UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 6, 2008

Mr. Richard L. Anderson
Vice President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT REGARDING CONTROL ROD NOTCH TESTING FREQUENCY AND CLARIFICATION OF A FREQUENCY EXAMPLE (TAC NO. MD7541)

Dear Mr. Anderson:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 271 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated December 20, 2007.

The amendment revises (1) the control rod notch surveillance frequency in Section 3.1.3, "Control Rod OPERABILITY," and (2) one example in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. These changes were done pursuant to the previously approved Technical Specification Task Force (TSTF) change traveler TSTF-475, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action," Revision 1.

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,

Peter S. Tam

Peter S. Tam, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures:
1. Amendment No. 271 to
   License No. DPR-49
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FPL ENERGY DUANE ARNOLD, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 271
License No. DPR-49

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

   A. The application for amendment by FPL Energy Duane Arnold, LLC dated December 20, 2007, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 271, 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

Lois M. James, 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 6, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 271

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following page of Renewed Facility Operating License DPR-49 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
Page 3	Page 3

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

REMOVE	INSERT
1.4-5	1.4-5
3.1-8	3.1-8
3.1-10	3.1-10
3.1-11	3.1-11
3.1-14	3.1-14

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:

Maximum Power Level

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) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 271, are hereby incorporated in the license. FPL Energy Duane Arnold, LLC shall operate the facility in accordance with the Technical Specifications.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Not required to be performed until 12 hours after ≥ 25% RTP. -----	
Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	B.3 Perform SR 3.1.3.2 for each withdrawn OPERABLE control rod. AND	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the Low Power Setpoint (LPSP) of the RWM.
	B.4 Perform SR 3.1.1.1	72 hours
E. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
F. One or more control rods inoperable for reasons other than Condition A or B.	C.1 -----------NOTE------------ RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ------------------------------ Fully insert inoperable control rod. AND	3 hours

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	-------------------------NOTE------------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than 20% RTP. -- Insert each withdrawn control rod at least one notch.	 31 days
SR 3.1.3.3	Verify each control rod scram time from fully withdrawn to notch position 04 is ≤ 7 seconds.	In accordance with SR 3.1.4.1 and SR 3.1.4.2
SR 3.1.3.4	Verify each withdrawn control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position <u>AND</u> Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

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Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

---NOTES---

3. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

4. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 04. These control rods are inoperable, in accordance with SR 3.1.3.3, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[a] (seconds) when REACTOR STEAM DOME PRESSURE ≥ 800 psig
46	0.44
38	0.93
26	1.83
06	3.35

(b) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 271 TO FACILITY OPERATING LICENSE NO. DPR-49

FPL ENERGY DUANE ARNOLD, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated December 20, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080020004), FPL Energy Duane Arnold, LLC (the licensee) submitted a license amendment request regarding the Duane Arnold Energy Center Operating License. The proposed amendment adopts Technical Specification Task Force (TSTF) change traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action." The proposed amendment would: (1) revise the Technical Specifications (TS) control rod notch surveillance frequency in TS 3.1.3, "Control Rod OPERABILITY," and (2) revise one example in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension (NUREG-1430 through NUREG-1434). The Licensee's letter states, "it should be noted that FPL Energy Duane Arnold does not use the same language as the Improved Standard TS (NUREG-1433) in the Note prior to [Surveillance Requirement] SR 3.1.3.2." Per the letter, the Duane Arnold Energy Center (DAEC) TS specifies that the frequency of SR 3.1.3.2, notch testing of fully withdrawn control rod, is performed "31 days after the control rod is withdrawn and THERMAL POWER is greater than 20% [Rated Thermal Power] RTP [reactor thermal power]" instead of "31 days after the control rod is withdrawn and THERMAL POWER is greater than the [Low Power Setpoint] LPSP of the [Rod Worth Minimizer] RWM." The letter further states, "this deviation from NUREG-1433 was incorporated into the DAEC TS by Amendment 223 dated May 22, 1998 (Accession No. ML021920121) in the conversion of the DAEC TS to the Improved Standard TS."

These changes are based on the NRC-approved TSTF change traveler TSTF-475, Revision 1, that revised the reference Standard Technical Specifications (STS) by: (1) revising the frequency of SR 3.1.3.2, notch testing of each fully withdrawn control rod, from 7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM" (NUREG-1433 and NUREG-1434); and (2) revising Example 1.4-3 in Section 1.4 "Frequency" to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in NOTES in the "SURVEILLANCE" column in addition to the time periods in the "FREQUENCY" column (NUREG-1430 through NUREG-1434).

The purpose of the surveillances is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. Control rods and the control rod drive (CRD) mechanism (CRDM) by which the control rods are moved are components of the CRD system (CRDS), which is the primary reactivity control system for the reactor. By design, the CRDM is highly reliable with a tapered design of the index tube which is conducive to control rod insertion.

A stuck control rod is an extremely rare event and industry review of plant operating experience did not identify any incidents of stuck control rods while performing a rod notch surveillance test.

