November 15, 2006

Mr. Gary Van Middlesworth
Vice President
Duane Arnold Energy Center
3277 DAEC Road
Palo, Iowa 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT REGARDING ELIMINATION OF MAIN STEAM LINE RADIATION MONITOR TRIP FUNCTION (TAC NO. MC8883)

Dear Mr. Van Middlesworth:

The U.S. Nuclear Regulatory Commission (NRC or the Commission), has issued the enclosed Amendment No. 261 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). This amendment revises Technical Specification (TS) Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," to eliminate the Main Steamline Radiation Monitor trip function.

The changes to the DAEC TSs are in response to the application by Nuclear Management Company, LLC (NMC, the former licensee) dated November 14, 2005, as supplemented by your letter of September 1, 2006. Amendment No. 260, issued on January 27, 2006, transferred the DAEC license from NMC to FPL Energy Duane Arnold, LLC (FPL Energy). By letter dated February 6, 2006, FPL Energy adopted all previous docketed requests before the NRC for review and approval.

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

Sincerely,

/RA/

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch III-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures:
1. Amendment No. 261 to License No. DPR-49
2. Safety Evaluation

cc w/encls: See next page

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Vice President
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Palo, Iowa 52324-9785

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October 12, 2006

FPL ENERGY DUANE ARNOLD, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 261
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[1] dated November 14, 2005, as supplemented by letter dated September 1, 2006, 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;

   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 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:

[1] The application was submitted by Nuclear Management Company, LLC (NMC, the former licensee). Amendment No. 260, issued on January 27, 2006, transferred license DPR-49 from NMC to FPL Energy Duane Arnold, LLC (FPL Energy, the current licensee). By letter dated February 6, 2006, FPL Energy adopted all previous docketed requests before the NRC for review and approval.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 261, are hereby incorporated in the license. FPL Energy Duane Arnold, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: November 15, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 261

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following page of the Facility Operating License No. DPR-49 with the attached revised page. The changed area is identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3

Replace the following page of Appendix A, Technical Specifications, with the attached revised page. The changed area is identified by marginal lines.

REMOVE	INSERT
3.3-57	3.3-57

2.B.(2) FPL Energy Duane Arnold, LLC, 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) FPL Energy Duane Arnold, LLC, 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) FPL Energy Duane Arnold, LLC, 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) FPL Energy Duane Arnold, LLC, 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:

<u>Maximum Power Level</u>

2.C.(1) FPL Energy Duane Arnold, LLC is authorized to operate the Duane Arnold Energy Center at steady state reactor core power levels not in excess of 1912 megawatts (thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 261, are hereby incorporated in the license. FPL Energy Duane Arnold, LLC shall operate the facility in accordance with the Technical Specifications.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 261 TO FACILITY OPERATING LICENSE NO. DPR-49

FPL ENERGY DUANE ARNOLD, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By application dated November 14, 2005, Nuclear Management Company, LLC (NMC, the former licensee), requested changes to the Technical Specifications (TSs) for the Duane Arnold Energy Center (DAEC). Amendment No. 260, issued on January 27, 2006, transferred the DAEC license (Facility Operating License No. DPR-49) from NMC to FPL Energy Duane Arnold, LLC (FPL Energy, the current licensee). By letter dated February 6, 2006, FPL Energy adopted all previous docketed requests before the U.S. Nuclear Regulatory Commission (NRC or the Commission) for review and approval. By letter dated September 1, 2006, FPL Energy provided a supplement pertaining to the requested license amendment. The supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 1, 2006 (71 FR 43533).

The proposed amendment would revise DAEC TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," to eliminate the Main Steam Line Radiation Monitor (MSLRM) trip function. Specifically, TS Table 3.3.6.1-1, Function 1.f, "Main Steamline Radiation - High," would be deleted and the Function 1.g, "Turbine Building Temperature - High," would be re-numbered as Function 1.f. The TS Bases would also be changed (under the TS Bases Control Program specified in TS 5.5.10) to reflect the above changes to Table 3.3.6.1-1.

