Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379-2000

February 7, 1997

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION - TECHNICAL SPECIFICATION (TS) CHANGE REQUEST 96-01 ON CONVERSION TO COGEMA FUEL (TAC NOS. M95144 AND M95145)

Reference: NRC letter to TVA dated January 8, 1997 on the above subject

In accordance with your request for additional information, please find enclosed, TVA's responses to the NRC questions on TS 96-01. Enclosure 1 provides the proprietary version of the response to the above reference. Enclosure 2 provides the nonproprietary version.

Since Enclosure 1 provides information which is proprietary to Framatome Cogema Fuel (FCF), Enclosure 3 contains the application for withholding and affidavit signed by FCF, the owner of the information. The application for withholding and the affidavit set forth the basis on which the information may be withheld from public disclosure by NRC and address with specificity the considerations listed in Section 2.790, Paragraph (b)(4) of the NRC regulations.

Accordingly, it is respectfully requested that information which is proprietary to FCF be withheld from public disclosure in accordance with 10CFR, Section 2.790, of the NRC regulations.

A001

ltr encl
1 1 w/o prop

change distribution: NRC PDR

Correspondence with respect to the proprietary aspects of Enclosure 1 or the supporting FCF affidavit should be addressed to J. H. Taylor, Manager of Licensing Services, Framatome Cogema Fuels, P.O. Box 10935, Lynchburg, Virginia 24506-0935.

Please direct questions concerning this issue to Jim Smith at (423) 843-6672.

Sincerely,

R. H. Shell

R. H. Shell
Site Licensing and Industry Affairs Manager

Enclosures
cc (Enclosures):
Mr. R. W. Hernan, Project Manager
Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy-Daisy, Tennessee 37379-3624

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323-2711

ENCLOSURE 2

NRC RAI ON TS 96-01

DATED

JANUARY 8, 1997

NONPROPRIETARY VERSION

OF

RESPONSE TO RAI

REQUEST FOR ADDITIONAL INFORMATION
TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NUMBERS 50-327 AND 50-328

1. Revision 2 to the BAW-10168 evaluation model was modified during the staff review with regard to the Moody break flow model and discharge coefficient. Please verify that the analysis that was used to support this Sequoyah Nuclear Plant (SQN) fuel change and Technical Specification (TS) amendment was performed using the approved model.

Response

Several modifications were made to FCF's small break LOCA evaluation model, BAW-10168, Revision 2, in response to the NRC review of the document. Chief among these modifications - with respect to the SQN small break LOCA analyses - are changes to the break discharge coefficient and the nodalization of the pump suction piping (the original evaluation model implemented a variable 0.7/1.0 discharge coefficient, the approved model requires the use of a constant 1.0 discharge coefficient and the upflow side of the pump suction piping is more finely divided to better predict the occurrence of loop seal clearing). The approved model was utilized to revise the SQN small break analyses. The results of the reanalysis are attached for your consideration. Note that the primary changes to the results are: (1) a shift in the worst case break from a 3-inch break to a 2.5-inch break and (2) an increase in PCT of about 25°F.

5.9.2 Small Break LOCA Evaluation Model

RELAP5 is used to predict the reactor coolant system thermal-hydraulic responses to a small break LOCA. The code has been approved by the NRC for licensing application and is documented in detail in Reference 5-3. RCS nodalization is based on the model described in Volume II of the NRC-approved BWNT RSG LOCA EM, Reference 5-1. Nodal diagrams of the SQN small break LOCA model are presented in Figures 5.9-1 and 5.9-2. The small break LOCA model is similar to that used in the large break analyses.

The reactor core is divided radially into two regions similar to that of the LBLOCA model; one region represents the hot fuel assembly and the other represents the remainder of the core. The core is further divided into twenty axial segments. Cross-flow junctions connect hot assembly fluid nodes with the adjacent "average" assembly nodes. This arrangement allows the computation of hot assembly cladding and vapor temperatures with limited influence by coolant from the average core and provides resolution of the mixture level to within approximately 0.5 foot. Initial fuel pin parameters are calculated with the TACO3 computer code (Reference 5-6). The reactor vessel downcomer and upper plenum regions are represented in finer axial detail than those of the large break LOCA model to give a better representation of the void distribution that affects the system hydrostatic balance.

In the small break LOCA model, the RCS is subdivided into two flow loops. One loop represents the broken loop and the other represents the three intact loops. The pressurizer is attached to the hot leg of the composite intact loop. The nodalization is similar to that of the large break LOCA model.

The steam generator tube region is divided into two radial regions. One region represents the shortest half of the tubes and the other region represents the remainder of the tubes. This provides sufficient modeling accuracy to simulate tube draining effects; tube draining can be sensitive to tube length.

The reactor coolant pump suction nodalization has been altered relative to the large break LOCA model to produce an accurate hydrodynamic representation of loop seal clearing. Two additional nodes are added to the downside of the pump suction pipe and three nodes to the riser section. This allows finer resolution of the void distributions and elevation heads that control the occurrence and timing of loop seal clearing.

The bottom elevation of the lowest node of the intact loop pump suction piping is artificially extended one foot below the corresponding node in the broken loop. This preferentially promotes the clearing of only the broken loop. An RCS

configuration characterized by a single clear loop and three intact loops conservatively restricts steam flow to the break. The added restriction can result in worsened core conditions and the potential uncovering of the core.

Both the broken loop and the intact loop reactor coolant pump discharge piping are modeled as four nodes. In the large break model, the intact loop is modeled as one node. Using four nodes provides an accurate simulation of the hydrodynamic effects of the ECCS injection.

The computer code options and generic input requirements used in the small break portion of the BWNT RSG LOCA EM are summarized in Volume II, Tables 9-1 and 9-2, of Reference 5-1. In addition, a minor BWNT RSG LOCA EM change is made to alleviate nonphysical conditions in the artificial leak sink volume: the selection of phase-equilibrium modeling in the leak volume. This change permits relatively stable conditions in the leak volume and has no significant impact on transient results.

The small break LOCA model is constructed within the guidelines of the BWNT RSG LOCA EM established in Reference 5-1. It has been developed utilizing the RELAP5 large break LOCA model, described in Section 5.2 of this report, as a basis. The small break model adheres to the requirements of 10CFR50 Appendix K and contains demonstrated conservatism for the evaluation of ECCS mitigation of a postulated small break at SQN.

5.9.3 Inputs and Assumptions

The major plant operating parameters used in the SQN small break LOCA analyses follow. These inputs are similar to those utilized in the large break analyses.

1. Power Level - The plant is assumed to be operating in steady-state at 3479 MWt (102% of 3411 MWt).

2. Total System Flow - The initial Reactor Coolant System (RCS) flow is 348,000 gpm.

3. Fuel Parameters - The initial fuel pin parameters are taken from TACO3 (Reference 5-6) runs performed for BOL fuel conditions.

4. ECCS - The ECCS flows are based on the assumption of a single active failure. A single train of ECCS is modeled as described in Volume II, Section 4.3.2.2, of the BWNT RSG LOCA EM, Reference 5-1. For the case of a centrifugal charging line break, charging flow is assumed to be spilled to the containment.

5. Total Peaking Factor (F_Q) - The maximum total peaking factor assumed by this analysis is 2.5. The hot assembly peaking for small break analysis is illustrated in Figure 5.9-3.

6. The moderator density reactivity coefficient is based on BOL conditions to minimize negative reactivity.

7. The cladding rupture model is based on NUREG-0630.

5.9.4 Analysis Results

In the SQN small break LOCA analyses, five break cases were considered independently to predict core and system responses over a spectrum of break sizes. Small break spectrum results in Volume II, Appendix A, Reference 5-1, indicate that break areas corresponding to 2- to 6-inch diameters produce the most severe core depression. Breaks of 1.5-, 2.5-, 3- and 5-inch diameters in the bottom of the reactor coolant pump discharge piping were analyzed for SQN. In addition, a 1.34-inch diameter centrifugal charging line break, located in the top of the piping, was analyzed.

Table 5.9-1 presents time sequence of events for each of the small break LOCA cases. Fuel thermal responses for the hot pin are included in Table 5.9-2. Parameters of interest to the small break analyses are shown in Figures 5.9-4 through 5.9-28. There are five sets of figures, each set contains five plots. The five figures of each set show (1) the RCS pressure, (2) the break flow rate, (3) loop seal levels in the pump suction downflow and upflow pipes, (4) core collapsed level, and (5) hot spot cladding temperature.

A relatively slow depressurization rate occurs in the 2.5-inch break case. The core does uncover, making the 2.5-inch break the most limiting case of the small break spectrum analysis. The resulting peak cladding temperature (PCT) is 1098°F. Core metal-water reaction for the 2.5-inch break is negligible because the cladding oxidation rate is not significant below about 1500°F.

For breaks smaller than 2.5 inches in diameter, core cooling is maintained by a combination of steam relief at the break and reflux cooling in the steam generator. The core does not uncover for these smaller breaks. For larger breaks such as 3- and 5-inch diameters, the rapid depressurization rate following loop seal clearing has two positive effects on the core level. One is increased ECCS flow, and the other is increased core level swell. No core heatup was predicted for these break sizes.

