NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

January 20, 2005

L-MT-05-001
10 CFR 50.90
10 CFR 50.67

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

Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Response to Request for Additional Information Related to Technical Specifications Change Request to Apply Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident," dated January 11, 2005 (TAC No. MC3299)

References:
1) NMC letter to NRC, "License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident," (L-MT-04-023) dated April 29, 2004.
2) NMC letter to NRC, "Supplement 1 to License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident," (L-MT-04-064) dated November 23, 2004, (TAC No. MC3299).
3) NRC letter to NMC, "Monticello Nuclear Generating Plant - Request for Additional Information Related to Technical Specifications Change Request to Apply Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident," dated January 11, 2005 (TAC No. MC3299).

On April 29, 2004, pursuant to 10 CFR 50.67 and 10 CFR 50.90, the Nuclear Management Company, LLC, (NMC) requested a selective scope application of an alternative source term (AST) to the fuel handling accident (FHA) for the Monticello Nuclear Generating Plant (MNGP) (Reference 1). NMC proposed to amend the MNGP licensing basis and Technical Specifications based on a revised FHA radiological consequence analysis with an AST. On November 23, 2004, NMC provided a supplemental letter discussing shutdown administrative controls for Secondary Containment, ventilation system and radiation monitor availability during refueling, and validation of the FHA radiological consequence analysis Control Room inleakage assumptions (Reference 2). The NRC staff requested additional information (RAI) on January 11, 2005 (Reference 3). Enclosure 1 provides the response to this RAI.

Summary of Commitments

This letter contains one new commitment and no revisions to existing commitments. The new commitment is:

> NMC will propose a Technical Specification for Spent Fuel Pool water level during irradiated fuel movement under separate correspondence.

John T. Conway for TJP

Thomas J. Palmisano
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

ENCLOSURE 1

RESPONSE TO AN NRC RAI FOR THE FUEL HANDLING ACCIDENT ALTERNATIVE SOURCE TERM SUBMITTAL

On April 29, 2004, pursuant to 10 CFR 50.67 and 10 CFR 50.90, the Nuclear Management Company, LLC, (NMC) requested a selective scope application of an alternative source term (AST) to the fuel handling accident (FHA) for the Monticello Nuclear Generating Plant (MNGP) (Reference 1). NMC proposed to amend the MNGP licensing basis and Technical Specifications (TS) based on a revised FHA radiological consequence analysis with an AST. On November 23, 2004, NMC provided a supplemental letter discussing shutdown administrative controls for Secondary Containment, ventilation system and radiation monitor availability during refueling, and validation of the FHA radiological consequence analysis Control Room inleakage assumptions (Reference 2). The NRC staff requested additional information (RAI) on January 11, 2005 (Reference 3). The NRC questions or requests are shown in 'bold' text below and the NMC response is provided in 'standard' text immediately after.

A. **Nuclear Management Company's (NMC's) April 29, 2004 License Amendment Request (LAR)**

1. **One of the proposed commitments associated with this license amendment request is to change the refueling procedures to require a minimum of 23 feet of water above stored fuel in the spent fuel pool during irradiated fuel movement. Such a commitment is usually linked with a technical specification (TS) requirement. Why wasn't a TS surveillance requirement proposed to require 23 feet of water above stored fuel?**

   NMC will propose a Technical Specification for Spent Fuel Pool water level during irradiated fuel movement under separate correspondence.

2. **Proposed changes to Table 3.2.4, "Instrumentation that initiates Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation" remove automatic isolation functions. Although the fuel handling accident (FHA) analyses predicts that the releases from the accident would be less than the guidelines presented in 10 CFR 50.67, the commitment to NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," requires the building to be isolated to contain the release of an accident and filter systems to be used to process and clean up the release, if required, in order to keep releases to a minimum. Please clarify if the exhaust through the reactor building ventilation is terminated manually or redirected through a filtered system as part of the secondary containment closure process defined by shutdown administrative controls.**

   In the event of a FHA, the Secondary Containment will be isolated in accordance with the proposed shutdown administrative controls. If automatic actuation and isolation of the Reactor Building vent was not available, plant procedures implementing the secondary containment closure process would direct that venting by the normal Reactor Building exhaust be manually terminated. The Standby Gas Treatment (SBGT) System, a safety related Engineered Safety Feature (ESF), which would provide filtration, would then be used to process the release.

