NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

December 6, 2002

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

10 CFR Part 50
Section 50.90

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request dated December 6, 2002
Revised Analyses of Long-Term Containment Response and
Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps

Attached is a request for amendment of the Monticello Operating License reflecting updates to the design basis loss of coolant accident (DBA-LOCA) containment response and containment overpressure for adequate net positive suction head (NPSH) for low pressure emergency core cooling (ECCS) pumps analyses.

The revised analyses are updates to analyses previously reviewed and approved by the NRC Staff in the following license amendments:

a. Monticello Nuclear Generating Plant – Issuance of Amendment Re: Updated Analysis of DBA Containment Temperature and Pressure Response and Reliance on Containment Pressure to Compensate for Potential Deficiency in NPSH for ECCS Pumps During DBA (TAC NO. M97781), Amendment No. 98 to Facility Operating License No. DPR-22, July 25, 1997

b. Monticello Nuclear Generating Plant – Issuance of Amendment Re: Power Uprate Program (TAC NO. M96238), Amendment No. 102 to Facility Operating License No. DPR-22, September 16, 1998

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The revised containment analyses incorporate updated inputs and alternative scenarios that include:

- Consideration of small breaks and reactor isolation events on long-term containment response
- Reactor decay heat revised to include 2σ uncertainty and additional terms described in General Electric Service Information Letter 636
- Updated K value for Residual Heat Removal (RHR) heat exchangers
- Effects of increased service water temperature
- Effects of increased wetwell attached piping design temperature
- Effect of delay in realigning RHR injection flow to containment cooling with flow through the RHR heat exchangers

The updated containment analyses involve differences in methods of evaluation currently described in the Monticello Updated Safety Analysis Report (USAR) and previously approved by the NRC. Paragraph c(2)(viii) of 10 CFR 50.59 requires changes of this type to be approved as a license amendment pursuant to the requirements of 10 CFR 50.90.

Exhibit A contains a description of the proposed license amendment, the reasons for requesting the amendment, a Safety Evaluation, a Significant Hazards Consideration Evaluation, and an Environmental Assessment. Exhibit B contains the current USAR pages annotated with the proposed changes. Exhibits C and E contain proprietary and non-proprietary versions, respectively, of General Electric report GE-NE-0000-0002-8817-01, Revision 1, September 2002, "Monticello Nuclear Generating Plant Long-term Containment Analysis which supports the requested amendment. Exhibit D is a GE affidavit requesting Exhibit C be withheld from public disclosure in accordance with 10 CFR 2.790-(a)(4). Exhibit F is a summary of Monticello Calculation CA-01-177, Revision 1, "Determination of Containment Overpressure Required For Adequate NPSH For Low Pressure ECCS Pumps Updated For Suction Strainer Debris Loading," which supports the requested amendment.

In addition to the proposed changes to the USAR, the revised long-term wetwell temperature response will be used to update the plant Environmental Qualification Program central file as necessary.

Please contact Mr. Doug Neve, Monticello Licensing Manager, at 763-295-1353 if you require additional information related to this request. This submittal contains no new NRC commitments, nor does it modify any prior commitments.

Proprietary Information

As noted above, Exhibit C contains information proprietary to the General Electric (GE) Company. GE requests that this document be withheld from publish disclosure in accordance with 10 CFR 2.790-(a)(4). An affidavit supporting this request is provided in Exhibit D.

By
Jeff S. Forbes
Site Vice President
Monticello Nuclear Generating Plant

Subscribed to and sworn before me this 6th day of December, 2002

KATHRYN I. KLEINE
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2005

Kathryn I Kleine
Notary

Attachments:

Exhibit A – Evaluation of Proposed Amendment

Exhibit B – USAR pages annotated with the proposed changes.

Exhibit C – General Electric report GE-NE-0000-0002-8817-01, Revision 1, September 2002, "Monticello Nuclear Generating Plant Long-term Containment Analysis (GE Proprietary Information).

Exhibit D – GE affidavit requesting Exhibit C be withheld from publish disclosure in accordance with 10 CFR 2.790-(a)(4).

Exhibit E – Non-proprietary version of Exhibit C

Exhibit F – Determination of Containment Overpressure Required for Adequate NPSH For Low Pressure ECCS Pumps Updated For Suction Strainer Debris Loading

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
Minnesota Department of Commerce
J Silberg, Esq.

Pursuant to 10 CFR Part 50, Section 50.90, Nuclear Management Company, LLC (NMC), hereby proposes the following amendment to the design basis containment response analyses for the Monticello Nuclear Generating Plant:

I. Proposed Amendment

I.1 Revise the Design Basis Loss of Coolant Accident (DBA-LOCA) Containment Response Analyses to Use Updated Plant Data and Additional Scenarios

A DBA-LOCA containment response analysis was performed in support of the 1998 Monticello power rerate project (Reference 1). The power rerate containment analysis was performed for a reactor thermal power of 1880 MWt. The long-term analysis used ANS 5.1 nominal decay values, assuming a service water temperature of 90°F and a Residual Heat Removal (RHR) heat exchanger K-value of 143.1 Btu/sec-°F. All applicable containment requirements associated with the responses analyzed were met at power rerate conditions. A ten-minute delay in initiating containment cooling following the accident was assumed in these analyses.

Reference 2 supplemented the results of the Reference 1 analysis, and provided DBA-LOCA and containment response evaluations at power rerate conditions for use in the analysis of net positive suction head (NPSH) requirements for the low pressure emergency core cooling system (ECCS) pumps.

The General Electric Company (GE) has updated the containment response analyses contained in References 1 and 2. The updated analyses are contained in Exhibits C (proprietary version) and E (non-proprietary version). It is requested that the updated methods and key results of these analyses be reviewed by the NRC Staff and approved for incorporation in the Monticello design basis and Updated Safety Analysis Report (USAR).

Proposed updates to appropriate sections of the USAR are provided in Exhibit B.

I.2 Revise the Analysis of Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps and Overpressure Credit Allowed By the Commission

Calculations incorporating the revised containment responses provided in Exhibits C and E, along with updates to other inputs as appropriate, were also completed to determine the impact on ECCS pump NPSH requirements for bounding accident scenarios.

It is requested that the updated NPSH analyses be reviewed by the NRC Staff and approved for incorporation in the Monticello design basis and USAR. A minor change in the long-term containment pressure credit for ECCS pump NPSH is requested.

Proposed updates to appropriate sections of the USAR are provided in Exhibit B.

II. Reasons for Proposed Amendment

II.1 Revise the DBA-LOCA Containment Response Analyses to Use Updated Plant Data and Additional Scenarios

Background

The reactor isolation event with the High Pressure Coolant Injection (HPCI) unavailable was recently determined to be the most limiting event with respect to the maximum time required to establish containment cooling. The maximum time required to initiate containment cooling for this event was calculated to be 48.6 minutes from initiation based on the General Electric SAFER/GESTR analysis of the event. This time consists of the time to reach the Low Pressure Coolant Inject (LPCI) injection valve permissive pressure, plus the five-minute LPCI interlock, plus the 400 seconds to complete system configuration line-up.

To determine the impact of the 48.6-minute delay in initiation of containment cooling on peak wetwell water temperature, the DBA-LOCA containment response was evaluated using the 48.6-minute delay, as compared with a 10-minute delay assumed for the Reference 1 power rerate analysis. This analysis was considered very conservative because the DBA-LOCA event was analyzed using the maximum time delay in initiation of containment cooling obtained for the isolation event.

The DBA-LOCA analysis with a 48.6-minute containment cooling initiation time resulted in peak wetwell pool temperature above 195°F (the original design temperature for piping system attached to the wetwell), when the service water (SW) temperature was assumed to be 90°F. With 85°F service water temperature, however, the same case resulted in peak wetwell pool temperature below 195°F. Consequently, a reduction in the maximum allowable ultimate heat sink temperature (UHS) from 90°F to 85°F was administratively imposed at the Monticello plant pending resolution of this issue.

Updated Containment Response Analyses for Issue Resolution

The revised containment response analyses in Exhibit C were completed by General Electric to justify restoration of the maximum acceptable UHS temperature to 90°F. A secondary purpose of the containment reanalyses were to incorporate other updates to analysis inputs and address additional scenarios and issues, including:

a. Revised decay heat uncertainty factor. The containment analyses of References 1 and 2 were calculated at 102% of 1880 MWt using a decay heat curve based on the nominal ANSI/ANS 5.1-1979 decay heat with no uncertainty adders.

However, the NRC currently requires a two-sigma adder to the ANS 5.1-1979 decay heat in containment analyses. The decay power time-history used in the Reference 1 analysis, using the nominal ANS 5.1-1979 decay heat at 102% of 1880 MWt, is roughly equivalent to the power corresponding to ANS 5.1-1979 with a 2-sigma adder at 102% of 1775 MWt. This provided justification at the time for using the results of the Reference 1 analysis to support a power rerate to 1775 MWt. However, it was understood that future analyses will include generation of a Monticello plant specific ANS 5.1-1979 + two sigma decay heat curve.

b. Consideration of additional contribution to reactor decay heat. In June, 2001, GE issued Service Information Letter (SIL) 636 (Reference 3) which describes a potential non-conservatism in decay heat calculations based on the ANSI/ANS-5.1-1979 standard. The non-conservatism results from failure to account for the cumulative effects of actinides other than ^{239}U and ^{239}Np, as well as activation products in structural materials. This non-conservatism can potentially impact the results of the Reference 1 and 2 analyses. Decay heat values that are calculated with additional terms, consistent with SIL 636, are used in the updated analyses.

c. Updated K-value of 147 Btu/sec-°F for the RHR heat exchanger. Long-term containment analyses were performed with an updated K-value of 147 Btu/sec-°F for the RHR heat exchangers as compared with the previous K-value of 143.1 Btu/sec-°F under accident conditions used in Reference 1. The value of 143.1 Btu/sec-°F was based on the original vendor heat exchanger design calculations. The updated value is based on state of the art heat exchanger performance predictions under accident conditions. The updated calculation was performed by Senior Engineering Company, the successor to the original heat exchanger vendor, using the computer code HTRI (Heat Transfer Research Institute).

   The calculation was independently verified using the PROTO-HX code developed by the Proto-Power Company. Calculation files are available on site for NRC Staff review.

d. Delayed cooling time for small breaks. SAFER/GESTR analyses to determine the Automatic Depressurization System (ADS) activation and containment cooling initiation times for a reactor isolation event, 0.01, and 0.1-ft^2 liquid line breaks at 102% of 1775 MWt core power were performed. The power level and decay heat values used in these analyses incorporated the SIL 636 and 2-sigma uncertainty additions discussed above. Long-term containment analyses for the isolation event and small breaks, using the ADS activation and containment cooling initiation times obtained from the SAFER/GESTR analyses, were performed using the GE SHEX code.

e. Long-term analysis of DBA-LOCA. An updated long-term containment analysis for the DBA-LOCA with direct wetwell pool cooling, using input values that maximize the containment pressure response, was performed.

f. Impact of low-pressure coolant injection through the RHR heat exchanger. The impact on the long-term containment response with a delay in realigning an RHR pump from injection to containment cooling was evaluated.

g. Evaluate increased wetwell attached piping design temperature. The maximum acceptable service water temperature that keeps peak wetwell water temperature below 196.7°F was determined by GE. A temperature of 196.7°F has recently been determined to be an acceptable replacement for the original 195°F design temperature limit for piping attached to the wetwell. Analyses using this service water temperature were performed for both: a) direct pool cooling with input values maximizing the pressure response, and b) containment spray cooling with input values minimizing the pressure response for input to a future NPSH evaluation.

h. Input for NPSH evaluation. Short-term (<600 seconds) and long-term (up to 12 days) containment analyses for the DBA-LOCA, using input values minimizing the containment pressure response, were performed. A service water temperature of 90 °F with an updated K-value of 147 Btu/sec-°F for the RHR heat exchangers were used. The analysis results were used for input to a separate evaluation of NPSH adequacy for low-pressure ECCS pumps taking suction from the wetwell (Exhibit F).

The GE report provided in Exhibit C describes the methodology, assumptions, and results of these analyses.

Exhibit B includes proposed revised pages for the Monticello USAR incorporating the results of these analyses.

II.2 Revise the Analysis of Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps and Overpressure Credit Allowed By the Commission

At Monticello, under certain conditions credit is required for containment pressure to assure adequate NPSH is available for operation of RHR and Core Spray pumps following a loss of coolant accident. Figures 5.2-15a (short term < 600 sec) and Figures 5.2-15b (long term ≥ 600 sec) of the USAR show the amount of containment pressure previously approved for this purpose by the NRC (Reference 4).

A small change in the shape of the plotted line representing the NRC approved containment pressure for NPSH purposes is being proposed in this License Amendment Request.

Bounding containment response cases for NPSH purposes are:

Short Term (< 600 sec)	-	DBA LOCA with LPCI Loop Select Logic Failure
Long Term (≥ 600 sec)	-	DBA LOCA with containment spray cooling and loss of offsite power and failure of one emergency diesel generator

Wetwell pressure and water temperature were recalculated by GE using the updated decay heat, updated RHR heat exchanger K values, and other updated inputs described above. Results are presented in Appendices C and D of Exhibit C.

NPSH requirements for RHR and core spray pumps were recalculated using the new containment temperature responses calculated by GE. The USAR design bases assumptions for head loss due to LOCA generated debris collecting on the ECCS suction strainers were used in these calculations. The methodology and results of these calculations are presented in Exhibit F, Cases 1, 2, and 3.

Additional cases presented in Exhibit F confirm the adequacy of available pump NPSH for a medium LOCA, shutdown from the Alternate Shutdown Panel (Appendix R requirement), and reactor isolation conditions.

Refer to Exhibit F for a summary of the calculations and results associated with the NPSH analyses.

The new long term DBA LOCA wetwell pressure response with containment spray cooling is reduced by a small amount in the new GE analysis during the period from 2000 to 4000 seconds. During this interval wetwell pressure falls below the previously approved NRC limit graph showing containment pressure NPSH credit. To accommodate this change, a reduction in the approved NPSH credit from 18.26 to 17.51 psia is requested in this time interval.

A conservative change in assumptions related to thermal equilibrium between wetwell air and water volumes in the GE model is the primary reason for the pressure reduction in the interval from 2000 to 4000 seconds. Additionally, the pressure difference required between the wetwell and drywell for vacuum breakers to fully open was conservatively reduced from 0.5 to 0.0 psid in the current NPSH analyses.

The proposed change in the previously approved NRC containment pressure NPSH credit graph is shown in Exhibit B, USAR Figure 5.2-15a.

III. Safety Evaluation

NMC has proposed changes to the Monticello design basis containment response analysis methodology and resulting containment response in the following areas:

- Incorporation of 2σ uncertainty in reactor decay heat to conform to NRC Staff requirements
- Increase in reactor decay heat to account for additional nuclides in accordance with GE SIL 636, Revision 1
- Update RHR Heat Exchanger K-value to 147 Btu/sec-°F resulting from updated analysis methods
- Analysis of increased times for initiating long-term containment cooling for reactor isolation and small break LOCA conditions on long-term containment response
- Update of DBA-LOCA long term containment analysis with new decay heat and RHR heat exchanger K-value parameters
- Analysis of impact on containment response with low pressure coolant injection through the RHR heat exchangers
- Analysis of effect of increased wetwell attached piping design temperature on allowable service water temperature
- Update of containment pressure requirements for adequate RHR and core spray pump NPSH.

Reactor coolant system response to reactor isolation and loss of coolant accidents has been evaluated by GE using SAFER/GESTER analysis. Containment response has been evaluated by GE using the SHEX code. Use of these analytical methods for Monticello is within the current licensing analysis basis in References 1, 2, and 4, and complies with NRC requirements.

A benchmarking analysis is included in Appendix B of Exhibit C. This analysis shows that the update of plant data and modeling results in a slightly more conservative prediction of peak suppression pool temperature.

The GE analyses include cases that minimize the calculated wetwell pressure. The results of these analysis have been combined with ECCS suction strainer design basis clogging assumptions to determine containment pressure requirements for adequate ECCS pump NPSH. Similar analyses have been reviewed and approved earlier (Reference 2) by the NRC Staff.

