From: Peter Tam
To: Loeffler, Rick; Salamon, Gabor
Date: 3/10/06 3:04PM
Subject: Monticello Draft Safety Evaluation for Proposed FHA AST Amendment (TAC MC7596)

Rick:

Attached please find the portions of the draft SE that we plan to use to support your proposed amendment to use the alternative source term (AST) for the postulated fuel handling accident (FHA). Please review the attached to ascertain that the factual information is accurate and complete. This draft is provided to you per the guidance of NRR Office Instruction COM-203, Revision 1.

Please call me to provide your comments.

This e-mail and the attached portions of the draft SE do not formally convey an NRC staff position or finding.

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

e-mail: pst@nrc.gov Tel.: 301-415-1451

3.0 TECHNICAL EVALUATION

3.1 The Licensee's Analyses

3.1.1 Radiological Analysis

To support the proposed change in MNGP licensing basis change, the licensee provided an analysis of the consequences of a fuel handling accident using the AST. The licensee's analysis assumed that the fuel handling accident occurred within the containment. This scenario was considered more limiting than the dropping of a fuel assembly over the spent fuel pool.

The licensee's analysis assumed that a fuel assembly was dropped on the top of the reactor core during refueling operations. The depth of water over a fuel bundle in the reactor cavity greatly exceeds 23 feet. In the spent fuel pool, there exists a low water alarm which corresponds to a depth of approximately 22 feet above the stored fuel. The decontamination afforded by the water in the spent fuel pool would thus be less than that which would be credited to the water in the reactor cavity due to this difference in water depth. The licensee stated that the drop over the reactor cavity would be more limiting because it would result in the damage of more fuel rods than the drop occurring over a spent fuel pool even with a 1-foot difference in water depth. By its April 12, 2005 letter, the licensee proposed to change the TSs to require a minimum water depth of 37 feet in the spent fuel pool during movement of irradiated fuel assemblies (an increase from the current requirement of 33 feet). In that letter, the licensee also presented a more detailed discussion of the bounding nature of the analysis of the FHA in the reactor cavity.

Specification 3.10.D of the MNGP TSs and the licensee's refueling procedures require that the reactor be shut down for a minimum of 24 hours prior to the movement of fuel within the reactor. Therefore, the licensee assumed a 24-hour decay period in determining the release of radioactivity.

The spent fuel pool at MNGP contains fuel assemblies which have 8x8, 9x9 and 10x10 array designs. The licensee indicated that the number and type of fuel rods in the reactor core may vary with each cycle. The number and type of fuel assemblies for each cycle are specified by the core nuclear design. The actual number of fuel rods which would fail in the event of a fuel handling accident would depend upon the fuel array and upon the fuel handling equipment involved. Section 14.7.6.3.1 of the MNGP USAR states that the radiological analysis for an FHA assumed failure of 125 rods of a GE 8x8 array. If the failed fuel involved a 9x9 or a 10x10 array, the activity associated with their failure would be 91 percent and 95 percent, respectively, of the activity associated with an 8x8 array. Therefore, the failure of an 8x8 array assembly was considered limiting.

The licensee's analysis assumed that the damaged fuel had a radial peaking factor (RPF) of 1.7. All of the gap activity of the damaged rods was assumed to be released instantaneously to the pool. The pool was assumed to retain all aerosols and particulate fission products. Noble gas activity released from the fuel was not assumed to be retained by the pool. All of the particulate iodine released from the fuel gap was assumed to be converted to the elemental form of iodine. A net decontamination factor of 200 was assumed for iodine.

The guidance in RG 1.183 allows an effective iodine decontamination factor of 200 when the depth of the water above the damaged fuel is at least 23 feet. This pre-condition is met for the reactor cavity, but not for the spent fuel pool, where the water depth is only 21 feet 4 inches. Nor is the pre-condition met for the drop of a fuel assembly over the reactor vessel flange, where the water depth is only 21 feet 8 inches. In its June 29, 2005 (Accession No. ML051960175, application to convert the MNGP TSs to improved Standard TSs) letter, the licensee provided calculations to show that the implied reduction in scrubbing efficiency is offset

by the reduced number of fuel rods that are projected to be damaged by either a fuel assembly drop in the spent fuel pool or over the reactor vessel flange.

The total effective dose equivalent (TEDE) includes contributions from both noble gases and iodine isotopes. The iodine scrubbing efficiency only applies to iodine isotopes, and mainly impacts the inhalation dose, or committed effective dose equivalent (CEDE). A decrease in the iodine scrubbing efficiency would increase the CEDE and, assuming the noble gas release remains the same, would also increase the TEDE to a lesser extent. The total radionuclide release (and the subsequent dose) is directly related to the number of fuel rods damaged in the drop. For the fuel drop in the spent fuel pool, the licensee calculated damage to 71 fuel rods. For the drop over the reactor vessel flange, only one assembly is involved with damage to all 60 of its fuel rods. These fuel damage estimates are compared to the damage and release from 125 rods assumed in the design basis analysis of the FHA in the reactor vessel.