The centrifugal charging injection line break is postulated to allow examination of a small break LOCA that is characterized by a degradation of high pressure injection. The break size is insufficient to allow significant depressurization of the RCS. Coolant addition in the progression of the transient is, therefore, governed by the high pressure injection alone. With a broken injection line, a large portion of the ECCS flow associated with the centrifugal charging flow is directed to the break. To ensure a conservative result, all of the charging flow is assumed lost to the break, and the transient is mitigated by safety injection pumps only. The results of the charging line break indicate that the core remains covered by the mixture level and that no core heatup occurs.

The small break analyses are terminated when the break flow rate is exceeded by the ECCS flow rate. Note that the collapsed liquid level at the end of the transient may still be below the top of the core. The core mixture levels at the end of the analyses are, however, above the top of the active core and the RCS pressure is still falling. A steady increase in ECCS injection and continued core cooling is therefore assured.

5.9.5 Compliance to 10CFR50.46

The small break calculations directly demonstrate compliance to two of the criteria of 10CFR50.46 and serve as the basis for demonstrating compliance with two others. As seen in the figures and in Table 5.9-2, the highest peak cladding temperature, 1098°F, and the highest local oxidation, about 0.008%, are well below the 2200°F and 17% criteria.

The whole-core oxidation criterion of 1% cladding reaction is met as well in the small break LOCA analyses. Whole-core oxidation will be much less than the peak local oxidation figure of 0.008%. Whole-core oxidation associated with small break LOCAs, utilizing the assumptions and inputs as documented above, is negligible.

The fourth acceptance criterion of 10CFR50.46 states that calculated changes in core geometry shall be such that the core remains amenable to cooling. The calculations in Section 5.9 directly assess the alterations in core geometry that result from the LOCA, at the most severe location in the core. These calculations demonstrate that the fuel pin cooled successfully. Further, for SQN, no hot assembly cladding ruptures occurred in any of the small break LOCA cases. Therefore, the assembly retains its pin-coolant channel-pin-coolant channel arrangement and is capable of being cooled.

The fifth acceptance criterion of 10CFR50.46 states that the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended

period of time required by the long-lived radioactivity remaining in the core. Successful initial operation of the ECCS is shown by demonstrating that the core is quenched and the cladding temperature is returned to near saturation temperature.

Compliance to the long-term cooling criterion is demonstrated for the systems and components specific to SQN in the FSAR and is not related to the fuel design. The initial phase of core cooling has been shown to result in low cladding and fuel temperatures. A pumped injection system capable of recirculation is available and operated by the plant to provide extended coolant injection. Therefore, compliance with the long-term cooling criterion of 10CFR50.46 has been demonstrated.

Table 5.9-1 Small Break LOCA Time Sequence Of Events

Events, Sec	1.5-inch	2.5-inch	3-inch	5-inch	CCI
Break Initiation	0.0	0.0	0.0	0.0	0.0
Reactor Scram	91.5	33.4	23.4	10.2	115.0
Coolant Pump Coastdown	91.5	33.4	23.4	10.2	115.0
Steamline Isolation	91.5	33.4	23.4	10.2	115.0
Feedwater Isolation	101.5	43.4	33.4	20.2	125.0
ECCS Injection	141.6	77.7	65.5	47.1	166.6
Loop Seal Clearing	NA	1129.4	704.2	236.4	NA
Peak Cladding Temperature	NA	3042.4	NA	NA	NA
Accumulator Injection	NA	2895.0	975.0	435.0	NA

Table 5.9-2 Small Break LOCA Results

Results	2.5-inch
Peak Cladding Temperature, °F	1098
Peak Temperature Location, ft	10.9
Rupture Time, sec	NA
Rupture Location, ft	NA
Maximum Local M/W Reaction, %	-0.008
Total M/W Reaction, %	<0.008

Figure 5.9-1 RELAP5 Small Break LOCA Model
Sequoyah Noding for Reactor Vessel and Core

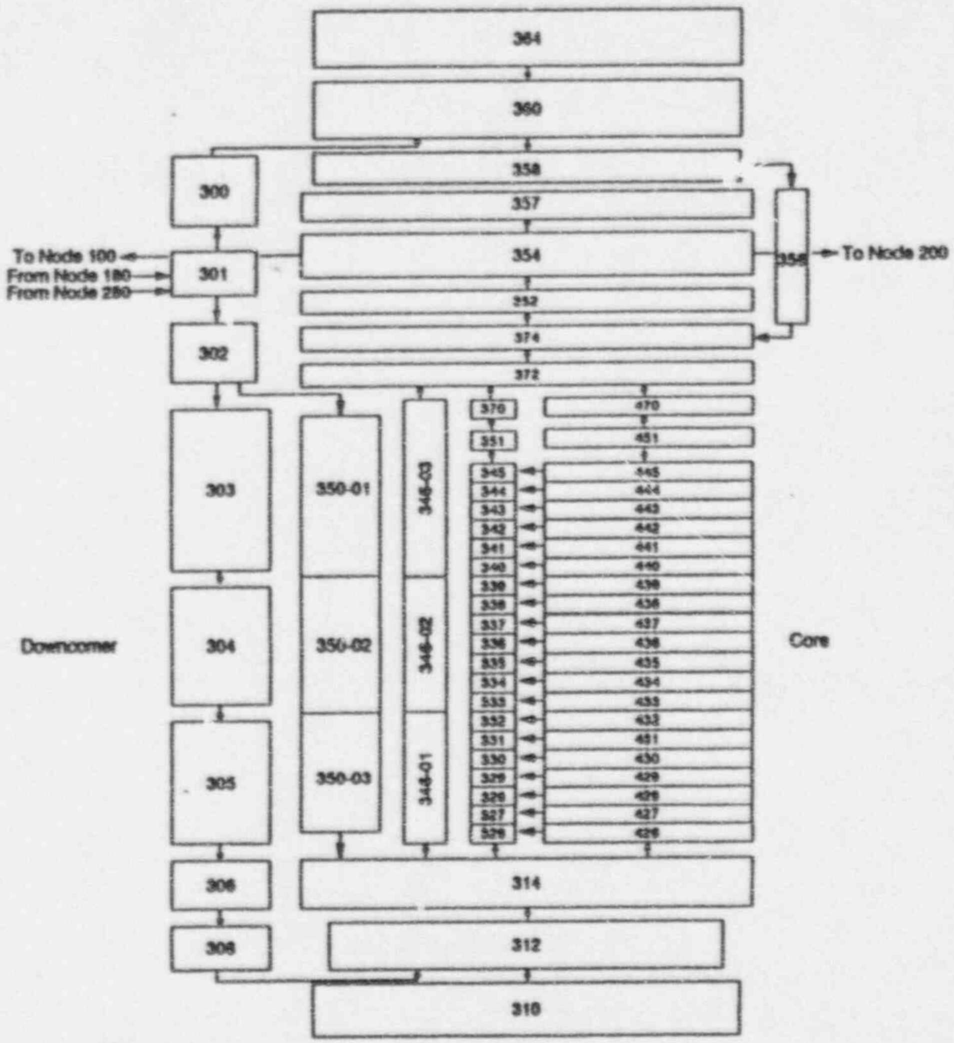


Figure 5.9-2 RELAP5 Small Break LOCA Model
Sequoyah Noding for Primary Loop with Model 51 SGs

