ENCLOSURE 1

STAFF EVALUATION OF THE MONTICELLO NUCLEAR GENERATING PLANT INDIVIDUAL PLANT EXAMINATION

(IPE)

(INTERNAL EVENTS ONLY)

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EXECUTIVE SUMMARY

The NRC staff completed its review of the internal events portion of the Monticello Nuclear Generating Plant (Monticello) Individual Plant Examination (IPE) submittal, and associated documentation which includes licensee responses to staff generated questions and comments. The licensee's IPE is based on a full scope level 2 PRA performed in fulfillment of Generic Letter 88-20 and is documented in the submittal. No specific Unresolved Safety Issues (USIs) or Generic Safety Issues (GSIs) were proposed for resolution as part of the IPE.

The findings presented in the submittal under the specific damage classes identify station blackout as the dominant contributor to overall core damage frequency 46%. The next largest overall contributor to core damage involved flooding 26%, followed by transients with loss of high pressure reactor inventory makeup 13%, with either subsequent failure to depressurize (contributing 12% of CD) or subsequent failure of low pressure reactor inventory makeup (contributing 1% of CD). These were followed by ATWS 10% and LOCA 5%. The IPE estimated the core damage frequency as 1.9E-5/yr for internal events excluding flooding and 2.6E-5/yr with internal flooding.

The licensee used the following questions to determine if vulnerabilities existed:

1. Are there any new or unusual means by which core damage or containment failure occur as compared to those identified in other PRA's?

2. Do the results suggest that the Monticello core damage frequency would not be able to meet the NRC's safety goal for core damage.

The Monticello IPE did not identify any severe accident vulnerabilities associated with either core damage or containment failure. However, the licensee identified hardware modifications and procedural changes which were implemented during the 1993 refueling outage, with the exception of one procedure which is scheduled for final approval in 1994. These improvements focus on both reducing core damage frequency and offsite release of radioactivity.

Based on the reviewed of the Monticello IPE submittals and associated documentation, staff concludes that the licensee met the intent of Generic Letter 88-20. This conclusion is based on the following findings: (1) the IPE is complete with respect to the information requested in Generic Letter 88-20 and associated supplement 1; (2) the analytic approach is technically sound and capable of identifying plant-specific vulnerabilities, including those associated with internal flooding; (3) the licensee employed a viable means to verify that the IPE models reflect the current plant design and operation at time of submittal to the NRC; (4) the IPE had been peer reviewed; (5) the licensee participated in the IPE process; (6) the IPE specifically evaluated the decay heat removal function for vulnerabilities; (7) the licensee responded appropriately to Containment Performance Improvement (CPI) program recommendations. In addition the licensee has indicated that, "NSP plans to have a living PRA Program to support Monticello licensing, training, engineering and operations."

It should be noted that the staff's review primarily focused on the licensee's ability to examine Monticello for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) which stem from the examination.

1.0 INTRODUCTION

On November 23, 1988, the NRC issued Generic Letter 88-20 which requires licensees to conduct an Individual Plant Examination in order to identify potential severe accident vulnerabilities at their plant, and report the results to the Commission. Through the examination process, a licensee is expected to (1) develop an overall appreciation of severe accident behavior, (2) understand the most likely severe accident sequences that could occur, (3) gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, reduce the overall probability of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

As stated in Appendix D of the IPE submittal guidance document NUREG-1335, all IPEs are to be reviewed by the staff to determine the extent to which each licensee's IPE process met the intent of Generic Letter 88-20. The IPE review itself is a two step process; the first step, or "Step 1" review, focuses on completeness and the quality of the submittal. Only selected IPE submittals, determined on a case-by-case basis, will be investigated in more detail under a second step or "Step 2" review. The decision to go to a "Step 2" review is primarily based on the ability of the licensee's methodology to identify vulnerabilities, and the consistency of the licensee's IPE findings and conclusions with previous PRA experience. Unique design may also warrant a "Step 2" to better understand the implication of certain IPE findings and conclusions. As part of this process, the Monticello IPE only required a "Step 1" review.

On February 27, 1992, the Northern States Power Company(NSP) submitted the Monticello IPE in response to Generic Letter 88-20 and associated supplements. The IPE submittal described the application of a Level 2 PRA to identify vulnerabilities, consistent with Generic Letter 88-20. The IPE submittal contains the results of an evaluation of internal events, including internal flooding. The licensee plans to provide a separate submittal on findings stemming from the IPE for external events (IPEEE). The staff will review the IPEEE separately, within the framework prescribed in Generic Letter 88-20 Supplement 4.

On December 17, 1992, the staff sent a request for additional information to the licensee. A conference call between the staff and the licensee was held on January 25, 1993, to clarify some of the questions. The licensee responded to the staff's request in a letter dated February 16, 1993.

2.0 STAFF'S REVIEW

2.1 Licensee's IPE Process

The Monticello IPE submittal describes the approach taken by the licensee to confirm that the IPE represents the current as-built as-operated plant. The date used for the plant configuration analyzed in the PRA/IPE is March 1, 1991. Additional information concerning the licensee's participation in the process was provided in NSP's response to questions. The type of documents used to support the IPE study included plant procedures, drawings, and operations manuals. In response to questions regarding the use of documents which control modifications, NSP indicated that the sources used captured the plant modifications since they are updated when an associated system is modified. In its response to staff questions, NSP also identified some modifications that had been installed between the time the IPE model was developed and the submittal was made. NSP stated that except for the fire water system crosstie to the RHR system the identified modifications were not expected to materially influence the IPE results.

In addition to the review of these documents, members of the NSP IPE staff also carried out walk-throughs during the analysis for familiarization with plant/system operations, equipment layout for origin and susceptibility to floods, and human reliability analysis. NSP indicated that individual system walkdowns were performed as necessary to answer questions that arose during fault tree construction. Based on the review of the information provided in the submittal and the response to questions, the staff concludes that these actions constituted a viable process capable of confirming that the IPE represents the as-built as-operated plant.

The IPE submittal contains a summary description of the licensee's IPE process, its participation in the process, and the subsequent in-house peer review of the final product. The staff reviewed the description in the submittal and the licensee's response to questions regarding the process. The licensee indicated that the NSP staff participated in all aspects of the PRA to obtain technology transfer in model development, reviews, data collection and quantification of the models. In addition to the IPE staff other department personnel were involved to ensure that the models accurately portrayed the plant. The submittal indicated that, "NSP plans to have a living PRA Program to support Monticello licensing, training, engineering and operations."

As part of the IPE process NSP established an in-house review team which consisted of personnel from all appropriate organizations including design, engineering, operations,and training. In addition a senior review team made up of three of NSP's consultants (Erin Engineering, Fauske & Associates, and Tenera L.P.) reviewed the PRA/IPE to "ensure the correctness of the methodology."

The licensee used the following questions to determine if vulnerabilities existed:

1. Are there any new or unusual means by which core damage or containment failure occur as compared to those identified in other PRA's?

2. Do the results suggest that the Monticello core damage frequency would not be able to meet the NRC's safety goal for core damage.

NSP stated that these criteria did not lead to identification of vulnerabilities. It appears that Monticello's IPE allowed them to identify particular items of significance (significant insights) at the system and component level. This ability is demonstrated in their characterization of important assumptions, and contributions for components and operator actions provided for the representative sequences in the accident classes in the submittal and in response to staff questions. From the results of the analysis and in response to staff questions NSP has identified changes and procedural enhancements which are identified in Section 2.5 of this report.

Consistent with the intent of the Generic Letter 88-20, the staff believes that the licensee's peer review process provided reasonable assurance that the IPE analytic techniques had been correctly applied and the documentation accurate.

2.2 Front-End Analysis and DHR Evaluation

The licensee has stated that the Monticello plant was initially examined in 1988 using IDCOR's IPE Methodology, but that in order to fulfill the requirements of the Generic Letter (88-20) a full scope Level 2 PRA was performed. The staff examined the front-end analysis as described in the IPE submittal for completeness and consistency with accepted PRA practices. The staff finds the employed methodology described and justified for selection. The chosen methodology as summarized below, is consistent with methods identified in Generic Letter 88-20.

The front-end IPE analysis used the small event tree/large fault tree (fault tree linking) methodology. The failure equations of the supporting systems were linked to the frontline system fault trees. Dependencies of the frontline systems on support systems are modelled explicitly in combining frontline system failures. The level 1 event trees were "functional" as opposed to "systemic" trees which NSP indicates were similar to those used in the IDCOR IPE Methodology. The computer software used for managing fault trees was CAFTA. PCSETS was used for sequence quantification. The front-end system model interfaces with the back-end (containment event trees) directly by using the cutsets for the Level 1 sequences as the initiating events in the containment event trees. The level 1 sequences were grouped into categories called accident classes which were based on characteristics such as the initiator, primary system conditions and containment conditions. The information carried from the level 1 to the level 2 analysis accounts for pre-existing conditions that would impact the back-end analysis and is consistent with other PSAs.

The licensee's listing of initiating events that were analyzed for Monticello closely follows (almost one for one) the IDCOR IPEM listing. As such they have identified four groupings: LOCAs, transients, loss of offsite power, and

ATWS, all of which encompass 26 events including internal flooding. Flooding events were analyzed under the transient grouping. Seven of the events are ATWS events with various initiators such as turbine trip.

In response to staff questions regarding the completeness of the set of events and the development of unique system specific initiators, NSP stated that it used the events in WASH 1400 and IDCOR BWR IPEM as a starting point. Additionally, NSP reviewed 18 years of the plant operating history and developed dependency matrices to examine the effects of support systems on important front-line systems. NSP indicated that no new events that had not already been included from the review of previous PRAs were identified.

NSP indicated in the submittal that insights derived from its application of the IPEM identified events that were of low frequency or had little impact. Loss of an AC bus was one of these and was eliminated from consideration due to low frequency and redundancy. In response to a question regarding the AC bus as an initiating event the frequency was not provided, but NSP responded that the effect of an AC bus would be encompassed by loss of offsite power. In order to determine the frequencies of plant-specific initiating events, plant transient occurrence data, including scram reports, LERs and monthly operating reports, were used. Generic data was applied to other events where specific plant data was insufficient. The licensee provided a list of initiating event categories with the representative events and frequencies, which the staff compared to other PSAs (Peach Bottom), and found them to be consistent.

As noted above the licensee used functional event trees similar to those specified in the IDCOR IPEM, in conjunction with system based fault trees (submittal section 3.2.2). The result of the analysis using this arrangement appears to provide sufficient information to identify significant contributors (section 3.4.2.1, important components) as requested by NUREG-1335. While NSP's presentation of results in the submittal is not always consistent (sometimes component contributions are addressed as types such as "HPCI and RCIC Random Failure Mechanisms" or as unspecified system components in a specific system) contributions for a number of components in specific accident classes were identified. In the response to staff questions regarding the contributions from components in the HPCI and the RCIC systems, NSP provided the percentage contribution for the major contributors including maintenance and test.

The IPE submittal contained the event and special trees developed from the plant response to transients, station blackout, LOCAs, and ATWS events. System success criteria were presented for each initiating event category. The bases for the success criteria for systems required to mitigate events were identified by NSP as MAAP, USAR, Operation Manual Description and the IPEM Analysis. NSP indicated that system success criteria and responses were evaluated using the Monticello simulator.

In general the staff finds the Monticello event and special trees to be consistent with regard to initiating events, associated success criteria and dependencies between top events.

The Monticello IPE analyzed frontline and support systems important to the prevention of core damage and mitigation of fission product release consistent with other PSAs. The IPE submittal addressed dependencies by providing dependency matrices which identify "support to support" and "support to frontline" system dependencies on a train basis, and by developing fault trees for support systems which linked directly into the frontline system fault tree logic for quantification.

The Monticello submittal addressed loss of pump room cooling. Based on time/temperature calculations and/or tests coupled with the implied small probability of random failure and failure to recover, the licensee concluded that no further evaluation was necessary. In response to staff questions regarding loss of cooling to AC and DC equipment rooms and the control room (CR), NSP indicated that due to room configuration (in the turbine building for AC) and lack of significant heat source (for the DC rooms), the temperature rise was not expected to present a problem. In addition, procedures are in place that have operators open doors and use dedicated portable fans if required, concluding that loss of HVAC to AC/DC rooms does not result in loss of AC or DC power. Loss of CR cooling was considered to have a negligible impact on CDF due to the availability of temporary means to cool the CR, and the necessity for multiple failures to exist to impact the CR, or both the CR and the alternate shutdown panel area if equipment problems in the CR became of concern.

The Monticello IPE used both generic and plant-specific data for quantification of the model. Mean values were employed where sufficient plant data existed. Plant-specific data included initiating event and component failure rates and test and maintenance unavailabilities collected between 1978 and 1987.

The submittal indicated that classical statistical methods of estimating component failure rates were used. However, when no failures were recorded in the plant data for the component operating hours or demands experienced, NSP stated that a value of 0.5 was used as the "number of failures" experienced allowing the "derivation of a conservative non-zero estimate." If the resultant failure rate was much higher than the generic estimates for similar components, then the generic estimate was used. In response to staff questions, NSP identified the components for which it used the 0.5 failure value vs. the generic values. Plant-specific data were used for a number of the important pumps such as HPCI, RCIC, RHR; the 0.5 approximation was used for components such as EDG, RHRSW pump, and SRVs.

The IPE treated common cause failures explicitly. Common cause events were defined and their probabilities were estimated "in the framework of the Multiple Greek Letter (MGL) Model." The primary data for common cause factor estimates were obtained from the following documents: EPRI NP-3967, NUREG/CR-2770, NUREG/CR-2098, NUREG-2099, EGG-EA-5623 and NPRDS Data. A search of plant data uncovered one common cause event which was included in the analysis. Of eight common cause events that contributed > 1% to CDF, five were related to EDG or EDG support systems(with Fussel Vesely values ranging from 1% to 23%).

The licensee's IPE flood analysis employed initiating events analysis, FMEA and walkdowns of each flooding zone to determine potential flood sources, locations, propagation paths, impact on plant operation and the ability of the operations staff to safely shut down the plant. This information was used to quantify the sequences with the existing logic models. Spray from high capacity systems was considered but screened out for low capacity systems. Maintenance activities causing flood were incorporated by distributing the differential between the estimates for total flood frequency(0.025/yr) and passive failure flood frequency (0.008/yr) to flood zones where maintenance activities would be performed. NSP indicated that its analysis used bounding and conservative assumptions and all flood events required additional random failures for inadequate core cooling to occur. Thus, while the contribution from all floods is approximately 26%(6.8E-6) of the total CDF (2.6E-5) due to all events, NSP concluded that "flood initiators do not contribute significantly to the risk of core damage." In response to staff questions NSP has stated that as a result no plant or procedure changes are planned but NSP has informed operators of flooding potential and consequences and has provided training on the IPE results.

Based on the review of the description of the internal flood analysis provided in the submittal and the response to questions the staff finds the IPE flood assessment to be consistent with Generic Letter 88-20.

The submittal identified the dominant accident sequences in accordance with the reporting guidelines in NUREG-1335. The submittal does not list all the system failures for each sequence, but represents sequences by accident class and within each class generally by the dominant initiating events. All functional sequences with frequencies greater than 1E-7/yr were presented. The sequences for internal events were discussed separately from the flooding sequences, and the contributions of the internal events as a percent of the CDF due to internal events(excluding flooding) were identified.

A point estimate was calculated for the total core damage frequency. The findings presented in the submittal under the specific damage classes identify station blackout as the dominant contributor to overall core damage frequency 46%. The next largest overall contributor to core damage involved flooding 26%, followed by transients with loss of high pressure reactor inventory makeup 13%, with either subsequent failure to depressurize contributing 12% of CD or subsequent failure of low pressure reactor inventory makeup contributing 1% of CD. These were followed by ATWS 10% and LOCA 5%. The IPE estimated the core damage frequency as 1.9E-5/yr for internal events excluding flooding and 2.6E-5/yr with internal flooding included.

In accordance with the resolution of USI A-45, the licensee has performed an evaluation of the DHR systems as an intrinsic part of the Monticello IPE to identify DHR vulnerabilities. The results of the IPE provide indications of the importance of the systems with respect to the decay heat removal function as a response to the initiating events postulated in the IPE. This is measured by the percentage of core damage frequency attributable to sequences that represent failure of DHR in the quantification.

The contribution to core damage of failure of long term heat removal identified as accident class 2 (Loss of containment heat removal) in the submittal is less than 1%.

The following systems associated with the decay heat removal functions were considered in the evaluation:

- Main Condenser
- RHR System
- Containment Venting

Of the systems included, NSP has identified the following as important contributors to their failure:

Main Condenser - Turbine bypass valves(71%), Instrument air failures(11%).

RHR System - common cause failure of the torus cooling valves(28%), random failure of components in both RHRSW loops(18%) and loss of one loop of RHR and opposite loop of RHRSW(11%).

Containment Vent - Instrument Air(33%), Operator Action to vent(18%), Service Water(13%), Instrument Panel(11%)

Based on the staff's review of the IPE front-end analysis and the finding that the employed analytical techniques are consistent with other NRC reviewed and accepted PSAs and capable of identifying potential core damage vulnerabilities, the staff finds that the Monticello IPE front-end analysis meets the intent of Generic letter 88-20.

3. Back-End Analysis and Containment Performance Improvements (CPI)

The staff examined the Monticello back-end analysis for completeness in regards to level of detail requested in NUREG-1335. The staff found the Monticello IPE submittal essentially complete with respect to the type of information requested.

The Monticello IPE back-end analysis utilized a methodology similar to that described in NUREG-2300 (PRA Procedures Guide), and employed Version 7.0, 7.02 and 7.03 of the MAAP-3.0B computer code to model the containment thermal response and radionuclide release characterization. The utilized approach is clearly described and is consistent with Generic Letter 88-20, Appendix 1 and NUREG-1335, Appendix A.

The results of the front-end analysis of the Monticello IPE identified six dominant classes of accidents. These classes include Class 1A — primarily transients with loss of high pressure injection and vessel depressurization failure; Class 1B — station blackout with loss of injection; Class 1C — ATWS; Class 1D — primarily transients with successful vessel depressurization, but with loss of all injection (high and low pressure); Class 2 — primarily transients with loss of containment heat removal; and Class 4 — ATWS with failure to insert negative reactivity. Class 3 (LOCAs) and Class 5 (containment bypass) did not meet any of the level 1 screening criteria. Each class that met the screening criteria was evaluated in the back-end analysis; that is, a CET was developed for each dominant accident class.

The top events for each CET were developed using fault trees. These trees were linked with the front-line results so that level 1 failures and dependencies could be transferred directly to the level 2 analysis. Therefore, the dependencies identified in the front-end analysis were treated in the back-end analysis. In addition, the systems credited (considered) in the back-end were checked for dependencies in the back-end. The Monticello IPE assumed that *"fault tree linking allows for dependencies and failures important to the level 1 results to be carried directly into the level 2 sequence analysis.....Because of the fault tree linking approach, an explicit check list accounting for the transfer of dependencies between the level 1 and level 2 sequences is not necessary."* Although Monticello staff felt a check was unnecessary, a limited one was performed. The IPE also considered environmental effects on equipment behavior. These effects included room temperature, pressure, humidity, and radiation.

The Monticello IPE back-end analysis examined each of the containment failure modes and mechanisms listed in Table A.4 of Appendix A, NUREG-1335 in regards to their potential challenge to containment integrity.

Plant-specific analyses were performed in conjunction with other reference sources as a basis for containment failure conclusions. Containment failure from vessel blowdown and from in-vessel steam explosion were not modeled in the CETs because it was concluded that they presented no challenge to containment integrity. Containment failure from overpressurization was examined considering challenges from steam generation, H_2 combustion, non-condensible generation, ex-vessel steam explosions, and high pressure vessel penetration. H_2 combustion, though small probability of occurrence, was assumed to fail the containment. Ex-vessel steam explosion and high pressure melt ejection were concluded to present no challenge to containment integrity. Containment overpressurization from steam generation and non-condensible generation was modeled in the CETs. Containment failure from liner melt-through was modeled in the CETs, but assigned a zero probability of occurrence. This value was based on the assumption that there would be insufficient amount of debris to flow out of the pedestal region and contact the drywell shell liner. This assumption was based on the large sumps located in the pedestal region. Core concrete interaction in terms of basemat melt-through and vessel structural failure from erosion were concluded to present no challenge to containment integrity and were not modeled in the CETs. Containment failure from thermal attack of the penetration seals was examined. Failure of the seals was assumed to be bounded by 700°F at which point it is assumed that mechanical failure of the drywell shell supersedes non-metallic performance. Thermal attack of the containment boundary was incorporated onto the CETs by assigning an overtemperature (>700°F) failure mode.

Failure to isolate lines penetrating containment was addressed in some detail in the Monticello IPE. Fluid piping lines, instrument lines, air lines, vent lines, cable penetrations, personnel locks, hatches and drywell head were considered for potential containment isolation failure. The failure of the personnel locks, hatches, drywell head and cable penetrations was assessed in the analysis of the containment failure modes (e.g., overpressurization, thermal attack, etc.). Examination of failure to isolate a fluid, air, vent, or instrument line resulted in the identification of six types of penetrations as potential contributors to containment isolation failure. These potential contributors include torus-reactor building vacuum breakers, torus vent supply and exhaust, recombiners, drywell vent supply and exhaust, sumps, and CRD scram discharge drains.

The CETs developed in the IPE are simplified with the critical safety functions identified as the top events that are required to provide vessel injection, decay heat removal and containment heat removal. The systems required to perform the safety functions are identified along with their success criteria.

The CETs accident progressions either result in a phenomena which leads directly to containment failure with a probability of 1.0 or result in no containment failure with a probability of zero. The location of containment failure, based on expected pressure loadings and structural analyses, was identified and probabilities were estimated for these locations. Containment failure from overpressurization was assumed to occur at a mean failure pressure of 118 psia at either the DW head or the WW and DW vent line bellows as a leak.

The CET quantification resulted in :

No Release	60%
WW Vent	1%
DW Vent	17%
Overtemperature	1%
Overpressure	18%
H_2 Combustion	1%
CCI (Over-Pressure, Due to Non-Condensible Gases	2%

Containment failure is dominated by drywell venting and overpressurization primarily because of the relative contribution of the core damage sequences and because of features that limit other failure modes. For example, liner melt-through is not a contributor at Monticello because the containment sump is large relative to the core size. Events primarily contributing to overpressurization involve service water failure that fails instrument air and subsequently prevents containment venting.

Each CET end-state was assigned a damage state that defined the reactor status, whether core damage was arrested or if not, whether vessel failure occurred at high or low pressure. The damage state also assessed the containment status. Ten different states based on the containment failure mode were identified. These designators are based on the accident progression, containment performance (i.e., containment loading), and therefore, the resultant failure types. In addition, the damage state also identified the type of release, whether no release occurred, late, intermediate or early release.

The back-end analysis examined the effect of certain uncertainties by performing both probabilistic and deterministic sensitivity analyses. Three issues were addressed for the probabilistic sensitivities: liner melt-through, molten core concrete interaction, and core damage arrest in vessel. In the first sensitivity, assuming that liner melt-through occurred with a probability of 0.5, the resulting impact was that approximately half of the releases from CCI and drywell vent were reclassified as a liner melt-through type release. In the second sensitivity, assuming that CCI always resulted in basemat melt-through and pedestal failure, there was little impact to the release modes because the large percent of "no releases" was core damage arrest before vessel failure, and those releases after vessel failure were early releases which would not be affected by CCI. For the core damage arrest sensitivity, assuming that core damage arrest was not possible, vessel failure always occurred; however, the release did not substantially change because only the location of the core debris changed in the analysis, not the releases. In addition to the probabilistic sensitivities, phenomenological sensitivities were also performed. These sensitivities included (1) channel blockage, (2) recovery of badly damage core, (3) fission product vaporization, (4) core material within original core boundary, and (5) coolability of debris in containment.

The licensee defined vulnerability as either *"any new or unusual means by which core damage or containment failure occur as compared to those identified in other PRAs or the results suggest that the Monticello core damage frequency would not be able to meet the NRC's safety goal for core damage."* In this context, no vulnerabilities were identified by the licensee. Insights from the IPE were, however, provided by the licensee along with recommendations for potential plant modifications in regards to containment performance. These modifications are of two types designed to either reduce the potential for a core damage event, or reduce the potential for a release given a core damage event. The licensee examined each damage class and identified both negative and positive factors influencing the results. These factors formed the bases for the insights and recommendations of plant modifications.