The effective iodine decontamination factor in RG 1.183 is based on an exponential function. Using this function, NMC calculated effective iodine decontamination factors for the water depths less than 23 feet. For the FHA in the spent fuel pool, the pool level of 21 feet 4 inches would result in a reduction in scrubbing efficiency of less than 25 percent (and a less than 25 percent increase in iodine species released from the water). This is less than the approximately 43 percent reduction in the amount of damaged rods, and, hence, the amounts of radionuclides released. For the FHA over the reactor vessel flange, similar reasoning can be used to show that the reduction in scrubbing efficiency of approximately 20 percent (and approximately 20 percent increase in iodine release) is more than compensated for by the 52 percent decrease in radionuclide release by fewer fuel rods assumed damaged. Although the licensee calculated minimum water levels that would still be bounded by the design basis analysis of the FHA in the reactor vessel, the licensee did not propose, nor does the staff approve, the use of the minimum water level values as anything other than information which shows evidence of analysis margin. Based on the preceding discussion, the NRC staff finds the licensee's conclusion that the consequences of an FHA over the reactor cavity bounds those for an FHA in the spent fuel pool or an FHA over the reactor vessel flange to be acceptable.
The licensee assumed that the primary and secondary containment were not isolated. All activity released from the pool was assumed to enter the reactor building and be released within two hours via the reactor building vent without credit for decay or dilution in the building. The licensee assumed that the standby gas treatment (SBGT) system did not operate to mitigate the consequences of the FHA.

3.1.2 Atmospheric Dispersion Factor Analysis

The licensee used onsite meteorological data collected during calendar years 1998-2002 to generate new control room, exclusion area boundary (EAB) and low-population zone (LPZ) atmospheric dispersion factors (χ/Q values) for use in this proposed license amendment. The licensee modeled ground level releases from the reactor building vent and elevated releases from the 100-meter-tall off-gas stack. Meteorological data input into the ARCON96 atmospheric dispersion computer code consisted of hourly records of wind speed and direction data from measurements made at a height of 10 meters and 43 meters above ground and stability class data calculated using the temperature difference between the 43-meter and 10-meter levels. The licensee provided a copy of these hourly data for NRC staff review. Meteorological data input into the PAVAN atmospheric dispersion computer code consisted of joint wind speed, wind direction, and atmospheric stability frequency distributions (joint frequency distributions). Three sets of joint frequency distributions were used: (1) 100-meter wind data with stability calculated using the temperature difference between the 100-meter and 10-meter levels, (2) 43-meter wind data with stability calculated using the temperature difference between the 43-meter and 10-meter levels, and (3) 10-meter wind data with stability calculated using temperature difference between the 43-meter and 10-meter levels.

In the February 28, 2005, response to an NRC staff request for additional information (RAI), the licensee stated that MNGP does not have a commitment to meet RG 1.23, "Onsite Meteorological Programs." However, the licensee stated that from 1998-2002, the

meteorological measurement program complied with RG 1.23, other than with respect to calibration frequency. The program met RG 1.23 recommendations regarding parameters to be measured; instrument siting, accuracy and maintenance; and data recording, reduction, and recovery. Data recovery exceeded 90 percent. The primary tower had two independent trains of instruments to measure wind speed, wind direction, and assess atmospheric stability. Wind speed and direction were also measured on the back-up tower. Data were evaluated, as specified in plant procedures, for consistency and to assure that the data appeared reasonable with respect to local conditions. Instruments were calibrated annually rather than semi-annually as recommended by RG 1.23. The towers and instruments were checked on a monthly basis to ensure that the instruments were functioning as expected and to identify problems. The licensee noted that calibration histories showed that the instruments were routinely within tolerance specifications. The NRC staff's assessment of the meteorological measurements is provided in Section 4.3 below.

The licensee calculated control room air intake χ/Q values using the 1998-2002 onsite meteorological data and the ARCON96 and PAVAN computer codes for two postulated release locations, a ground level release from the reactor building vent and an elevated release from the off-gas stack. ARCON96 (see NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") implements guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." PAVAN (see NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Plants") implements guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." Specific areas of note are as follows:

- The licensee generated χ/Q values for a postulated elevated release from the 100-meter-tall off-gas stack using guidance in RG 1.194 that states that comparative calculations should be made using both the PAVAN and ARCON96 computer codes. Wind measurements in the form of joint frequency distribution data at the 100-meter level were input into the PAVAN calculations. The licensee assumed fumigation during the first half hour of the postulated accident when using the PAVAN computer code. Wind measurements in the form of hourly meteorological data at the 43-meter level were input into the ARCON96 calculations.