B. Enclosure 1 of NMC's Submittal [April 29, 2004 LAR]

1. **NMC provided only one dose analysis for a FHA. That was the dose associated with fuel which was not "recently" irradiated. If NMC ever intends to handle fuel which is "recently" irradiated, then NMC needs to provide an analysis that demonstrates acceptable dose results, both offsite and in the control room, in the event of an FHA.**

   The AST FHA analysis assumes a refueling accident occurs 24 hours after the reactor is shutdown. The proposed TS do not permit fuel that has been "recently" irradiated (i.e., defined in the TS Bases as having been within a critical reactor core within the preceding 24 hours), to be handled. NMC does not propose, or intend, to handle recently irradiated fuel and therefore a FHA analysis was not performed for this scenario.

2. **The FHA analysis assumes that 125 fuel rods of an 8x8 array assembly are damaged. How is it ensured that this analysis is bounding for each operating cycle?**

   Global Nuclear Fuel (GNF) performs licensing analyses for NMC of fuel designs used at MNGP in accordance with the NRC approved topical report GESTAR II (Reference 4). The NRC safety evaluation for Amendment 22 of GESTAR II (Reference 5), Section 2.13, "Refueling Accident Analysis Licensing Evaluation," states, "The consequence of a refueling accident as presented in the country-specific supplement GESTAR II or the plant FSAR shall be confirmed as bounding or a new analysis shall be performed." The NRC staff evaluation of Amendment 22 concluded:

   > "The consequence of the fuel handling accident is mainly dependent on the amount of fuel rods in a bundle. If there is a change to the number of fuel rods or a new fuel design is proposed, the effect on the refueling accident must be reconfirmed or reanalyzed; therefore, this approach is acceptable."

   Thus, the AST FHA analysis assumption that the equivalent to 125 fuel rods of an 8x8 array assembly are damaged is re-evaluated as new fuel designs are proposed for use at MNGP consistent with the GESTAR II, Amendment 22 process. This assumption is confirmed by NMC as new fuel designs are adopted for use at the MNGP, and appropriate reanalysis are performed, as required, in accordance with regulatory requirements.

3. **NMC states that the MNGP control room ventilation system normally operates only in the recirculation mode of operation with no makeup flow due to concerns of leakage past the normal makeup air intake dampers. Consequently, the normal makeup air dampers were blanked off. It would seem that a stagnant air problem would develop in the control room envelope (CRE) without normal makeup. If there is not a problem, this would imply that inleakage during normal operation is substantial. What is the inleakage to the CRE in the normal mode of operation?**

   There are two normal modes of Control Room Ventilation-Emergency Filtration System (CRV-EFT) System operation, differentiated by whether an EFT System train is running to provide fresh air makeup to the CRE or in standby. In both of these normal modes one CRV train is in operation for air circulation and conditioning.

   a. Normal Mode with a CRV Train and an EFT Train in Operation

      With one EFT train running the CRE configuration is the same as that tested in the worst-case tracer gas test (refer to Supplement 1, Reference 2). The simulated multiple equipment failures in this test alignment included the applicable CRE components in their normal mode of operation. In this mode one EFT Train is running providing a nominal 1000 cubic feet per minute (cfm) of filtered makeup flow. Inleakage in this configuration was measured during tracer gas testing as 100 ± 25 cfm. The CRV-EFT System is operated in this configuration approximately 85 percent of the time.

   b. Normal Mode with only a CRV Train in Operation

      With the EFT trains in standby, there is no forced makeup flow to balance the forced exhaust flows so the CRE is generally at a negative pressure[1] with respect to adjacent areas. The low inleakage results obtained by tracer gas testing demonstrate that the CRE has good integrity. Significant outleakage would not be expected from a negative envelope with good integrity; therefore, inleakage in this configuration may effectively be determined by measuring the forced exhaust fan flows from the envelope. Field measurements were performed with each train in operation and determined an inleakage of 404 and 278 cfm for the “A” and “B” CRV train respectively.

