February 28, 2005

L-PI-05-014
10 CFR 50.90

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

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

License Amendment Request (LAR) to Incorporate Revisions to Small Break Loss of Coolant Accident Methodology into the Prairie Island Nuclear Generating Plant Licensing Basis

Pursuant to 10 CFR 50.90, Nuclear Management Company, LLC (NMC), hereby requests the following amendment to the Operating Licenses for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2. This LAR proposes changes to the PINGP licensing basis and does not include any material changes to the Facility Operating License, Technical Specifications (TS) or TS Bases. Upon approval, the licensing basis changes proposed in this LAR will be incorporated into the Updated Safety Analysis Report (USAR).

The proposed amendment would allow the use, for PINGP, of the Small Break Loss of Coolant Accident (SBLOCA) methodology described in Westinghouse WCAP 10054-P-A Addendum 2 Revision 1, *Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model,* dated July 1997. This revised methodology determines the core response following a SBLOCA event and will be used to assure compliance with the post Loss of Coolant Accident (LOCA) acceptance criteria specified in 10 CFR 50.46.

NRC letter, dated August 12, 1996, forwarding its Safety Evaluation Report for WCAP-10054-P, Addendum 2, Revision 1, states the model is "...acceptable for referencing in NOTRUMP SBLOCA applications for operating reactors." PINGP uses the NOTRUMP methodology for its SBLOCA analyses.

The NRC has approved this methodology for use at a similar plant, Kewaunee Nuclear Power Plant, by letter dated April 4, 2003.

Based on the discussion in Exhibit A, which contains the licensee's evaluation of this proposed change, NMC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and attachments to the designated State Official.

NMC requests approval by February 20, 2006. NMC requests 60 days for implementation following approval.

Summary of Commitments
This LAR contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on:
FEB 28 2005

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

Exhibit A: License Evaluation

Exhibit A

License Amendment Request to Incorporate Revisions to Small Break Loss of Coolant Accident Methodology into the Prairie Island Nuclear Generating Plant Licensing Basis

Licensee Evaluation

1.0 DESCRIPTION

This license amendment request (LAR) is a request to change the licensing basis of the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 operating under Licenses DRP-42 and DPR-60.

The Nuclear Management Company (NMC) is requesting that the Nuclear Regulatory Commission (NRC) review and approve the use, for PINGP, of an addendum to the Small Break Loss of Coolant Accident (SBLOCA) methodology described in Reference 1 (which has been found acceptable by the NRC for referencing in NOTRUMP SBLOCA applications for operating reactors). This addendum is documented in Westinghouse WCAP 10054-P-A Addendum 2 Revision 1, *Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model* dated July 1997. This revised methodology determines the core response following a SBLOCA event and will be used to assure compliance with the post LOCA acceptance criteria specified in 10 CFR 50.46.

In summary, this change will allow the use of the changes to the NOTRUMP evaluation model as described in WCAP 10054-P-A Addendum 2 Revision 1 in all Prairie Island SBLOCA licensing basis analyses.

Additionally, future SBLOCA analyses for PINGP will apply "loop seal restrictions" consistent with Westinghouse approved methodology as applied in other plant analyses. Discussion of this change in modeling is provided for completeness since it relates to the proposed NOTRUMP SBLOCA methodology change. The change in modeling of the loop seal restrictions is not part of this LAR and discussions of the change are provided for information only.

2.0 PROPOSED CHANGE

A brief description of the proposed changes to the PINGP Updated Safety Analysis Report (USAR) is provided below.

USAR Section 14.7.3 "Small Break LOCA Evaluation Model" contains a statement listing the NOTRUMP topicals applicable to the Prairie Island Analysis. Specifically, WCAP 10054-P-A and WCAP 10079-P-A are presently listed. WCAP 10054-P-A Addendum 2 Revision 1 will be added to that list.

3.0 BACKGROUND

The NOTRUMP computer code is a one dimensional general thermal-hydraulic network code developed to address the Nuclear Regulatory Commission (NRC) concerns expressed in NUREG-0611. The computer code, "SBLOCTA," is used in conjunction with NOTRUMP to determine the clad temperatures, cladding oxidation and hydrogen generation. Together, these codes are the backbone of the Westinghouse NOTRUMP SBLOCA Evaluation Model (EM). The Prairie Island licensing basis SBLOCA analysis has historically been completed using the NOTRUMP SBLOCA EM as described in Reference 2.

NMC is requesting NRC approval for utilization of the approved revised NOTRUMP SBLOCA EM described in Reference 1 for application to the Prairie Island Unit 1 and 2 SBLOCA analyses. This revised NOTRUMP EM was developed by Westinghouse in 1995 to address the effects due to injecting Emergency Core Cooling System (ECCS) fluid directly into the broken loop rather than spilling it to the containment floor for breaks smaller than the ECCS injection line. In addition, the effects due to an improved injection condensation model, COSI, were included. This revised NOTRUMP SBLOCA EM is the standard for SBLOCA analyses presently supported by Westinghouse. These changes to the NOTRUMP SBLOCA EM have been previously reviewed and approved by the NRC as documented in the NRC Safety Evaluation Report dated August 12, 1996 as included in Reference 1.

A "loop seal restriction" discussion follows. The most recent NOTRUMP SBLOCA licensing basis analyses completed by Westinghouse for Prairie Island has included a conservative assumption for the Reactor Coolant System (RCS) intermediate leg water inventory. NMC has directed Westinghouse to conservatively implement modeling options which simulate maintaining the intermediate leg full creating a loop seal which results in much higher peak clad temperatures. It was NMC's position that the original NOTRUMP evaluation model documented in Reference 2 required the loop seal to be simulated and maintained for all break sizes, for additional conservatism. However, more recent reviews of the applicable topical reports and communications with the NRC concluded that the evaluation model, as approved by the NRC, allows removal of this loop seal restriction under specific loop break flow conditions. Removal of the loop seal restriction will result in reduced peak clad temperatures for the larger breaks where the loop seal will clear for long durations. The basis for the application of this analysis criteria is documented in Reference 3 and is the standard for Westinghouse SBLOCA analysis using the NOTRUMP SBLOCA EM.

4.0 TECHNICAL ANALYSIS

PINGP is a two unit plant located on the west bank of the Mississippi River approximately 6 miles northwest of the city of Red Wing, Minnesota. The facility is owned by Northern States Power Company (NSP) and operated by the NMC. Each unit at PINGP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The initial PINGP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC) in April 1967. The Final Safety Analysis Report (FSAR) was submitted for application of an Operating License in January 1971. Prairie Island Unit 1 began commercial operation in December 1973 and Unit 2 began commercial operation in December 1974.

The PINGP was designed and constructed to comply with NSP's understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967.

PINGP was not licensed to NUREG-0800, "Standard Review Plan".

4.1 APPLICABILITY OF THE REVISED NOTRUMP EM TO PRAIRIE ISLAND

The revised NOTRUMP model described in Reference 1 was submitted to the NRC for review and approval in 1995. This report was approved by the NRC in 1996 and the resultant NRC Safety Evaluation Report (SER) is included in the front of Reference 1. The NRC SER concluded the revised method provides conservative results and thus is acceptable for application to Westinghouse plants with two implied limitations as described on page 3 of the NRC SER. These two plant configuration specific implied limitations are the ECCS injection jet velocities and the range of loop pressures used to correlate the loop steam condensation model, COSI.

An analysis completed for Prairie Island conservatively determined the ECCS injection jet velocities are between 3 ft/sec to 12 ft/sec prior to accumulator injection, depending on the limiting ECCS configurations possible. This is well within the range of test conditions performed to justify application of the COSI steam condensation correlation as summarized in the NRC SER. In addition, Westinghouse determined that the RCS loops remain water solid following a SBLOCA until the system pressure decreases below that where the loop seals begin to clear and steam condensation becomes significant. This conclusion was based on generic studies for Westinghouse plants. Based on the Prairie Island specific analysis presented in Section 4.4 of this request, the loop seals do not begin to clear until the loop pressures are less than 1200 psia, which is the limit of the pressure range for applicability of the COSI correlations as discussed in the NRC SER. This provides additional assurance that the system pressures following clearing of the loop seals, when steam condensation becomes significant, are within the upper range of the test conclusions based on the COSI tests. Also, the accumulators in the Prairie Island facility are pressurized to a minimum of 710 psig as required by Technical Specification SR 3.5.1.3. This is well above the lower end of the COSI correlation limit of applicability (550 psia). As discussed in Reference 1, once the accumulators inject, the importance of the loop steam condensation, due to Safety Injection (SI) injection, becomes less significant since the accumulator injection dominates the peak cladding temperature reduction. In conclusion, the Prairie Island design maintains a configuration which is bounded by the implied limitations stated in the NRC SER for the revised NOTRUMP evaluation model.

Additionally, the Prairie Island ECCS and RCS configurations are typical of Westinghouse designs. The RCS is configured with two RCS loops. Having only two loops assures realistic modeling of each loop's hydraulic conditions since there is no need to lump intact loops for modeling simplification. The ECCS is configured so that either of the two Safety Injection pumps are aligned to a common header that can inject into either RCS cold leg as has been assumed in the NOTRUMP model. In addition, the revised NOTRUMP model assumes the ECCS cold leg injection nozzle is located above the bottom of the loop. This assumption is utilized to eliminate the need to evaluate the scenario where no ECCS injection into the loop occurs for the smaller breaks (See References 1 and 2 for the basis of this analysis assumption). At Prairie Island, the ECCS cold leg injection occurs above the bottom of the loop and thus this configuration assumption is applicable to Prairie Island.

Section 4 of Reference 1 presents typical analyses for Westinghouse 3-loop and 4-loop PWRs. These analyses were completed specifically to provide representative changes to peak cladding temperature (PCT) for the two modifications to the NOTRUMP evaluation model (COSI and SI injection into the broken loop). There were no specific terms and conditions created as a result of these sample cases except for the conclusion that the limiting break size does not change between the original and revised NOTRUMP SBLOCA EM. Since no sample analyses were completed for a 2-loop plant, this LAR does not request adoption of this conclusion for Prairie Island. Therefore, the revised analysis completed with the new NOTRUMP SBLOCA EM performs a full break spectrum to assure the limiting break size is identified.

4.2 LOOP SEAL RESTRICTION

NOTE: The following technical analysis is a summary of the applicable discussions documented in Reference 3. Proprietary information was omitted to simplify this license amendment request submittal.

The following discussion addresses the Westinghouse Small Break LOCA methodology assumption associated with the imposition of/removal of the artificial loop seal restriction utilized in the NOTRUMP SBLOCA EM. This discussion documents some background information as well as the Westinghouse traditional practice of the removal of the artificial loop seal restriction under specified conditions. The following discussion is a summary of the applicable sections from Reference 3 which was provided to the NRC in June 2000.

In a SBLOCA transient, at least one loop seal would eventually vent steam (and possibly more depending on loop-to-loop interactions). Without sufficient steam flow to ensure that all loops will vent steam for an extended period of time, venting of steam was chosen to occur in the broken loop only via the imposition of an artificial loop seal restriction. The loop seal restriction in the NOTRUMP SBLOCA EM is an artificiality imposed on calculations to restrict steam flow to ensure that venting of steam flow through the loop seal of the broken loop would occur first. This is more important in 3-loop and 4-loop plants modeled with lumped loops where loop to loop interactions cannot be modeled in sufficient detail to predict system responses. The reasons for the imposition of this restriction and the justification for its conservative effect on calculations are described in more detail in Reference 2 and Reference 4. For 3-loop and 4-loop plants with an explicit N-loop noding scheme, as well as for 2-loop plants (where the standard model represents explicit loop noding), the technical reasons for restricting the steam flow in any loop are not applicable. Although artificial, the restriction has routinely been applied for these cases, when steam flow is not sufficient to vent through all loops for an extended period of time, to maintain consistency with the licensing documentation. In application, the artificial loop seal restriction may only be removed for breaks for which steam flow is sufficiently high as described in Reference 4.

Reference 2 describes the conditions for which loop seal unpredictability and loop-to-loop interactions may result in non-conservative results. To address these conditions, Westinghouse identified a model that would ensure conservative behavior for these conditions. The conditions for when the model must be applied are described by a threshold break size below which the loop seal restriction is required. The following are pertinent excerpts from Reference 2 related to the loop seal restriction:

Page 5-101

"A method to ensure the conservative behavior for appendix K analysis is discussed and break spectrum calculations using the evaluation model with appendix K modification are presented. When the loop seal steam venting was limited to the broken loop, limiting core uncovery and cladding heatup results were calculated with results well below the limits of 10CFR50 part 46 and appendix K."

Page 5-45:

"....This modification [loop-seal restriction] is used in the evaluation model to ensure conservative behavior for break sizes below the threshold break size. Reiterating, break sizes larger than the threshold break size will realistically vent steam through more than one loop seal and in doing so will result in minimal core uncovery. The modification to assure conservative behavior is also applied to those breaks to ensure a continuum of response in terms of peak cladding temperature when only the broken loop is artificially forced to vent steam."

While there is some reason associated with maintaining a "continuum of response", especially in the presentation of the generic model application in Reference 2, this would rarely be the case in practice. Typical NOTRUMP SBLOCA analysis results in other reasons for "discontinuous" results between break cases, independent of the loop seal restriction modeling considerations.

The NRC SER for Reference 2 does not specifically address the loop seal restriction. However, TABLE VI-1 (p. 37 of the SER) identified the analysis assumptions for the SBLOCA audit calculations. Included in this table is reference to "Westinghouse conservative assumptions" in item 16, which states, "Loop seals in the intact loops are not permitted to clear prior to clearing of the loop seal in the broken loop." This condition is met even for larger breaks where the loop seal restriction is removed as described in more detail in Reference 3.

Reference 4 (WCAP 11145-P-A) was approved by the NRC in 1986. Reference 4 provides further clarification on the Westinghouse loop seal restriction modeling. The following are pertinent excerpts from this topical report related to the loop seal restriction:

Pages 2-11 and 2-12

"...........The loop seal clearing behavior may be delineated by defining threshold and critical break sizes. The threshold break size is the break size at which the transient loop seal perturbations are large enough to always result in more than one loop seal venting steam for a period of time. Break sizes below the threshold break size tend to vent steam through only one loop. For break sizes above the threshold break size, but below the critical break size, there are multiple loop seal clearings which will be oscillatory in nature and may involve loop-to-loop interactions. At the critical break size and above, the loop seal perturbations are large enough to always result in all loop seals venting steam for a period of time.... Consequently, for breaks below the critical break size, the NOTRUMP SBLOCA EM only allows the broken loop seal to clear and to vent significant amounts of steam.... Restricting the intact loop seal from clearing for breaks above the critical break size is unnecessarily conservative. Consequently, for breaks above the critical break size, the loop seal restriction can be removed (See Reference 3 for more detail)". Cases for which the loop seal restriction has been removed are also presented in Reference 4. The NRC SER for Reference 4 does not specifically address the loop seal restriction.