The purpose of these revisions is to reduce the number of control rod manipulations and, thereby, reduce the opportunity for reactivity control events.

The purpose of the change to Example 1.4-3 in Section 1.4 "Frequency" is to clarify the applicability of the 25% allowance of SR 3.0.2 to time periods discussed in NOTES in the "SURVEILLANCE" column as well as to time periods in the FREQUENCY" column.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix A, General Design Criterion (GDC) 29, Protection against anticipated occurrence, requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences. The design relies on the CRDS to function in conjunction with the protection systems under anticipated operational occurrences, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRDS provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during anticipated operational occurrences. Meeting the requirements of GDC 29 for the CRDS prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during anticipated operational occurrences. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

3.0 TECHNICAL EVALUATION

The NRC staff previously reviewed the following information provided by the TSTF to support the staff's review and approval of TSTF-475, Revision 1. Specifically, the following documents were reviewed during the NRC staff's evaluation:

- TSTF letter TSTF-04-07 (Reference 1) - Provided a description of the proposed changes in TSTF-475 that changes the weekly rod notch frequency to monthly and clarify the applicability of the 25% allowance in Example 1.4-3.

- TSTF letter TSTF-06-13 (Reference 4) - Provided responses to the NRC staff's request for additional information (RAI) on (1) industry experience with identifying stuck rods, (2) tests that would identify stuck rods, (3) continue compliance with Service Information Letter (SIL) 139, (4) industry experience on collet failures, and (4) applying the 25% grace period to the 31 day control rod notch SR test frequency.

- BWROG letter BWROG-06036 (Reference 5) – Provided the GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," in which CRD notching frequency and CRD performance were evaluated.

- TSTF letter TSTF-07-19 (Reference 6) - Provided response to the NRC staff's RAI on CRD performance in Control Cell Core (CCC) designed plants, including TSTF-475, Revision 1.

The CRD System at DAEC is the primary reactivity control system for the reactor. The CRD System, in conjunction with the Reactor Protection System, provides the means for the reliable control of reactivity changes to ensure under all conditions of normal operation, including anticipated operational occurrences that specified acceptable fuel design lirnits are not exceeded. Control rods are components of the CRD System that have the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System.

The CRD System consists of a CRDM, by which the control rods are moved, and a hydraulic control unit (HCU) for each control rod. The CRDM is a mechanical hydraulic latching cylinder that positions the control blades. The CRDM is a highly reliable mechanism for inserting a control rod to the full-in position. The collet piston mechanism design feature ensures that the control rod will not be inadvertently withdrawn. This is accomplished by engaging the collet fingers, mounted on the collet piston, in notches located on the index tube. Due to the tapered design of the index tube notches, the collet piston mechanism will not impede rod insertion under normal insertion or scram conditions.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD which houses the collet mechanism which consists of the locking collet, collet piston, collet return spring and an unlocking cam. The collet mechanism provides the locking/unlocking mechanism that allows the insert/withdraw movement of the control rod. The CRT has three primary functions: (a) to carry the hydraulic unlocking pressure to the collet piston, (b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings, and (c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

The NRC staff approved TSTF-475 which revised the TS SR 3.1.3.2, "Control Rod OPERABILITY" in the STS (NUREG-1433 and NUREG-1434) from seven days to monthly based on the following: (1) slow crack growth rate of the CRT; (2) the improved CRT design; (3) a higher reliable method (scram time testing) to monitor CRD scram system functionality; (4) GE chemistry recommendations; and (5) no known CRD failures were detected during the notch testing exercise. The NRC staff, therefore, concluded that the changes would reduce the number of control rod manipulations thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the extremely high reliability of the CRD system. The following paragraphs describe the bases for the staff's approval of TSTF-475:

According to the Boiling Water Reactor Owners Group (BWROG), at the time of the first CRT crack discovery in 1975, each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Control rod insertion capability was demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal.

It was determined that during scrams, the CRT temperature distribution changes substantially at reactor operating conditions. Relatively cold water moves upward through the inside of the CRT and exits via the flow holes into the annulus on the outside. At the same time, hot water from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through-wall and circumferential temperature gradients during scrams which contribute to the observed CRT cracking.

Subsequently, many boiling-water reactors (BWRs) have reduced the frequency of notch testing for partially withdrawn control rods from weekly to monthly. The notch test frequency for fully withdrawn control rods are still performed weekly. The change for partially withdrawn control rods was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested on a weekly basis. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods.