   The inleakage in the normal mode with a CRV and an EFT train in operation (used approximately 85 percent of the time) was determined to be 100 ± 25 cfm during tracer gas testing. The maximum measured inleakage in the normal mode with only a CRV train in operation was 404 cfm.

---

1 For example, the Reactor Building is maintained at a significant negative pressure during plant operation. Thus the CRE is normally positive with respect to the Reactor Building.

Therefore, the maximum inleakage for the normal modes is less than 500 cfm, which is half the 1000 scfm inleakage assumed in the AST FHA radiological consequence analyses. The total flows (makeup plus unfiltered inleakage) discussed above are considerably less than the total flow of 8440 scfm (7440 scfm unfiltered air intake plus 1000 scfm inleakage) assumed in the AST FHA radiological consequence analyses.

C. Enclosure 2 to NMC's April 29, 2004 LAR

NMC states on page 2 that it used a peaking factor of 1.7 in the analysis even though MNGP does not specify a radial peaking factor in the TSs or the COLR [Core Operating Limits Report] and that the value was considered conservative. What core parameter(s) are monitored to ensure that the FHA analysis remains relevant? How are these parameter(s) used to conclude that the core remains within the assumed 1.7 value for radial peaking factor? If it is determined that a value greater than 1.7 should be used, will MNGP be resubmitting a FHA for staff review and approval?

The radial peaking factor (RPF) is a core design parameter. The RPF is not directly monitored during reactor operation. Maintaining reactor operation within the core operating limits indirectly assures compliance with the RPF design criterion. Core operating limits are determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met. Compliance with the operating limits described in the COLR demonstrates that the licensing basis analyses remain relevant. The monitored parameters are critical power ratio (CPR), linear heat generation rate (LHGR) and maximum average planar linear heat generation rate (MAPLHGR).

Core design parameters are controlled by the core design and reload analysis process. The core design and reload analysis procedures and design documents will be revised to clearly specify the connection between RPF as an AST FHA analysis assumption and reload design. The specific RPF value of 1.7 is considered conservative based on conceptual core designs from the NMC Nuclear Analysis Department and review of previous calculation assumptions.

10 CFR 50.59 provides guidance for determining when a proposed activity requires prior NRC approval. A change in RPF for an FHA resulting in more than a minimal increase in radiological consequences would require submittal of a license amendment for NRC review and approval.

D. <u>Enclosure 3 to NMC's April 29, 2004 LAR</u>

1. **The BASES section associated with TS Table 3.2.4 does not address effluent monitoring for the various modes of operation.**

   Radiological effluent controls including monitoring and surveillance requirements are located in the MNGP Offsite Dose Calculation Manual (ODCM). The ODCM contains the Radiological Effluent Technical Specifications (RETS) and their bases. Effluent monitoring provisions (the RETS) were relocated to the ODCM in accordance with Generic Letter 89-01 (Reference 6) by TS License Amendment 120. MNGP TS Section 6.8.D, "Radioactive Effluent Controls Program," specifies the requirements and controls for the ODCM. The ODCM controls for effluent monitoring and monitoring instrumentation apply at all times and would therefore be applicable to all modes of operation including refueling.

<u>Proprietary Calculation 2004-02104, Rev. 0 - Sargent & Lundy Project No. 11163-013</u>

NMC states that the calculation is conservative regardless of whether the reactor building normal ventilation or standby gas treatment fans are operating or not. If a fan was not operating within the reactor building, would this result in the release occurring over a period longer than 2 hours, and would it result in a higher control room operator dose than the analysis provided? If the release occurred over 2 hours without the reactor building fan operating, would it result in a larger dose?

The 2-hour release period used in the FHA analysis was based on the requirements of Regulatory Guide 1.183, Appendix B, Item 5.3, which specifies, "…the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period." The calculated dispersion factors decrease significantly after two hours. Therefore, a higher Control Room dose for a longer release period would not be expected since less activity reaches the Control Room air intake.