The Monticello IPE examined the containment performance improvement issues: (1) alternate water supply for drywell spray and vessel injection; (2) enhanced ADS; and (3) implementation of revision 4 of the EPGs. These modifications would potentially reduce the core damage frequency by less than a factor of two (i.e., reduce to 9.3E-6) and would also reduce the importance of the dominant contributors. For alternate water supply for drywell spray or vessel injection, this type of system was credited in the IPE but had little effect. The IPE credited RHRSW cross-tie to RHR and the licensee has modified the plant to connect the fire system to RHR. A dominant failure mode, however, in the IPE is common failure of the RHR injection valves. For enhanced depressurization reliability, the licensee indicated that plans are under consideration to modify the power supply for the nitrogen bottles to the SRVs.

For emergency operating procedures, the licensee examined Revision 4 of the EPGs relative to Revision 3. The review indicated that the changes did not affect the actions modeled in the IPE nor did it affect the success criteria already included in the IPE. The staff concludes that the licensee's response to the containment performance improvement program recommendations is reasonable and consistent with the intent of Generic Letter 88-20.

In summary, the licensee employed a process in the back-end analysis to understand and quantify severe accident progression. The process led to a determination of containment failure and is consistent with the intent of Generic Letter 88-20, Appendix 1. Sensitivity studies were performed to better understand the impact of both data and phenomenological uncertainties. Dominant contributors to containment failure were found to be consistent with insights from other PSAs of similar design and differences were properly accounted for. The IPE characterized containment performance for each of the CET end-states by assessing containment loading. The licensee's IPE addressed each of the severe accident phenomena normally associated with a Mark I containment. The overall assessment of the back-end analysis is that the licensee has made reasonable use of PSA techniques in performing the back-end analysis, and that the techniques are capable of identifying severe accident vulnerabilities. Based on these findings, the staff concludes that the licensee's back-end IPE process is consistent with the intent of Generic Letter 88-20

2.4 Human Factors Consideration

The Monticello Nuclear Generating Station IPE submittal describes the human reliability analysis performed. The licensee's IPE submittal categorized human errors as: 1) errors made in restoring systems to normal operating status following test or maintenance activities; 2) activities in progress at the time of the initiating event which influence its outcome such as maintenance or testing; and 3) actions performed in responding to an accident. This human action taxonomy is logical and representative of those used in other PRAs, and it supports the identification of important human actions.

The human reliability analysis methodology employed in the IPE utilized the SHARP framework (EPRI NP-3583) and the quantification approach described in NUREG/CR-4772 (i.e., ASEP method). Importance measures (i.e., Risk Reduction Worth and Risk Achievement Worth) were used in the IPE to identify operator actions that contributed significantly to the baseline core damage probability or for which a change in the failure rate could cause a significant increase in overall core damage probability. A detailed HRA was then performed on the most significant operator actions and recovery actions identified using the EPRI SHARP framework and the ASEP quantification method.

The licensee compared the HEP results from the detailed HRA to the results from the screening analyses. Since none of the screening HEP values used in the accident sequence quantification for the important human actions were underestimated, the licensee did not requantify the accident sequences with the lower HEP values from the detailed HRA. Consequently, the licensee considers the current Monticello accident sequences incorporating the important human actions to be bounding.

The licensee indicated that its staff were directly involved in the conduct of the IPE to insure the knowledge gained from the examination would become an integral part of plant procedures and training program and allow future activities to be performed with limited involvement by consultants. A walkdown of the simulator, and areas outside the control room in which operator actions are required, to support the derivation of human error probabilities (HEPs) included both the licensee analyst responsible for deriving HEPs and the consultant responsible for HEP guidelines. Of note, the licensee used existing information compiled as part of the Monticello control room design review to identify human factors considerations that may affect the quantification of the HEPs. The licensee also reviewed the operator actions in the EOPs to identify human factors concerns.

The quantified HEPs for those human events occurring after the onset of core damage appear to be low. However the licensee indicated that performance shaping factors were evaluated in the quantification and that importance measures were calculated to determine the potential contribution of these human events. The importance measures indicated that these events were not significant and led to no additional insights.

In summary, and based on a review of the licensee's IPE submittal and responses to staff questions, the staff finds the licensee's assessment of human reliability, conducted as part of the IPE of the Monticello Nuclear Generating Station, capable of discovering severe accident vulnerabilities from human errors consistent with the intent of Generic Letter 88-20. The HRA methodology described in the licensee's IPE submittal supports the quantitative understanding of the overall probability of core damage during plant operations, as well as an understanding of the contribution of human actions to that probability. Human-related plant improvements that have been implemented (including, for example, changes to training and procedures to enhance operator reliability associated with reactor depressurization and feedwater system recovery) or under review are expected to enhance human reliability and plant safety.

In addition, the licensee's intent to maintain a living PRA program to support Monticello licensing, training, engineering and operations indicates that a mechanism will exist for the licensee to continue to identify and evaluate the risk significance of potentially important human actions during plant operation and maintenance.

2.5 Licensee's Actions and Commitments from the IPE

As part of the IPE process and in response to staff questions for clarification, the licensee identified the following changes to the plant, procedures, and training. Some of these changes were presented in the form of recommendations, and some have been implemented or are to be implemented. They are presented by accident classes.