- The postulated release from the reactor building vent was modeled as a ground level release. Consideration was given to other possible release scenarios, including releases from other penetrations, but the licensee determined that dispersal from the reactor building vent was the most limiting case for the FHA. The licensee made calculations for two "taut string" distances as described in RG 1.194 and the χ/Q value for the more limiting case was selected for comparison with the χ/Q value calculated for the release from the off-gas stack.

- The licensee compared the reactor building vent and off-gas stack χ/Q values and found the reactor building vent χ/Q value to be more limiting. Consequently, the reactor building vent χ/Q value was used to model all release scenarios for the control room FHA dose assessments.

The NRC staff's assessment of the licensee's control room atmospheric dispersion analysis is provided in Section 3.2.3 below.

The licensee calculated EAB and LPZ χ/Q values for two postulated release pathways, a ground level release from the reactor building vent, and an elevated release from the off-gas stack. Specific areas of note are as follows:

- Direction-dependent χ/Q values were calculated using the actual EAB and LPZ distances. To calculate the site limit χ/Q values, the licensee also assumed a circular EAB distance of 500 meters, which is the shortest distance in any direction to the EAB.

This resulted in a more limiting estimate than using the actual EAB distances. Since the actual LPZ distance does not vary by direction, a similar assumption was not made for the LPZ calculations. For both the EAB and LPZ assessments, as recommended in RG 1.145, the licensee compared the highest directional χ/Q value with the site limit χ/Q value to identify the higher of the two values for use in its dose assessment.

- The licensee used wind measurements at the 100-meter level to calculate χ/Q values for postulated releases from the off-gas stack. Fumigation was assumed to occur during the first half hour of the two-hour release.

- For the release from the reactor building vent, the licensee initially used wind measurements from the 43-meter level, which is the height of the reactor building vent, and extrapolated the measurements to the 10-meter level. However, typically, 10-meter level wind measurements are used for ground level releases. In the February 28, 2005, letter the licensee provided revised χ/Q values based upon joint frequency distribution data from the 10-meter level and noted that these values were only slightly higher than those based upon wind data at the 43-meter level.

- The licensee compared the reactor building vent and off-gas stack χ/Q values and found the reactor building vent χ/Q value to be more limiting. Consequently, the reactor building vent χ/Q values were used to model all release scenarios for the EAB and LPZ FHA dose assessments.

NRC staff assessment of the licensee's EAB and LPZ atmospheric dispersion analysis is provided in Section 3.2.3 below.

3.1.3 Control Room Mode of Operation

The applicable modes of operation for the control room heating and ventilation - emergency filter treatment system for the fuel handling accident are the normal mode and pressurization mode. The combined inleakage/makeup flows for these modes range from about 280 to 1200 cubic feet per minute (cfm). The licensee assumed that when the fuel handling accident occurred, the control room emergency filter train (EFT) system was not operating and was not initiated even after the accident had occurred. In support of this application for amendment, the licensee submitted a number of analyses for the control room operators' dose, which assumed combined inleakage/makeup flows ranging from 300 to 8440 standard cubic feet per minute. The analysis which the licensee presented as the limiting case assumed 7440 cfm of makeup flow and 1000 cfm of unfiltered inleakage into the control room envelope (CRE).

The value of 7440 cfm was based upon the maximum capacity of one control room ventilation system fan with the outside air blanking plate removed. This is not the normal mode of operation for the control room ventilation system. In the normal mode, there is no outside air supplied to the control room. None of this air is filtered or adsorbed. In the normal mode of operation control room EFT trains would be in standby. There would be no forced makeup flow to balance the forced exhaust flows. The CRE would generally be at a negative pressure with respect to adjacent areas. With control room air being recirculated in this operating mode, makeup air would be provided to the CRE on an as-needed basis through the operation of one of the control room EFT trains.

The licensee conducted American Society for Testing and Materials E741 testing of the Monticello CRE in June 2004 to determine its inleakage characteristics. The CRE was tested with both the 'A' and 'B' EFT trains operating and areas adjacent to the CRE pressurized, the 'A' EFT train operating and the areas adjacent to the CRE not pressurized and 'B' control room ventilation (CRV) train operating in the recirculation mode of operation. Of the configurations tested, the latter had the greatest amount of inleakage, 188 ± 9.5 cfm.

3.1.4 Proposed Technical Specification Changes

To support the implementation of the AST for the postulated FHA, the licensee proposed a number of changes to the MNGP TSs. Details of these changes are described and evaluated in Section 3.2.5 below.

3.2 NRC Staff Assessment

The licensee's April 29, 2004, letter presented acceptable results for the consequences of a postulated fuel handling accident based upon the use of AST. These results also used new offsite atmospheric dispersion factors for the EAB and LPZ and a new onsite atmospheric dispersion factor for control room intake. It was the licensee's intent to use the results of the consequences of a fuel handling accident to justify not maintaining secondary containment integrity and not isolating the containment in the event of a fuel handling accident and to not use the SBGT and the control room EFT. However, the NRC determines that It is insufficient to rely solely upon the dose consequences of an FHA for this purpose; it is also necessary that a licensee demonstrate that, with such a proposed operating mode, the facility still meets GDCs 60, 61, and 64 of Appendix A to 10 CFR Part 50 for plants licensed to the GDC or to their equivalent criteria for plants licensed to the principle design criteria. For Monticello, the NRC staff concluded that the appropriate principle design criteria would be Criterion 17, Monitoring Radioactivity Releases (Category B), Criterion 69, Protection Against Radioactivity Release from Spent Fuel and Waste Storage (Category B), and Criterion 70, Control of Release of Radioactivity to the Environment.