In response to NRC's RAIs and to support their position to reduce the CRD notch testing frequency, the BWROG provided plant data and a GE Nuclear Energy report entitled, "CRD Notching Surveillance Testing for Limerick Generating Station". The GE report provided a description of the cracks noted on the original design CRT surfaces. These cracks, which were later determined to be intergranular, were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness. Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360 degree severance of the CRT that would render the CRD inoperable which would prevent insertion, withdrawal or scram. Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the Technical Specifications. To a lesser degree, cracks have also been noted at the welded joint of the interim design CRT but no cracks have been observed in the final improved CRT design. Neither the BWROG nor the NRC staff was able to find evidence of a collet housing failure since 1975. To date, operating experience data shows no reports of a severed CRT at any BWR. No collet housing failures have been noted since 1975. On a numerical basis for instance, based on the BWROG assumption that there are 137 control rods for a typical BWR/4 and 193 control rods for a typical BWR/6, the yearly performance would be 6590 rod notch tests for a BWR/4 plant and 9284 for a BWR/6 plant. For example, if all BWRs operating in the United States are taken into consideration, the yearly performances of rod notch data would translate into approximately 240,000 rod notch tests without detecting a failure.

In addition, the intergranular stress corrosion cracking (IGSCC) crack growth rates were evaluated, at Limerick Generating Station, using GE's PLEDGE model with the assumption that the water chemistry condition is based on GE recommendations. The model is based on fundamental principles of stress corrosion cracking which can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. It was determined that the additional time of 24 days represented an additional 10 mils of growth in total crack length. The small difference in growth rate would have little effect on the behavior between one notch test and the next subsequent test. Therefore, from the materials perspective based on low crack growth rates, a decrease in the notch test frequency would not affect the reliability of detecting a CRDM failure due to crack growth.

Also, the BWR scram system has extremely high reliability. In addition to notch testing, scram time testing can identify failure of individual CRD operation resulting from IGSCC-initiated cracks and mechanical binding. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator, as well as operation of the control rods. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod.

Also, the HCUs, CRD drives, and control rods are tested during refueling outages, approximately every 18-24 months. Based on the data collected during the preceding cycle of operation, selected control rod drives are inspected and, as required, their internal components are replaced. Therefore, increasing the CRD notch testing frequency to monthly would have a very minimal impact on the reliability of the scram system.

FPL Energy stated in its application that it has reviewed the basis for the NRC staff's acceptance of TSTF-475, Revision 1, and concluded that the basis is applicable to DAEC, and supports its adoption of the TSTF-475 changes into the DAEC TS. The NRC staff also reviewed the TSTF-475, Revision 1 basis, and similarly concluded that the basis for the TSTF is applicable to DAEC, and therefore, the TSTF is appropriate for adoption by the licensee. In addition, the NRC staff reviewed the licensee's proposed changes against the corresponding changes made to the Standard Technical specifications by TSTF-475, Revision 1, which the NRC staff has found to satisfy applicable regulatory requirements, as described above. The proposed changes would: (1) revise the Technical Specifications (TS) control rod notch surveillance frequency in TS 3.1.3, "Control Rod OPERABILITY," and (2) revise one Example in Section 1.4 "Frequency" to clarify the applicability of the 1.25 surveillance test interval extension. The NRC staff found that the proposed changes are consistent with the changes approved by the NRC staff in TSTF-475, Revision 1. The NRC staff, therefore, finds these changes acceptable.

The NRC staff has reviewed the licensee's proposal to amend existing DAEC TS sections SR 3.1.3.2, "Control Rod OPERABILITY," and Example 1.4-3, "Frequency" applicable to SR 3.0.2. The NRC staff has concluded that the TS revisions will have a minimal effect on the high reliability of the CRD system while reducing the opportunity for potential reactivity events; thus, the proposed amendment meets the requirement of 10 CFR Part 50, Appendix A, GDC 29, and will clarify the applicability of the 1.25 provision in SR 3.0.2. Therefore, the NRC staff concludes that the proposed amendment is acceptable.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to use of facility components 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 the amounts, and no significant change in the types, of any effluent 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 (73 FR 10298). 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.

7.0 CONCLUSION

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

7.0 REFERENCES

1. Letter TSTF-04-07 from the Technical Specifications Task Force to the NRC, TSTF-475 Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," August 30, 2004, ADAMS accession number ML042520035.

2. NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4, Revision 3," August 31, 2003

3. Letter TSTF-07-19, Response from the Technical Specifications Task Force to the NRC, "Request for Additional Information (RAI) Regarding TSTF-475 Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," dated February 28, 2007, (TSTF-475 Revision 1 is an enclosure), ADAMS accession number ML071420428

4. Letter TSTF-06-13 from the Technical Specifications Task Force to the NRC, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0," dated July 3, 2006, ADAMS accession number ML061840342

5. Letter BWROG-06036 from the BWR Owners Group to the NRC, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0," dated November 16, 2006, with Enclosure of the GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," dated November 2006, ADAMS accession number ML063250258

Principal Contributor: R. Grover, NRR

Date: November 6, 2008

November 6, 2008

Mr. Richard L. Anderson
Vice President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT REGARDING CONTROL ROD NOTCH TESTING FREQUENCY AND CLARIFICATION OF A FREQUENCY EXAMPLE (TAC NO. MD7541)

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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/

Peter S. Tam, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures:
1. Amendment No. 271 to License No. DPR-49
2. Safety Evaluation

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