Class 1A - Loss of high press.injection and failure to depress.

- o Modification to provide power to solenoid valves for bottled nitrogen (used to operate the SRVs to depressurize) from an instrument panel that can be powered by an essential power supply or batteries. Applicable to class 1B also. Modification was completed during the 1993 refueling outage.

- o Modification to assure faster operation of the condensate demineralizer bypass valve on loss of air was implemented to help prevent loss of condensate/feedwater on loss of air.

Class 1B - SBO and failure of HPCI and RCIC

- o Operator training on station blackout including shedding of DC loads. Completed

- o Improve battery load shedding procedures. Expected to be completed in 1993

- o Provide AC independent power supply to battery chargers. Under evaluation at time of response (Feb. 93)

- o Modification to allow crosstie of diesel fire water pump to the RHR system. Completed

- o Modify power supply for SRV pneumatics (same as class 1A)

Class 1C - Loss of high and low pressure injection systems

- o A recommendation was considered for development of procedures for the use of low pressure backup injection systems such as RHRSW through LPCI, condensate service water, service water to the hotwell.

- o Operators trained on operation of RHR and core spray pumps under certain operating conditions such as when cavitation is possible.

Class 2 - Loss of Decay Heat Removal

Recommendations were being considered for:

- o Operator training on recovery of failed RHR system and failed condenser with degraded or failed support systems.

- o Elimination of locked open condition for discharge valves for air receiver tanks to allow isolation to prevent loss of air.

- o A procedure for the replenishment of the water in the CSTs.

Class 4 - ATWS

- o Operator training on the significant insights for ATWS was performed.

Recommendations were being considered for:

- o Moving operator actions for mechanically bound CRDs to a contingency procedure to allow operators to focus on reactor shutdown with SLC.

- o Testing the boron injection hoses.

3.0 CONCLUSION

The staff finds the licensee's IPE submittal for internal events including internal flooding consistent with the information requested in NUREG-1335. Based on the review of the submittal, the licensee's response to questions, and associated information, the staff finds the licensee's IPE conclusion that no fundamental weakness or severe accident vulnerabilities exist at Monticello to be reasonable. The staff notes that:

(1) NSP personnel were involved in the development and application of PSA techniques to the Monticello facility, and that the associated walkdowns and documentation reviews constituted a viable process for confirming that the IPE represent the as-built, as-operated plant.

(2) The licensee performed an in-house peer review to ensure that the IPE analytic techniques had been correctly applied and that the documentation is accurate.

(3) The front-end IPE analysis is complete with respect to the level of detail requested in NUREG-1335. In addition, the analytical techniques were found to be consistent with other NRC reviewed and accepted PSAs.

(4) The back-end analysis addressed the most important severe accident phenomena normally associated with the Mark I containment type. No obvious or significant problems or errors were identified.

(5) The HRA allowed the licensee to develop a quantitative understanding of the contribution of human errors to CDF and containment failure probabilities.

(6) The employed analytical techniques in the front-end analysis, the back-end analysis, and the HRA are capable of identifying potential plant-specific vulnerabilities.

(7) The licensee's IPE process searched for DHR vulnerabilities consistent with the USI A-45 (Decay Heat Removal Reliability) resolution.

(8) The licensee responded to CPI Program recommendations, which included searching for vulnerabilities associated with containment performance during severe accidents.

Based on the above findings, the staff concludes that the licensee demonstrated an overall appreciation of severe accidents, has an understanding of the most likely severe accident sequences that could occur at the Monticello facility, has gained a quantitative understanding of core damage and fission product release, and responded appropriately to safety improvement opportunities identified during the process. The staff, therefore, finds the Monticello IPE process acceptable in meeting the intent of Generic Letter 88-20. The staff also notes that the licensee's plan to have a living PRA program to support Monticello licensing, training, engineering and operations will enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the lifetime of the plant.

APPENDIX
MONTICELLO DATA SUMMARY SHEET*
(INTERNAL EVENTS)

Total core damage frequency (TCDF) with flooding: 2.6E-5/yr
core damage without flooding: 1.9E-5/yr

Major initiating events:	Contribution (% of TCDF)
Loss of Offsite Power	50.8%
Turbine Building 931' Service Water Flood	9.4%
Reactor Building 896' and higher Service Water Flood	9.3%
Turbine Trip	7.0%
Turbine Building 911' Service Water Flood	6.6%
Loss of Feedwater	3.8%
Manual Shutdown	2.6%
MSIV Closure	2.6%
Medium LOCA	1.9%
Large LOCA	1.2%
Small LOCA	0.9%

Major contributions by functional group:	Contribution (% of TCDF)
Station Blackout	46%
Floods	26%
Loss of Injection, without Depressurization	12%
Anticipated Transient Without Scram (ATWS)	10%
LOCA	5%
Loss of Injection, with Depressurization	1%
Loss of Heat Removal	<1%

Major contributions to dominant core damage sequences:

- Station Blackout - Loss of offsite power and diesel generators and (1) failure to recover offsite and onsite power leading to failure of high pressure makeup after 4 hours as a result of battery depletion and failure to depressurize; (2) failure to recover offsite power within 30 min. and independent failure of HP makeup and failure to depressurize.