The licensee's application provided the following background information regarding why the change is being requested:

> The purpose of the MSLRM is to generate an isolation signal on conditions of high radiation in the Main Steam Lines (MSL) that are indicative of a Design Basis Accident (DBA) - Control Rod Drop Accident (CRDA). The isolation signal will cause a trip and isolation of the Mechanical Vacuum Pump (MVP), which is used during plant startup to initially establish a vacuum condition in the main condenser, and will close the MSL drains. In addition, the reactor coolant sample valves in the Main Recirculation System also receive a signal to close.

Originally, the MSLRM also generated a Reactor Protection System [RPS] trip (SCRAM) and a Nuclear Steam Supply Shutoff System (NSSSS) trip (Main Steam Isolation Valve (MSIV) closure). The [boiling-water reactor] BWR Owners' Group (BWROG) developed a licensing topical report [LTR] (NEDO-31400-A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," 1991) that justified the removal of the RPS and NSSSS/MSIV trips on the MSLRM high radiation signal. NRC subsequently approved the BWROG LTR. The associated license amendment for the DAEC was granted as License Amendment # 182 in 1992.

Because not all BWRs have the MVP trip and sample valve isolations on the MSLRM signal, the BWROG topical report did not address these specific isolations. In addition, the topical report did not include the isolation of the MSL drains. Thus, these isolations were not removed during implementation of Amendment # 182 at the DAEC.

Due to obsolescence issues with the current MSLRM instruments, NMC seeks to eliminate this trip function as a cost-beneficial solution to replacement of the current instruments with those of a newer design. In addition to the immediate cost savings from not having to replace the existing instruments, there is a long-term cost savings from avoided maintenance costs (both corrective maintenance and surveillance testing/calibrations).

2.0 REGULATORY EVALUATION

As discussed in DAEC Updated Final Safety Analysis Report (UFSAR) Section 6.2.4, “Containment Isolation System,” the safety objective of the primary containment system is to provide the capability in conjunction with other safeguard features, including the containment isolation system to limit the release of fission products in the event of a postulated DBA so that offsite doses are held to a practical minimum, and do not exceed the values set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67.

The current DAEC design and licensing basis for the MSLRM containment isolation function is discussed in UFSAR Sections 7.3.1.1.8, 11.5.1, and 15.2.4. High radiation in the vicinity of the MSLs could indicate a gross release of fission products from the fuel. On indication of such a release, a containment isolation signal is generated. The isolation signal trips and isolates the MVP, closes the MSL drains, and closes the reactor coolant sample valves in the main recirculation system. The MSLRM containment isolation function is currently credited in the CRDA analysis in order to reduce the radiological consequences of the accident. In support of the proposed amendment, the licensee has reanalyzed the radiological consequences of a CRDA without taking credit for the MSLRM containment isolation function.

Based on a review of UFSAR Section 3.1; NUREG-0800, “Standard Review Plan” (SRP), Sections 6.2.4, 6.4, 13.2.1, 13.2.2, 13.5.2.1, 15.0.1, 15.4.9, and 18.0; and the licensee’s application dated November 14, 2005, the NRC staff identified the following regulatory requirements and guidance as being applicable to the proposed amendment:

- 10 CFR 50.67, “Accident source term,” insofar as it sets standards for radiological consequences of postulated accidents.

- 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 13, “Instrumentation and control,” insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety.

- 10 CFR Part 50, Appendix A, GDC 19, “Control room,” insofar as it requires that the control room be maintained in a safe, habitable condition under accident conditions by providing adequate protection against radiation.

- 10 CFR Part 50, Appendix A, GDC 20, “Protection system functions,” insofar as it requires the protection system to be designed to sense accident conditions and to initiate the operation of systems and components important to safety.

- 10 CFR Part 50, Appendix A, GDC 28, “Reactivity limits,” insofar as it requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core.

