressure Injection, Condensate Service Water and Demineralized Water Systems at rates exceeding the boil off rates. The calculated minimum time from the loss-of-pool cooling until the SFP water level drops to the active fuel height, which is the time available to align the makeup water, for the worst case was 3.8 hours for Revision 4, which was a slight decrease from 4.5 hours of Revision 2, but remained substantially longer than 2 hours to align the ESW water. The licensee stated that there were operating instructions in place to ensure alignment of the ESW system to the SFP when needed. Based on the information the licensee provided, and review of the calculated results, BNL concludes that in the unlikely event that there is a complete loss of cooling, the licensee is capable of aligning the makeup water from various sources to the SFP before the water level drops to the active fuel height, and that make-up water will be supplied at a rate which exceeds the boil-off rate.

In view of the decreased time available to provide makeup water, the licensee provided a "Basis for No Significant Hazard Consideration." BNL's evaluation was limited to the review of the calculated results, and did not include the review of the "Basis for No Significant Hazard Consideration."

Based on this review, BNL finds that in the unlikely event that there is a complete loss of cooling, the licensee is capable of aligning the makeup water from various sources to the pool before the water level drops to the active fuel height and makeup water can be supplied at a rate which exceeds the boil-off rate. BNL further concludes that cooling the SFP at DAEC by adding makeup water during an unlikely event resulting in a complete loss of SFP cooling confirms with the guidance described in the SRP and is, therefore, acceptable.

4.0 CONCLUSIONS

DAEC has proposed a revision to its Operating License DPR-49. The purpose of this revision is to revise the previously submitted, and approved, thermal-hydraulic analysis. Specifically, this request proposed advanced core designs including GE-14 fuel, increased fuel burnup, increased cycle length, and increased reload batch size. To support these changes, the licensee submitted a licensing report entitled "Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie (Revision 4)", prepared by Holtec International.

Based upon our evaluation of the thermal hydraulic aspects related to the proposed Operating License revision, BNL concludes that the proposed revision complies with the applicable regulatory documents (i.e., NUREG-0800, RG 1.13 and the Office Technical Position). This will allow for the continued safe storage of spent fuel. The BNL review was limited to the thermal-hydraulic aspects associated with the proposed operating license amendment request.

5.0 REFERENCES

1. Letter, Middlesworth (NMC) to NRC, dated November 17, 2000, Request for Operating License Change (TSCR-040) - Revised Thermal Hydraulic Analysis for the Spent Fuel Pool.

2. Letter, Putnam (NMC) to NRC, dated April 9, 2001, Response to Request for Additional Information (RAI) to Request for Operating License Change (TSCR-040) - Revised Thermal Hydraulic Analysis for the Spent Fuel Pool.

3. Letter, Liu (IES) to Murley (NRC), dated March 26, 1993, Spent and New Fuel Storage.

4. Letter, Pulsifer (NRC) to Liu (IES), dated February 24, 1994, Duane Arnold Energy Center License Amendment No. 195.

5. Letter, Franz (IES) to Collins (NRC), dated October 3, 1997, Refueling Operations.

6. Letter, Laufer (NRC) to Liu (IES), dated April 2, 1998, Duane Arnold Energy Center License Amendment No. 222.

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-331

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 242 to Facility Operating License No. DPR-49 issued to Nuclear Management Company, LLC (the licensee), which revised the license for operation of the Duane Arnold Energy Center located in Linn County, Iowa. The amendment is effective as of the date of issuance.

The amendment modified the license to allow refueling activities in accordance with a revised thermal-hydraulic analysis based upon use of advanced core designs employing advanced fuel, increased fuel burnup, increased cycle length, and increased reload batch size. The revised analysis also corrects several input parameter discrepancies in the existing analysis.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on March 7, 2001 (66 FR 13793). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (66 FR 38442).

For further details with respect to the action see (1) the application for amendment dated November 17, 2000, and supplemented February 16, and April 9, 2001, (2) Amendment No. 242 to License No. DPR-49, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, http://www.nrc.gov/NRC/ADAMS/index.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 21st day of September 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Brenda L. Mozafari, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation