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

April 16, 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 REGARDING SECONDARY CONTAINMENT OPERABILITY DURING MOVEMENT OF IRRADIATED FUEL AND CORE ALTERATIONS (TAC NO. MB1569)

Dear Mr. Van Middlesworth:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 237 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 a portion of your application dated October 19, 2000, as supplemented November 16, 2000, and April 9, 2001, and as limited in scope by letter dated March 23, 2001.

The amendment revises the TS regarding operability requirements during core alterations and while moving irradiated fuel assemblies within the secondary containment.

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

Sincerely,

Darl S. Hood

Darl S. Hood, Senior 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. 237 to License No. DPR-49
2. Safety Evaluation

cc w/encls: See next page

NRR-058

cc

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

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
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), October 19, 2000, as supplemented November 16, 2000, and April 9, 2001, and as limited in scope by letter dated March 23, 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 5l 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. 237 , are hereby incorporated in the license. The licensee 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of Issuance: April 16, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 237

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
3.3-65	3.3-65
3.6-35	3.6-35
3.6-36	3.6-36
3.6-37	3.6-37
3.6-39	3.6-39
3.6-41	3.6-41
3.6-42	3.6-42
3.6-43	3.6-43
3.6-44	--- (deleted)

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 165.6 inches
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.2 psig
3. Reactor Building Exhaust Shaft – High Radiation	1,2,3, (a)	1	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 12.8 mR/hr
4. Refueling Floor Exhaust Duct – High Radiation	1,2,3, (a)	1	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 10.6 mR/hr

(a) During operations with a potential for draining the reactor vessel.

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During Operations with a Potential for Draining the Reactor Vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.2 Be in MODE 4.	36 hours
C. Secondary containment inoperable during OPDRVs.	C.1 --------NOTE-------- LCO 3.0.3 is not applicable. ------------------- Initiate action to suspend OPDRVs.	Immediately

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed.	31 days
SR 3.6.4.1.2	-----------------NOTE------------------ Doors in high radiation areas may be verified by administrative means. ------------------------------------- Verify that either the outer door(s) or the inner door(s) in each secondary containment access opening are closed.	31 days
SR 3.6.4.1.3	Verify each SBGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment at a flow rate ≤ 4000 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves/Dampers (SCIV/Ds)

LCO 3.6.4.2 Each SCIV/D shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During Operations with a Potential for Draining the Reactor Vessel (OPDRVs).

ACTIONS

-----------------------------------NOTES-----------------------------------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIV/Ds.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV/D inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve/damper, closed manual valve, or blind flange. AND	8 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during OPDRVs.	D.1 ---------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SBGT) System

LCO 3.6.4.3 Two SBGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During Operations with a Potential for Draining the Reactor Vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SBGT subsystem inoperable.	A.1 Restore SBGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- C.1 Place OPERABLE SBGT subsystem in operation. OR	 Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Initiate action to suspend OPDRVs.	Immediately
D. Two SBGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately
E. Two SBGT subsystems inoperable during OPDRVs.	E.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Initiate action to suspend OPDRVs.	Immediately

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SBGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	-----------------NOTE-------------------- When a SBGT subsystem is placed in an inoperable status solely for the performance of VFTP testing required by this Surveillance <u>on the other subsystem</u>, entry into associated Conditions and Required Actions may be delayed for up to 1 hour. -- Perform required SBGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	 In accordance with the VFTP
SR 3.6.4.3.3	Verify each SBGT subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.6.4.3.4	Verify each SBGT filter cooler bypass damper can be opened and the fan started.	24 months

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 237 TO FACILITY OPERATING LICENSE NO. DPR-49

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated October 19, 2000, Nuclear Management Company, LLC (NMC and the licensee) requested an amendment to the operating license for the Duane Arnold Energy Center (DAEC). The amendment would address replacing the current accident source term used in design basis radiological analyses with an alternative source term (AST) pursuant to 10 CFR 50.67, "Accident Source Term." The alternate source term is applied as discussed in NMC's application for amendment, by letter dated November 16, 2000, to increase the maximum power level authorized by Section 2.C(1) of the operating license from 1658 thermal megawatts (MWt) to 1912 MWt. In the October 19, 2000, letter, NMC requested several changes to the DAEC licensing basis and various technical specification (TS) changes. In a subsequent letter dated March 23, 2001, providing additional supporting information, NMC requested that the Nuclear Regulatory Commission (NRC) review and address a portion of the application separately to facilitate an earlier approval consistent with an upcoming refueling outage. By letter dated April 9, 2001, NMC forwarded the typed TS replacement pages reflecting the TS changes proposed in the March 23, 2001, letter.