- 10 CFR 50.36, “Technical specifications,” insofar as it specifies the regulatory requirements related to the content of the TSs. Specifically, 10 CFR 50.36(c)(2)(ii), which requires that a TS limiting condition for operation (LCO) be established for each item meeting one or more of the following criteria:

  Criterion 1: Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

  Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

  Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

  Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

- NRC Information Notice (IN) 97-78, “Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times,” insofar that it provides guidance related to evaluating the acceptability of substituting manual actions for automatic actions.

- Regulatory Guide (RG) RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Regulatory Position 4.4.

- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants.”

3.0 TECHNICAL EVALUATION

3.1 Operator Performance

The NRC staff reviewed the proposed amendment with respect to operator performance. As discussed in the licensee’s application, under the current DAEC design if a CRDA were to occur during MVP operation, any resulting fission product release from the reactor would travel through the open MSIVs to the main condenser where the MVP would exhaust them to the plant Offgas Stack, resulting in an elevated release until the MSLRM trip automatically isolates this pathway. Under the proposed amendment, tripping the MVP and terminating the elevated release from the Offgas Stack would be a manual action performed by operators.

In the request, the licensee referenced NRC IN 97-78, and stated that it used this guidance to review the following factors to determine whether substituting manual for automatic actions is appropriate: (1) whether there are sufficient alarms or other indications that can be used to diagnose the event; (2)

whether plant procedures and training exist to guide the operator response or actions; and (3) the sensitivity of the timing of these actions (including the consequences of any errors in executing those actions).

Alarms and Other Indications

The licensee stated that there are several alternate indications of fuel damage and offsite release. For example, the normal range/post-accident range radiation monitoring system (KAMAN) alarms in the main control room, and the alarm settings, which are in accordance with the Offsite Dose Assessment Manual (ODAM), are well below radiation levels anticipated during any analyzed accident. Backup monitoring is provided by the Offgas Stack radiation monitors. These monitors alarm in the control room and cause Engineered Safety Feature (ESF) trips on a high radiation signal. The licensee stated that during a CRDA, the alarm settings would be exceeded quickly (estimated time approximately 2-5 minutes) and prompt the operator action to trip the MVP.

Plant Procedures, Training, and Operator Actions

Regarding procedures, the alarms mentioned above correspond to a section in an Annunciator Response Procedure (ARP). The ARPs direct operators to respond to the alarms, including taking actions in associated Abnormal Operating Procedures (AOPs) and Operating Instructions (OIs). The AOP for the alarms discussed above directs the operator to secure the MVP. Training is provided to operators on ARPs, AOPs, and OIs as part of the annual license re-qualification program. In addition, the licensee states that, if NRC approval for the request is received, operating procedures (ARPs, AOPs, and OIs) will be consolidated to reflect the removal of the MSLRMs and to streamline the instructions for manually securing the MVP upon a confirmed high radiation condition on the remaining offgas radiation monitors. Associated training for operators will be conducted.

As discussed in DAEC UFSAR Section 15.0.4, the plant licensing basis is that for the initial response to an accident (first 10 minutes), all actions shall be automatic and require no decision or manipulation of controls by plant operations personnel. Therefore, credit for manual actions may be taken after 10 minutes. The licensee's supplement dated September 1, 2006, provided a detailed best estimate time line of the sequence of events and actions beginning with the CRDA event and ending with the operator verifying the MVP has been tripped. The time line assumes that the MVP is tripped after 10 minutes. Securing the MVP involves turning the associated hand switch to the "STOP" position. To verify the MVP trip, the operator will check the indicator light located directly above the hand switch. Since the controls for securing the MVP are in the main control room, the operator does not need to enter the plant to perform the action, and no support personnel are needed. Based on the actions shown in the time line, and the location of the MVP controls, the NRC staff finds that the assumption of 10 minutes is reasonable. In addition, the NRC staff finds that the 10 minute time frame for manual action is consistent with the licensing basis as stated in UFSAR Section 15.0.4.