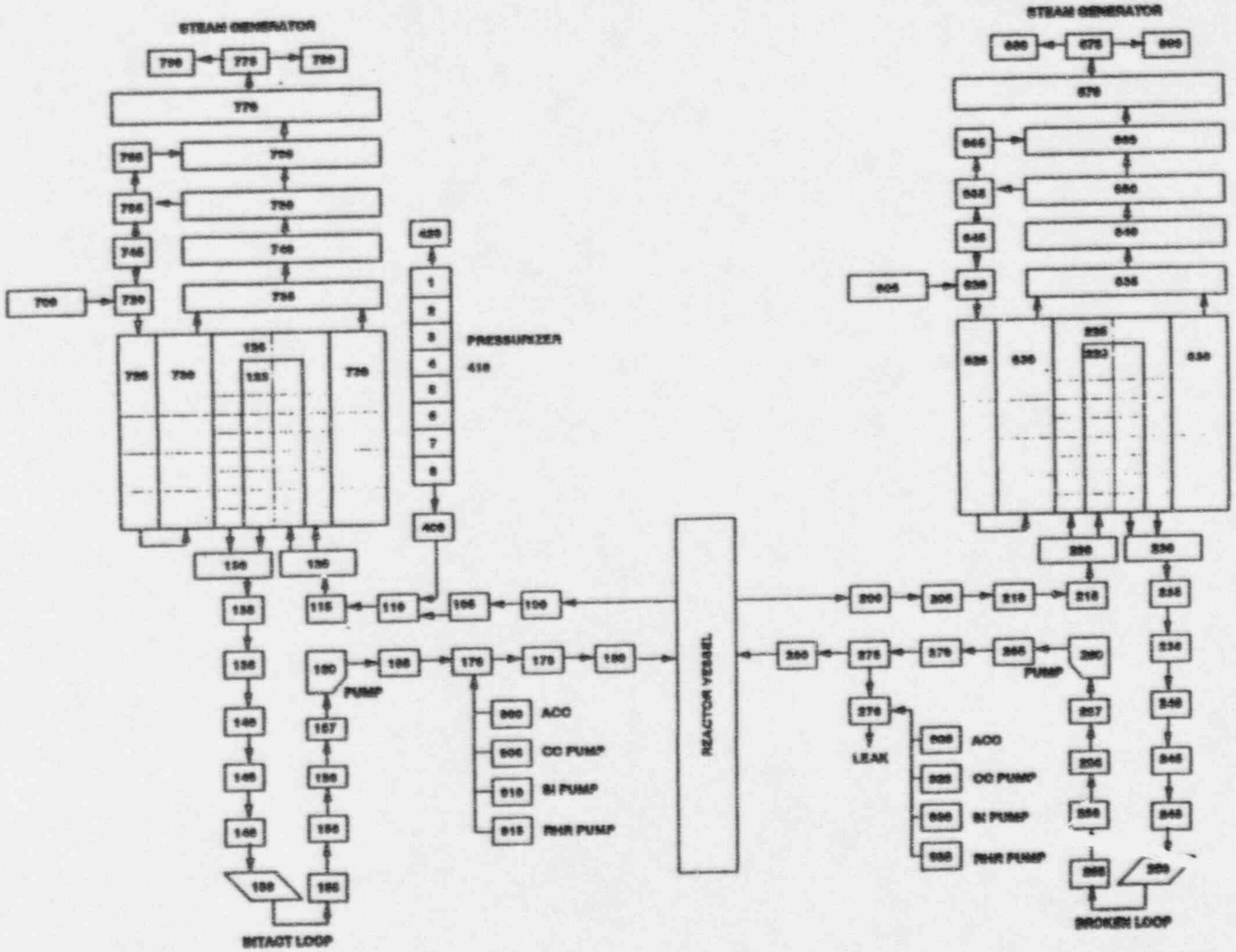


Figure 5.9-3 Small Break LOCA Study
Hot Channel Power Profile

Figure 5.9-4 1.5-inch Pump Discharge Break
Primary System Pressure

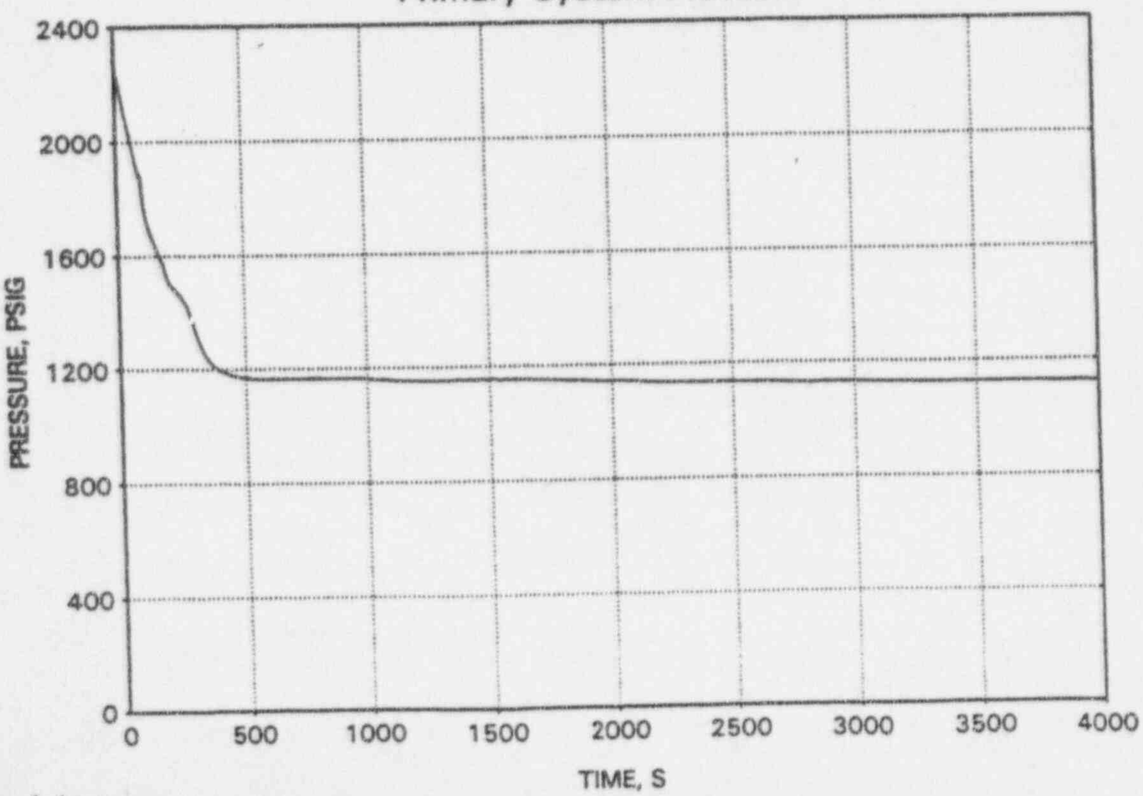


Figure 5.9-5 1.5-inch Pump Discharge Break
Leak Flow Rate

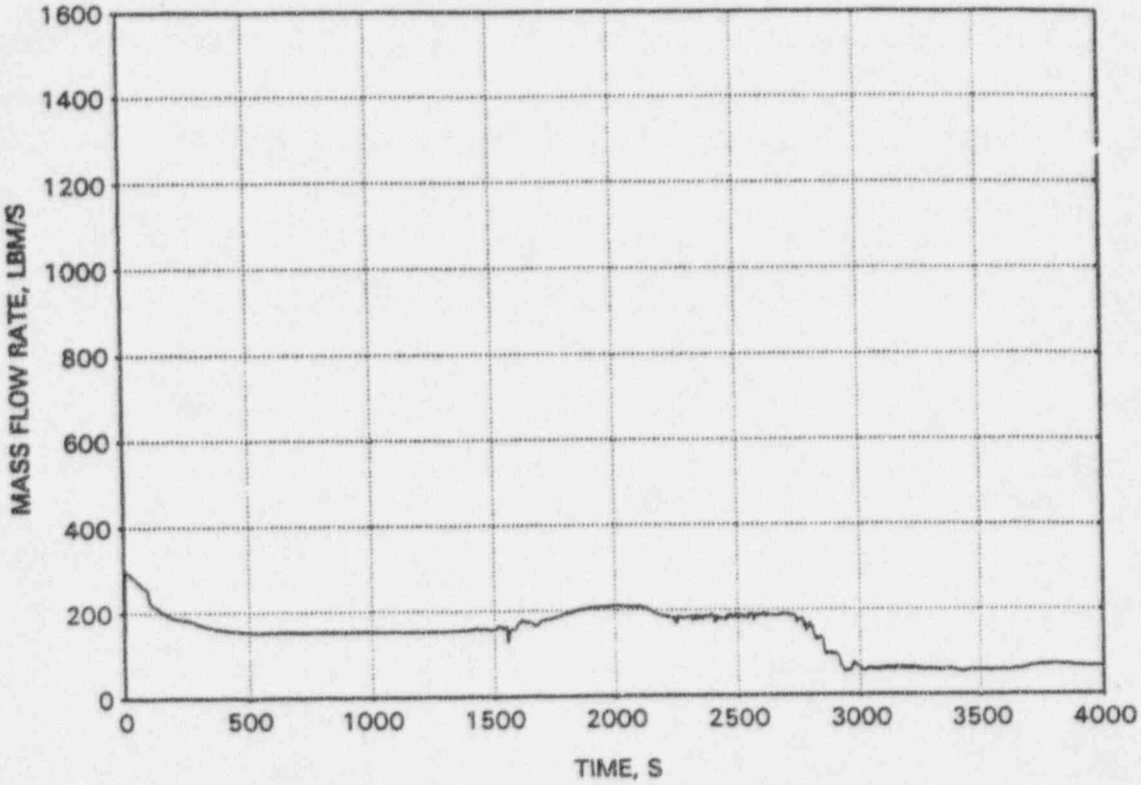


Figure 5.9-6 1.5-inch Pump Discharge Break
