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Duane Arnold Energy Center
Operated by Nuclear Management Company, LLC

November 14, 2005

NG-05-0594
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Duane Arnold Energy Center
Docket 50-331
License No. DPR-49

Technical Specification Change Request (TSCR-074): "Elimination of Main Steam Line Radiation Monitor Trips"
Affected Technical Specifications: Section 3.3.6.1

Pursuant to 10 CFR 50.90, Nuclear Management Company, LLC (NMC) hereby requests revision to the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC). The proposed Amendment revises the table of Primary Containment Isolation Instrumentation (Table 3.3.6.1-1) to eliminate the trip generated by the Main Steam Line Radiation Monitors (MSLRM).

The proposed Amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c). Associated TS Bases changes will be completed per the TS Bases Control Program (TS 5.5.10).

NMC requests approval of the proposed amendment by November 30, 2006. Once approved, the amendment will be implemented within 120 days. This schedule will permit the removal of the existing instrumentation during the next refuel outage, tentatively in February 2007.

This application has been reviewed by the DAEC Operations Committee. A copy of this submittal, along with the 10CFR50.92 evaluation of "No Significant Hazards Consideration," is being forwarded to our appointed state official pursuant to 10 CFR Section 50.91.

This letter makes no new commitments or changes to any existing commitments.

If you have any questions or require additional information, please contact Mr. Tony Browning at (319) 851-7750.

A001

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 14, 2005.

Gary Van Middlesworth
Site Vice President, Duane Arnold Energy Center
Nuclear Management Company, LLC

Exhibits: A) EVALUATION OF PROPOSED CHANGE
B) PROPOSED TECHNICAL SPECIFICATION AND BASES CHANGES (MARK-UP)
C) PROPOSED TECHNICAL SPECIFICATION PAGES (RE-TYPED)

cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
Resident Inspector, DAEC, USNRC
D. McGhee (State of Iowa)

EXHIBIT A

EVALUATION OF PROPOSED CHANGE

Subject: TSCR-074 - Elimination of Main Steam Line Radiation Monitor Trips

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1. DESCRIPTION

This letter is a request to amend Operating License DPR-49 for the Duane Arnold Energy Center (DAEC). The proposed Amendment would modify the Limiting Condition for Operation (LCO) 3.3.6.1, "Primary Containment Isolation Instrumentation" by deleting the isolation on Main Steam Line Radiation – High, (Function 1.f in Table 3.3.6.1-1). The subsequent line item in the Table would be re-numbered accordingly.

2. PROPOSED CHANGE

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend the Technical Specifications (TS) by deleting the referenced pages and replacing them with the enclosed new pages.

SUMMARY OF CHANGES:

TS Pages	BASES Pages
3.3-57	B 3.3-151 B 3.3-161 B 3.3-162 B 3.3-185

The proposed Amendment revises the Table of Primary Containment Isolation Instrumentation in LCO 3.3.6.1 to remove the Main Steam Line Radiation Monitor (MSLRM) trip on high radiation. Specifically, Table 3.3.6.1-1, Function 1.f would be removed and the subsequent line item in the Table would be re-numbered accordingly.

Technical Specification Bases are also modified to reflect the above changes (see Exhibit B). The Bases changes are included for information only. Bases changes will be completed per the TS Bases Control Program (TS 5.5.10).

3. BACKGROUND

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 trip (SCRAM) and a Nuclear Steam Supply Shutoff System (NSSSS) trip (Main Steam Isolation Valve (MSIV) closure). The BWR Owners' Group (BWROG) developed a licensing topical report (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).

4. TECHNICAL ANALYSIS

Compliance with Current Regulations and Design Basis

Design Basis

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.

The MSLRM alarm and trip setpoints are set at 150% and 300% increases in radiation above that expected level at normal, full-power operation, respectively. Thus, the MSLRMs are not expected to alarm or trip due to minor fuel defects/failures, but only on significant core damage.

The MVP is used during the early part of plant startup to evacuate the main condenser until enough steamflow is achieved such that the Steam Jet Air Ejectors can be put into service to maintain condenser vacuum. The MVP is secured from operation prior to exceeding 10% rated thermal power. 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 offsite release, until the

MSLRM trip occurs to isolate this pathway. At which point, the offsite release is a ground-level release for the remainder of the release period.

During initial phases of plant startup (i.e., reactor coolant temperature < 200 °F), the MSL drain lines are isolated. Thus, these lines would not represent a release path if a CRDA were to occur during this period. During later periods of the startup when reactor coolant temperatures exceed 212 °F, the MSL drain lines are opened, as needed, to maintain heatup within the TS limits. However, the MSIVs are also open during this period, i.e., the Main Steam Lines are open to the condenser. The MSL drain lines are a smaller piping than the MSL and both the Main Steam Line and MSL drain lines exhaust to the Main Condenser. Thus, both sets of piping see the influence of the MVP operation. Consequently, the MSLRM isolation of the MSL drain lines has a negligible effect on the overall release during a CRDA, given the smaller piping size compared to the MSL and the fact that they exhaust to the same location.

The sample line on the main recirculation system piping is for drawing the necessary coolant samples to confirm plant water chemistry is within required limits. The sample piping is very small (0.75 inch diameter) where it penetrates the primary containment. The piping size is reduced down to 0.50 inch diameter tubing to connect to the associated equipment, which is outboard from the containment isolation valves. This tubing connects to the Crack Arrest Verification System (CAVS), used to monitor reactor coolant chemistry conditions during plant startup and power operation. As part of putting the CAVS into service during plant startup activities, a leak check of the equipment and associated tubing connections is performed. So, this is not considered to be a likely release path. Because the CAVS is located in the Reactor Building (i.e., Secondary Containment), any small system leakage would be into the Secondary Containment. Secondary Containment Integrity, Standby Gas Treatment System (SGTS) and Secondary Containment Isolation Valves/Dampers (SCIV/Ds) are all required to be Operable by TS during plant startup (Modes 1 & 2). So, any leakage via the Recirculation Sample line pathway would be retained within the Reactor Building (Secondary Containment). A high radiation condition in the Offgas Stack, which would be expected during a CRDA with the MVP in operation, will result in a Secondary Containment isolation and initiation of the SGTS, as the Offgas Stack radiation monitor would normally be in service. If the Offgas Stack monitor is not in service, any significant radiation release in the Reactor Building would trip the building exhaust radiation monitors, which also cause a Secondary Containment isolation and initiation of SGTS. Thus, any release via this pathway would be an elevated, filtered release. Therefore, a MSLRM trip and isolation of the Recirculation Sample Line would not significantly impact the overall release during a CRDA.