Accordingly, as requested in NMC's letter dated March 23, 2001, this safety evaluation addresses only the following portions of the original October 19, 2000, amendment request:

1. The AST implementation as limited to the design basis fuel handling accident (FHA) radiological consequence analysis performed to show compliance with 10 CFR 50.67(b)(2).

2. Revised atmospheric dispersion factors for radiological releases related to release points and human receptors associated with an FHA.

3. Revised TS requirements for operability of secondary containment. TS for secondary containment isolation instrumentation (i.e., TS 3.3.6.2), secondary containment (TS 3.6.4.1), secondary containment isolation valves/dampers (TS 3.6.4.2), and the standby gas treatment system (TS 3.6.4.3), establish operability requirements during certain operating modes and activities. NMC proposes to relax these requirements by eliminating their applicability during core alterations and movement of irradiated fuel assemblies.

The NRC will address the remaining changes requested in NMC's letter of October 19, 2000, as supplemented November 16, 2000, by a separate amendment.

In support of this request, NMC committed to revise DAEC guidelines for assessing systems removed from service during the handling of irradiated fuel assemblies or core alterations to implement provisions of Section 11.3.6.5 of NUMARC 93-01, Revision 3. These provisions address the capability of restoring the secondary containment.

NMC's letters dated March 23 and April 9, 2001, are within the scope of the changes proposed in NMC's letter of October 19, 2000, that was noticed in the *Federal Register* on March 6, 2001 (66 FR 13598).

2.0 EVALUATION

2.1 Alternative Source Term

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, which provides a mechanism for licensed power reactors to replace the traditional accident source term used in the design-basis accident analyses with alternative source terms (ASTs). Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Section 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected design-basis accidents. NMC's application addresses these requirements in proposing selectively to use the AST described in RG 1.183 in evaluating the radiological consequences of a FHA. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67 (b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11, "Reactor Site Criteria--Determination of Exclusion Area, Low Population Zone, and Population Center Distance," and General Design Criterion (GDC)-19 of 10 CFR Part 50, Appendix A, which (based upon NMC's selective application) is limited to the FHA only.

The NRC staff reviewed NMC's implementation of the AST to the FHA and finds that it meets the requirements of 10 CFR 50.67 and the guidance provided in RG 1.183. Therefore, the NRC staff finds NMC's selective AST implementation acceptable.

2.2 Radiological Consequences of a Design-basis Fuel Handling Accident

The objective of a design-basis accident (DBA) radiological analysis is to evaluate the performance of various plant safety systems intended to mitigate the postulated release of radioactive materials from the plant to the environment. Specifically, the DBA FHA is evaluated to demonstrate compliance with GDC-61, "Fuel Storage and Handling and Radioactivity Control." The NRC staff reviewed NMC's analysis description and performed confirmatory calculations. The analysis assumptions used by the NRC staff are consistent with those used by NMC and are tabulated in Table 1 attached to this safety evaluation. The radiation doses reported by the licensee are tabulated in Table 2 attached to this safety evaluation.

This accident analysis postulates that a spent fuel assembly is dropped during refueling. The kinetic energy developed during this drop is conservatively assumed to be dissipated upon impact, resulting in damage (breaching) to the cladding on 151 fuel rods.[1] The fission product inventory in the core is largely contained within the fuel pellets that are enclosed in the fuel rod cladding. However, the volatile constituents of this inventory will migrate from the pellets to the gap between the pellets and the fuel rod cladding. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. This activity is assumed to be released from the damaged fuel and the spent fuel pool to the secondary containment building (aka, the reactor building), from whence it is assumed to be released to the environment. NMC assumes that the activity is released to the environment from the reactor building vent as a ground level release. Although radiation monitors in the exhaust ducts from the refueling floor would automatically isolate the reactor building vent and actuate the standby gas treatment system (SGTS), these features may not be operable due to the TS changes requested in the application for amendment. If the reactor building vent fans were to be inoperable, there would be a lesser force driving the release to the environment.

In modeling this accident, NMC assumed that the release from the pool is mixed in an arbitrary 2400 ft^3 volume of the secondary containment above the refueling floor and is released to the environment at a rate equal to 2.4×10^7 percent per day. NMC modeled the release from the fuel as occurring at a linear rate over two hours. Regulatory guidance provides for an instantaneous release from the fuel, followed by a release to the environment over two hours. Thus, NMC's model is functionally equivalent in that the overall release and the overall release rate are unchanged and, therefore, acceptable.

NMC assumed no credit for filtration by the SGTS. Fission products released from the damaged fuel rods are decontaminated as they pass through the pool water, depending upon their physical and chemical forms. NMC assumed no decontamination for noble gases, a decontamination factor of 200 for radioiodines, and full retention of all aerosol and particulate fission products by the pool water.

NMC used atmospheric dispersion (χ/Q) values that are different from those in the current licensing basis for DAEC. These are addressed in Section 2.3 of this safety evaluation.

As shown in Table 2 attached to this safety evaluation, the results of NMC's analyses indicate that the dose at the exclusion area boundary would be no more than 0.94 rem total effective dose equivalent (TEDE)[2] and the dose at the low-population zone would be no more than 0.23

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[1] The breaching of 151 fuel rods is assumed to result from dropping the fuel assembly 30 feet over the top of the reactor core. This is more limiting (produces more damaged fuel) than the shorter drops that could occur over the spent fuel pool. Assuming 151 damaged fuel rods is also more conservative than the existing design and licensing basis value of 125 rods.

[2] As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11, "Reactor Site Criteria--Determination of Exclusion Area, Low Population Zone, and Population Center Distance," and General Design Criterion (GDC)-19 of 10 CFR Part 50, Appendix A, which (based upon NMC's selective application) is limited to the FHA only.

rem TEDE. These results are less than the TEDE criterion of 6.3 rem set forth in RG 1.183 (Table 6) and are a small fraction of the dose criteria in 10 CFR 50.67(b)(2)(i) and (ii). Therefore, these results are acceptable.

NMC evaluated the dose to operators in the control room and personnel in the technical support center (TSC). NMC assumed that, once the event started, the operators would manually actuate control room isolation within ten minutes. This ten-minute period is consistent with that assumed for other manual operator actions addressed in the current licensing basis. The control room isolation can be actuated within the control room with a small number of operator actions. During refueling operations, continuous communications are maintained between the control room personnel and the refueling crew. During a FHA, there are only limited actions required of the control room operators to contend with this event. As such, the NRC staff finds the assumed ten-minute delay in isolation to be acceptable. The TSC is also isolated manually, however, the delay time was assumed to be 30 minutes in this case.

NMC analyzed the control room and TSC doses over a thirty-day period. Although the TSC and control room are both designed to be pressurized during an accident, NMC assumes that unfiltered inleakage occurs. Since this inleakage has not been quantified, NMC analyzed two cases, one with 1000 cfm unfiltered inleakage and one with no unfiltered inleakage. The zero inleakage case was considered because, for the loss-of-coolant and the main steamline break accidents, the limiting case was at 0 cfm. NMC based the maximum inleakage value on 100 percent of the emergency mode ventilation flow rate from the standby filter units. NMC asserts that control building positive pressure surveillance tests performance in accordance with TS surveillance requirement (SR) 3.7.4.4 would be unsatisfactory if such a large leakage pathway existed. NMC states that no other inflow or pressurization source exists for the control building that could mask the performance of the standby filter units during SR 3.7.4.4. The NRC staff accepts the licensee's position that 1000 cfm unfiltered inleakage is limiting for control room and TSC doses given (1) the margin between the projected dose and the dose criterion, and (2) the relative change in dose between the 0 cfm and 1000 cfm cases.[3]

As shown in Table 2 of this safety evaluation, the results of NMC's analyses indicate that the control room operators would receive no more than 3.16 rem TEDE and TSC personnel would receive no more than 2.83 rem TEDE. These doses are less than the TEDE limit ot 5 rem contained in 10 CFR 50.67(b)(2)(iii) and GDC-19, "Control Room." They are, therefore, acceptable.

On the basis of its review of the NMC analysis as described above, and as confirmed by its own independent analysis, the NRC staff finds the NMC analyses and the reported results to be acceptable.

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[3] The NRC staff notes, however, that it is currently developing regulatory guidance regarding control room habitability, including surveillance testing of unfiltered inleakage. In addition, the Nuclear Energy Institute is developing an industry initiative document (NEI-99-03) on control room habitability. The NRC staff's acceptance of NMC's unfiltered inleakage assumption does not foreclose any future generic regulatory actions that may become applicable to DAEC.