The NRC staff's assessment of the acceptability of the proposed amendment is based upon the ability of the Monticello to continue to meet the above noted criteria, the acceptability of the (1) re-calculated atmospheric dispersion factors, (2) consequences of a fuel handling accident, and (3) proposed technical specification changes. The following sections provide the results of the NRC staff's assessment in these areas.

3.2.1 Adherence to Principal Design Criteria 17, 69, and 70

The General Electric Principal Design Criteria (PDC) are the design and licensing basis of MNGP (see the MNGP USAR. Accordingly, the NRC staff expects that the proposed amendment would comply with those PDCs. However, the licensee's original application did not address the licensee's adherence to PDC 17, 69, and 70. Consequently, the NRC staff asked the licensee to address the manner in which effluents would be monitored during fuel handling operations as a result of the proposed change in operations and technical specifications. Specifically, would the licensee's monitoring be consistent with its licensing basis (i.e., PDC 17, 10 CFR Part 20, and Appendix I of 10 CFR Part 50)?

In its November 23, 2004, letter the licensee indicated that radiological effluent controls, including monitoring and surveillance requirements, are contained in the Monticello Offsite Dose Calculation Manual (ODCM). The licensee stated that the ODCM controls implement the requirements of 10 CFR 20, 10 CFR 50.36a, GDC 60 of Appendix A to 10 CFR 50 and are consistent with PDC 17 and the design objectives of Appendix I to 10 CFR 50. The licensee also indicated that the ODCM controls for effluent monitoring and monitoring instrumentation apply at all times. Since the ODCM controls for plant gaseous effluents are applicable at all times, they would also apply during fuel handling operations. The licensee also indicated that the manner in which effluents will be monitored during fuel handling operations, even after issuance of the proposed amendment and the resulting change in operations, will remain unchanged. Wide range gas monitors installed at the plant stack and reactor building ventilation duct stacks will continue to perform effluent monitoring functions.

Based upon the above assessment, the NRC staff concludes that the licensee will continue to meet PDCs 17, 69, and 70, Appendix I of 10 CFR Part 50 and 10 CFR Part 20.

3.2.2 Control Room Mode of Operation

The NRC staff assessed the licensee's assumption for the manner of operation of the control room ventilation system in the event of an FHA. The NRC staff's assessment focused on whether the assumption used in the licensee's dose assessment reflected the manner in which the system would actually be operated. If it did not, then was the assumption used by the licensee more limiting in terms of dose consequences?

The NRC staff concluded that the licensee's assumption for the manner of operation for the control room ventilation system during an FHA was not realistic. It does not appear that the manner of operation resembles, in any fashion, a likely configuration of the control room ventilation system. Specifically, the licensee needs to evaluate the control room ventilation system configured with different makeup and inleakage flow rates.

The NRC staff concluded that the most realistic mode of operation would be with the control room ventilation system operating in the recirculation mode. There would be no fresh air makeup in this mode. The only source of contaminated flow would be that which leaked into the CRE. The NRC staff performed its own assessment with the CRE inleakage at the value measured during the June 2004 E741 test (i.e., 198 cfm) and at 1000 cfm. The latter was a case which was analyzed by the licensee and included in its application for amendment. The licensee did not submit an analysis based upon 198 cfm of unfiltered inleakage. The licensee also included cases with 300 and 500 cfm of unfiltered inleakage. The licensee's results showed the dose to the control room operators increased slightly as inleakage increased from 300 to 1000 cfm. The licensee's calculations showed control room operators' doses, assuming no makeup air, were just slightly less than the dose assumed with 7440 cfm of makeup flow and 1000 cfm of inleakage.

3.2.3 NRC Staff's Atmospheric Dispersion Factor Assessment

The NRC staff performed a quality review of the 1998-2002 ARCON96 hourly meteorological data using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. The NRC staff's examination of the data confirmed that recovery of each parameter was in the upper 90 percentiles each year. With respect to atmospheric stability measurements, the time of occurrence and duration of stable and unstable conditions were consistent with expected meteorological conditions. Stable and neutral conditions were reported to occur at night and unstable and neutral conditions during the day, with neutral or near-neutral conditions predominating during each year. Wind speed, wind direction, and stability class frequency distributions for each measurement channel were reasonably similar from year to year and when comparing measurements at the 10-meter and 43-meter levels. A comparison of joint frequency distributions derived by the NRC staff from the ARCON96 hourly data with the joint frequency distributions developed by the licensee for input into PAVAN code and the 1980 historical data in Chapter 2.3 of the Monticello USAR showed a slightly higher occurrence of light winds in the ARCON96 hourly data. In the February 28, 2005, letter the licensee attributed differences between the 1980 historical data and the 1998-2002 period to differences in sample size, potential changes due to construction and vegetation in the area surrounding the site, and improvements in instrumentation and data recording. The licensee attributed departures between the ARCON96 and PAVAN data files to differences in the data selection process used to create the files.