Current Regulations

The proposed change is consistent with the current regulations and thus, an exemption pursuant to 10 CFR 50.12 is not required. The current regulations do not specifically require the consideration of the CRDA as a "design basis accident," and consequently, there is no specific requirement for the MSLRMs to mitigate this event. Conformance to the current regulations will be maintained, in particular, 10 CFR 50.67[1], with the elimination of the MSLRM from the plant design and Technical Specifications.

From footnote 1 of 10 CFR 50.67:

> *The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. {emphasis added}*

This application will demonstrate that the dose consequences are in conformance with §50.67, assuming the MSLRM are no longer available to mitigate the consequences of a CRDA. In addition, maintaining the DAEC licensing basis for assumptions on manual operator actions keeps the event "credible."

10 CFR 50.36

In this application, NMC will demonstrate that the MSLRM and its associated trip function to isolate the MVP, MSL drains and recirculation system sample valves do not satisfy the 10 CFR 50.36 criteria and can be eliminated from the Technical Specifications. Specifically, the dose analysis will demonstrate that the current designation of satisfying Criterion 3 (§50.36(c)(2)(ii)(C)) is no longer applicable, as offsite dose consequences remain within regulatory limits, assuming no MSLRM isolation function, consistent with the DAEC licensing basis for assumptions of manual operator actions. In addition, a qualitative risk assessment will demonstrate that Criterion 4 (§50.36(c)(2)(ii)(D)) is also not applicable, as the MSLRM are not risk significant.

10 CFR 50, Appendix A, General Design Criteria

The DAEC Construction Permit was issued in 1970, prior to the issuance of 10 CFR 50, Appendix A, General Design Criteria (GDC), and the DAEC was not specifically licensed to them (Ref. SECY-92-223). The following describes the DAEC UFSAR commitment to the GDCs pertinent to this application and the impact of the requested change on those commitments.

> *Criterion 13—Instrumentation and control.* Instrumentation shall 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, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall

[1] The DAEC has adopted the Alternate Source Term as its licensing basis for dose consequences (full conversion) in License Amendments # 237 and 240.

be provided to maintain these variables and systems within prescribed operating ranges. {emphasis added}

DAEC commitment to GDC 13 (UFSAR Section 3.1.2.2.4) – Amendment 182 to the DAEC Operating License previously removed the primary trip functions (SCRAM and NSSSS isolation) of the MSLRMs; only the isolation of the MVP, MSL drains, and recirculation system sample valves remain. This application will demonstrate that these residual isolations are not needed to achieve the overall purpose to "assure adequate safety," per GDC 13. The primary signal used to assure adequate safety is the reactor SCRAM on high neutron flux initiated by the Neutron Monitoring System (NMS), which is not being affected by this application.

> *Criterion 20—Protection system functions.* The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) <u>to sense accident conditions and to initiate the operation of systems and components important to safety</u>. {emphasis added}

DAEC commitment to GDC 20 (UFSAR Section 3.1.2.3.1) – this application will demonstrate that the MSLRMs and their associated trip function are not needed to mitigate the consequences of an accident, and thus, are no longer SSC "important to safety" and can be removed from the facility. Again, as stated above, the NMS trip on high neutron flux is the primary safety action necessary to mitigate the CRDA.

> *Criterion 28—Reactivity limits.* The reactivity control systems shall 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. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), <u>rod dropout</u>, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition. {emphasis added}

DAEC commitment to GDC 28 (UFSAR Section 3.1.2.3.9) – While NMC proposes to eliminate the MSLRMs and their associated trip function, other design features will be retained that will limit the consequences of a CRDA. Specifically, the Control Blades will retain their velocity limiter design to limit the rate of reactivity addition from the highly unlikely event of a rod dropout (i.e., a CRDA), and the Rod Worth Minimizer (RWM) and its associated Banked Position Withdrawal Sequence (BPWS) rod pattern controls will be retained in TS to limit the individual control rod worth to values below those used in the CRDA analysis. Thus, the results of the CRDA analysis in Chapter 15 of the DAEC UFSAR that demonstrate that the reactor coolant pressure boundary remains intact and that a core geometry amenable to cooling is maintained after a CRDA, will remain valid.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. {emphasis added}

DAEC commitment to GDC 64 (UFSAR Section 3.1.2.6.5) – The DAEC will retain the capability to perform this function utilizing other installed equipment (e.g., RG 1.97 accident range effluent monitors and Offgas Stack radiation monitors), after the removal of the MSLRMs. In addition, the removal of the MSLRMs will not impact the ability to monitor releases during normal operation, as that function is performed by other instrumentation, maintained in accordance with the DAEC Offsite Dose Assessment Manual (ODAM), pursuant to 10 CFR Part 20 and 10 CFR Part 50, App. I.

10 CFR 50, Appendix B, Quality Assurance

Introduction

Nuclear power plants and fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying. {emphasis added}

Again, NMC will demonstrate that eliminating the MSLRMs as a SSCs "important to safety" will not result in "undue risk to the health and safety of the public."

Standard Review Plan (SRP) and associated Regulatory Guides

While the DAEC was not originally licensed to the SRP, applicable portions of the SRP will be used in this application to assist the Staff in its review.

SRP Chapter 15.4.9, (Spectrum of Rod Drop Accidents (BWR)) establishes the specific guidelines to meet the basic acceptance criteria outlined in GDC 28. As stated above for GDC 28, while NMC proposes to eliminate the MSLRMs and their associated trip function, other design features will be retained that will limit the consequences of a CRDA. Thus, the results of the CRDA analysis in Chapter 15 of the DAEC UFSAR that demonstrate that the reactor coolant pressure boundary remains intact and that a core geometry amenable to cooling is maintained after a CRDA, will remain valid.