2.3 Atmospheric Dispersion (χ/Q) Changes

NMC assumed revised χ/Q values in performing the FHA analysis. NMC states that the historical χ/Q data did not meet their expectations for level of documentation with regard to the DAEC design-basis and for this reason, new values for various combinations of release points and receptors were generated using NMC's variant of the PAVAN code[4] (PAVAN-PC) and the ARCON96 code.[5] These were tabulated in NMC's application for amendment dated October 19, 2000 (see Section 3.1.3 of Attachment 4 thereto, which includes several tables). This safety evaluation will address only those data applicable to the FHA and tabulated in Tables 1 and 8 of the NMC application for amendment. The hourly observation meteorology data used in these analyses were obtained from DAEC's meteorological program over the period of January 1, 1997 to December 31, 1999. The DAEC program, which is described in Section 2.3.3 of the updated final safety analysis report (UFSAR), meets the guidance of RG 1.23 and is subject to DAEC's quality assurance program in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." Over 90 percent of the available meteorology observations were collected (i.e., data recoverability exceeded 90 percent) during this period. The NRC staff compared the wind speed and wind direction distributions with the historical distributions documented in the DAEC UFSAR and found acceptable consistency.

The PAVAN-PC and ARCON96 codes are acceptable methodologies. In response to NRC staff requests, NMC provided copies of the input data printouts from PAVAN-PC and ARCON96 and provided the meteorological data on magnetic media. The NRC staff qualitatively reviewed the inputs to the codes and found them to be consistent with UFSAR site configuration drawings and NRC staff practice. The NRC staff also performed a confirmatory calculation of the reactor building ground level χ/Q using the PAVAN code (of which PAVAN-PC is a variant), and found acceptable correlation. Based on this review, the NRC staff finds the revised χ/Q values acceptable.

2.4 Proposed Technical Specification Changes

NMC requested several technical specification changes that will relax requirements for the secondary containment and supporting systems to be operable during core alterations or movement of irradiated fuel. The following technical specifications are affected by the proposed changes:

TS 3.3.6.2, "Secondary Containment Isolation Instrumentation"
TS 3.6.4.1, "Secondary Containment"
TS 3.6.4.2, "Secondary Containment Isolation Valves/Dampers"
TS 3.6.4.3, "Standby Gas Treatment (SBGT) System"

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[4] "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG-2858, November 1982. RSICC Computer Code Collection No. CCC-445.

[5] "Atmospheric Relative Concentrations in Building Wakes," NUREG-6331, Revision 1, May 1997. RSICC Computer Code Collection No. CCC-664.

The FHA radiological consequence analysis described above did not assume operability of these four systems or structures. Thus, the inoperability of these systems/structures during a FHA does not increase the potential radiation doses estimated by this revised analysis. Although the projected radiological exposures (doses) at the exclusion area boundary (EAB) and the low population zone (LPZ) are projected to be greater than those currently documented in the DAEC licensing basis, the doses are still a small fraction of the dose criteria in 10 CFR 50.67(b)(2)(i) and (ii) and are, therefore, acceptable. The projected dose to a control room operator or to TSC personnel was not previously analyzed. The present analysis projects doses for the control room operators and TSC personnel that meet the dose criteria in 10 CFR 50.67(b)(2)(iii) and GDC-19.

The NRC staff has traditionally and conservatively required the secondary containment systems to be operable during core alterations and movement of irradiated fuel within the secondary containment as a defense-in-depth measure to mitigate the consequences of the postulated FHA. Eliminating these operability requirements provides operational flexibility during refueling periods. Since the postulated doses, without these systems being operable, remain within regulatory criteria, the provisions of GDC-61, which provide in part for appropriate containment, confinement, and filtering systems, continue to be satisfied. The NRC staff has approved similar relaxations for other boiling and pressurized water reactor facilities and the NRC staff has approved a related generic Boiling Water Reactor (BWR) TS change (Technical Specification Task Force (TSTF) Standard TS Change Traveler, TSTF-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2). In these previous amendments for other facilities, the NRC staff has requested the licensees to make appropriate commitments to implement administrative means to facilitate restoration of the secondary containment should an FHA occur. Although NMC did not base its initial application for amendment upon TSTF-51, NMC subsequently stated in its letter dated March 23, 2001, that it adopts the associated commitment in TSTF-51 to follow the guidelines in NUMARC 93-01, Revision 3, Section 11, regarding restoration of the secondary containment. The NRC staff finds this commitment acceptable, and as such, finds the proposed TS changes acceptable.