With regard to control room, EAB, and LPZ χ/Q values, the NRC staff qualitatively reviewed the input data to the ARCON96 and PAVAN computer runs and found them generally consistent with site configuration drawings and NRC staff practice or acceptable for the following reasons. In the control room χ/Q assessment, the licensee's assumption of fumigation for the first half hour of the release from the 100-meter tall off-gas stack to the control room is more limiting than using the non-fumigation 0-2 hour χ/Q value for the entire 2-hour time period as recommended by RG 1.194. Further, while it would have been preferable to use wind data from the 100-meter level, ARCON96 extrapolates wind data to the input height of release. Calculated χ/Q values using the PAVAN code were much more limiting than that calculated using ARCON96 such that, in the NRC staff's judgement, use of extrapolated data does not impact the conclusion that the PAVAN χ/Q values are more limiting. Further, the NRC staff agrees that the ground level reactor building vent χ/Q value used by the licensee in the FHA control room dose assessments is more limiting than the off-gas stack χ/Q values. Although the EAB and LPZ dose assessments were initially based upon ground level release χ/Q calculations using wind measurements from the 43-meter level, the licensee revised the dose calculations to use wind data from the 10-meter level, thus following standard practice which is acceptable.

In summary, the NRC staff reviewed the available information relative to the onsite meteorological measurements program and the resulting ARCON96 and PAVAN meteorological data input files provided by the licensee. On the basis of this review, the NRC staff concludes that the 1998-2002 data provide an acceptable basis for making estimates of ARCON96 χ/Q values for the FHA assessment addressed in this application for license amendment. However, the PAVAN joint frequency distribution data should not be considered acceptable for use in other dose assessments without further review to ensure that light wind speed conditions are adequately considered. The NRC staff reviewed the licensee's assessment of control room, EAB, and LPZ post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. On the basis of this qualitative review and its independent estimates, the NRC staff concluded that the χ/Q values presented in Table 1 are acceptable for use in this FHA dose assessment. These values represent a change from those used in the current Monticello USAR Chapter 14 accident analysis.

3.2.4 Specifics of the Postulated Fuel Handling Accident

The only dose analysis provided by the licensee for a postulated FHA involved fuel which is not "recently" irradiated. Consequently, the NRC staff asked whether the licensee intended to handle fuel which has been "recently" irradiated. In response, the licensee stated that the proposed TS requirements do not permit fuel that has been "recently" irradiated to be handled and that the licensee had no intention to handle recently irradiated fuel. Therefore, an FHA analysis was not performed for this scenario. Based upon this information, the NRC staff concluded that it is not necessary to perform an analysis of the consequences of a postulated FHA involving recently irradiated fuel. The NRC staff also concluded that the MNGP licensing basis did not cover the handling of recently irradiated fuel.

The NRC staff's assessment of the consequences of a postulated FHA also encompassed a determination of the assumption that damage to 125 fuel rods from 8x8 array assemblies is bounding for each operating cycle. The licensee indicated that the validity of assuming 125 damaged fuel rods from an 8x8 array will be re-evaluated as new fuel designs are proposed for use at MNGP. If this re-evaluation shows that the fuel design is no longer valid, then appropriate re-analyses will be performed, as required, in accordance with regulatory requirements. The licensee's response addressed the staff's concern regarding the assumption of 125 damaged rods from 8x8 array assemblies.

In its April 29, 2004, application the licensee stated that it used an RPF of 1.7 in the analysis even though MNGP does not specify an RPF in the TSs or in the Core Operating Limits Report (COLR). The licensee also stated that the value of 1.7 was conservative. The NRC staff asked the licensee what core parameter(s) are monitored to ensure that the FHA analysis remains relevant and how these parameter(s) are used to conclude that the core remains within the assumed 1.7 value for RPF. The licensee was also asked if it is determined that a value greater than 1.7 should be used, will Monticello be re-submitting an FHA analysis for NRC staff review and approval. In response to these two questions, the licensee stated that while the RPF is a core design parameter, the RPF is not directly monitored during reactor operation. By maintaining reactor operation within the core operating limits, the licensee indirectly assures compliance with the RPF design criterion. The licensee has established core operating limits such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, emergency core cooling system 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 licensee committed to revising the core design and reload analysis procedures and design documents to clearly specify the connection between RPF as an AST FHA analysis assumption and reload design. The licensee considers the specific RPF value of 1.7 as conservative based on conceptual core designs from the Nuclear Management Company, LLC, Nuclear Analysis Department and the review of previous calculation assumptions. The licensee indicated that a change in RPF for an FHA resulting in more than a minimal increase in radiological consequences would require approval via a license amendment. The NRC staff has no more concern regarding the RPF.