The licensee also stated that the existing RG 1.97 instrumentation for monitoring offsite releases and assessing core damage will remain adequate for directing emergency response to a CRDA event. Therefore, no new or upgraded instrumentation is needed to compensate for the removal of the MSLRMs.

Sensitivity and Timing of Operator Actions

The licensee's submittal dated September 1, 2006, stated that the best estimate time line for accomplishing the assumed operator actions in the dose analysis supporting the amendment request is not considered to constitute "time critical operator actions," because a sensitivity study performed to support the submittal demonstrated that the "no operator action" case (i.e., worst-case operator error) still achieves acceptable results in terms of dose rates. The NRC staff's evaluation of the radiological consequences associated with the proposed amendment is discussed in safety evaluation (SE) Section 3.2.

Conclusion for Operator Performance

The NRC staff finds that, consistent with the guidance in IN 97-78, the licensee has given adequate consideration to the alarms and other indications available to the operators to diagnose a CRDA, to the procedure changes and associated training needed for implementation of the change, to the operator actions related to tripping the MVP, and to sensitivity and timing of the operator actions. Based on these considerations, the NRC staff concludes that the proposed amendment is acceptable with respect to operator performance related to substituting the automatic tripping of the MVP with manual operator actions.

3.2 Radiological Consequences

The NRC staff performed independent confirmatory dose calculations, for the affected CRDA event, using the NRC-sponsored radiological consequence computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in NUREG/CR-6604. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The findings of this SE are based on the description of the licensee's analysis and other supporting information docketed by the licensee.

As discussed above in SE Section 2.0, the MSLRM isolation function (i.e., tripping of MVP and closing of MSL drains and reactor coolant sample valves) is currently credited in the CRDA analysis in order to reduce the radiological consequences of the accident. In support of the proposed amendment, the licensee has reanalyzed the radiological consequences of a CRDA without taking credit for the MSLRM isolation function. Specifically, the licensee's reanalysis assumes that the MSLRM will not automatically trip the MVP, this will be a manual action by the operators. In addition, the analysis assumes that the MSL drains and reactor coolant sample valves are not isolated following a CRDA. The impact of the proposed amendment on potential DBA radiological consequences is assessed and analyzed below.

Control Rod Drop Accident

CRDA is the only DBA event affected by the proposed TS change. The key assumptions of the submitted analysis are listed in Table 1. The licensee's re-analysis of the CRDA event is consistent with the current licensing basis, except for the following:

- release fractions, currently based on draft DG-1081, are based on the final RG 1.183, Appendix C;
- use of the 5 seconds release time, as suggested in the RG 1.183 Section 3.3, instead of the 2 hour release assumed in the current calculation; and
- performing calculations of the control room (CR) and technical support center (TSC) doses (current licensing basis includes an engineering judgement that the CRDA is not a limiting event for the DAEC).

These changes, made to the current licensing basis analysis, are consistent with the regulatory guidance stated in the RG 1.183, and are, therefore, acceptable.

The radionuclide core inventory used in this re-analysis was previously reviewed and approved by the NRC staff in License Amendment No. 240 (ADAMS Accession No. ML011660142).

The licensee performed a bounding analysis assuming an operator failure to isolate the MVP until 24 hours after the onset of the MSLRM signal, rather than isolating the system after the 10 minute time frame allowed for manual operator actions (as discussed in SE Section 3.1). The results, presented in Table 2, indicate that the calculated bounding doses are within the regulatory limits. However, the calculated bounding dose at the exclusion area boundary (EAB) of 7.4 rem of total effective dose equivalent (TEDE) exceeds the suggested limit of 6.3 rem.

The suggested EAB limit of 6.3 rem for fuel rod drop accidents, stated in the RG 1.183, was established in SRP 15.4.9 to represent the NRC staff's guidance to be "well within" (defined as 25 percent of) the 10 CFR 50.67 regulatory limit of 25 rem. Since the calculated bounding EAB dose is based on a conservative assumption (as reviewed and accepted by the NRC staff in License Amendment No. 240), and since there are additional conservatisms embedded in the licensee's analysis, the NRC staff considers such a deviation as insignificant from the point of view of the intent of the regulatory guidance, and, therefore, is acceptable.

Atmospheric Dispersion

As presented in the November 14, 2005, license amendment request, the licensee generated new atmospheric dispersion factors (χ/Q - read as "CHI over Q" values) for elevated releases from the plant stack to the CR and TSC. These new χ/Q values were generated partially based on the guidance in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." The licensee also used existing χ/Q values for postulated elevated releases to the EAB and low-population zone (LPZ) and ground level releases to the CR, TSC, EAB, and LPZ. The existing χ/Q values were previously generated by the licensee in support of DAEC License Amendment No. 240.

Meteorological Data

All of the χ/Q values used in the dose assessment for the current license amendment request were generated using the site meteorological data collected from 1997–1999 which the licensee previously provided in support of DAEC License Amendment No. 240. Based upon the discussion of atmospheric dispersion factors in the SE associated with Amendment No. 240, the NRC staff has concluded that the 1997–1999 site meteorological database provides an acceptable basis for making atmospheric dispersion estimates for use in support of this license amendment request.

Control Room Atmospheric Relative Concentrations

The licensee used previously approved χ/Q values for the CR and TSC dose assessments for postulated ground level releases from the main condenser. These χ/Q values are discussed in the SE associated with DAEC License Amendment No. 240 (see ML0116601420 dated July 31, 2001).

The licensee generated new CR χ/Q values for postulated releases from the plant stack to the CR and the TSC partially based on the guidance in RG 1.194 for elevated releases. RG 1.194 states that χ/Q values for postulated elevated releases to the CR should be based on χ/Q values generated from both the ARCON96 (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes" and PAVAN (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accident Releases of Radioactive Material from Nuclear Power Stations") computer codes. The resulting ARCON96 and PAVAN χ/Q values are then combined as a function of time period to determine the effective χ/Q values to be used in the dose assessments.

In generating the effective χ/Q values used in the dose assessments, the licensee provided new PAVAN χ/Q values and used previously generated ARCON96 χ/Q values approved in DAEC Amendment No. 240. The meteorological input for the new PAVAN χ/Q values consisted of 1995–1997 joint frequency distributions compiled from wind data measured at the 10-meter level on the onsite meteorological tower. The meteorological input to the previous ARCON96 χ/Q values consisted of 1995–1997 hourly wind speed and direction data from the 10-meter and 45.7-meter levels on the onsite meteorological tower. Stability class input to both the PAVAN and ARCON96 χ/Q values was based on delta-temperature measurements made between the 45.7-meter and 10-meter levels on the onsite meteorological tower.

The NRC staff qualitatively reviewed the inputs to the new PAVAN calculations and found them generally consistent with site configuration drawings and NRC staff practice with two exceptions. When calculating χ/Q values for an elevated release using PAVAN, RG 1.194 states that calculations should be made for several distances in each wind direction sector with the objective of identifying the maximum χ/Q value. The licensee input the actual distances and directions from the stack to the CR and TSC. This would result in χ/Q values less than the maximum χ/Q values due to the relative closeness of the DAEC stack to the CR intake and TSC. The second exception regards input of terrain heights. Use of terrain heights is not applicable to the CR χ/Q calculations and would result in an increase in χ/Q values in this scenario due to a reduction in effective release height.