Pump Suction Loop Seal Levels

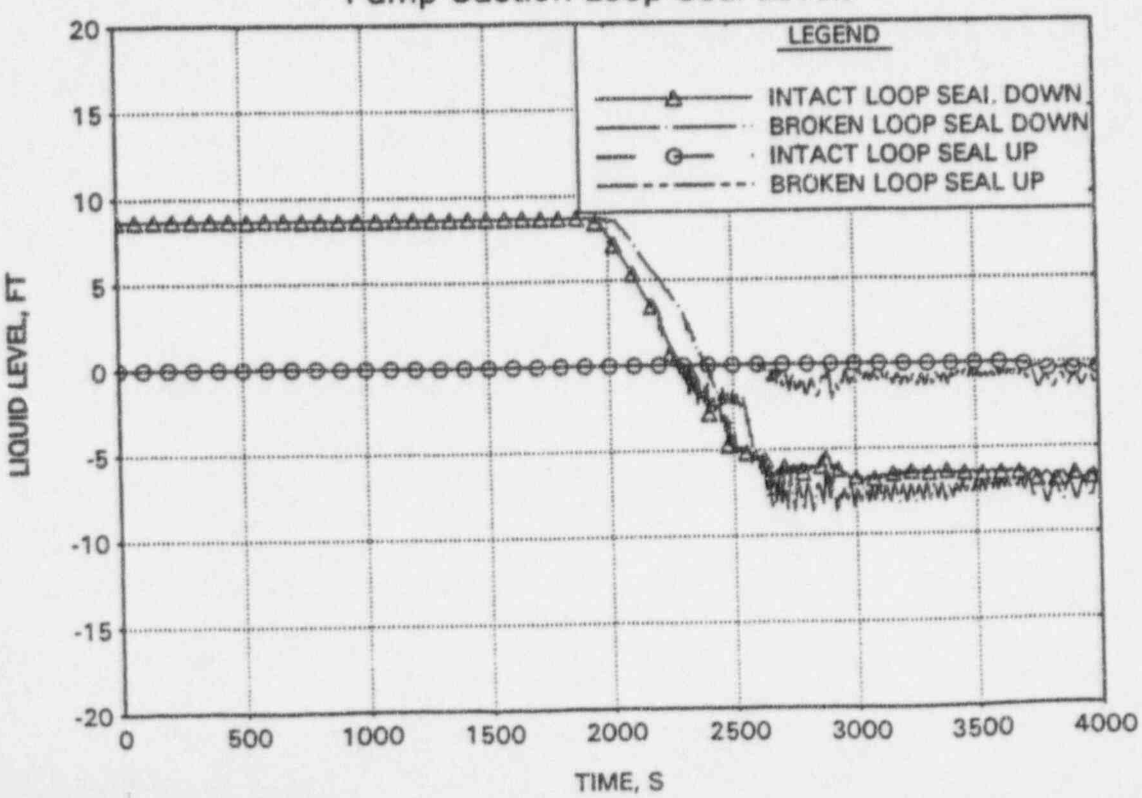


Figure 5.9-7 1.5-inch Pump Discharge Break
Core Collapsed Level

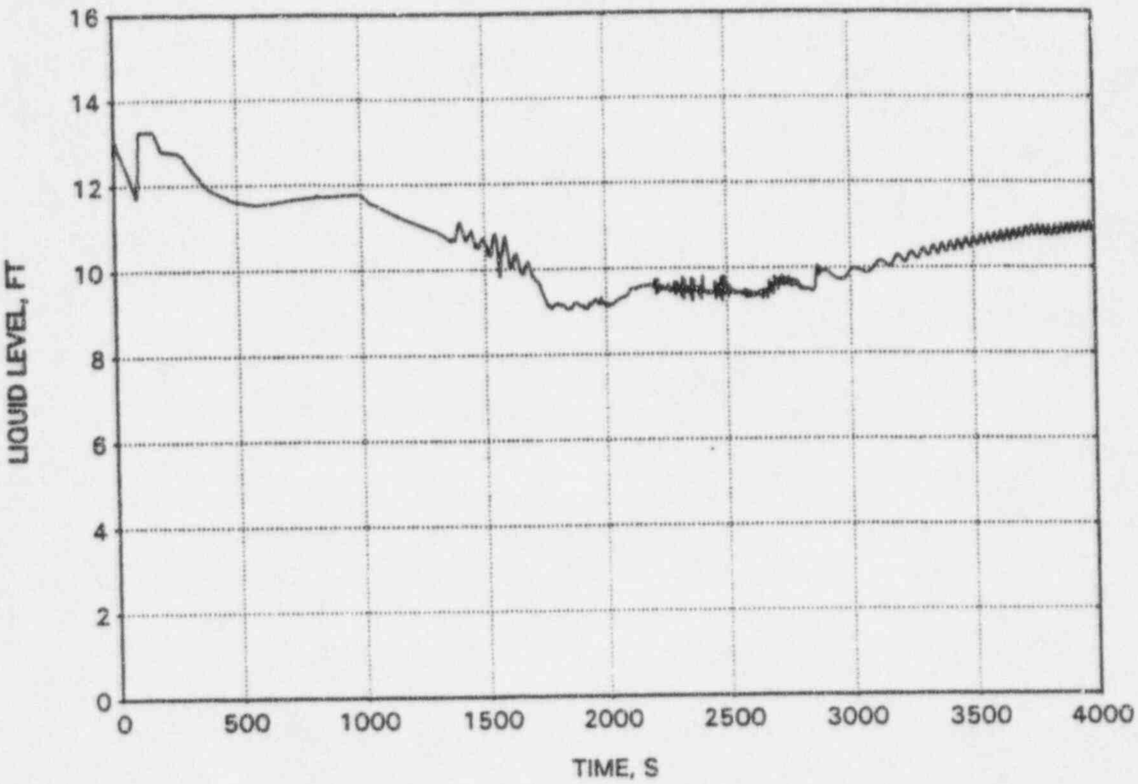


FIGURE 5.9-8 1.5-inch Pump Discharge Break
Hot Rod Clad Temperature

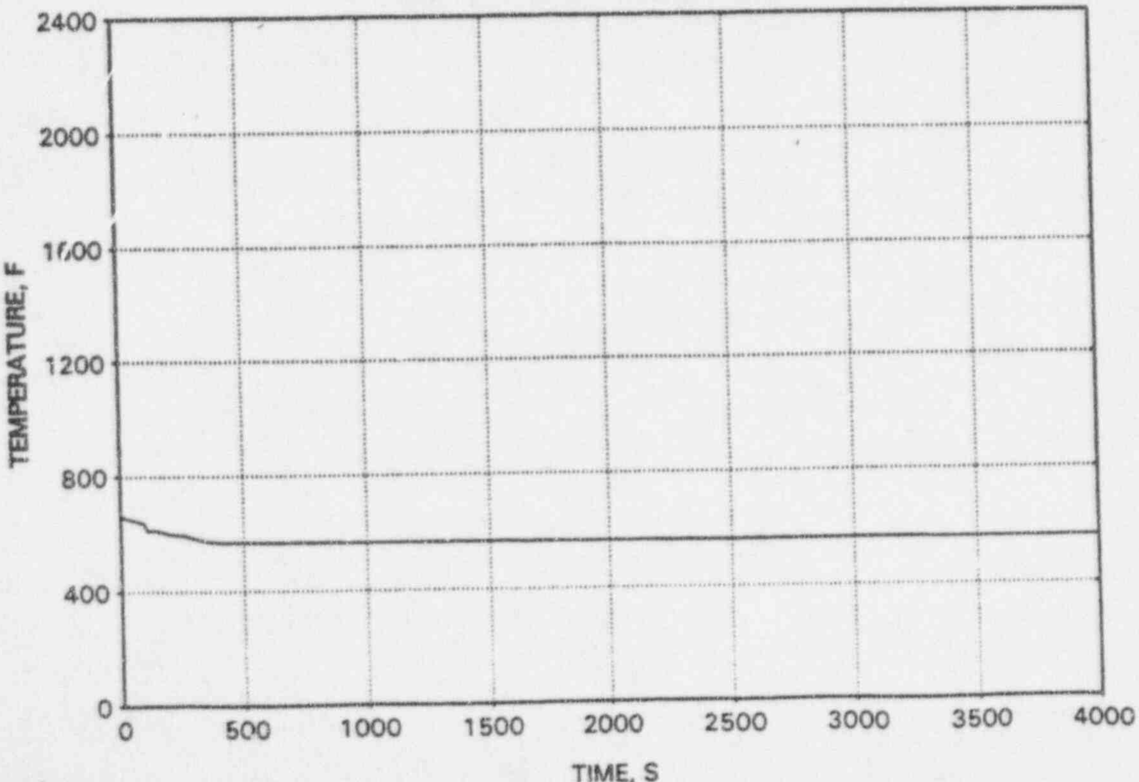


Figure 5.9-9 2.5-inch Pump Discharge Break
Primary System Pressure