* Information has been taken from the Monticello IPE and the NSP response to staff questions and has not been validated by the NRC staff.

- Internal flooding from (1) a service water line break in reactor building > el. 896', failure of all high pressure injection systems and failure to depressurize; or (2) a service water line break in the 931' el. east turbine building which propagates to motor control centers and main access control and station batteries rendering high pressure injection system inoperable, and failure to depressurize; or (3) service water flood at 911' el.turbine building disabling Div 1 4KV room, failure of high and low pressure makeup.

- Transient - loss of offsite power, manual shutdown, turbine trip or loss of feedwater and subsequent failure of all high pressure sources of makeup and failure to depressurize.

Major operator action failures (% contribution to the total core damage frequency):

Failure to depressurize reactor (45 minutes)	22.0%
Failure to inject SLC/turbine trip initiator	3.1%
Failing to control level; dilutes boron	1.8%
Failure to inject SLC/MSIV closure initiator	1.2%
Failure to manually open SV-4234/35 (alternate nitrogen supply)	1.0%

Significant IPE findings:

- The assumption that the RHR and core spray pumps will continue to provide adequate flow after containment failure reduced the calculated core damage frequency by about 1E-5/yr.

- Loss of high pressure injection and failure to depressurize stems from the dependency of long-term operation of the SRVs on a key instrument panel. Instrument panel Y20 powers AC solenoid valves which provide nitrogen to the SRVs for the purpose of depressurizing the reactor. The solenoid valves isolate on SBO because panel Y20 is not powered from the battery supplied essential AC. Accumulators assure operation of SRVs until Y20 is repowered by a diesel or until offsite power is restored.

- Station Blackout coping is negatively influenced by the potential inability to provide high pressure makeup and depressurization of the reactor vessel following battery depletion after 4 hours. The benefit of the crosstie between the diesel fire pump and the RHR as an injection source is limited unless the ADS capability is increased by extending battery life and modifying the power supply to the SRV pneumatic solenoid valves as recommended in the IPE.

- NSP took credit for quick recovery of loss of feedwater events based on occurrences which it used to estimate that the failure to recover would be only 11%. NSP indicated that this had the effect of significantly reducing the total contribution of accident class 1A, characterized by loss of high pressure makeup and failure to depressurize (CDF = 3E-6). Loss of feedwater accounted for 20% of this class.

- An anticipated transient without scram (ATWS) is exacerbated by the small capacity of the turbine bypass at Monticello. A turbine trip with bypass could result in heatup of the containment at a rate similar to MSIV closure until the operator takes action to initiate power/level control as specified by EOPs.

Important support systems (% contribution to the total core damage frequency):

AC	58.0%
Emergency Service and Emergency Diesel Generator Service Water (mostly EDG-ESW)	19.0%
DC	5.0%

Important plant hardware

RCIC - In 73% of cutsets of Class 1A(CDF = 3E-6).
36% of failure due to failure of pumps to start or run.
23% of failure due to valve failure

HPCI - In 69% of cutsets of class 1A(CDF = 3E-6).
61% of failure due to failure of HPCI or aux oil pump to start or run.

Enhanced procedures, hardware, and operator actions:

In response to the accident class 1A, concerning a loss of high pressure injection and failure to depressurize:

Modifications are under consideration to supply power to the bottled nitrogen supply for the solenoid valves from an instrument panel that can be powered by an essential power supply or batteries.

In response to accident class 1B, concerning Station Blackout:

1) Operator training was conducted covering such items as shedding DC loads, operating HPCI and RCIC to minimize battery drain, and plant response for at least 4 hours.

2) Procedure changes were drafted to upgrade the steps to loadshed the station batteries after an SBO to extend battery life if the diesels are not available, which would provide 2 extra hours of capacity.

3) Recommendations made to add procedural steps to commence a controlled cooldown as soon as possible after an SBO.

4) Recommendations made to supply station battery chargers with an AC independent power supply to extend battery life.

5) Plant was modified to allow the diesel fire pump to be aligned to RHR as a reactor vessel injection source.

In response to accident class 1C, concerning loss of high and low pressure injection systems:

1) A recommendation was considered for development of procedures for the use of low pressure backup injection systems such as RHRSW through LPCI, condensate service water, service water to the hotwell.

2) Operators trained on operation of RHR and core spray pumps under certain operating conditions such as when cavitation is possible.

In response to accident class 2, concerning Loss of Decay Heat Removal:

Recommendations were being considered for:

1) Operator training on recovery of failed RHR system and failed condenser with degraded or failed support systems.

2) Elimination of locked open condition for discharge valves for air receiver tanks to allow isolation to prevent loss of air.

3) Writing a procedure for the replenishment of the water in the CSTs.

In response to the accident class 4, concerning ATWS:

1) Operator training was conducted on the significant insights regarding ATWS.

2) Actions for mechanically bound CRDs were moved to a contingency procedure in the EOPs, so that the operator will focus on reactor shutdown with SLC.