The NRC staff accepts use of the licensee's 0-2 hour, 2-8 hour, and 8-24 hour χ/Q values presented in Exhibit A of their November 14, 2005 submittal because staff generated χ/Q values following the guidance in RG 1.194 and found the licensee's χ/Q values for these time intervals to be conservative. Although they were not used as part of this license amendment request, staff does not find the χ/Q values for the 1-4 day and 4-30 day χ/Q values presented in Exhibit A acceptable for use in other design-basis accident control room analyses. The elevated release control room χ/Q values should be regenerated by fully implementing the guidance in RG 1.194 or using another methodology found acceptable by the NRC staff before being used in any other design-basis control room dose analysis.

On the basis of the discussion presented above, the review described in the SE associated with Amendment No. 240, and a review of the licensee's use of these χ/Q values in this license amendment request, the NRC staff has concluded that the CR and TSC χ/Q values presented in Table 3 are acceptable for use in the DBA assessments described in this SE.

EAB and LPZ Atmospheric Dispersion Factors (χ/Q values)

The licensee used existing ground level and elevated release χ/Q values that were accepted by the NRC staff in Licensing Amendment No. 240 to evaluate the impact of the DAEC postulated releases to the EAB and LPZ. Based on the review described in the SE associated with DAEC License Amendment No. 240 and a review of the licensee's use of these χ/Q values in this license amendment request, the NRC staff has concluded that the EAB and LPZ χ/Q values presented in Table 4 are acceptable for use in the design basis accident assessments described in this SE.

Summary of Revised Analysis

The licensee re-evaluated the radiological consequences resulting from the postulated CRDA event, affected by the proposed TS changes, and concluded that the radiological consequences at the EAB, LPZ, CR and TSC are within the dose criteria provided in 10 CFR 50.67 and accident dose guidelines specified in SRP 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE.

Deviations from the current licensing basis were found acceptable to the NRC staff and are discussed in SE Section 3.2, subsection "*Control Rod Drop Accident*." The NRC staff performed independent calculations using the licensee's assumptions. These NRC staff analyses confirmed the licensee's calculated results. The NRC staff finds that the EAB, LPZ, CR and TSC doses, estimated by the licensee for the CRDA, presented in Table 2, meet the applicable dose criteria and are, therefore, acceptable.

Conclusion for Radiological Consequences

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of a postulated CRDA event affected by the proposed TS modification. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0, with the exceptions noted for RG 1.194 as discussed in SE Section 3.2, subsection *"Control Room Atmospheric Relative Concentrations."*

The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, CR and TSC doses will comply with these criteria, except for the RG 1.183 EAB dose criterion as discussed in Section 3.2, subsection *"Control Rod Drop Accident."*

The NRC staff finds reasonable assurance that, as modified by the proposed TS changes, DAEC will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression and in analysis assumptions and parameters. Therefore, the proposed license amendment is acceptable with respect to radiological consequences.

Table 1
Selected parameter values and assumptions for CRDA

Power level (102%)	1950 MWt
MVP rated flow rate	1800 cfm
Condenser volume	55,000 ft^3
Condenser leakage	1% volume per day
CR building volume	155,000 ft^3
TSC volume	68,300 ft^3
CR intake flow rate	3150 cfm
TSC unfiltered intake flow rate	900 cfm
No. of failed fuel rods	1200
Melted fuel percentage	0.77%

Table 2
Revised calculated CRDA doses (TEDE)
(rem)

Receptor	Base Case (10 min Delay)	Bounding Case (24 hrs Delay)	Regulatory Limit	Accident Dose Criteria (from RG 1.183)
EAB	2.85	7.4	25.0	6.3
LPZ	1.29	3.52	25.0	6.3
CR	0.48	0.88	5.0	5.0
TSC	0.54	0.69	N/A	5.0

Table 3
DAEC Control Room χ/Q Values (sec/m³)

	Control Room		Technical Support Center	
Time Period	Ground Level	Elevated	Ground Level	Elevated
0-2 hrs	1.48×10^{-3}	1.68×10^{-5}	2.14×10^{-3}	1.37×10^{-5}
2-8 hrs	1.27×10^{-3}	3.75×10^{-7}	1.86×10^{-3}	2.16×10^{-7}
8-24 hrs	5.56×10^{-4}	1.33×10^{-7}	8.44×10^{-4}	8.00×10^{-7}