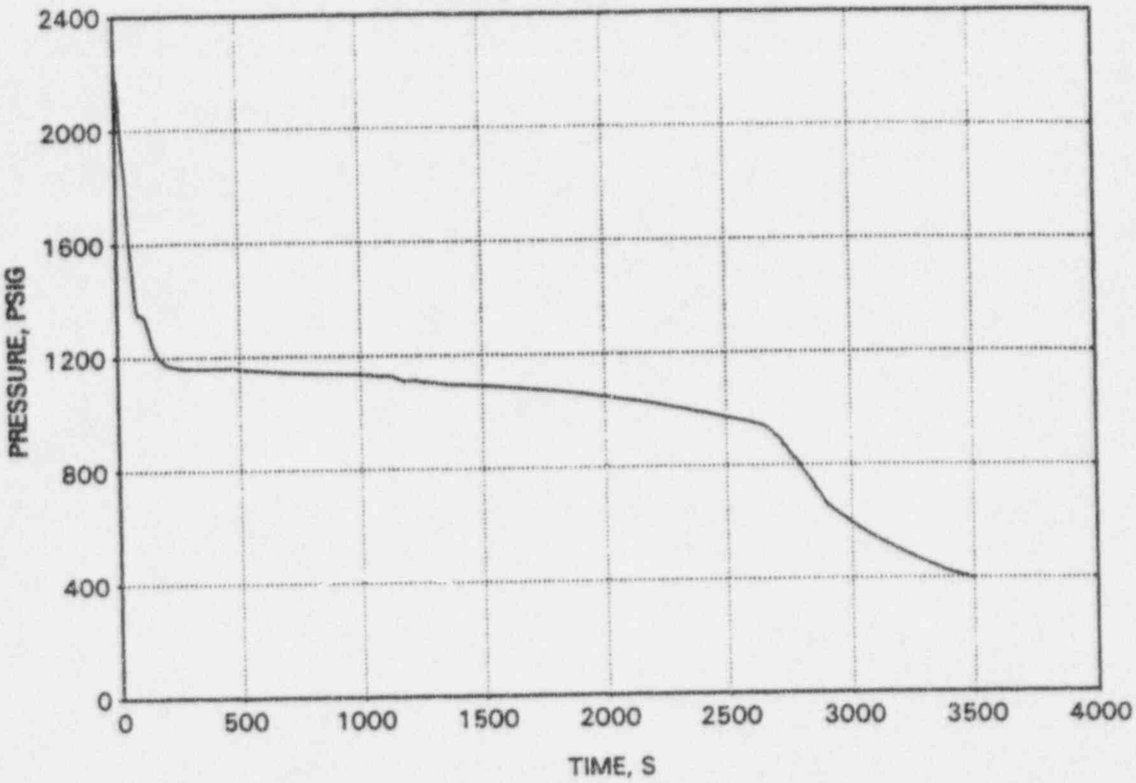


Figure 5.9-10 2.5-inch Pump Discharge Break
Leak Flow Rate

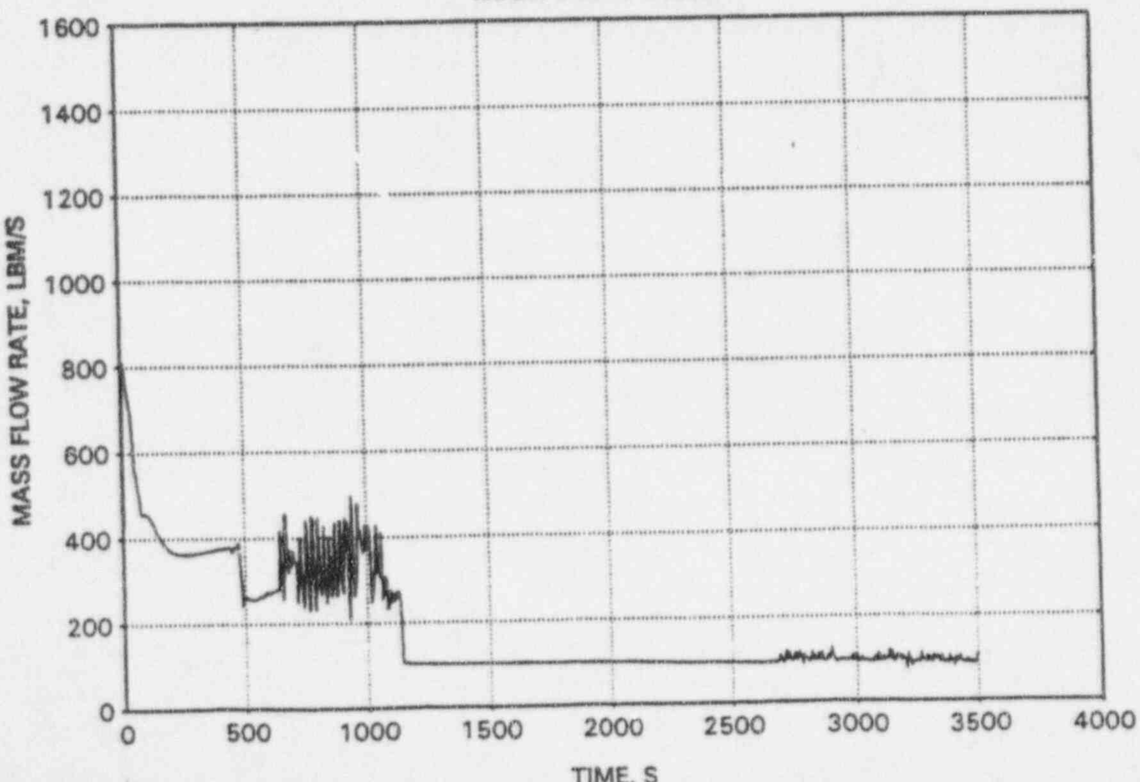


FIGURE 5.9-11 2.5-inch Pump Discharge Break
Pump Suction Loop Seal Levels

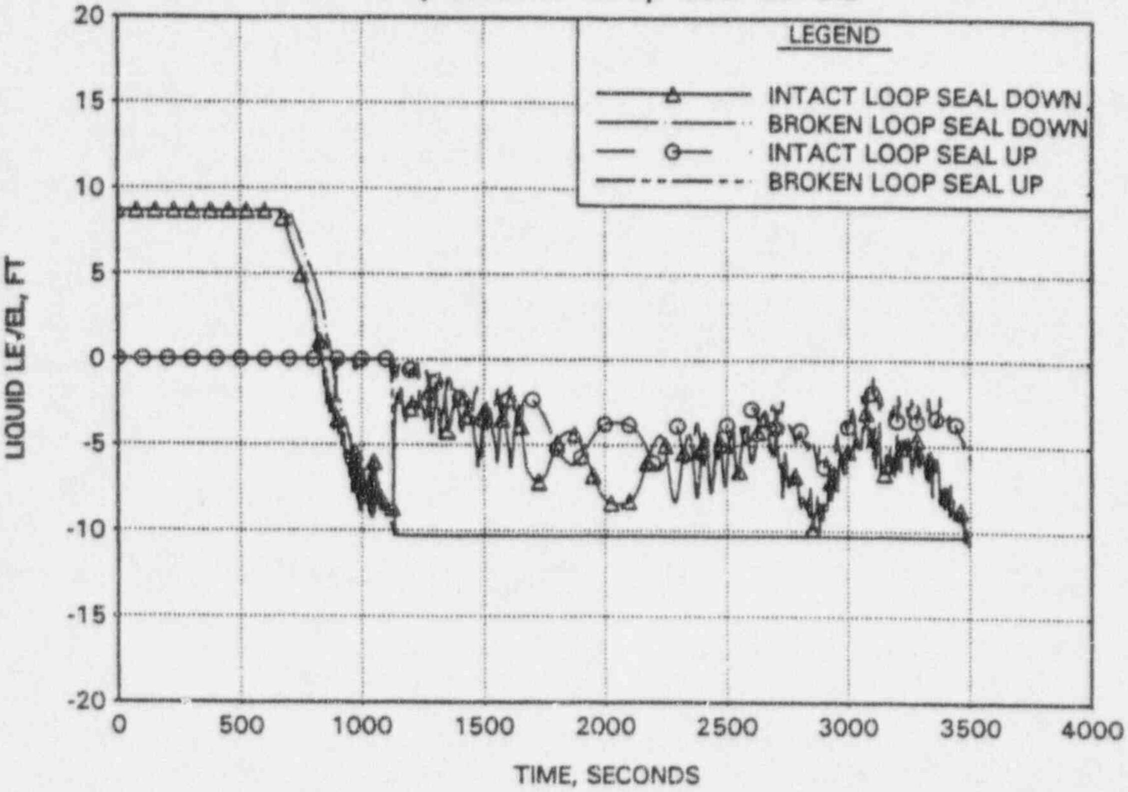


Figure 5.9-12 2.5-inch Pump Discharge Break
Core Collapsed Level