The Reactor Building vent source provides a bounding value for the AST FHA analysis. A larger dose would not result if the Reactor Building fans were not operating. The Reactor Building vent was chosen as a representative and conservative release point based on its location with respect to the Control Room air intake. The other significant Secondary Containment opening, the Reactor Building Railroad Door airlock, was considered to be a less limiting source since it is located almost 50 percent further away from the Control Room air intake. There would be minimal driving force for any release if the Reactor Building fans were not in operation.

NMC's November 23, 2004 Supplement 1 to the LAR

E. Enclosure 1 to NMC's November 23, 2004, Supplement 1

1. **Address the manner in which effluents are monitored during fuel handling operations as a result of this change in operations and plant TSs. Is the monitoring consistent with your licensing basis i.e., principle design criterion 17, 10 CFR Part 20 and Appendix I of 10 CFR Part 50?**

   Effluent monitoring at MNGP is performed in accordance with the MNGP ODCM. The ODCM controls for plant gaseous effluents are applicable at all times and would apply during fuel handling operations. The controls implement the requirements of 10 CFR 20, 10 CFR 50.36a, General Design Criteria 60 of Appendix A to 10 CFR 50, are consistent with Principle Design Criterion 17 and the design objectives of Appendix I to 10 CFR 50.

   The manner in which effluents are monitored during fuel handling operations as a result of this change in operations and plant TSs remains unchanged. Wide Range Gas Monitors (WRGM's) installed at the Plant Stack and Reactor Building Ventilation duct stacks perform effluent monitoring functions. The WRGM's are described in MNGP USAR Section 7.5, "Plant Radiation Monitoring Systems," and the operability requirements for the WRGM's are described in Section 03.01 of the ODCM.

2. **Enclosure 1 said that one train of the control room ventilation system will be operating during refueling operations. Control room air is being recirculated in this operating mode. Makeup air to the control room envelope (CRE) is provided on an as-needed basis through the operation of one of the control room emergency filtration treatment filter banks. The analysis provided in support of this amendment did not assume the control room ventilation systems would be operating in the manner described above. Rather, it was assumed that when the FHA occurred, makeup air was being provided to the control room envelope at a rate of 7440 cubic feet per minute (cfm) and CRE inleakage was 1000 cfm. None of this are [sic] was filtered or adsorbed. NMC stated that the dose to control room operators was insensitive to inleakage or makeup flows for the range of 300-8500 cfm. Inleakage or makeup flows less than 300 cfm are not addressed. The actual mode of operation during refueling operations will involve no makeup flow. Based upon NMC's November 18, 2004 response to Generic Letter 2003-01, it is indicated that the CRE inleakage while operating the B train in the recirculation mode of operation is 188 cfm ± 10. No value is provided for the A train because the A train was not tested in this configuration. Instead, the A train was tested in the pressurization mode of operation as was the B train. The A train was found to have more inleakage than B train. What is the inleakage rate for the A train operating in the recirculation mode of operation? What are the dose consequences with the limiting train operating in the recirculation mode of operation?**

Clarification of the Recirculation Mode of Operation

As discussed in the response to Request B.3, the CRV-EFT System operates in one of two normal modes, differentiated by whether an EFT System train is in standby or running to provide fresh air makeup to the CRE. In both normal modes one CRV train is in operation. In the LAR and Supplement (References 1 and 2), the CRV System was described as operating as a “recirculation-only” system following the blanking-off of the air intakes. This terminology was meant to describe how the system operates and was not intended to be synonymous with the ‘Recirculation Mode’ of CRV-EFT System operation.

As reported in the response to Generic Letter 2003-01 (Reference 7), an “A” Train CRV Recirculating Mode test was not deemed necessary since all recirculating boundary dampers and system ductwork outside the CRE are common to both trains. Therefore, the inleakage rate with the “A” CRV train operating in the Recirculation Mode would be similar to that measured for the “B” CRV train (188 ± 10 cfm).

CRV-EFT System operation in the Recirculation Mode differs from that in the normal modes of operation in that the CRE is isolated, the EFT System trains do not operate, and there is no forced makeup or exhaust. The Recirculation Mode is used only in the event of a toxic chemical release. This mode is not applicable to normal operation or to operation during a radiological event. Therefore, there are no radiological dose consequences associated with operation of the CRV-EFT System in the Recirculation Mode.