SRP Chapter 15.4.9, App. A, (Radiological Consequences of Control Rod Drop Accident (BWR)), as updated by RG 1.183 (Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors), is used in this application for performing the dose assessments.

SRP Chapters 4.2, 4.3 and 4.4, while peripheral to the subject of CRDA, are not specifically applied here, as the traditional reactor physics analysis of the CRDA is not modified by this application.

Deterministic Evaluation:

Re-Baseline Analysis

As noted above, the DAEC current licensing basis is based upon a full conversion to the Alternative Source Term of §50.67 and RG 1.183. The analyses performed for this evaluation are consistent with that licensing basis, including the changes due to the final issuance of RG 1.183 from those of Draft Guide 1081, as NMC committed in Reference 1. The three changes implemented in this re-baseline calculation are as follows:

- Change in release fractions from DG-1081 to those approved by the NRC in RG 1.183, Appendix C.
- Change in release time to satisfy the instantaneous release assumption approved by the NRC in RG 1.183 Section 3.3. This re-baseline evaluation used a five second fission product release period for both the gap release and release from melted fuel, rather than the two hour release assumed in the original calculation. A shorter release period results in a larger early release of radioactivity, higher radiological exposures, and is more realistic since the CRDA causes fuel damage within a matter of seconds.
- Calculation of Control Room (CR) and Technical Support Center (TSC) doses is performed in accordance with approved NRC guidance in RG 1.183. These calculations were not performed in the original calculation based on engineering judgment that the CRDA was not a limiting event for the DAEC.

The radiological dose consequences are summarized in Table 1 below: Results are shown for two sources (Gap Release and Pellet [fuel melt] Release), and Total Dose for each of three sets of assumptions for release period and fission product release fractions. The first set of results is from the original analysis, assuming DG-1081 release fractions and the 2-hour release. The second set assumes RG 1.183 release fractions, but retains the 2-hour release assumption of the original analysis. The third set assumes a 5 second release period and the RG 1.183 release fractions.

Table No. 1: Control Rod Drop Accident Radiological Consequences

Accident Type		Exclusion Area Boundary (EAB) (2 hr)	Low Population Zone (LPZ) (30 day)	Control Room (30 Day)	TSC (30 Day)
		TEDE (rem)			
Gap Release	Original Analysis (2 hr Release, DG-1081 Release Fractions)	[5.78E-02]	[3.85E-02]	[N/A]	[N/A]
	2 hr Release RG 1.183 Release Fractions	5.5017E-02	3.5896E-02	3.4405E-01	5.0040E-01
	5 sec Release RG 1.183 Release Fractions	6.3661E-02	3.8113E-02	3.5203E-01	5.1157E-01
Fuel Melt Release	Original Analysis (2 hr Release, DG-1081 Release Fractions)	[1.56E-03]	[1.07E-03]	[N/A]	[N/A]
	2 hr Release RG 1.183 Release Fractions	1.8135E-03	1.3304E-03	9.9444E-03	1.4389E-02
	5 sec Release RG 1.183 Release Fractions	3.4633E-03	1.8833E-03	1.1262E-02	1.6193E-02
Total Release	Original Analysis (2 hr Release, DG-1081 Release Fractions)	[5.94E-02]	[3.96E-02]	[N/A]	[N/A]
	2 hr Release RG 1.183 Release Fractions	5.6831E-02	3.7226E-02	3.5399E-01	5.1479E-01
	5 sec Release RG 1.183 Release Fractions	6.7124E-02	3.9996E-02	3.6329E-01	5.3770E-01
Total Dose*		**6.8E-2**	**4.0E-2**	**3.7E-1**	**5.4E-1**

*Numerical precision of results is shown based on results of software. Some inputs are only specified to 2 significant digits, therefore results are not considered to be accurate beyond two significant digits.

Comparing the results in Table 1 for the 2-hour assumption with RG 1.183 release fractions to the same 2-hour assumption with DG-1081 release fractions (the original calculation) shows that this change would have resulted in a slight decrease in predicted dose consequences from the original calculations. However, the results for the 5-second release period with RG 1.183 release fractions represent small increases in dose consequences over those predicted in the original calculations.

New Analysis for MSLRM Trip Elimination

ANALYSIS DESCRIPTION

The key difference in this new analysis over the above re-baseline evaluation is that the MSLRM will not automatically trip the MVP and terminate the elevated release from the Offgas Stack. This will now be a manual action by the Operators.

Consistent with the DAEC licensing basis, credit may be taken under accident conditions for Operator actions after 10 minutes (DAEC UFSAR Section 15.0.4).

NRC has published guidelines for substituting Operator manual actions for automatic actions in Information Notice 97-78 (Ref. 2). The Staff's guidelines were originally intended to guide licensees in making proper determinations of whether prior NRC approval was required for such a substitution under the provisions of §50.59. Because NMC is asking for prior approval, the guidelines will be used to demonstrate that this substitution is appropriate. Per the guidelines, the key aspects for determining whether substituting manual for automatic actions is appropriate are whether there are sufficient alarms or other indications that can be used to diagnose the event, that plant procedures and training exist to guide the Operator response or actions, the sensitivity of the timing of these actions, and the consequences of any errors in executing those actions.

In this case, there are several alternate indications of fuel damage and that an offsite release is taking place. Specifically, the normal range/post-accident range radiation monitoring system (KAMAN) installed to meet NUREG-0737, Item II.F.1 and RG 1.97, Category 3 (Variable C13). The KAMAN system alarms in the main Control Room. The alarm settings on the normal-range KAMAN monitor is set in accordance with the Offsite Dose Assessment Manual (ODAM), for normal effluent releases per 10 CFR Part 20 and 10 CFR 50, App. I, i.e., well below those radiation levels anticipated during any analyzed accident. In addition, there is a backup monitoring system on the offgas stack; the General Electric (GE) Offgas Stack radiation monitors. These monitors would normally be in service and alarm in the Control Room and would also cause Engineered Safety Feature (ESF) trips (selective Primary Containment penetration isolations and a Secondary Containment Group III isolation) on a high radiation signal. (Note: this ESF trip does not trip the MVP or otherwise secure the CRDA release path.) Consistent with the accident analysis assumption of a 5 second transport time for the fission products released during the CRDA to reach the main condenser, the alarm settings on both the primary and backup radiation monitors would be exceeded very quickly into the event. Thus, the Operator would be prompted by these alarms shortly after the event occurred.