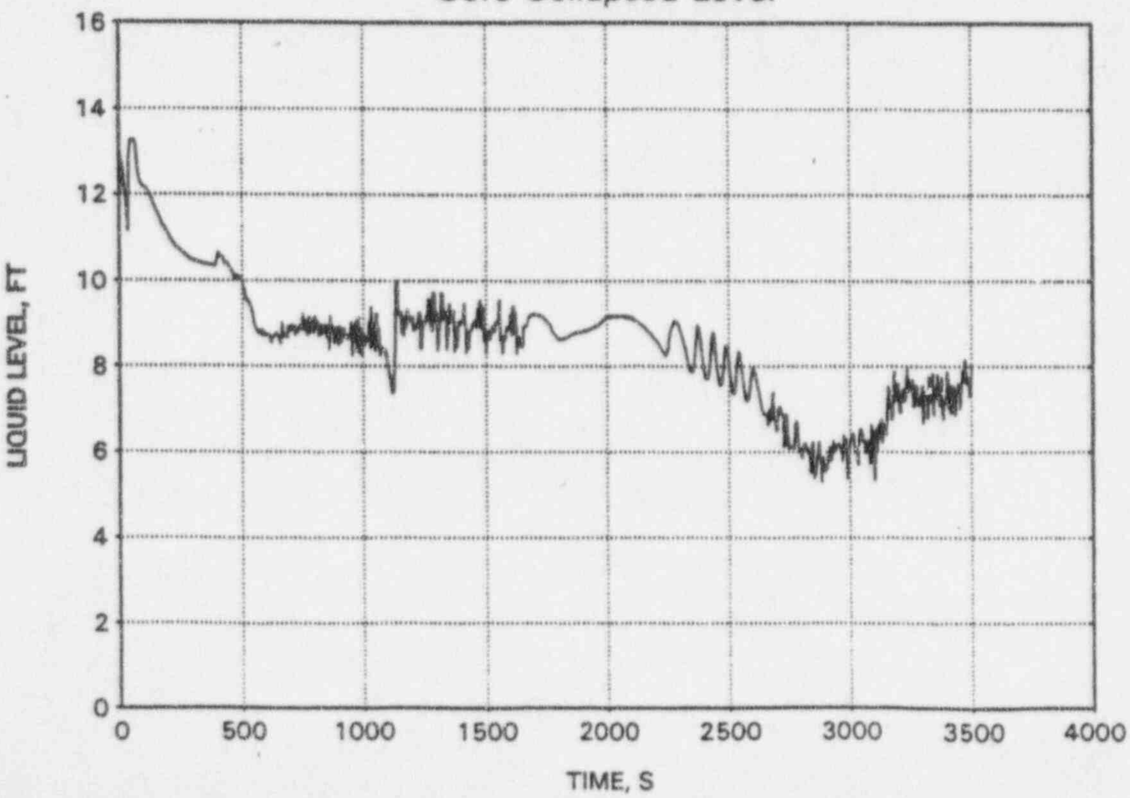


Figure 5.9-13 2.5-inch Pump Discharge Break
Hot Rod Clad Temperature

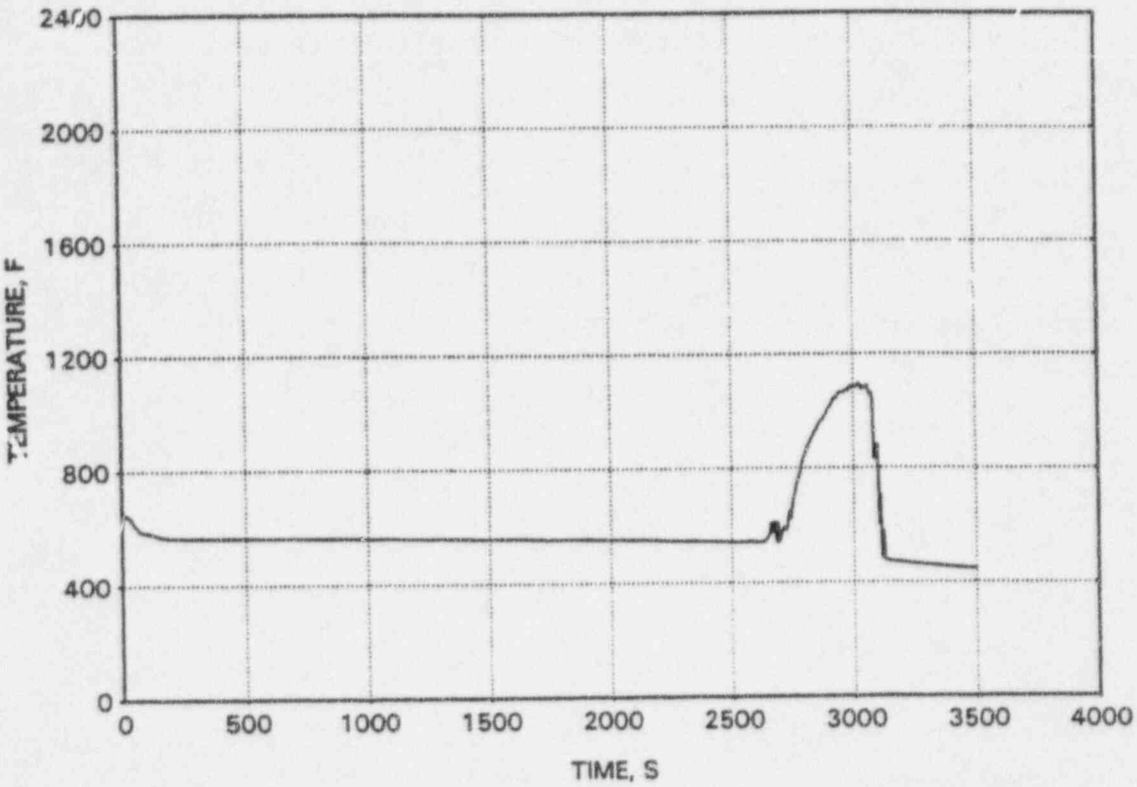


Figure 5.9-14 3-inch Pump Discharge Break
Primary System Pressure

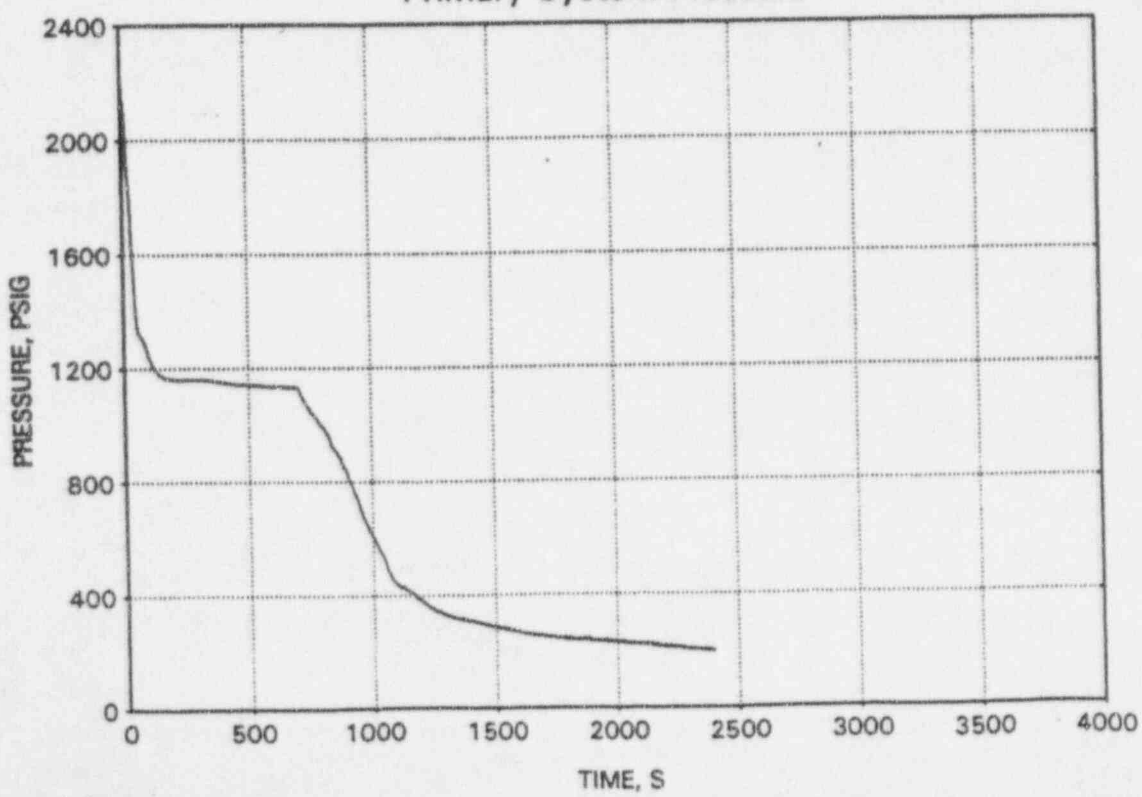


Figure 5.9-15 3-inch Pump Discharge Break
Leak Flow Rate

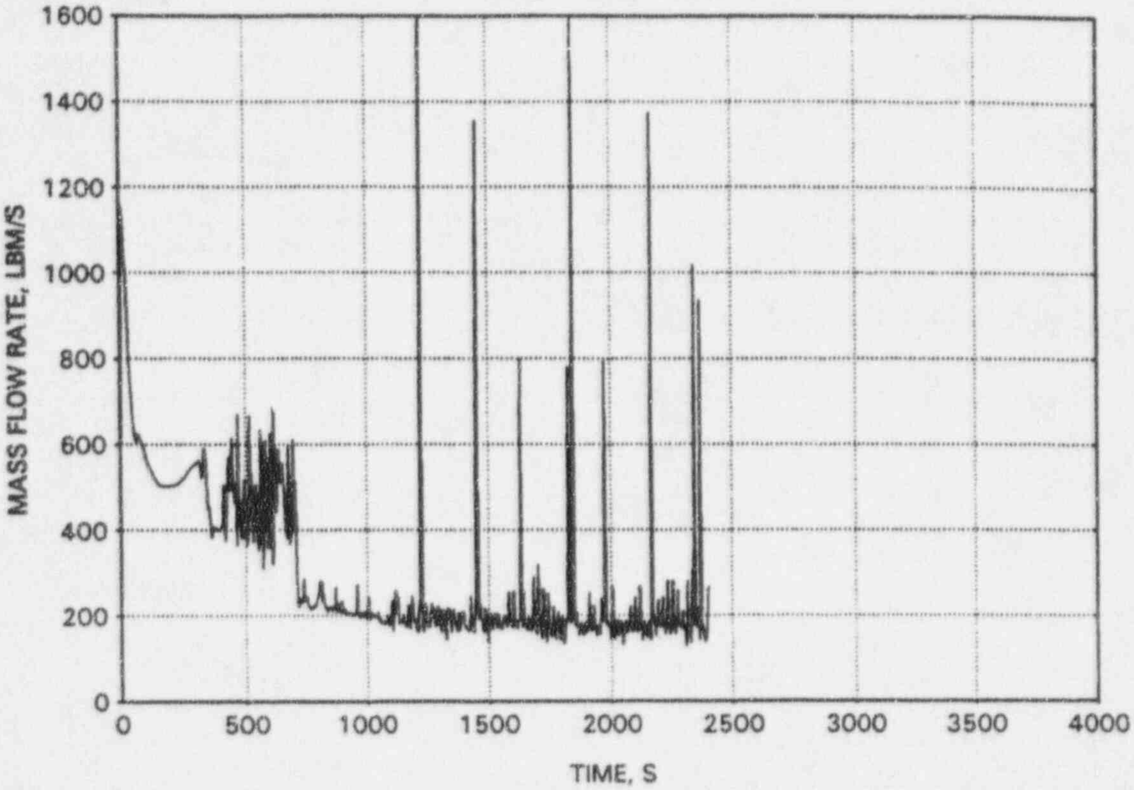