2.5 Summary of NRC Staff Review and Conclusions

The NRC staff has reviewed the selective AST implementation and the secondary containment TS changes proposed by NMC for DAEC. In performing this review, the NRC staff relied upon docketed information provided by NMC, NRC staff experience in doing similar reviews and, to a limited extent, upon selective NRC staff confirmatory calculations.

The NRC staff reviewed the assumptions, inputs, and methods used by NMC to assess the radiological impacts of the proposed changes. The NRC staff finds that NMC used analysis methods and assumptions consistent with the conservative guidance of RG 1.183. The NRC staff compared the doses estimated by NMC to the applicable criteria and to the results of confirmatory analyses by the NRC staff. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, control room, and TSC total effective dose equivalent due to a FHA will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183.

The NRC staff finds reasonable assurance that DAEC's selective AST implementation 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, as they apply to the design-basis FHA. The NRC staff concludes that the proposed selective AST implementation and the proposed secondary containment TS changes are acceptable.

This licensing action is considered a selective implementation of the AST. With the approval of this amendment, the AST, the TEDE criteria, and the analysis methods, assumptions and inputs, become the design-basis for the assessment of radiological consequences of the design-basis FHA accident. All future radiological analyses associated with the design-basis FHA accident for DAEC shall use this approved design-basis. This approval is limited to this specific application. The NRC staff continues to review the additional analyses provided by NMC in support of the requested full-scope AST implementation. Until the NRC staff completes this review and approves the remainder of NMC's amendment request, the AST and TEDE criteria of 10 CFR 50.67 shall not be extended to other aspects of plant design or operation.

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 CONSIDERATION

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 April 16, 2001 (66 FR 19586), 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 Commission has concluded based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance With the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachments: 1. FHA Analysis Assumptions
2. FHA Analysis Dose Results

Principal Contributor: S. LaVie

Date: April 16, 2001

TABLE 1

FUEL HANDLING ACCIDENT ANALYSIS ASSUMPTIONS

Reactor Power, MWt, (x 102%)	1950
Radial Peaking Factor	1.46
Fuel Decay Period, hours	60
Number of Assemblies in Core	368
Number of Fuel Rods in an Assembly (*equivalent full and part length rods*)	87.3
Number of Damaged Rods	151
Fraction of Gap Activity Released from Damaged Rods	1.0
Fraction of Core Inventory in Gap	
I-131	0.08
Kr-85	0.10
Other Iodine and noble gases	0.05
Pool Decontamination Factor, Effective	200
Iodine Species Fraction Above Pool Water	
Elemental	0.57
Organic	0.43
Release Duration, hours	
From fuel and pool	Instantaneous
From secondary containment	2
Release Rate to Environment, %/day	2.4E7
Collection and Filtration by SBGT	None
Assumed Release Point	Reactor Building Vent
Atmospheric Dispersion, 0-2 hours, sec/m^3	
EAB	5.57E-4
LPZ	1.34E-4
Control Room	2.85E-3
TSC	2.66E-3
Control Room Volume, ft^3	155,000
Control Room Normal Makeup, cfm	3150
Control Room Emergency Flow, cfm	1000
Control Room and TSC Filter Efficiency, %	
Elemental	90
Organic	30
Aerosol	99
Control Room Unfiltered In-Leakage, cfm	1000

TABLE 1 (Continued)

FUEL HANDLING ACCIDENT ANALYSIS ASSUMPTIONS

Control Room Isolation Delay. Minutes	10
TSC Volume, ft^3	68,300
TSC Normal Makeup, cfm	900
TSC Emergency Flow, cfm	200
TSC Recirculation Flow, cfm	800
TSC Emergency Ventilation Actuation, minutes	30
TSC Unfiltered Inleakage, cfm	1000

TABLE 2

FUEL HANDLING ACCIDENT ANALYSIS DOSE RESULTS

	EAB TEDE *rem*	LPZ TEDE *rem*	Control Room TEDE *rem*	TSC TEDE *rem*
Analysis Result	0.94	0.23	3.16	2.83
Acceptance Criteria	6.3	6.3	5.0	5.0