Each of these alarms in the Control Room has a corresponding section in an Annunciator Response Procedure (ARP). These ARPs, including the current ARPs associated with the MSLRMs, direct the Operator response to these alarms, which include taking the actions in the associated Abnormal Operating Procedure (AOP) and Operating Instructions (OI). The pertinent AOP for the above alarms directs the Operator to secure the MVP if it is operating. The Operators receive training on the ARPs, AOPs and OIs as part of their annual license re-qualification program. Note: as part of implementation of this submittal, when approved by the Staff, these current 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, either the KAMAN or GE system. Associated operator training will also be conducted as part of this implementation.

The controls for securing the MVP are in the Main Control Room. The Operator need not enter the plant to perform this operation.

Thus, it is reasonable to assume that the Operator will take the appropriate action to secure the MVP consistent with the licensing basis action time of 10 minutes.

Later in this evaluation, NMC will present the results of a sensitivity study that demonstrate that the timing of the Operator actions is not critical and that the worst operator error – failure to manually trip the MVP, does not result is a significant increase in offsite dose consequences.

Existing RG 1.97 instrumentation for monitoring offsite releases (e.g., KAMANs) and assessing core damage (Drywell Radiation Monitors) remain adequate for directing the emergency response to a CRDA event. No new or upgraded instrumentation is needed to compensate for the removal of the MSLRMs. Nor does the instrumentation that directs that Operator response meet the definition for a Type A variable, per RG 1.97, as the Operator action is not essential to achieving acceptable consequences.

MODELING

Radiological consequences were calculated using version 3.03 of the RADTRAD code.

Atmospheric dispersion was calculated with PAVAN and ARCON96.

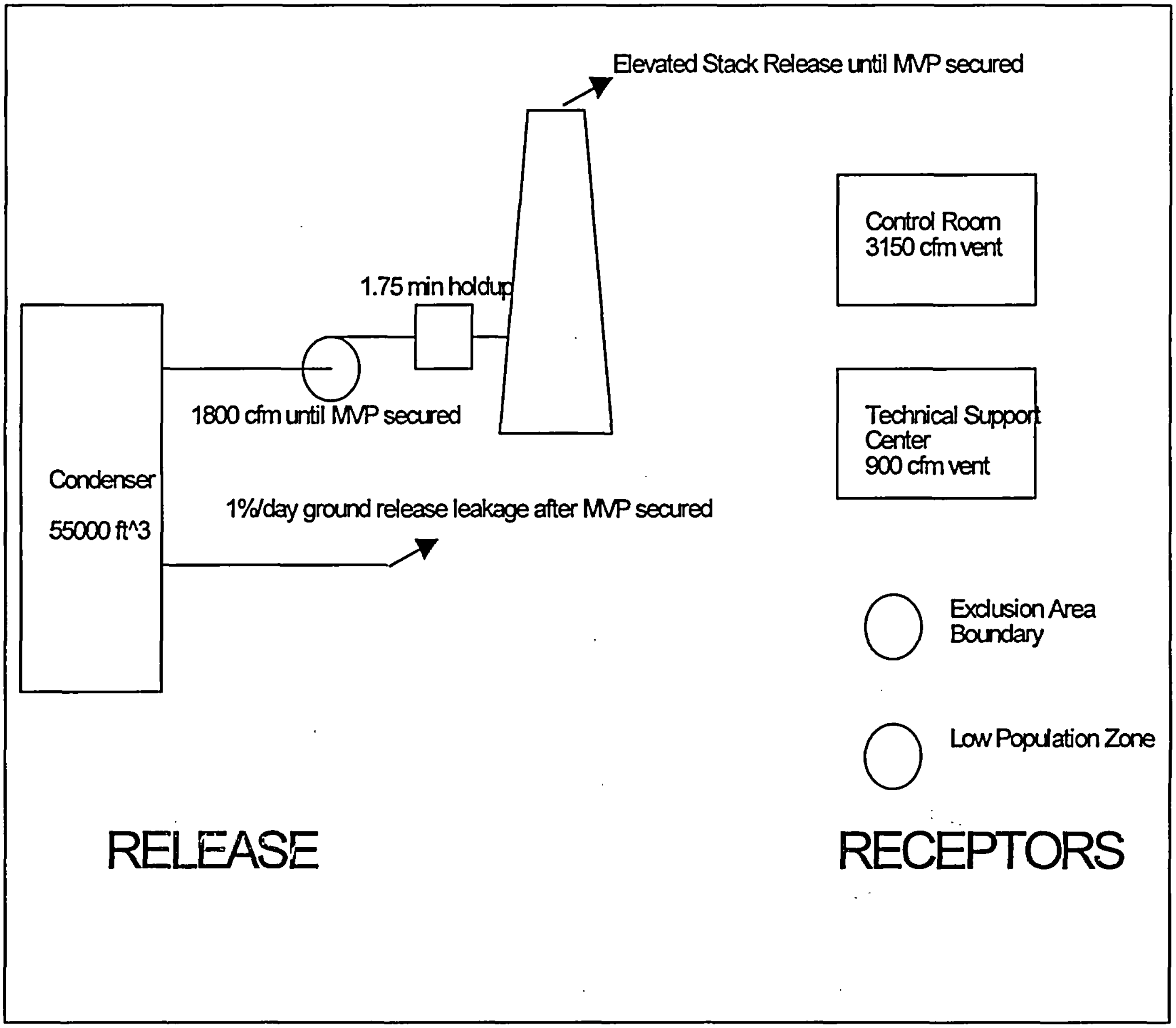
The basic model of the CRDA event is shown in Figure 1.

As a conservative assumption, the MSL drain flowpath to the condenser is not modeled and all the fission product release is transported directly to the turbine and condenser through the larger main steam piping. This maximizes the release, both in magnitude and timing, as the holdup in these drain lines is ignored.