Figure 5.9-16 3-inch Pump Discharge Break
Pump Suction Loop Seal Levels

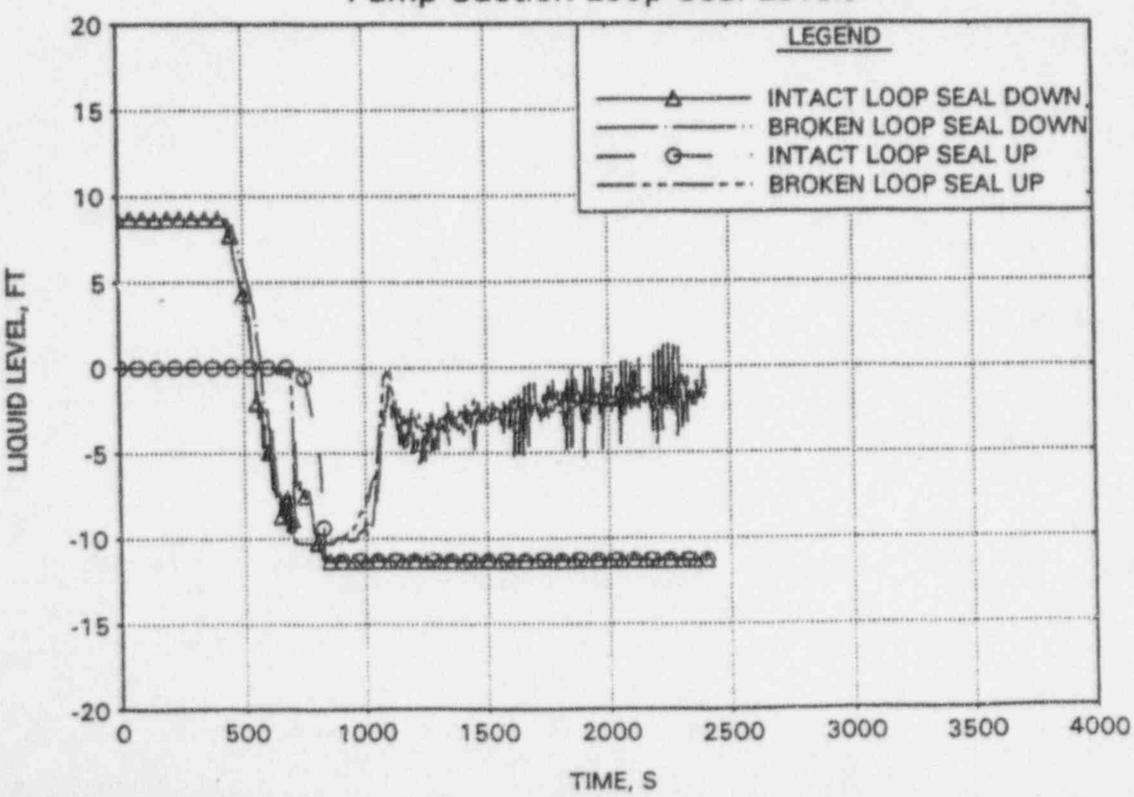


Figure 5.9-17 3-inch Pump Discharge Break
Core Collapsed Level

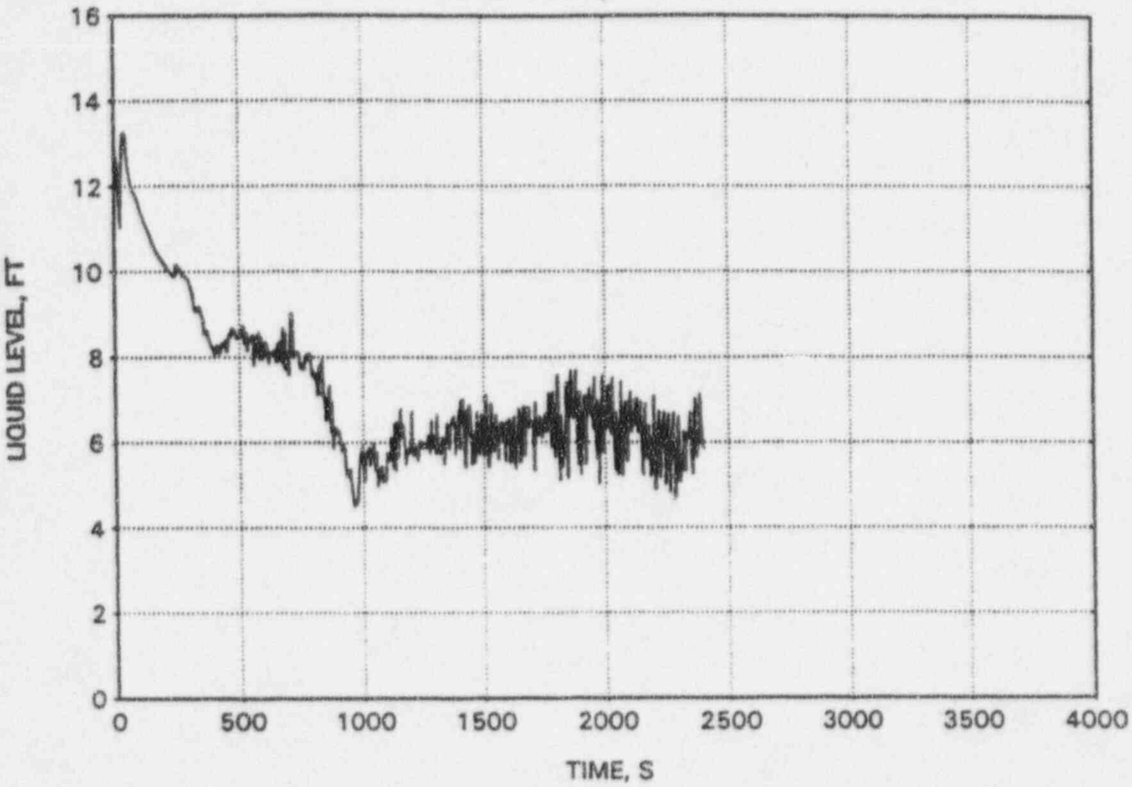


Figure 5.9-18 3-inch Pump Discharge Break
Hot Rod Clad Temperature

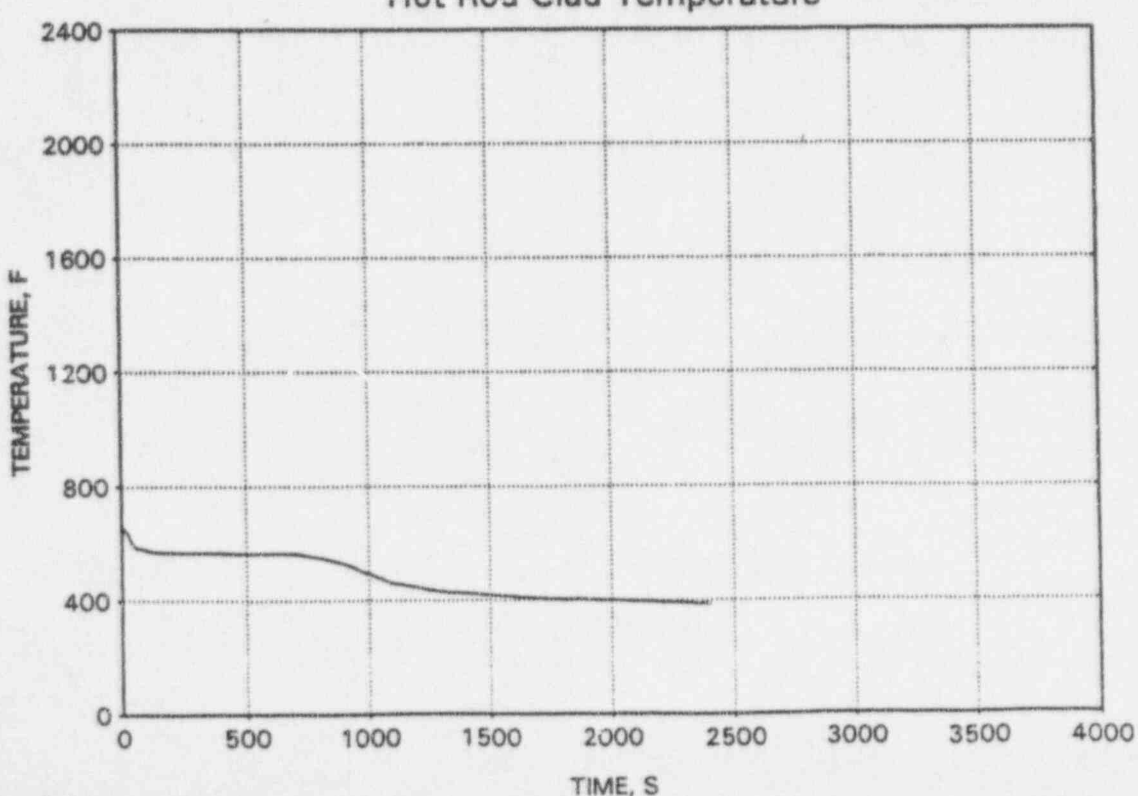


Figure 5.9.-19 5-inch Pump Discharge Break