Table 4
DAEC EAB and LPZ χ/Q Values (sec/m³)

Time Period	Receptor	Ground Level	Elevated
0-0.5 hrs	EAB	5.57×10^{-4}	7.03×10^{-5} (fumigation)
0.5-2 hrs	EAB	5.57×10^{-4}	6.95×10^{-6}
0-0.5 hrs	LPZ	1.34×10^{-4}	3.15×10^{-5} (fumigation)
0.5-2 hrs	LPZ	1.34×10^{-4}	6.69×10^{-6}
2-8 hrs	LPZ	6.43×10^{-5} *	3.58×10^{-6} *
8-24 hrs	LPZ	4.46×10^{-5}	2.61×10^{-6}

* The 0-8 hour LPZ χ/Q values are being conservatively applied to the 2-8 hour time period.

3.3 Deletion of MSLRMs

The NRC staff reviewed the proposed amendment regarding the impact of the deletion of the MSLRMs. Specifically, the NRC staff evaluated whether the proposed change complied with GDCs 13, 20, and 28, and 10 CFR 50.36.

GDC 13 requires, in part, that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operation occurrences, and for accident conditions as appropriate to assure adequate safety. The licensee's application stated that the primary signal used to assure adequate safety (with respect to a CRDA) is the reactor scram on high neutron flux initiated by the neutron monitoring system, which is not affected by the subject application. This statement is consistent with the DAEC accident analysis for a CRDA as discussed in UFSAR Section 15.2.4 which shows that, for the CRDA sequence of events, the average power range monitor 120 percent power signal scrams the reactor and the scram terminates the accident. UFSAR Section 15.2.4 also indicates that the MSLRMs trip and isolation function reduces the radiological consequences of a CRDA. However, based on the discussion above in SE Sections 3.1 and 3.2, replacing the automatic actions performed by the MSLRMs with manual operator actions is acceptable with respect to maintaining the radiological consequences within regulatory limits. The NRC staff finds that the MSLRMs are not required to assure adequate safety in mitigating a CRDA based on the scram function performed by the neutron monitoring system. Therefore, the NRC staff concludes that the proposed deletion of the MSLRMs does not affect DAEC's conformance with the intent of GDC 13.

GDC 20 requires, in part, that the protection system be designed to sense accident conditions and to initiate the operation of systems and components important to safety. The licensee's application stated that the MSLRMs and their associated trip function are not needed to mitigate the consequences of an accident, and thus, are no longer systems and components important to safety. The licensee also reiterated that the neutron monitoring system trip on high neutron flux is the primary safety action necessary to mitigate a CRDA. The NRC staff finds that, for a CRDA, the neutron monitoring system is sufficient to meet the intent of GDC 20. Therefore, the NRC staff concludes that the proposed deletion of the MSLRMs does not affect DAEC's conformance with the intent of GDC 20.

GDC 28 requires, in part, that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. As discussed in UFSAR Section 15.2.4, the initiator for a CRDA is when a control blade becomes decoupled from its control rod drive mechanism, and sticks inside the core at the fully-inserted position, such that it does not follow its drive when withdrawn from the core. Later in the startup sequence, when the stuck control rod is at its maximum possible control rod worth, it breaks free and drops at its maximum velocity, causing a prompt, super critical reactivity event. The licensee's application stated that design features will be retained that will limit the consequences of a CRDA. Specifically, the mechanical design of the control blades for General Electric BWRs includes a velocity limiter. This device is located on the bottom of the control blade and is designed to provide flow resistance such that, should the blade come uncoupled from its drive mechanism, the blade falls slowly out of the core (i.e., limits rate of reactivity addition). The licensee's application also stated that the rod worth minimizer and its associated banked position withdrawal sequence rod pattern controls, would be retained in the TSs to limit individual control rod worths below those used in the CRDA analysis. In consideration of the design of the control blades, and the rod worth minimizer TSs, the NRC staff concludes that the intent of GDC 28 is satisfied with respect to the CRDA without consideration of the trip signals received from the MSLRMs.