The NRC staff performed an independent calculation of the offsite and onsite consequences of an FHA. The assumptions which were used by the NRC staff in its calculations are contained in Table 2. As noted in Section 3.2.2 above, the NRC staff performed calculations to determine doses to control room operators with the control room ventilation system in the recirculation mode of operation and inleakage values of 198 and 1000 cfm. An analysis was also performed with makeup flow to the control room at 7440 cfm and inleakage at 1000 cfm. The NRC staff's calculations determined that the limiting case was with no makeup flow and an unfiltered inleakage of 1000 cfm. For this case, the NRC staff calculated a dose of approximately 4.3 rem TEDE. A similar case with 198 cfm of unfiltered inleakage resulted in a slightly lower dose (slightly greater than 4.2 rem to the thyroid) to the control room operators. The lowest dose calculated by the NRC staff was with 7440 cfm of makeup and 1000 cfm of unfiltered inleakage. For that case the dose was approximately 4.1 rem TEDE. Therefore, the NRC staff did not conclude that the limiting case was the case proposed by the licensee. The licensee should re-assess its limiting cases in light of the NRC staff's assessment. Nevertheless, for the limiting case determined by the NRC staff, acceptable dose consequences were obtained. The results of the NRC staff's calculations are presented in Table 3.

3.2.5 Technical Specification Changes

To support implementation of the AST for the postulated FHA, the licensee proposed a number of TS changes. The NRC staff had reviewed these TS changes and found that they reflect the implementation of AST for the FHA as evaluated above in Sections 3.2.1 thru 3.2.4. These TS changes are found acceptable by the NRC staff; details are described below:

<u>Table 3.2.4 - Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment [SBGT] Initiation</u>

The licensee proposed changes to allow the applicable modes or operating conditions for each instrument function to be specified individually. Currently, the table is sorted by four sets of instruments which initiate the reactor building ventilation and SBGT systems. This is based on the analysis result of the postulated FHA using the AST has demonstrated that initiation of the SBGT is only required during operations with the potential for draining the reactor vessel and during the movement of recently irradiated fuel assemblies in secondary containment. The licensee's results showed that a 24-hour decay period was sufficient such that the SBGT, the control room EFT system and secondary containment integrity are not required if fuel has decayed for 24 hours or longer.

The licensee proposed to implement these system conditions with the following changes to Table 3.2.4:

a. Addition of a column entitled, "Applicable Modes or Other Specified Conditions for Which the Function Must be Operable or Operating#" - This new column allowa the applicable modes or operating conditions to be specified individually for each instrument function, and clarifies the applicability requirements.

b. Addition of a footnote to explain "#" in the new column - The footnote specifies other conditions for which the function must be operable or operating. These conditions include operation with the potential for draining the reactor vessel, and during movement of recently irradiated fuel in secondary containment. These conditions were consistent with the applicability paragraphs and action statement paragraphs being added to the SBGT system technical specification (Technical Specification 3.7.B.1).

c. Specifying in the new column the conditions of Hot Shutdown, StartUp and Run for the table functions designated as Low Low Reactor Water Level, High Drywell Pressure, Reactor Building Plenum Radiation Monitors, and Refueling Floor Radiation Monitors. For these functions, the Hot Shutdown, Startup and Run modes were specified because these are times of operation when considerable energy exists in the reactor coolant system (RCS). Therefore, if an

reactor coolant system pipe break would occur during one of these modes, there is a probability of a significant release of radioactive steam and gases. Refuel and cold shutdown modes were not specified because the probability of a pipe break during these modes would be low, and the consequences would be low due to the reactor coolant system temperature and pressure limitations associated with these modes.

d. The Low Low Reactor Water Level, Reactor Building Plenum Radiation Monitors, and Refueling Floor Radiation Monitors functions are qualified with a note (a). This note specifies that these functions are required to be operable during operations with the potential for the draining of the reactor vessel. During these operations the capability to isolate the potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded should core damage occur.

e. The Reactor Building Plenum Radiation Monitors and the Refueling Floor Radiation Monitors function is qualified with a note (b). This note specifies that these instruments are required to be operable during the movement of recently irradiated fuel assemblies in the secondary containment because the capability of detecting radiation releases due to fuel failures from a dropped fuel assembly must be provided to ensure that offsite dose limits are not exceeded. Following 24 hours of decay, this isolation capability would not be required.

<u>Specification 3.3 - Control Rod Systems</u>

Section 3.3.G currently provides an action to be taken when the requirements for shutdown margin are not met, stating:

> If Specifications 3.3.A through 3.3.D above are not met, an orderly shutdown shall be initiated and have reactor in the cold shutdown condition within 24 hours.