A slight reduction in the graph showing approved NRC credit for containment over-pressure for NPSH purposes is requested. This change is small and reflects additional conservatism included in the reanalysis of wet well pressure long term.

The results of the updated containment response and ECCS pump NPSH analyses will be included in the Monticello USAR as shown in Exhibit B. The end result of these USAR changes will be a more accurate description of containment performance under design basis accident conditions.

IV. Significant Hazards Consideration Evaluation

The proposed changes have been evaluated to determine whether they constitute a significant hazards consideration as required by 10 CFR Part 50, Section 50.91 using the standards provided in Section 50.92. This analysis is provided below:

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The amendment involves changes in the Monticello design basis containment response analysis methodology and resulting containment response as described in the Monticello USAR. Changes are proposed in the following areas:

a. Incorporation of 2σ uncertainty in reactor decay heat to conform to NRC Staff requirements

b. Increase in reactor decay heat to account for additional nuclides in accordance with GE SIL 636, Revision 1

c. Update RHR Heat Exchanger K-value to 147 Btu/sec-^{o}F resulting from updated analysis methods

d. Analysis of increased times for initiating long-term containment cooling for reactor isolation and small break LOCA conditions on long-term containment response

e. Update of DBA-LOCA long term containment analysis with new decay heat and RHR heat exchanger K-value parameters

f. Analysis of impact on containment response with low pressure coolant injection through the RHR heat exchangers

g. Analysis of effect of increased wetwell attached piping design temperature on allowable service water temperature

h. Update of containment pressure requirements for adequate RHR and core spray pump NPSH.

These changes update parameters used in the Monticello containment safety analyses and expand the range and scope of the analyses. The results of the updated analyses will be included in the Monticello USAR. This will result in a more accurate description of containment performance under design basis accident conditions.

The methods used in the updated analyses have, in large part, been previously reviewed and accepted by the NRC Staff. The updated analyses affect only the evaluation of previously reviewed accidents. No plant structure, system, or component (SSC) is physically affected by the updated and expanded analyses. No method of operation of any plant SSC is affected. Therefore there is no significant increase in the probability or consequence of a previously evaluated accident.

2. <u>The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

The proposed amendment involves only the updating and expansion in scope of the existing design bases analysis of the Monticello containment. No new failure mode or mechanism have been created for any plant SSC important to safety nor has any new limiting single failure been identified as a result of the proposed analytical changes.

Therefore the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. <u>The proposed amendment will not involve a significant reduction in a margin of safety.</u>

Proposed changes to containment response analytical methods and scope include additional contributions to, and allowance for error, in the reactor decay heat input; longer delays in initiation of long-term containment cooling for reactor isolation and small break events; and reducing the credit during a short time interval for containment pressure to assist in ECCS pump NPSH. These changes are conservative with respect to their effect on analytical results. A small improvement in RHR heat exchanger performance and a small increase in wetwell attached piping design temperature based on updated analyses are also proposed for inclusion in the updated containment design basis. These changes do not constitute a significant reduction in the margin of safety.

Based on the above evaluation, and pursuant to 10 CFR 50.91, the operation of Monticello in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR 50.92.

V. Environmental Assessment

NMC has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration, or

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51 Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51 Section 51.22(b), an environmental assessment of the proposed changes is not required.

VI. References

1. Exhibit E, Section 4.1, "Revision 1 to License Amendment Request Dated July 26, 1996, Supporting the Monticello Nuclear Generating Plant Power Rerate Program," December 4, 1997. Approved by NRC Staff September 16, 1998, as Amendment No. 102 to Facility Operating License No. DPR-22.

2. GE-NE-T23-00731-2, "Monticello Nuclear Generating Plant LOCA Containment Analyses for Use in Evaluation of NPSH for the RHR and Core Spray Pumps," June 1997. Submitted for NRC review as Exhibit D of Monticello License Amendment Request dated June 19, 1997. Approved by NRC Staff in Reference 4.