During the development and early applications of the NOTRUMP SBLOCA EM, Westinghouse worked closely with the Westinghouse Owners' Group (WOG) and the NRC to ensure that the new model would adequately and appropriately address the requirements of the Three Mile Island action plan. While not documented, this topic was presented to the NRC in April 1985. Close participants in the process were aware of the refinement of key assumptions leading to the application of NOTRUMP in addressing the requirements of NUREG-0737, II.K.3.31 under the umbrella of the WOG. It was in this environment that Westinghouse's clarification of the intended removal of the loop seal restriction, under clearly defined conditions, was placed in the public record with the publication of WCAP-11145-P-A (Reference 2). Since that time, it has been the

Westinghouse business practice to allow the analysts to remove the loop seal restriction when the appropriate technical conditions are satisfied.

Based on the technical justifications and identifications of intended application of the loop seal restriction in the NRC approved topical reports, References 2 and 4, which describe the NOTRUMP EM and its application, Westinghouse believes that its long-standing business practice of removal of the artificial loop seal restriction is appropriate under the conditions identified in Reference 3.

4.3 KEY INPUT ASSUMPTIONS

At a minimum, the Westinghouse SBLOCA NOTRUMP EM has incorporated the following input assumptions to comply with the requirements of 10 CFR 50 Appendix K:

1) 2% Power Measurement Uncertainty
2) All radial and axial peaking factors simultaneously at their most limiting value.
3) The Baker-Just zirc water reaction rate
4) 120% of the 1971 American Nuclear Society infinite decay heat load
5) Moody correlation for saturated break flow
6) Modified Zaloudek correlation for sub-cooled break flow

Other major assumptions inherent in the Westinghouse SBLOCA NOTRUMP EM are:

1) The break area is less than 1 ft^2.
2) The event initiates from Mode 1 or 2.
3) All rod control cluster assemblies insert except the single most reactive rod.
4) The worst case single active failure scenario is the Loss of Offsite Power with an emergency diesel failure resulting in loss of one train of emergency power and associated Safety Injection pumps, RCP trip and coast-down and Main Steam-line Isolation with no steam dump.
5) Standard two-loop ECCS spilling assumptions where breaks larger than the SI injection line are assumed to spill to the containment floor against containment backpressure. This results in a significant reduction in ECCS flow to the intact loop which conservatively impacts peak clad temperature for these larger break sizes.

4.4 PRAIRIE ISLAND ANALYSIS

NOTE: Reference 1 contains detailed sample analyses for 3-loop and 4-loop plants. The following is a summary of an analysis for the Prairie Island Unit 1 plant using the NOTRUMP Evaluation Model described in References 1 and 2, provided as an example only.

The analysis was completed using the NOTRUMP SBLOCA EM described in References 1 and 2. A complete SBLOCA break spectrum, including 2", 3", 4", 6" and 8" breaks, was completed to assure a limiting case was identified. The following are the plant specific inputs used in the Prairie Island NOTRUMP Evaluation Model:

Core Parameters	
100% Licensed Core Power	1650 MWt
Fuel Type	14x14, OFA V+, ZIRLO™ and/or ZIRC4 Cladding, 250 psig fill pressure, Annular Pellets
Total Core peaking Factor, F_Q	2.8
Hot Channel Enthalpy Rise Peaking Factor, $F_{\Delta H}$	2.0
Hot Assembly Average Power Factor P_{HA}	1.81
Axial Offset	+25%
K(z) Limit	1.0 for 0 ft $\leq z \leq$ 12 ft
Core Power Calorimetric Uncertainty	2%
Reactor Coolant System	
Thermal Design Flow	86,300 gpm/loop
Total Core Bypass Flow	6.0%
Nominal Vessel Average Temperature	560°F
Vessel Average Temperature Uncertainty	±4.0°F
Pressurizer Pressure	2250 psia
Pressurizer Pressure Uncertainty	±40 psi
Reactor Coolant Pump Type	Model 93A 6000HP
Reactor Coolant Pump Effective Weir Height	0.0 ft (no weir)
Reactor Protection System	
Low Pressurizer Pressure Reactor Trip Setpoint	1700 psia
Reactor Trip Signal Processing Time (Includes Rod Drop Time)	4.4 seconds
Auxiliary Feedwater System	
Maximum AFW Temperature	100°F
Minimum AFW Flow Rate	90 gpm/SG
Initiation Signal	LPP/SI

AFW Delivery Delay Time	60 seconds
Purge Volume	31.6 ft^3/loop
Steam Generators (RSGs)	
Steam Generator Tube Plugging	10%
MFW Isolation Signal	LPP/SI
MFW Isolation Delay Time	1.0 second
MFW Flow Coastdown Time	7.0 seconds
Feedwater Temperature	437.5°F
Steam Generator Safety Valve Flow Rate vs. Pressure (full open includes 3% uncertainty and 3% accumulation)	716,600 lbm/hr to 754,800 lbm/hr per valve at full open pressure
Steam Generator to MSSV ΔP	<100 psi
Initial Secondary Side Liquid Mass at 102% Power	103,555 lbm
Safety Injection	
Limiting Single Failure	1 ESF Emergency Bus
Safety Injection Water Temperature	120°F
Low Pressurizer Pressure SI Setpoint	1700 psia
Safety Injection Delay Time	28 seconds
Safety Injection Flow Rate vs. RCS Pressure (only a representative sample of pressures are presented)	**SI to loop (smaller breaks):** Pressure (psig) / Intact Loop (gpm) / Broken Loop (gpm) 0 / 310.6 / 328.4 600 / 262.2 / 277.2 1200 / 204.1 / 215.8 1800 / 116.9 / 123.6 2100 / 0 / 0 **SI to Containment (larger breaks)** Pressure (psig) / Intact Loop (gpm) / Broken Loop (gpm) 0 / 310.7 / 0 300 / 185.5 / 0 600 / 0 / 0

Accumulators	
Maximum Initial Temperature	120°F
Initial Water Volume	1270 ft^3
Total Tank Volume	2000 ft^3
Minimum Cover Gas Pressure	699.7 psia

For the analysis results summarized below, the artificial loop seal restriction was conservatively maintained for all break sizes.

The sequence of events for all break sizes is summarized below:

Event Time (sec)	2 Inch	3 Inch	4 Inch	6 Inch	8 Inch
Break initiation	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	30.6	13.5	8.8	6.7	6.1
S-Signal	30.6	13.5	8.8	6.7	6.1
Safety Injection Begins (1)	58.6	41.5	36.8	34.7	34.1
Loop Seal Clearing (2)	572	262	153	28	16
Core Uncovery	(3)	642	(3)	126	77
Accumulator Injection Begins	2352	704	351	158	87
Core Recover	(3)	851	(3)	244	140(4)

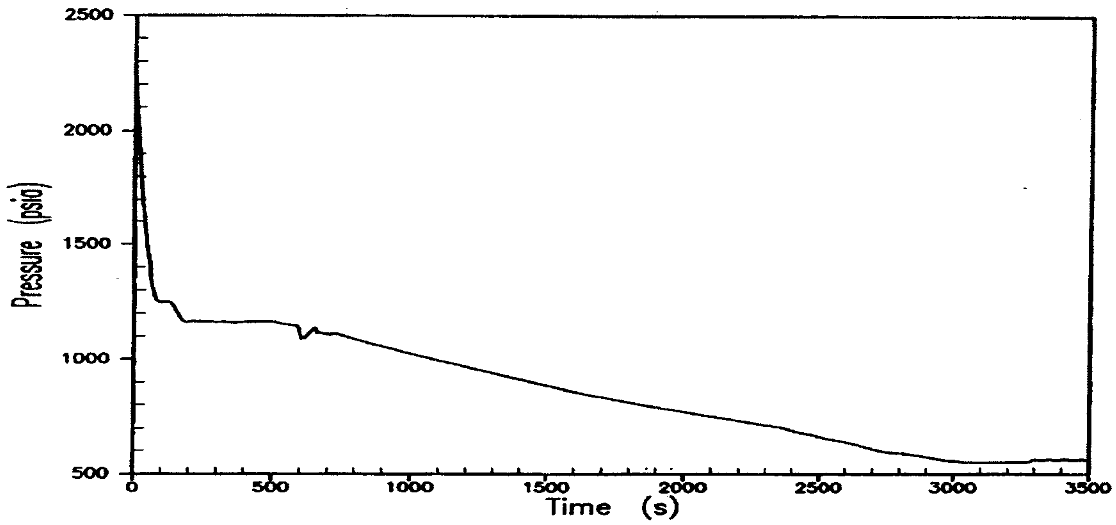
Notes:

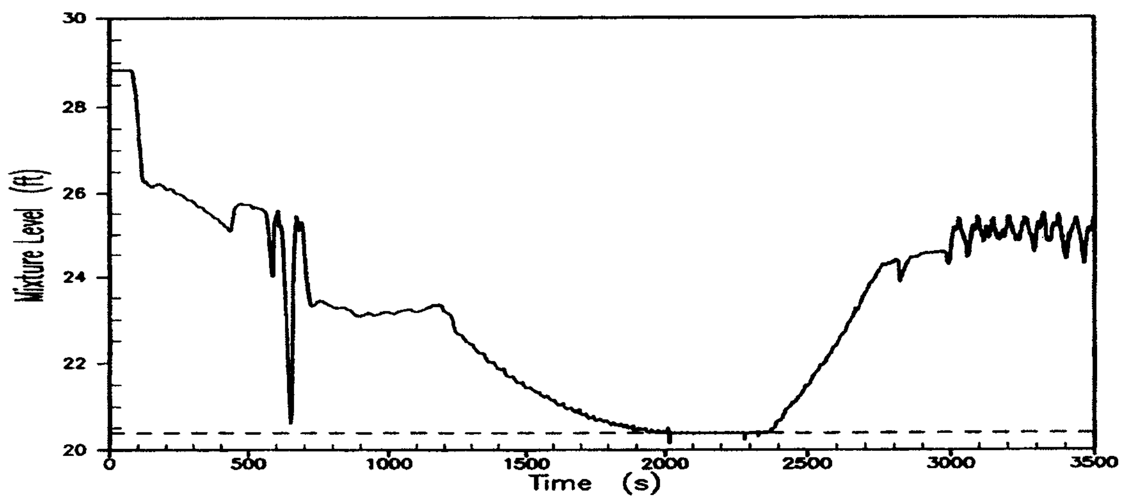
1. Safety injection (SI) begins 28.0 seconds (SI delay time) after the safety injection signal is reached.
2. Loop seal clearing is considered to occur when the broken loop, loop seal vapor flow rate is sustained above 1 lbm/s.
3. No appreciable core uncover occurred during the boil-off period, resulting in no appreciable core heatup.
4. Spurious core uncovery occurs. However, all other parameters suggest that the core will remain covered if low head safety injection is credited

The fuel rod heat-up results are summarized below:

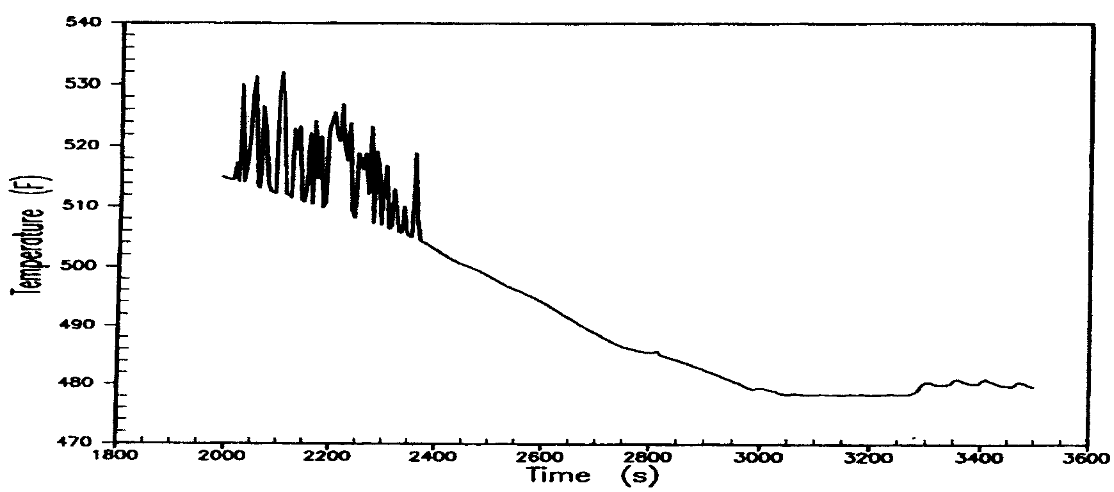
	2 Inch	3 Inch	4 Inch	6 Inch	8 Inch
Time-in-Life (MWD/MTU)	0.08	0.08	0.08	0.08	0.08
PCT (°F)	536.5	973.8	626.8	1409.2	1175.2
PCT Time (s)	2075.2	789.4	160.3	197.9	121.4
PCT Elevation (ft)	12.0	11.25	11.5	10.75	10.5
Hot Rod Burst Time (s)	N/A	N/A	N/A	N/A	N/A
Hot Rod Burst Elevation (ft)	N/A	N/A	N/A	N/A	N/A
Max. Local ZrO_2 (%)	<17%	<17%	<17%	<17%	<17%
Max. Local ZrO_2 Elev (ft)	12.0	11.25	12.0	10.75	10.5
Hot Rod Axial Avg. ZrO_2 (%)	<1%	<1%	<1%	<1%	<1%

Plots of system pressure, core mixture level and peak clad temperature for the 2 inch break are shown below:

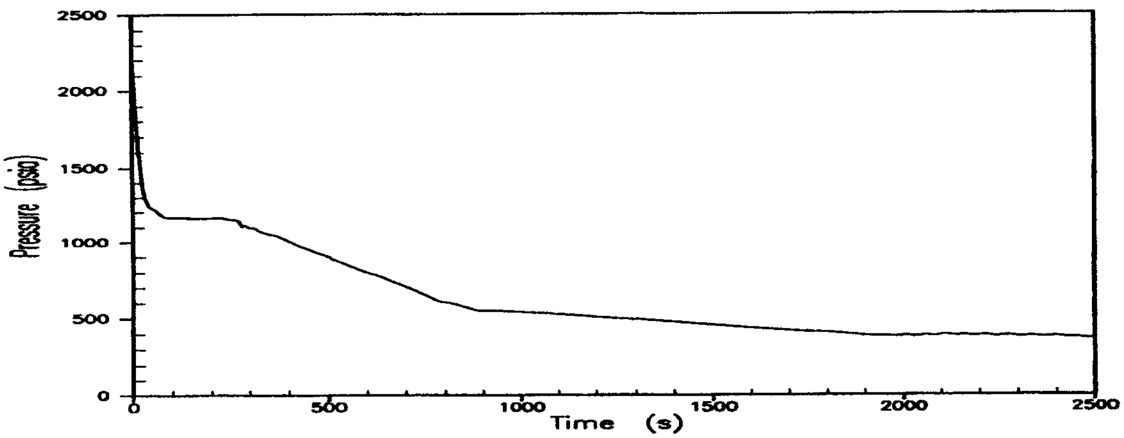


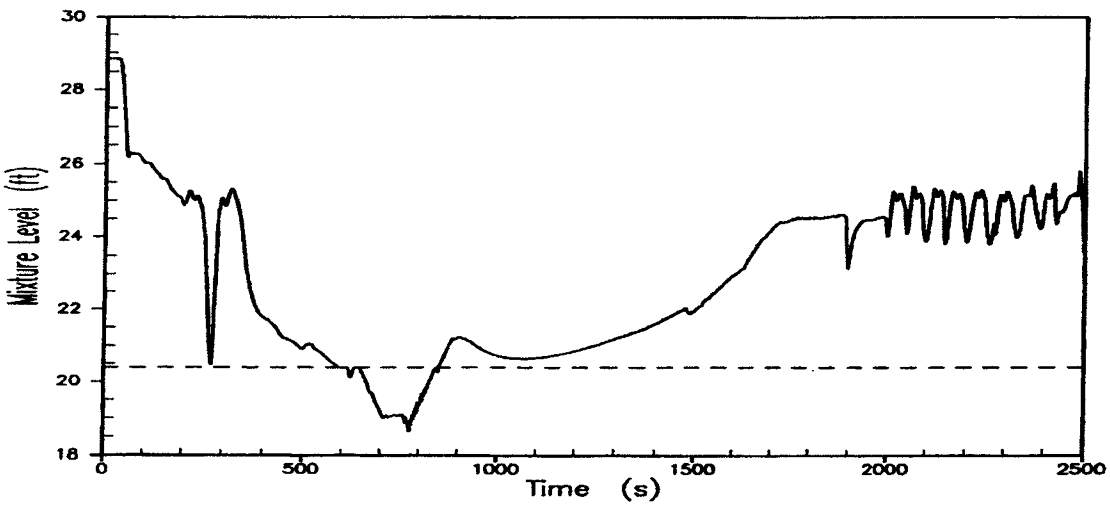


------- designates the top of the core

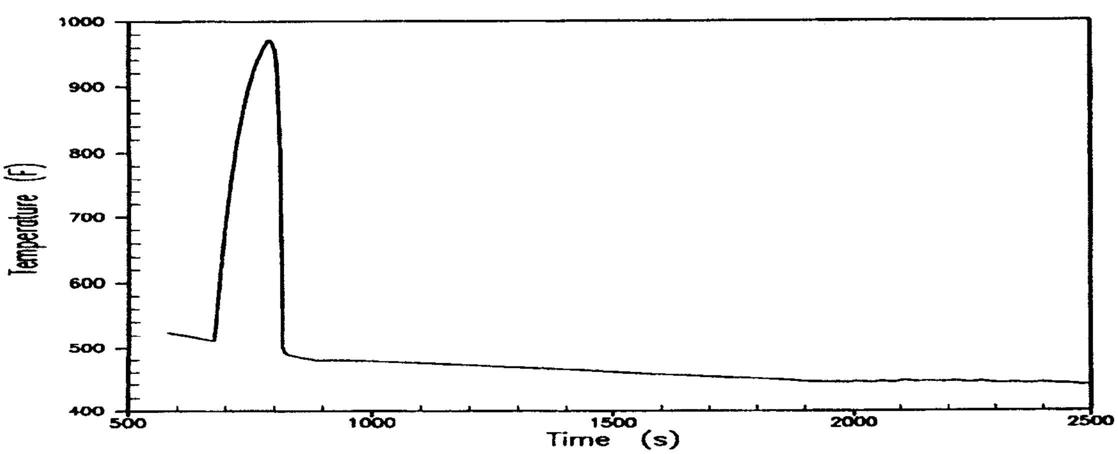


Plots of system pressure, core mixture level and peak clad temperature for the 3 inch break are shown below:

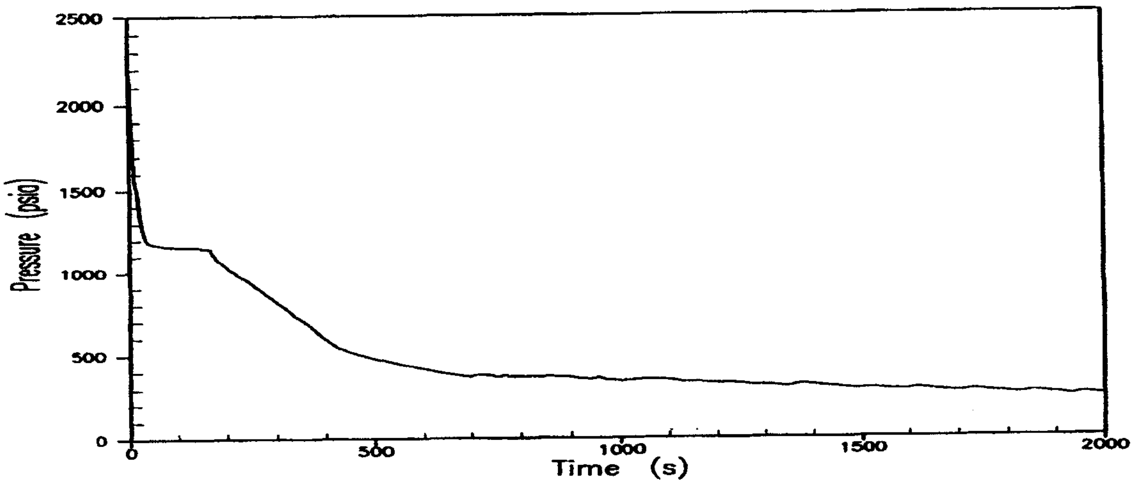


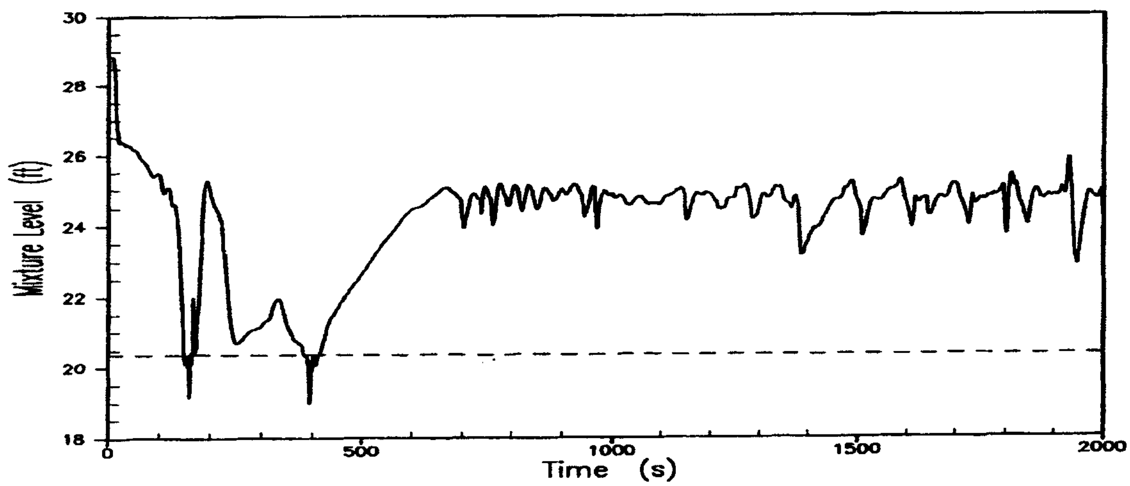


------- designates the top of the core

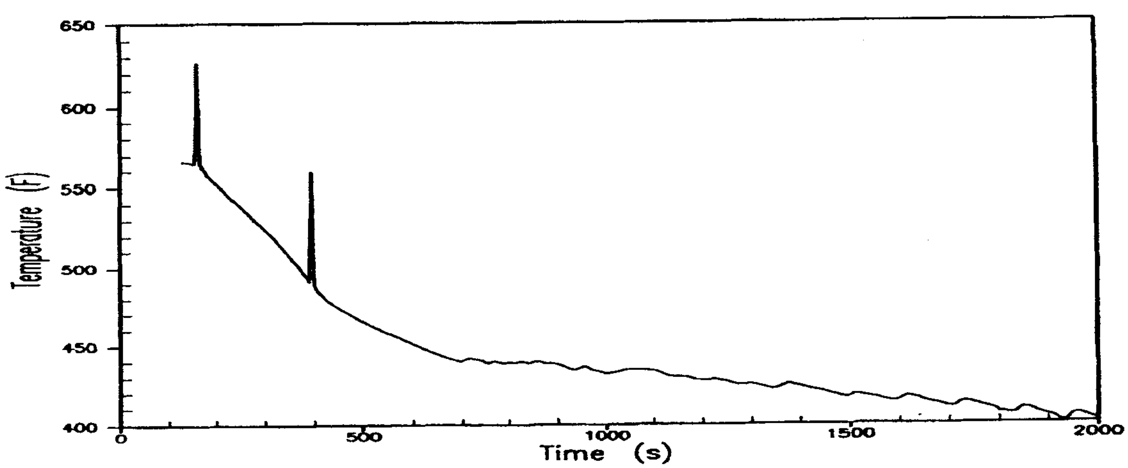


Plots of system pressure, core mixture level and peak clad temperature for the 4 inch break are shown below:

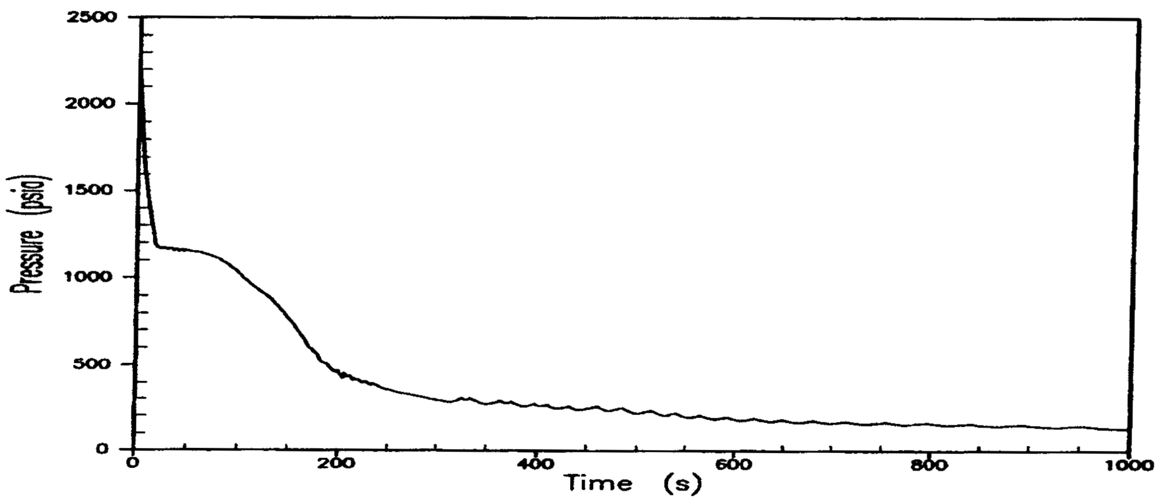


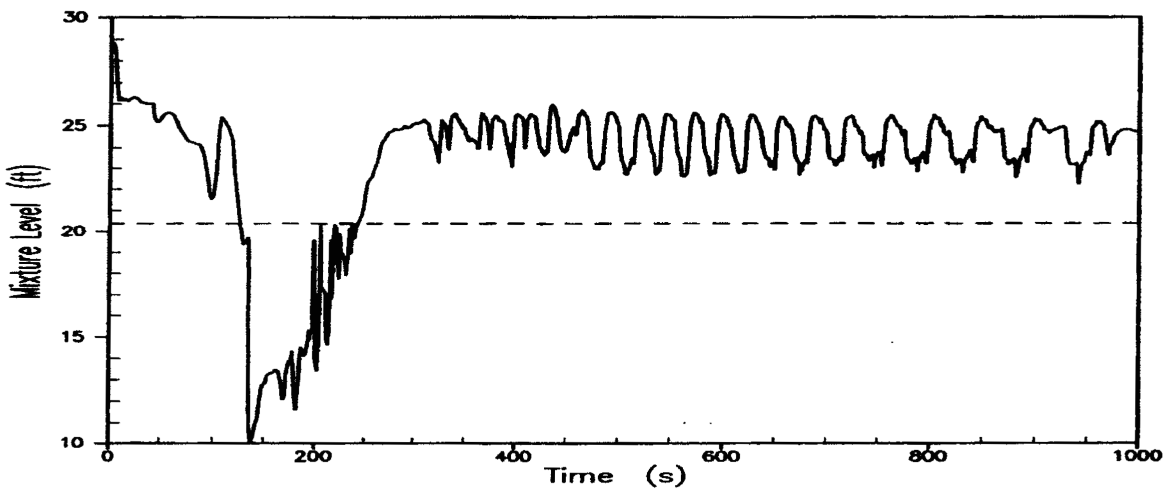


------- designates the top of the core

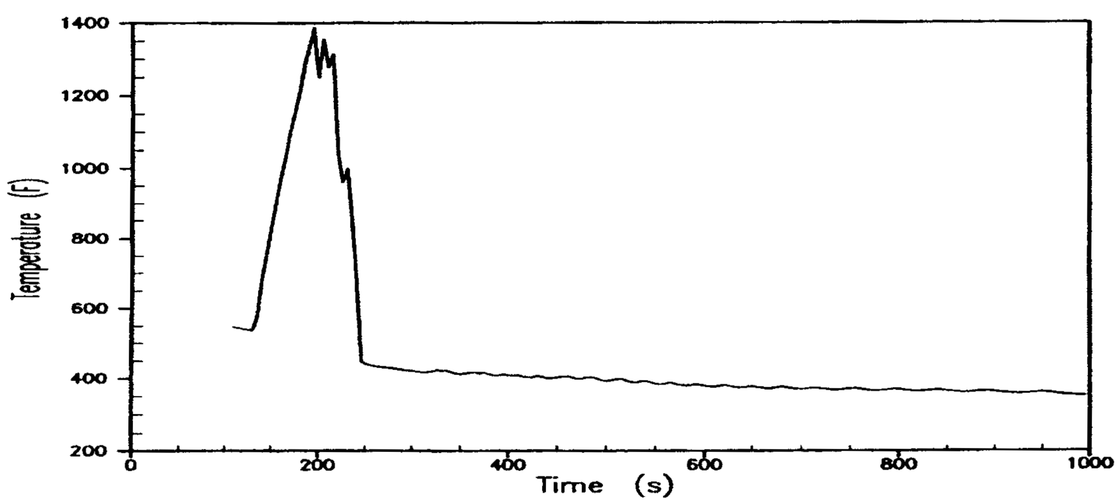


Plots of system pressure, core mixture level and peak clad temperature for the 6 inch break are shown below:

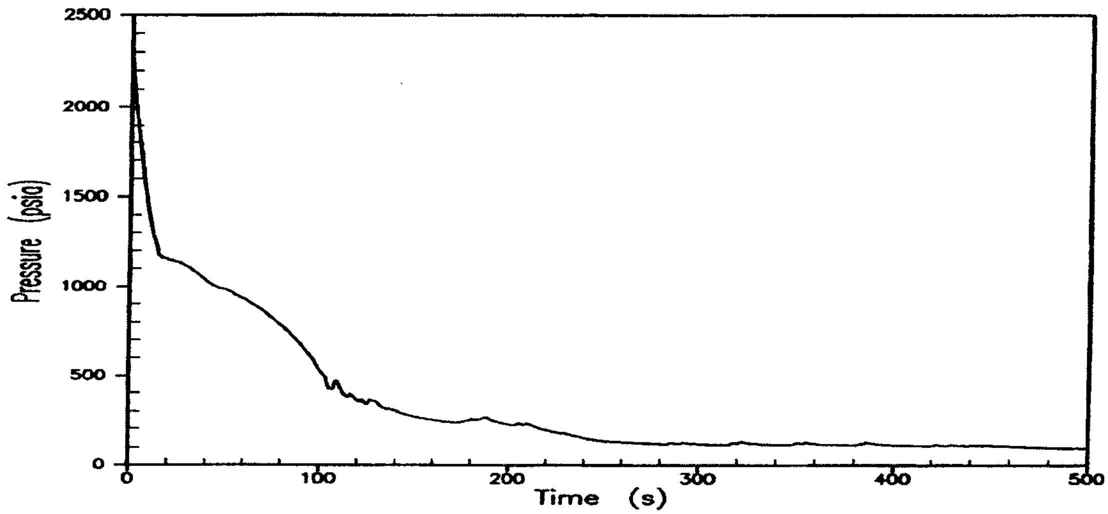


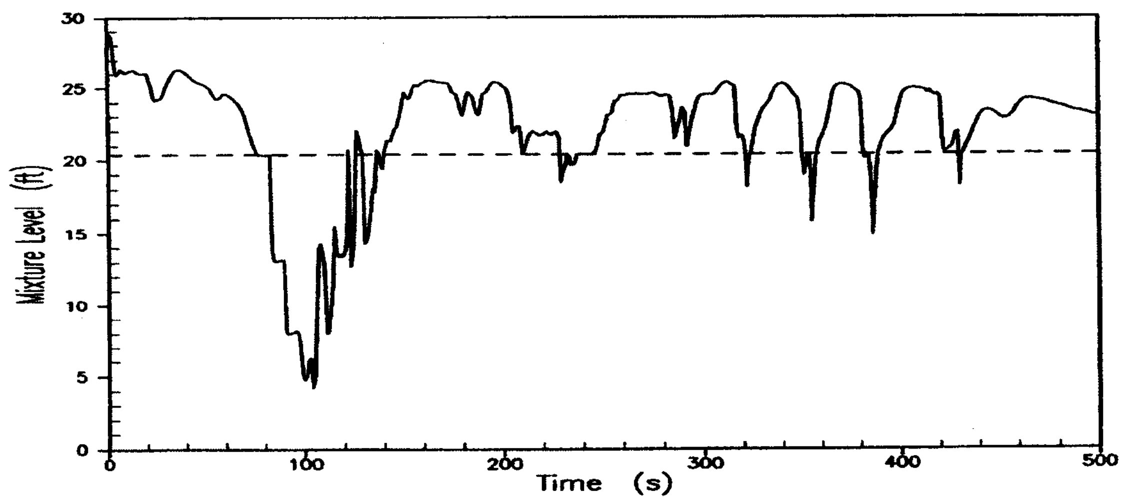


------- designates the top of the core

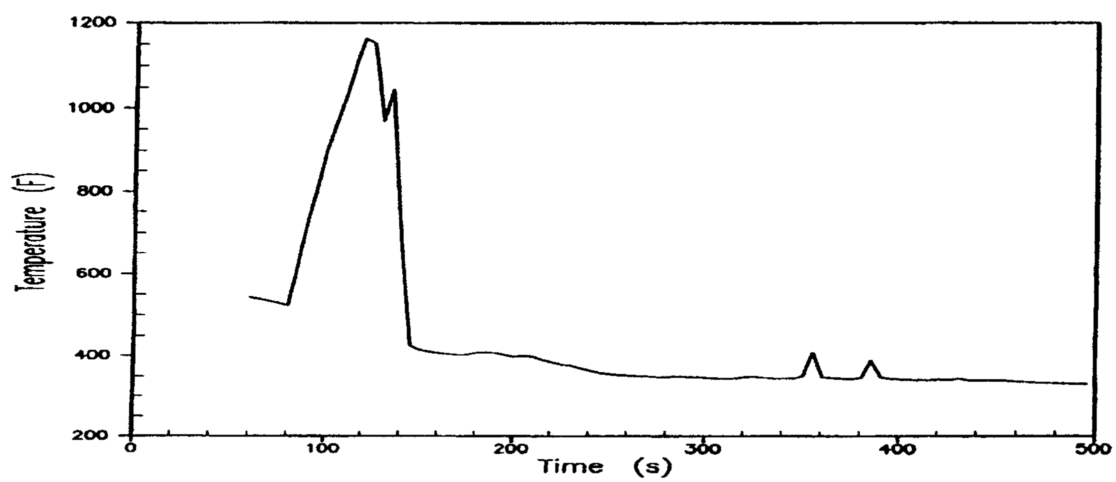


Plots of system pressure, core mixture level and peak clad temperature for the 8 inch break are shown below:





------- designates the top of the core



For comparison, an additional analysis was completed with the artificial loop seal restriction removed for the limiting larger 6 inch break. The results of this analysis are shown below:

The sequence of events is summarized below:

Event Time (sec)	6 Inch
Break initiation	0.0
Reactor Trip Signal	6.7
S-Signal	6.7
Safety Injection Begins (1)	34.7
Loop Seal Clearing (2)	28
Core Uncovery	(3)
Accumulator Injection Begins	154
Core Recovery	(3)

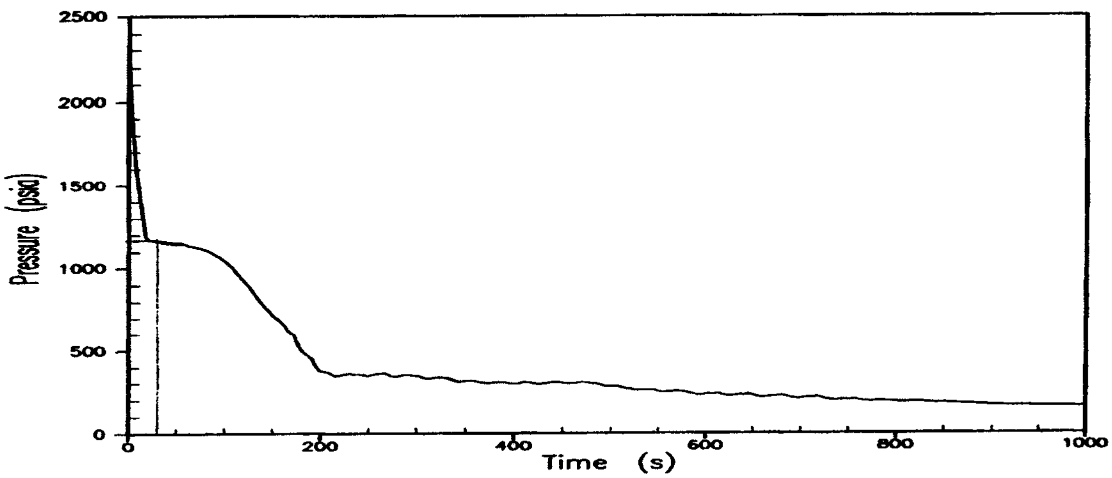
Notes:

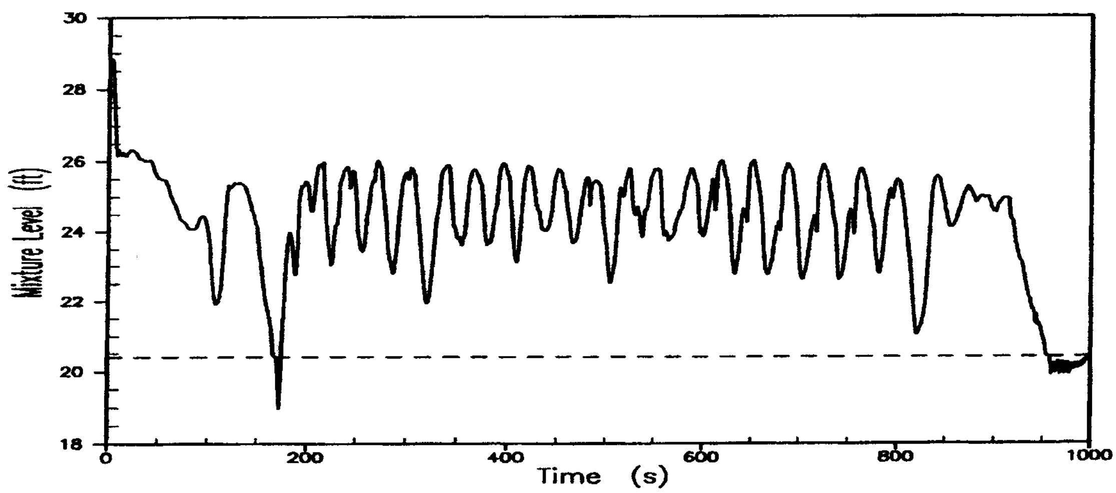
1. Safety injection (SI) begins 28.0 seconds (SI delay time) after the safety injection signal is reached.
2. Loop seal clearing is considered to occur when the broken loop, loop seal vapor flow rate is sustained above 1 lbm/s.
3. No appreciable core uncovery is expected to occur if low head safety injection is credited.

The fuel rod heat-up results are summarized below:

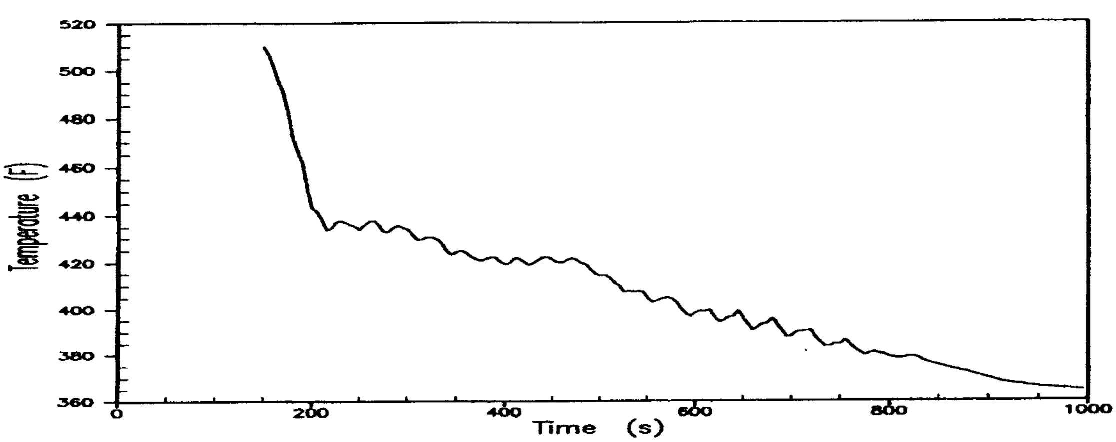
	6 Inch
Time-in-Life (MWD/MTU)	0.08
PCT (°F)	556.5
PCT Time (s)	173.5
PCT Elevation (ft)	11.5
Hot Rod Burst Time (s)	N/A
Hot Rod Burst Elevation (ft)	N/A
Max. Local ZrO_2 (%)	<17%
Max. Local ZrO_2 Elev (ft)	12.0
Hot Rod Axial Avg. ZrO_2 (%)	<1%

Plots of system pressure, core mixture level and peak clad temperature for the 6 inch break without the artificial loop seal restriction are shown below:





------- designates the top of the core



5.0 REGULATORY ANALYSIS

5.1 Significant Safety Hazards Assessment

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

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

   Response: No

   The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the approved NOTRUMP SBLOCA Evaluation Model described in Westinghouse WCAP 10054-P-A Addendum 2 Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model".

   The methodology used to perform small break loss of coolant accident (SBLOCA) analyses is not an accident initiator, thus changing the methodology does not increase the probability of an accident.

   The fuel heat-up results generated by the proposed methodology will be utilized to demonstrate that the loss of coolant accident (LOCA) criteria for design basis for fission product barriers as described in 10 CFR 50.46 are not exceeded. The proposed methodology does not alter the nuclear reactor core, reactor coolant system, or equipment used directly in mitigation of a Small Break LOCA, thus radioactive releases due to a SBLOCA accident are not affected by the proposed change in analysis methodology. Therefore, this change does not increase the consequences of an accident previously evaluated.

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

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

   Response: No

   The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the approved NOTRUMP SBLOCA Evaluation Model described in Westinghouse WCAP 10054-P-A Addendum 2 Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the

NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model".

The analysis of a SBLOCA accident using the proposed methodology does not alter the nuclear reactor core, reactor coolant system, or equipment used directly in mitigation of a Small Break LOCA.

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

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the approved NOTRUMP SBLOCA Evaluation Model described in Westinghouse WCAP 10054-P-A Addendum 2 Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model".

The methodology in the proposed licensing basis change has previously been reviewed and approved by the Nuclear Regulatory Commission as a conservative methodology. The Prairie Island configuration is representative of the modeling used in the methodology. Therefore, the proposed licensing basis change will result in a conservative calculation of fuel conditions following a SBLOCA event. This will ensure that there is no reduction in the margin of safety for Prairie Island SBLOCA analyses that utilize this methodology.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, the Nuclear Management Company concludes that the proposed submittal presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements

General Design Criteria

The construction of the Prairie Island Nuclear generating Plant was significantly complete prior to the issuance of 10 CFR 50 Appendix A General Design Criteria. The Prairie Island Nuclear Generating Plant was designed and constructed to comply with the Atomic Energy Commission General Design Criteria as proposed on July 10, 1967 as described in the Updated Safety Analysis Report. The proposed Atomic Energy Commission General Design Criteria 44 provides emergency core cooling system (ECCS) design guidance for the Reactor Core Response to a Loss of Coolant Accident.

> Criterion 44 – Emergency Core Cooling Systems Capability
>
> "At least two emergency core cooling systems, preferably of different design principles, each with a capability of accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes and breaks in the reactor coolant pressure boundary, including the double ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss of coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss of coolant accident and is not lost during the entire period this function is required following the accident."

The proposed amendment does not physically impact the engineered safeguards equipment or core design. The proposed amendment will alter the methodology used to determine the core response to a small break LOCA. The new SBLOCA methodology will conservatively demonstrate that one train of an emergency core cooling system can adequately cool the core for a range of limiting smaller break LOCA events.

10 CFR 50.46

The revised methodology will be used to determine the core response following a SBLOCA event and to assure compliance with the post LOCA acceptance criteria specified in 10 CFR 50.46.

NUREGs

The proposed changes to the SBLOCA analyses methodology were not implemented to address any new NUREGs. The original NOTRUMP evaluation model documented in Reference 2 was developed to address NUREG-0611, NUREG-0623 and NUREG-0737. The original modeling changes completed to address these NUREG's have not changed with the implementation of the amended SBLOCA methodology described in Reference 1.

Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the commissions regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

The Nuclear Management Company has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1) WCAP 10054-P-A, Addendum 2, Rev 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

2) WCAP 10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code" and WCAP 10079-P-A, "NOTRUMP; A Nodal Transient Small Break and General Network Code," August 1985.

3) Westinghouse Letter NSBU-NRC-00-5972 from H.A.Sepp of Westinghouse to Samuel J. Collins of the NRC, dated June 30, 2000.

4) WCAP 11145-P-A "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," October 1986.