Modes of CRV-EFT System Operation Versus AST FHA Analysis Assumptions

The applicable modes of operation for the CRV-EFT System for the FHA are the Normal Mode and Pressurization Mode. As discussed in the response to Request B.3, combined inleakage/makeup flows for these modes range from about 280 to 1200 cfm. The AST FHA analysis submitted in conjunction with the LAR (Reference 1), assumed combined inleakage/makeup flows ranging from 300 to 8440 scfm[(2)].

Additional parametric cases were performed at 75 and 150 scfm inleakage with no additional makeup. Calculated doses were slightly less than doses calculated for the flow range of 300 to 8500 scfm. This confirms that low combined flow rates do not result in an increase in dose consequences. Thus, the AST FHA analysis demonstrates that the dose consequences for operation in either the Normal or Pressurization Mode are acceptable.

2 Because the X/Q for the source of outside air intake and unfiltered inleakage (the CR air intake) and the timing (throughout the accident) are the same, the AST FHA analysis treats them equivalently.

3. NMC's letter of November 23, 2004, contains an Assessment of Ventilation System and Radiation Monitor Availability. The NRC staff does not consider the submittal to be risk informed. No probability risk assessment is provided, and no basis for risk established. Please clarify what is meant by "risk" or "acceptable risk" and give NMC's basis for determining when and where systems need to be available to monitor or control the ventilation during movement of irradiated fuel after the period "recently" has passed.

Risk encompasses what can happen, its likelihood, and its level of damage. In plant operations, use of the term risk is not limited to only activities for which a probabilistic risk assessment (PRA) has been performed. The MNGP AST FHA analysis (and associated TS changes) is not a "Risk Informed" submittal. Risk management during an outage is an integrated process of assessing and reducing the likelihood and/or consequences of an adverse event. NUMARC 93-01 (Reference 8) provides guidelines for the risk assessment of maintenance activities:

> "The assessment method may use quantitative approaches, qualitative approaches, or blended methods. In general, the assessment should consider:
>
> - Technical specifications requirements
> - The degree of redundancy available for performance of the safety function(s) served by the out-of-service [system, structure or component] SSC
> - The duration of the out-of-service or testing condition
> - The likelihood of an initiating event or accident that would require the performance of the affected safety function.
> - The likelihood that the maintenance activity will significantly increase the frequency of a risk-significant initiating event (e.g., by an order of magnitude or more as determined by each licensee, consistent with its obligation to manage maintenance-related risk).
> - Component and system dependencies that are affected.
> - Significant performance issues for the in-service redundant SSCs"

The MNGP Plant Risk Management Program implements the NUMARC 91-06 guidance, which also satisfies NUMARC 93-01 (Section 11) requirements for shutdown conditions. This program ensures that systems and components that perform key safety functions, including secondary containment functions, are available when needed. Adhering to this programmatic guidance attains acceptable risk levels. Performance of the safety assessment for shutdown conditions generally involves a qualitative assessment with regard to key safety functions. MNGP also performs a shutdown PRA analysis; this evaluation is used as an input to the site shutdown assessment.

4. **There are numerous references to "outage schedule" or "outage schedule design" in NMC's letter of November 23, 2004. Outage schedule should not be a consideration in the mitigation of the accident. Also, on page 2 of 9 of Enclosure 1 to NMC's letter of November 23, 2004, NMC quotes the following NUMARC 93-01 guidance:**

   > **The goal of maintaining ventilation and system radiation monitor availability is to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases.**

   Please clarify how outage schedule impacts the mitigation of an FHA with respect to controlling releases. How do the shutdown administrative controls demonstrate that the NUMARC goal will be achieved?