The licensee proposed to change Section 3.3.G by replacing it with two subparagraphs, one to address action in non-refueling mode and one to address action in the refueling mode. The proposed subparagraphs read:

1. If Specifications 3.3.A (except when the reactor mode switch is in the Refuel position) through 3.3.D above are not met, an orderly shutdown shall be initiated and the reactor placed in the cold shutdown condition within 24 hours.

2. If Specification 3.3.A is not met when the reactor mode switch is in the Refuel position, immediately suspend core alterations except for fuel assembly removal and immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

In the April 29, 2004, letter the licensee further clarified these subparagraphs by stating that shutdown margin is not met during refueling the operator must immediately suspend operations that could reduce shutdown margin. Inserting control rods or removing fuel from the core will reduce the total reactivity and are thus excluded from the suspended actions.

<u>Specification 3.7 - Containment Systems</u>

The licensee proposed to add action statements 3.7.B.1.c and 3.7.B.1.d. to Section 3.7.B.1, regarding the Standby Gas Treatment System. These action statements define the actions to be taken when one or both trains of the SBGT system are inoperable during the movement of recently irradiated fuel in secondary containment or during operations with the potential for draining the reactor vessel. Action statement 3.7.B.1.c addresses one inoperable SBGT train. ActionProposed Action 3.7.B.1.d addresses two inoperable trains.

In 3.7.B.1.c, the licensee proposed that if one SBGT system train remained inoperable after seven days, then either the operable train must be placed in operation or the movement of recently irradiated fuel assemblies in secondary containment must immediately cease as well as any operations with the potential to drain the reactor vessel. In 3.7.B.1.d, if both trains of the SBGT system are inoperable, then the movement of recently irradiated fuel assemblies in secondary containment and operations with the potential for draining the reactor vessel must immediately be suspended. The condition regarding "operations with the potential to drain the reactor vessel" was added for consistency with current industry guidance.

With the proposed additions of 3.7.B.1.c and 3.7.B.1.d, the SBGT system would no longer be required to be operable if the fuel had decayed for longer than 24 hours.

The addition of action statements 3.7.B.1.c and 3.7.B.1.d allows for the removal of the term "and fuel handling" from action statement 3.7.B.1.a since the proposed action statements cover more definitive actions to be taken during fuel handling operations.

Speciofication 3.7 - Containment Systems

Section 3.7.C establishes requirements for the secondary containment. Subsections 3.7.C.1 and 3.7.C.2 define applicability of this limiting condition for operation (LCO). Subsections 3.7.C.3 and 3.7.C.4 provide actions to be taken when the LCO cannot be met.

The licensee proposed changes to the applicability portions of the LCO, deleting the current applicability paragraph 3.7.C.2.c (due to redundant requirement already in paragraph 3.7.C.4), and dividing the applicability paragraph 3.7.C.2.d into two separate paragraphs, 3.7.C.2.c and 3.7.C.2.d. The new paragraph 3.7.C.2.c would pertain only to the fuel cask while 3.7.C.2.d would apply to movement of recently irradiated fuel. The term "recently" was added to "irradiated fuel" in the new applicability paragraph.

With 3.7.C.2.d, the absence of secondary containment would be allowed if recently irradiated fuel is not being moved in the secondary containment. A new applicability Item 3.7.C.2.e is added to require the establishment of secondary containment during operations with the potential for draining the reactor vessel.

Section 3.7.C directs compliance with Specification 3.3.A via Specifications 3.7.C.2.a and 3.7.C.2.c (which is being deleted as explained above) and provides the action to take if compliance cannot be maintained, since individual action statement paragraphs are not provided under Specification 3.3.A.1, “Reactivity Limitation, Reactivity Margin - Core Loading.” Since the MNGP TSs are presented in a manner different than the presentation of TSs in Revision 3 of NUREG-1433, “Standard Technical Specifications, General Electric Plants, BWR/4,” the licensee proposed actions pertaining to the movement of recently irradiated fuel and operations with the potential for draining the reactor vessel which were separate from those required for shutdown margin considerations. In the April 29, 2004, application the proposed actions were embodied in a new action statement 3.7.C.5. In the April 12, 2005 letter, the licensee deleted the request for the new action statement 3.7.C.5 and, in its stead, proposed a new action statement in Section 3.3.G. See above for the evaluation regarding Section 3.3.G.

The licensee proposed the removal of the term "alterations of the reactor core" from action statement 3.7.C.4, and divide this statement into sub paragraphs a and b to clarify the required actions based on the operational mode. The licensee proposed to add the word "recently" before "irradiated fuel" in action statement 3.7.C.4 to clarify that secondary containment is not required during the handling of irradiated fuel that has decayed for longer than 24 hours. The licensee also proposed to revise action statement 3.7.C.4 to require establishment of secondary containment integrity during operations with the potential for draining the reactor vessel.