As discussed in SE Section 2.0, 10 CFR 50.36(c)(2)(ii), which requires that a TS LCO be established for each item meeting one or more of the following criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

With respect to Criterion 1, the MSLRMs are not used to detect leakage in the reactor coolant pressure boundary. As such, the MSLRMs do not satisfy Criterion 1.

With respect to Criterion 2, the MSLRM trip function is not an initial condition assumed in the CRDA analysis. As such the MSLRMs do not satisfy Criterion 2.

With respect to Criterion 3, as discussed in SE Section 3.2, the MSLRM trip is not necessary to ensure that the radiological consequences of a CRDA remain within regulatory limits. As such, the MSLRMs do not satisfy Criterion 3.

With respect to Criterion 4, the licensee's application provided a qualitative risk assessment. The licensee stated that various existing TS provisions that preclude this event are being maintained to ensure that the probability of a CRDA is maintained at a very low value (e.g., control rod coupling checks in TS 3.1.3.5, stuck rod provision in TS 3.1.3, "slow" control rod separation criteria in TS 3.1.4). The licensee also stated that the MSLRM trip does not prevent a CRDA; its sole function is in response to the event after it occurs. As such, the elimination of the MSLRMs will not increase the risk of core damage from a CRDA event. In addition, since the elimination of the MSLRM trip function would not result in unacceptable dose consequences, the large early release frequency would not be increased. Thus, the licensee concluded that the overall risk of a CRDA event would remain extremely low after removal of the MSLRM trip function. The NRC staff concurs with the licensee's assessment that the MSLRM trip function is not significant to public health and safety. As such, the MSLRMs do not satisfy Criterion 4.

The NRC staff concludes that since none of the four criteria in 10 CFR 50.36(c)(2)(ii) are satisfied, a TS LCO for the MSLRM trip function is not required.

3.4 Technical Evaluation Conclusion

Based on the considerations discussed in SE Sections 3.1 through 3.3, the NRC staff finds the proposed changes to TS Table 3.3.6.1-1 to be acceptable.

The licensee stated that the TS Bases would also be changed (under the TS Bases Control Program specified in TS 5.5.10) to reflect the changes to Table 3.3.6.1-1. The NRC staff considered the proposed TS Bases changes as information only. The NRC staff did not review or make a finding with respect to these changes.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATIONS

The 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. The NRC staff has determined that the amendment involves no significant increase in 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 (71 FR 43533). 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

The Commission has concluded, based on the considerations discussed in this document, 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.

7.0 REFERENCES

1. Van Middlesworth, G., Duane Arnold Energy Center, "Technical Specifications Change Request: 'Elimination of Main Steam Line Radiation Monitor Trips," letter to U.S. Nuclear Regulatory Commission, November 14, 2005. ADAMS Accession No. ML053260438.

2. Van Middlesworth, G., Duane Arnold Energy Center, "Disposition of Pending NRC Actions," letter to U.S. Nuclear Regulatory Commission, February 6, 2006. ADAMS Accession No. ML060530621.

3. Van Middlesworth, G., Duane Arnold Energy Center, "Response to Request for Additional Information Related to the Proposed Amendment Requesting Elimination of Main Steamline Radiation Monitor Trips (TAC NO. MC8883)," letter to U.S. Nuclear Regulatory Commission, September 1, 2006. ADAMS Accession No. ML062890143.

4. General Electric Company, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," NEDO-31400-A, October 1992.

5. U.S. Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 182 to Facility Operating License No. DPR-049, Iowa Electric Light and Power Company, Central Iowa Power Cooperative, Corn Belt Power Cooperative, Duane Arnold Energy Center, Docket No. 50-331," March 24, 1992.

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