   NMC places primary emphasis on reactor and personnel safety when preparing an outage schedule. As described in NUMARC 91-06 (Reference 9), outage safety can be improved by focusing on the availability of systems that provide and support key safety functions as well as on measures that can reduce both the likelihood and consequences of adverse events. The outage schedule identifies systems and features available for the mitigation of the accident at any given point in time during the outage. It is recognized that the complexity, diversity, and number of activities that take place during an outage require a high degree of coordination in order to maintain defense-in-depth. Defense-in-depth entails providing for the backup of key safety functions, optimization of safety system availability, stipulating administrative controls that support the functionality of key equipment, and utilizing procedures designed to mitigate the loss of key safety functions. MNGP procedures for outage schedule design and risk minimization implement this NUMARC guidance. These practices may be applied directly to the mitigation of the FHA. For example:

   Secondary Containment, Emergency Filtration (EFT), and Standby Gas Treatment (SBGT) availability during the outage – Secondary containment, EFT, and SBGT System availability is optimized during the outage to maintain defense-in-depth. A clear result of this approach is that minimizing the time when these systems are impaired, maximizes the time they are available to contain, filter, and monitor a release due to a potential FHA.

   Minimization of concurrent activities that utilize openings in Secondary Containment during refueling – Minimizing the number of open Secondary Containment penetrations minimizes the time required to promptly close those open penetrations. This enhances assurance that the ventilation systems are able to draw the release from a postulated FHA in the proper direction such that it can be treated and monitored.

Development of contingency plans for secondary containment closure – Contingency plans are developed during outage planning for circumstances identified by the schedule that have the potential to lessen defense-in-depth and worsen the consequences of the FHA.

The shutdown administrative controls to be implemented upon approval of this proposed LAR are of two general types. The first type consists of planning and scheduling controls; the second type includes instructions for Secondary Containment closure.

Planning and scheduling controls contribute to achievement of the NUMARC goal. Evaluation of the availability of the Secondary Containment and ventilation systems and associated radiation monitors occurs during development of an outage schedule. This evaluation considers impacts on filtering, monitoring, and minimizing potential releases in the event of a FHA. Employment of the administrative controls in development of the outage schedule maximizes the availability of these filtering, monitoring and containment equipment over the course of the refueling outage. It follows that achievement of the purpose of the NUMARC goal, to reduce doses below that provided by the natural decay and to avoid unmonitored releases, is enhanced by these use of these controls.

Plant procedures for monitoring Secondary Containment openings and executing Secondary Containment closure requirements contribute to achievement of the NUMARC goal. These procedures assure the ongoing identification of open penetrations (with associated contingency or closure plans) and assure prompt closure of open Secondary Containment penetrations. This enables ventilation systems to draw the release from a postulated FHA in the proper direction such that it can be treated and monitored. Prompt Secondary Containment closure will reduce the dose effects of the FHA below the natural decay levels referred to in the NUMARC document.

References

1. NMC letter to NRC, "License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident," (L-MT-04-023) dated April 29, 2004.

2. NMC letter to NRC, "Supplement 1 to License Amendment Request: Selective Scope Application of an Alternative Source Term Methodology for Re-evaluation of the Fuel Handling Accident," (L-MT-04-064) dated November 23, 2004, (TAC No. MC3299).

3. NRC letter to NMC, "Monticello Nuclear Generating Plant - Request for Additional Information Related to Technical Specifications Change Request to Apply Alternative Source Term (AST) Methodology to Re-Evaluate the Fuel-Handling Accident," dated January 11, 2005 (TAC No. MC3299).

4. Global Nuclear Fuels − Americas Report, NEDE-24011-P-A-14, "General Electric Standard Application For Reactor Fuel," (GESTAR II), current revision, and U.S. Supplement, NEDE-24011-P-A-14-US, current revision.

5. Global Nuclear Fuels − Americas Report, NEDE-24011-P-A-14-US, General Electric Standard Application For Reactor Fuel, (GESTAR II), U.S. Supplement, current revision, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Relating to Amendment 22 to General Electric Topical Report NEDE-24011-P-A General Electric Standard Application for Reactor Fuel."

6. U.S. Nuclear Regulatory Commission, Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program (Generic Letter 89-01)," dated January 31, 1989.

7. NMC letter to NRC, "Generic Letter 2003-01: Control Room Habitability – Design Bases, Licensing Bases and Inleakage Testing Results," (L-MT-04-049) dated November 18, 2004.

8. Nuclear Energy Institute, NUMARC 93-01, Revision 3, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated July 2000.

9. Nuclear Energy Institute, NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," dated December 1991.