In addition, the release path from the Recirculation Sample Valves is also not modeled, as any small system leakage would be into the Secondary Containment. Secondary Containment Integrity, Standby Gas Treatment System (SGTS) and

Secondary Containment Isolation Valves/Dampers (SCIV/Ds) are all required to be Operable by TS during plant startup (Modes 1 & 2). So, any leakage via the Recirculation Sample Valves pathway would be retained within the Reactor Building (Secondary Containment). A high radiation condition in the Offgas Stack, which would be expected during a CRDA with the MVP in operation, will result in a Secondary Containment isolation and initiation of the SGTS, as the Offgas Stack radiation monitor would normally be in service. If the Offgas Stack monitor is not in service, any significant radiation release in the Reactor Building would trip the building exhaust radiation monitors, which also cause a Secondary Containment isolation and initiation of SGTS. Thus, any release via this pathway would be an elevated, filtered release. Therefore, assuming all the fission products are released via the Condenser - MVP pathway is conservative, as this is not a filtered release.

Figure 1



Elevated Stack Release until MVP secured
Control Room
3150 cfm vent
1.75 min holdup
1800 cfm until MVP secured
Technical Support
Center
900 cfm vent
Condenser
55000 ft^3
1%/day ground release leakage after MVP secured
Exclusion Area
Boundary
Low Population Zone
RELEASE
RECEPTORS

KEY ASSUMPTIONS:

- When the CRDA occurs, the source term from the damaged fuel is transferred to the reactor coolant over a 5 second period and is immediately transported to the main condenser. This assumption conservatively ignores transport time and assumes the full source term is transported to the condenser during the 5 second release period.
- The MVP is assumed to be operating to draw a vacuum in the condenser until manually isolated by the Operators at 10 minutes after the event initiation.
- The MVP pumps the contents of the Main Condenser through a 1.75-minute delay line to the Offgas Stack where it is released to the environment.
- Once the MVP is secure, the release continues due to leakage from the condenser at 1% volume change per day for the remainder of the 24 hour release duration. All condenser leakage is immediately released to the environment via direct leakage out of the TB without holdup, plateout, or dilution.
- No credit is taken for isolation or filtration systems for the CR or TSC. Normal ventilation is assumed for the duration of the event. Control room ventilation also assumes 1000 cfm of unfiltered in-leakage.

KEY INPUTS:

- The available source term is the same as that used in the original analysis (Amendment 240) and accounts for the portion of fuel which experiences clad breach (1200 rods) and the portion of damaged fuel which experiences melting (0.77%).

Group	Release Fraction		
	Gap Release	Fuel Melt Release	Total
Noble Gases	0.10	0.9	1.0
Halogens	0.10	0.4	0.5
Alkali Metals	0.12	0.13	0.25
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

- MVP rated flow of 1800 cfm.
- Condenser volume of 55,000 ft^3.
- Control Building Volume of 155,000 ft^3. (Note: the Control Building includes the Main Control Room.)
- TSC Volume of 68,300 ft^3.
- Control Building Intake flow rate of 3150 cfm.
- TSC Intake flow rate of 900 cfm.
- Condenser leakage is 1% volume change per day.

The following are the χ/Q's for the various release paths and dose receptor locations:

Ground Level Release χ/Q's to EAB and LPZ (sec/m^3)
Calculated Using PAVAN code

Time Period	EAB (629m, ENE)	LPZ (3218m, NE)
0 – 2 hours	5.57E-04	1.34E-04
2 - 8 hours	3.42E-04	6.43E-05
8 – 24 hours	2.69E-04	4.46E-05
1 - 4 days	1.59E-04	2.01E-05
4 – 30 days	7.43E-05	6.42E-06

Ground Level Release χ/Q's to CR and TSC (sec/m^3)
Calculated Using ARCON96 code

Includes Occupancy Adjustment Factors for ARCON96 Values		
Time Period	CR	TSC
0 – 2 hours	1.48E-03	2.14E-03
2 – 8 hours	1.27E-03	1.86E-03
8 – 24 hours	5.56E-04	8.44E-04
1 – 4 days	2.04E-04	3.66E-04
4 – 30 days	1.06E-04	1.88E-04

Elevated (Stack) Release χ/Q's to EAB and LPZ (sec/m3) (RG 1.194)

Time Period	EAB	LPZ
0 – 30 min (Fumigation)	7.03E-05	3.15E-05
30 min to 2 hrs	6.95E-06	6.69E-06
2 - 8 hours	3.61E-06	3.58E-06
8 – 24 hours	2.61E-06	2.61E-06
1 - 4 days	1.28E-06	1.32E-06
4 - 30 days	4.64E-07	4.99E-07

Post-accident meteorology atmospheric dispersion factors (χ/Q's) for the Control Room and TSC were calculated using inputs from the original analysis (Ref. 1) and the methodology described in RG 1.194 (Reference 5), Section C.3.2.2. The maximum χ/Q using PAVAN or

ARCON96 is used for the 0 – 2 hour interval. ARCON96 values are used for the intervals from 2 hours to 24 hours. The "1 – 4 Day" and "4 – 30 Day" intervals are calculated using a weighted average assuming 1 hour at the PAVAN value and 23 hours at the ARCON96 value per day. Results are summarized as follows:

Elevated (Stack) Release χ/Q's to CR (sec/m3)

Time Period	ARCON96	PAVAN	RG 1.194
0 – 30 Min (Fumigation)	Not Calculated	2.62E-4	N/A
0 – 8 Hours	3.80E-07	4.70E-06	N/A
0 – 2 Hours	3.93E-07	1.68E-05	1.68E-05
2 - 8 Hours	3.75E-07	Not Calculated	3.75E-07
8 – 24 Hours	1.33E-07	2.49E-06	1.33E-07
1 - 4 Days	6.24E-08	3.74E-07	7.54E-08
4 - 30 Days	3.75E-08	3.42E-08	3.74E-08

Elevated (Stack) Release χ/Q's to TSC (sec/m3)

Time Period	ARCON96	PAVAN	RG 1.194
0 – 30 Min (Fumigation)	Not Calculated	2.38E-04	N/A
0 – 8 Hours	2.20E-07	3.65E-06	N/A
0 – 2 Hours	2.32E-07	1.37E-05	1.37E-05
2 - 8 Hours	2.16E-07	Not Calculated	2.16E-07
8 – 24 Hours	8.00E-08	1.89E-06	8.00E-07
1 - 4 Days	3.69E-08	2.71E-07	4.67E-08
4 - 30 Days	2.16E-08	2.31E-08	2.17E-08

RESULTS:

The results from this analysis are as follows:

	Dose Receiver Location (REM TEDE)			
Case	**EAB**	**LPZ**	**CR**	**TSC**
Gap	2.699	1.217	0.460	0.527
Pellet	0.151	0.0679	0.0151	0.0169
Totals*	**2.85**	**1.29**	**0.48**	**0.54**

*Numerical precision of results is shown based on results of software. Some inputs are only specified to 2 significant digits, therefore results are not considered to be accurate beyond two significant digits.