Specification 3.10 - Refueling

Currently Section 3.10.C states:

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level of greater [than] or equal to 33 feet.

The licensee proposed to revise Section 3.10.C to read as follows:

C. Spent Fuel Storage Pool Water Level

During movement of irradiated fuel assemblies, the spent fuel storage pool water level shall be maintained $37 ft above the bottom of the spent fuel storage pool.

If the spent fuel storage pool water level is made or found not to be within limits, immediately suspend movement of irradiated fuel assemblies.

The licensee also proposed to revise Surveillance Requirement 4.10.C to read as follows:

C. Spent Fuel Storage Pool Water Level

Verify that the spent fuel storage pool water level is $ 37 ft above the bottom of the spent fuel storage pool:

1. Once every 24 hours, during movement of irradiated fuel assemblies, or

2. Once every 7 days, when irradiated fuel assemblies are stored in the spent fuel storage pool.

The purpose of this change is to assure sufficient water depth to validate the assumptions made in the FHA analysis with respect to decontamination factor.

Specification 3.17 - Control Room Habitability

The licensee proposed to modify the CRV system specification applicability paragraph 3.17.A.1, and action statements 3.17.A.2.c and 3.17.C.3.c to remove the term "core alterations." The licensee also proposed that action statement 3.17.C.3.c be revised to require that it be entered immediately when both CRV trains are inoperable.

The licensee proposed to modify the control room EFT system specification applicability paragraph 3.17.B.1, and action statements 3.17.B.1.c and 3.17.B.1.d to remove the term "core alterations." The licensee also proposed to add the word "recently" before the term "irradiated fuel assemblies" to paragraph 3.17.B.1 and action statement paragraphs 3.17.B.1.c and 3.17.B.1.d; this modification clarifies that these specifications do not apply during the handling of irradiated fuel assemblies that have decayed for longer than 24 hours.

Technical Specification Bases

The licensee proposed changes to the TS Bases associated with the TS sections evaluated above. The TS Bases are not part of the TS (see 10 CFR §50.36(a)) but currently exist in the same book holding the TS. The NRC staff reviewed the licensee's proposed TS Bases changes and found that they reflect the proposed implementation of AST for the FHA as evaluated above in Sections 3.2.1 thru 3.2.4.

Table 1 - Monticello Atmospheric Dispersion Factors

Time	Source	Receptor	χ/Q values (s/m^3)
0 - 2 hours	Reactor building vent	Exclusion area boundary	7.51×10^{-4}
0 - 2 hours	Reactor building vent	Low-population zone	1.53×10^{-4}
0 - 2 hours	Reactor building vent	Control room	2.48×10^{-3}

Table 2 - NRC Staff Assumptions for Monticello Fuel Handling Accident

Parameter	Value
Power (megawatts thermal)	1.918
Fuel Burnup (gigawatt days per metric ton)	60
Radial Peaking Factor	1.7
Number of Damaged Fuel Rods	125
Total Number of Fuel Rods in the Core	29,040
Fraction of Fission Product Inventory in the Gap	^{131}I = 0.08
	^{85}Kr = 0.10
	Other Halogens & Noble Gases = 0.05
Decay Time (hrs)	24
Reactor Cavity & Spent Fuel Pool Water Depth (ft)	23
Reactor Cavity & Spent Fuel Pool DF	200
Containment ESF Filter System Efficiencies (%)	0
Chemical Form of Iodine in the Water	Particulate = 0
	Organic = 0.0015
	Elemental = 0.9985
Chemical Form of Iodine in Release to Environment	Particulate = 0
	Organic = 0.43
	Elemental = 0.57
Release Period (hrs)	2
Release Location	Reactor Bld. Vent
Control Room Emergency Ventilation System (CREVS) Initiated	No
CREVS Intake Flow Rate	NA
Control Room Normal Ventilation System Flowrate (cfm)	7400
CRE Inleakage During CREVS Operation (cfm)	NA
CRE Inleakage During Control Room Normal Ventilation System Operation (cfm)	1000
CREVS Filter & Absorber Efficiencies (%)	0

Table 3 - Onsite and Offsite Doses Resulting from a Fuel Handling Accident

Accident	EAB	LPZ	Control Room Operators
Fuel handling accident	1.8	0.36	4.3
Regulatory Limit	6.3	6.3	5

Mail Envelope Properties (4411DBD1.72D : 12 : 35330)

Subject:	Monticello Draft Safety Evaluation for Proposed FHA AST Amendment (TAC MC7596)
Creation Date:	3/10/06 3:04PM
From:	Peter Tam
Created By:	PST@nrc.gov

Recipients	**Action**	**Date & Time**
nmcco.com	Transferred	03/10/06 3:05 PM
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MESSAGE	1545	03/10/06 03:04PM
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Options

Auto Delete:	No
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Notify Recipients:	Yes
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Return Notification:	None
Concealed Subject:	No
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Status Tracking:	Delivered & Opened