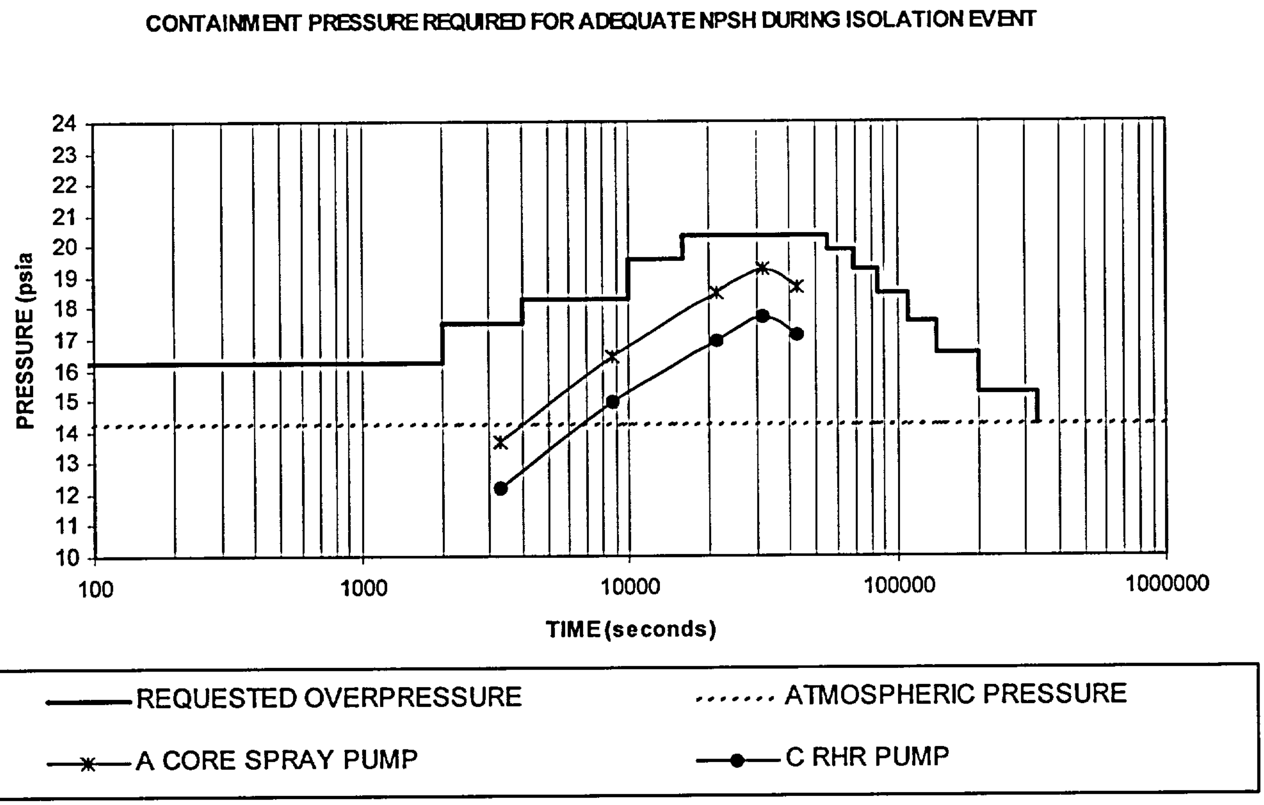
3. General Electric Service Information Letter (SIL) Number 636, Revision 1, "Additional Terms Included in Reactor Decay Heat Calculations," June 6, 2001.

4. "Issuance of Amendment RE: Updated Analysis of DBA Containment Temperature and Pressure Response and Reliance on Containment Pressure to Compensate for Potential Deficiency in NPSH for ECCS Pumps During DBA (TAC No. M97781)," Amendment 98 to DPR-22, July 25, 1997.

Exhibit F

Monticello License Amendment Request dated December 6, 2002

FIGURE 11
CASE 6



References

1. GE Report GE-NE-0000-0002-8817-01, "Monticello Nuclear Generating Plant Long-term Containment Analysis."

2. NRC Safety Evaluation Report, "Monticello Nuclear Generating Plant – Issuance of Amendment RE: Updated Analysis of DBA Containment Temperature and Pressure Response and Reliance on Containment Pressure to Compensate for Potential Deficiency in NPSH for ECCS Pumps during DBA (TAC No. M97781)," dated July 25, 1997.