As can be seen from these results, the dose at each of the receptor points is well below the regulatory guidelines of 6.3 REM TEDE for the CRDA (RG 1.183, Table 6).

SENSITIVITY ANALYSIS:

Operator Response Time

As part of the in-house feasibility study for this change, several scenarios were analyzed with varying MVP isolation times. As this is a short-term release of fission products (i.e., the fuel failure is prompt and abrupt), the entire fission product inventory is assumed to be transported to the main condenser within 5 seconds of event initiation. Over the release duration, there is both an elevated release (during MVP operation), and a subsequent ground level release, via condenser leakage and turbine building leakage (after MVP operation is secured). As expected, the offsite dose results were most sensitive to the length of time the MVP operates; the longer the MVP operates, the longer the elevated release period. Because of the limited nature of the release (quantity and duration), sustained MVP operation evacuates the condenser of fission products to the point where almost all, ~99.9%, of the offsite dose occurs within the first 2 hours of MVP operation.

To bound this event, specifically to demonstrate the scenario where the Operator fails to take the assumed actions to manually isolate the MVP, a case was performed assuming 24-hour MVP operation. The 24-hour duration is consistent with the guidelines of App. C of RG 1.183. The results of that case follow:

Dose (REM TEDE)			
EAB	LPZ	CR	TSC
7.40	3.52	0.88	0.69

The dose rates to the LPZ, Control Room, and Technical Support Center remain within regulatory guidelines.

The limiting 2-hour EAB dose does exceed the regulatory guidelines of RG 1.183, but by only 17.5%. However, this value remains well below the regulatory limit of 25 REM TEDE in 10 CFR 50.67. This is considered to meet the "well within the exposure guideline values," criterion stated in SRP Chapter 15.4.9, App. A. The 25% value in SRP 15.4.9 and RG 1.183, for defining "well within" was arbitrarily chosen. The above result (~30%) can easily be considered to be "well within the exposure guideline values" and therefore, this result is deemed to be acceptable.

More importantly, this case demonstrates that the assumption of the Operator response time of 10 minutes is not critical to achieving acceptable results and the associated instrumentation used to detect the offsite release need not be upgraded to a Type A variable, per RG 1.97.

Safety Margins:

The following lists the major conservatisms in this analysis, which demonstrate additional safety margins:

- No credit is taken for reduced source term inventory from radioactive decay during shutdown or operation at lower than full power. Although the CRDA is assumed to occur at <10% rated power, i.e., during plant startup, a full-power (102% of rated power) inventory is used in the analysis.
- The RG 1.183 *assumption for fuel damage during a CRDA is also* conservative. The expected fuel enthalpy during a CRDA is well below the 280 cal/gm acceptance limit[2]. Thus, a CRDA is not expected to result in any significant fuel damage.
- The source term inventory assumes all damaged fuel is equivalent to the highest power peaked bundle in the core. Since 1200 rods are assumed to fail and the average GE14 bundle has 87.3 equivalent rods, this would require at least 13 fuel bundles to fail. A core loading pattern with that many highly peaked bundles concentrated in a single area of the core (i.e., clustered around the dropped control rod) is not feasible under other fuel constraints (e.g., shutdown margin).
- The assumption that the entire source term is released to the condenser volume in the first 5 seconds is conservative both for duration and quantity of radioisotopes transported.
- Both the MSL drain lines and Recirculation Sample flow paths have been excluded. This is conservative in that the entire fission product inventory is promptly transported to the Main Condenser via the larger MSL piping.
- The assumption that MVP flow rate is constant at 1800 cfm for up to 24 hours is conservative. The CRDA will not result in damage to MSL piping or to the condenser walls and seals. The MVP will be drawing a vacuum on the condenser and, consequently its flow rate will decrease accordingly over time, thus reducing the release rate through the offgas stack.
- No holdup time or transport delay is considered for condenser leakage into the turbine building.
- No holdup time, platecut, or transport delay is considered for leakage from the turbine building to the environment.
- 1000 scfm of unfiltered in-leakage into the Control Room is very conservative relative to measured in-leakage (<100 scfm).

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[2] From Ref. 3, Section S.2.2.3.1.2 - Control rod drop accident (CRDA) results from BPWS plants have been statistically analyzed and documented in Reference S–15. The results show that, in all cases, the peak fuel enthalpy in an RDA would be much less than the 280 cal/gm design limit even with a maximum incremental rod worth corresponding to 95% probability at the 95% confidence level. Based on these results, it was proposed to the NRC, and subsequently found acceptable, to delete the CRDA from the standard GE BWR reload package for the BPWS plants.

The analysis, performed consistent with the DAEC licensing basis for Operator actions, demonstrates that the regulatory guidelines are met with considerable margin. The sensitivity case performed, and the major conservatisms in the analysis listed above, demonstrate that there is a considerable safety margin in the analysis results.

Defense-in-Depth Considerations:

While NMC proposes to eliminate the MSLRMs and their associated trip function, other design features will be retained that will limit the consequences of a CRDA. Specifically, the Control Blades will retain their velocity limiter design to limit the rate of reactivity addition from the highly unlikely event of a rod dropout (i.e., a CRDA), and the Rod Worth Minimizer (RWM) and its associated Banked Position Withdrawal Sequence (BPWS) rod pattern controls will be retained in TS (LCO 3.3.2.1 and LCO 3.1.6, respectively) to limit the individual control rod worth to values below those used in the CRDA reactor physics analysis. Thus, the results of the CRDA analysis in Chapter 15 of the DAEC UFSAR that demonstrate that the reactor coolant pressure boundary remains intact and that a core geometry amenable to cooling is maintained after a CRDA, will remain valid. Thus, preserving the defense in depth of the overall plant design.

Qualitative Risk Assessment

The CRDA is acknowledged to be an extremely unlikely event in a Boiling Water Reactor (Ref. 4), as numerous, independent events (combinations of equipment failures and operator errors) are required to result in fuel damage. The various, existing TS provisions that preclude this event are being maintained to ensure that the probability of the CRDA is maintained at a very low value. For example, the control rod coupling checks (SR 3.1.3.5), stuck rod provision (LCO 3.1.3), "slow" control rod separation criteria (LCO 3.1.4), are not being revised as part of this license amendment request.

Again, the MSLRM trip does not prevent a CRDA; its sole function is in response to the event once it occurs. Thus, elimination of the MSLRM trip function will not increase the risk of core damage from a CRDA event.

As demonstrated by the deterministic dose assessment above, elimination of the MSLRM trip function does not result in unacceptable dose consequences. Thus, the elimination of the MSLRM trip will not increase the large early release frequency of a CRDA.

Thus, the overall risk of a CRDA event is still extremely low, after the removal of the MSLRM trip function.

Conclusion

The above evaluation demonstrates that the MSLRM do not satisfy any criterion in 10 CFR 50.36(c)(2)(ii) for inclusion in TS:

Criterion 1: The MSLRM trip function is not an initial condition or assumption in any accident analysis. The only event that credits the MSLRM trip function is the CRDA. The MSLRM trip is assumed to occur as a consequence of this event, it does not preserve an initial condition.

Criterion 2: The MSLRM trip function does not detect leakage in the reactor coolant pressure boundary, but detects the significant fission product release during a CRDA and to isolate the release path via the MVP (which bounds the MSL drains and recirculation sample system release paths).

Criterion 3: The enclosed dose assessment demonstrates that the MSLRM trip of the MVP (which bounds the MSL drains and recirculation sample system release paths) is not necessary to ensure that on-site and off-site dose consequences of a CRDA remain within published guidelines. Thus, the MSLRM trip function is not critical to the mitigation of any analyzed accident.

Criterion 4: The enclosed dose assessment demonstrates that eliminating the MSLRM trip function does not result in unacceptable consequences, i.e., contribute to the large early release frequency. And, the MSLRM trip function does not preclude the CRDA event, i.e., does not contribute to the core damage frequency. Thus, the MSLRM trip is not a risk-significant function.

Thus, the MSLRM trip function can be deleted from the TS, as the above criteria for inclusion in TS are not met.

Furthermore, NMC concludes that the elimination of the MSLRM and their associated trip functions from the plant design will not result in undue risk to the health and safety of the public. Thus, the associated costs for replacing the existing equipment and maintaining it is not justified by the minimal safety benefit.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Nuclear Management Company (NMC), LLC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No.

The proposed change deletes the Main Steamline Radiation Monitor (MSLRM) trip function from TS. The MSLRM is not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The consequences of an accident previously evaluated, specifically the Control Rod Drop Accident (CRDA), have been evaluated consistent with the DAEC licensing basis utilizing the Alternative Source Term (10 CFR 50.67). As demonstrated by the dose calculations, the consequences of the accident are within the regulatory acceptance criterion. As a result, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

No new or different accidents result from utilizing the proposed change. The changes do not involve a change in the methods governing normal plant operation. The equipment proposed to be removed from the plant, the MSLRM, is only credited in the CRDA analysis and no other event in the safety analysis. The proposed changes are consistent with the revised safety analysis assumptions for a CRDA included in this application.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

Response: No.

The proposed change deletes the requirement for the MSLRM isolation function. Analyses performed consistent with the DAEC licensing basis, demonstrate that the removal of this isolation will not cause a significant reduction in the margin of safety, as the resulting offsite dose consequences are being maintained within regulatory limits.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the preceding 10 CFR 50.92 evaluation NMC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

Attorney for Licensee: Jonathan Rogoff, Esquire, General Counsel, NMC, LLC, 700 First St., Hudson, WI, 54016

5.2 Applicable Regulatory Requirements/Criteria

By letter dated November 14, 2005, Nuclear Management Company, LLC (NMC) submitted a request for revision of the Technical Specifications for the Duane Arnold Energy Center (DAEC). The proposed amendment revises the Technical Specifications by eliminating the isolation function generated by the Main Steamline Radiation Monitors (MSLRM) on a high radiation signal.

Evaluation:

As demonstrated in Section 4 above, the elimination of the MSLRM trip function from the plant design and Technical Specifications is consistent with current regulations. Thus, an exemption pursuant to 10 CFR 50.12 is not required. Further, the change is wholly consistent with the DAEC current licensing basis for dose assessment (§50.67) and for crediting operator actions in response to analyzed accidents.

In conclusion, based on the considerations discussed above, (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. Therefore, NMC has concluded that the proposed revision to the DAEC Technical Specifications is acceptable.

6.0 ENVIRONMENTAL CONSIDERATION

10 CFR Section 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (3) result in a significant increase in individual or cumulative occupational radiation exposure. NMC has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows.

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9) for the following reasons:

1. As demonstrated in the 10 CFR 50.92 evaluation included in this exhibit, the proposed amendment does not involve a significant hazards consideration.

2. The proposed changes do not result in an increase in power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. As demonstrated by the dose calculations performed in accordance with the DAEC licensing basis, and presented in the application, the consequences of the accident are well within the regulatory acceptance criterion. Thus, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

3. The proposed changes do not result in changes in the level of control or methodology used for processing of radioactive effluents or handling of solid radioactive waste nor will the proposal result in any change in the normal radiation levels within the plant. There is no significant increase in individual or cumulative occupational radiation exposure.

Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment.

7. REFERENCES

1. NMC letter, "Technical Specification Change Request TSCR-037: Alternative Source Term," NG-00-1589, October 19, 2000. (ADAMS Accession # ML003762396)

2. NRC Information Notice (IN) 97-78: Crediting Of Operator Actions In Place Of Automatic Actions And Modifications Of Operator Actions, Including Response Times," October 23, 1997.

3. General Electric Standard Application for Reactor Fuel (Supplement for United States), NEDE–24011–P–A–14–US, June 2000.

4. Thadani (NRC) to Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel", Revision 8, Amendment 17," December 27, 1987.

5. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants", dated June 2003.

EXHIBIT B

PROPOSED TECHNICAL SPECIFICATION

AND

BASES CHANGES

(MARK-UP)

5 Pages to Follow

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 38.3 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	≥ 821 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	≤ 138% rated steam flow
d. Condenser Backpressure - High	1, 2[a], 3[a]	2	D	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 7.2 inches Hg vacuum
e. Main Steam Line Tunnel Temperature - High	1,2,3	4	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	≤ 205.1°F
~~f. Main Steam Line Radiation - High~~	~~1,2,3~~	~~2~~	~~F~~	~~SR 3.3.6.1.1~~ ~~SR 3.3.6.1.4~~ ~~SR 3.3.6.1.8~~ ~~SR 3.3.6.1.9~~	~~≤ 5530 mR/hr~~
~~g.~~ f Turbine Building Temperature - High	1,2,3	4	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	≤ 205.1°F

(continued)

a) When any turbine stop valve is greater than 90% open or when the key-locked bypass switch is in the NORM Position.

B 3.3 INSTRUMENTATION

B 3.3.6.1 Primary Containment Isolation Instrumentation

BASES

BACKGROUND

The primary containment isolation instrumentation automatically initiates closure of appropriate Primary Containment Isolation Valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary containment and Reactor Coolant Pressure Boundary (RCPB) isolation. Most channels include equipment (e.g., on-off sensors or bi-stable trip circuits) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a primary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logics are (a) reactor vessel water level, (b) area ambient and differential temperatures, (c) Main Steam Line (MSL) flow measurement ~~and high radiation~~, (d) Standby Liquid Control (SLC) System initiation, (e) condenser vacuum, (f) main steam line pressure, (g) High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) steam line flow, (h) drywell pressure, (i) HPCI and RCIC steam line pressure, (j) HPCI and RCIC turbine exhaust diaphragm pressure, (k) Reactor Water Cleanup (RWCU) differential flow, (l) reactor steam dome pressure, (m) Offgas Vent Stack radiation, (n) Reactor Building Exhaust Shaft radiation, and (o) Refueling Floor Exhaust Duct radiation. Redundant sensor input signals from each parameter are provided for automatic initiation of isolation. The only exceptions are SLC System initiation and RWCU differential flow. In addition, manual isolation of certain logics is provided. Primary containment isolation instrumentation has inputs to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.e and 1.~~g~~ f. Area Temperature—High

Area temperature is provided to detect a leak in the RCPB and provides diversity to the high flow instrumentation. The isolation occurs when a very small leak has occurred. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. However, credit for these instruments is not taken in any transient or accident analysis in the UFSAR, since bounding analyses are performed for large breaks, such as MSLBs.

Area temperature signals are initiated from Resistance Temperature Detectors (RTDs) located in the area being monitored. Sixteen channels of Main Steam Tunnel Temperature—High Function are available and 8 channels (2 per main steam line) are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function for a break of the size for which protection is necessary. Eight channels of Turbine Building Area Temperature—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each channel consists of a RTD and its contacts in the corresponding temperatures indicating switch.

The Area Temperature-High Allowable Value is set far enough above the temperature expected during operations at rated power to avoid spurious isolation, yet low enough to provide early indication of a steamline break.

These Functions isolate the Group 1 valves.

~~1.f. Main Steam Line Radiation - High~~

~~The Main Steam Line Radiation - High isolation signal has been removed from the MSIVs (Ref. 9); however, this isolation Function has been retained for other valves (e.g., Main Steam Line (MSL) Drains) to ensure that the assumptions utilized to determine that acceptable offsite doses resulting from a CRDA are maintained.~~

~~Main Steam Line Radiation - High signals are generated from four radiation elements and associated monitors, each of which is located near one of the MSLs in the steam tunnel.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~1.f. Main Steam Line Radiation - High (continued)~~

~~The Main Steam Line Radiation - High Allowable Value is chosen to be low enough so that the assumptions utilized to determine that acceptable offsite doses resulting from a CRDA are maintained.~~

~~Four Main Steam Line Radiation - High channels are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.~~

~~This function isolates the MSL drains and the recirculation sample valves, and causes the mechanical vacuum pump to trip, if operating, and then the suction valves to the vacuum pump to close.~~

Primary Containment Isolation

2.a. Reactor Vessel Water Level — Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Reactor Vessel Water Level-Low supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Reactor Vessel Water Level — Low Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level — Low signals are initiated from level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level — Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level — Low Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level - Low scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Primary Containment Isolation instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions for Functions other than 5.a may be delayed for up to 6 hours provided the associated Function maintains isolation capability and for up to 6 hours for Function 5.a. For Functions 1.c, 1.e, and 1.~~g~~ f, the Allowed Outage Time (AOT) is applied at the instrument channel level, since the associated trip function and isolation capability are maintained via the companion logic channel. This is consistent with the "normal" trip arrangements with one instrument channel feeding each trip logic. Thus, a six hour AOT is applied to each instrument channel undergoing required testing. Upon completion of the Surveillance, or expiration of the applicable 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

This Note is based on the reliability analysis (Refs. 5 and 6) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the PCIVs will isolate the penetration flow path(s) when necessary. Because the Ref. 5 and 6 analyses made no assumptions regarding the elapsed time between testing of consecutive channels in the same logic, it is not necessary to remove jumpers/relay blocks or reconnect lifted leads used to prevent actuation of the trip logic during testing of logic channels with instruments in series solely for the purpose of administering the AOT clocks, provided that the AOT allowance is not exceeded on a per instrument channel basis.

(continued)

EXHIBIT C

PROPOSED TECHNICAL SPECIFICATION PAGES

(RE-TYPED)

1 Page to Follow

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 38.3 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	≥ 821 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	≤ 138% rated steam flow
d. Condenser Backpressure - High	1, 2[(a)], 3[(a)]	2	D	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 7.2 inches Hg vacuum
e. Main Steam Line Tunnel Temperature - High	1,2,3	4	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	≤ 205.1°F
f. Turbine Building Temperature - High	1,2,3	4	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	≤ 205.1°F

(continued)

(a) When any turbine stop valve is greater than 90% open or when the key-locked bypass switch is in the NORM Position.