Primary System Pressure

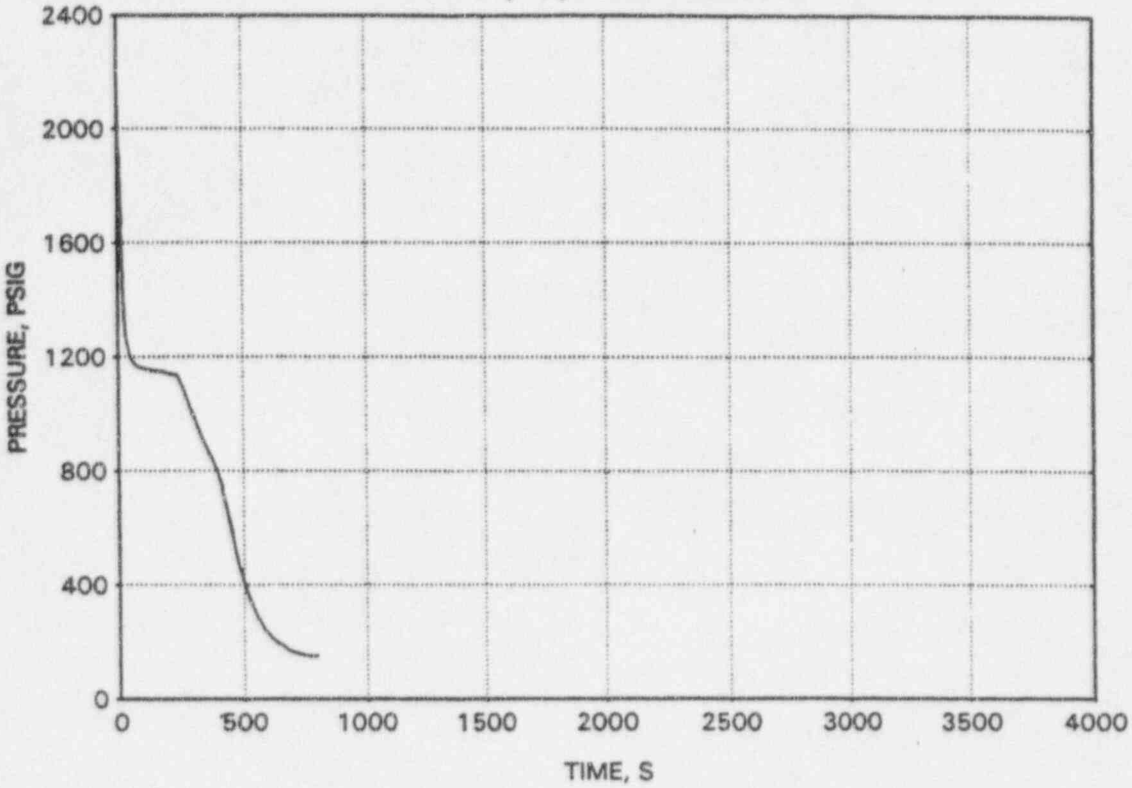


Figure 5.9-20 5-inch Pump Discharge Break
Leak Flow Rate

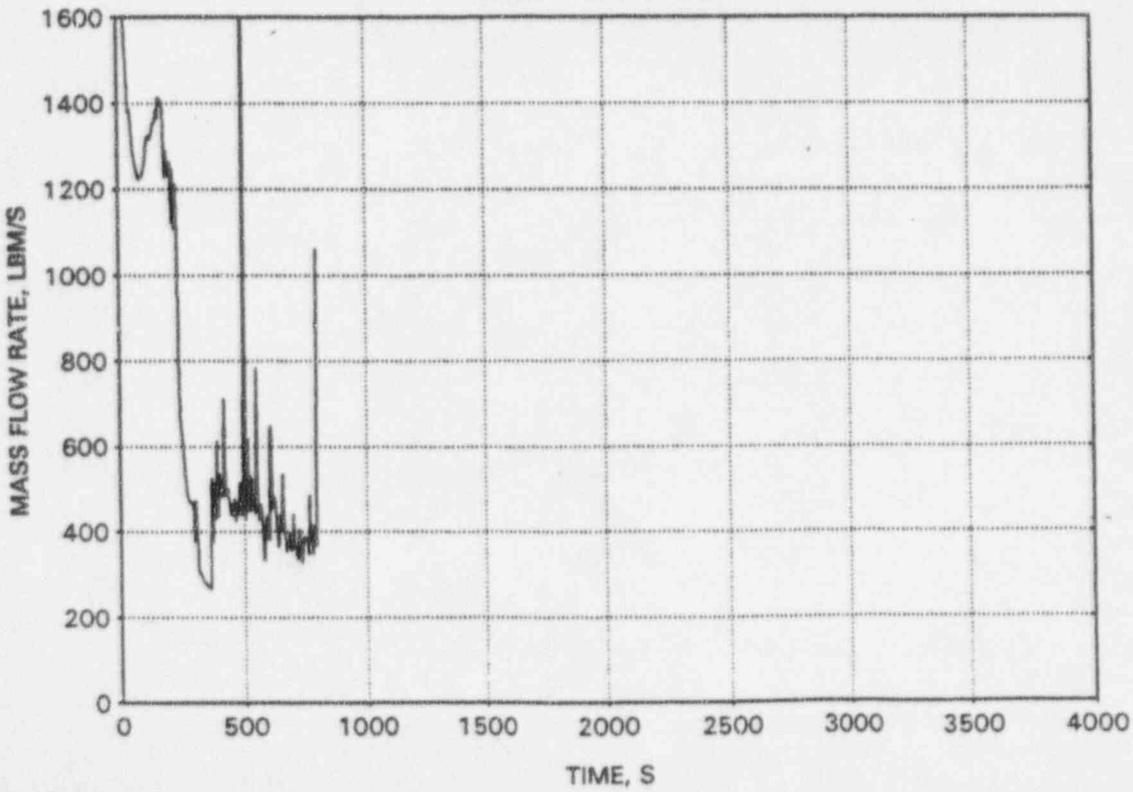


FIGURE 5.9-21 5-inch Pump Discharge Break
Pump Suction Loop Seal Levels

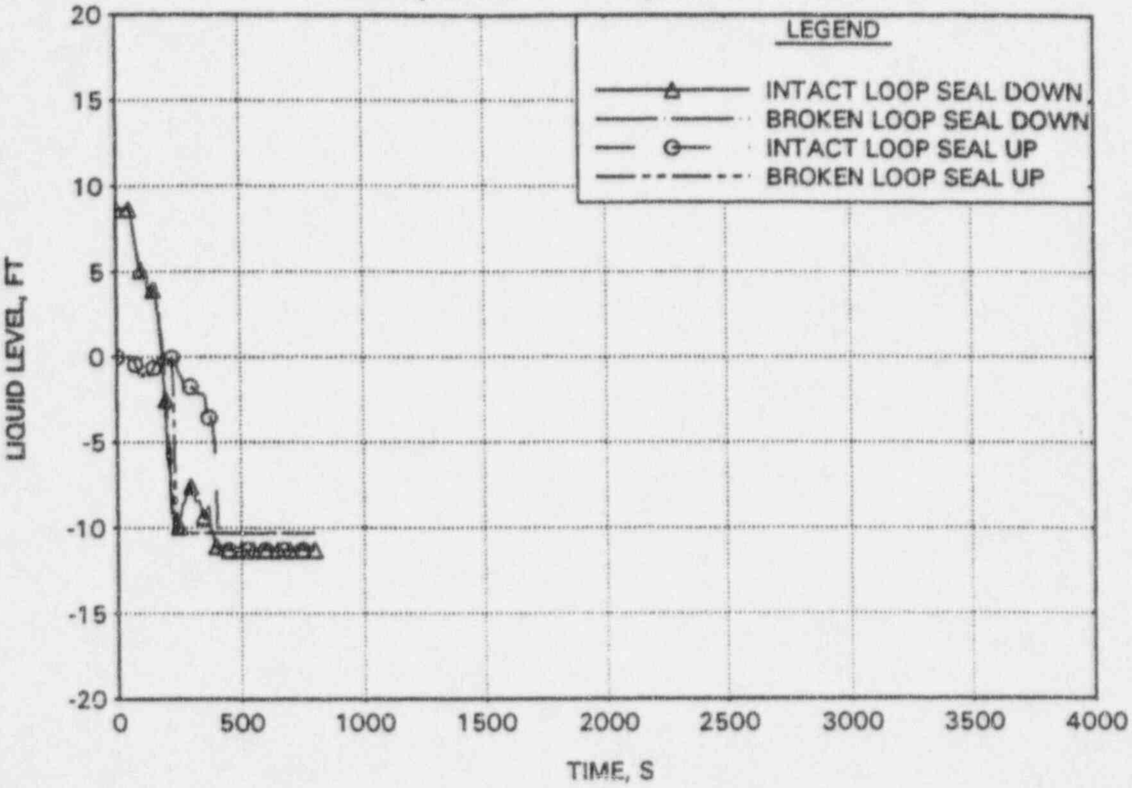


FIGURE 5.9.-22 5-inch Pump Discharge Break
Core Collapsed Level

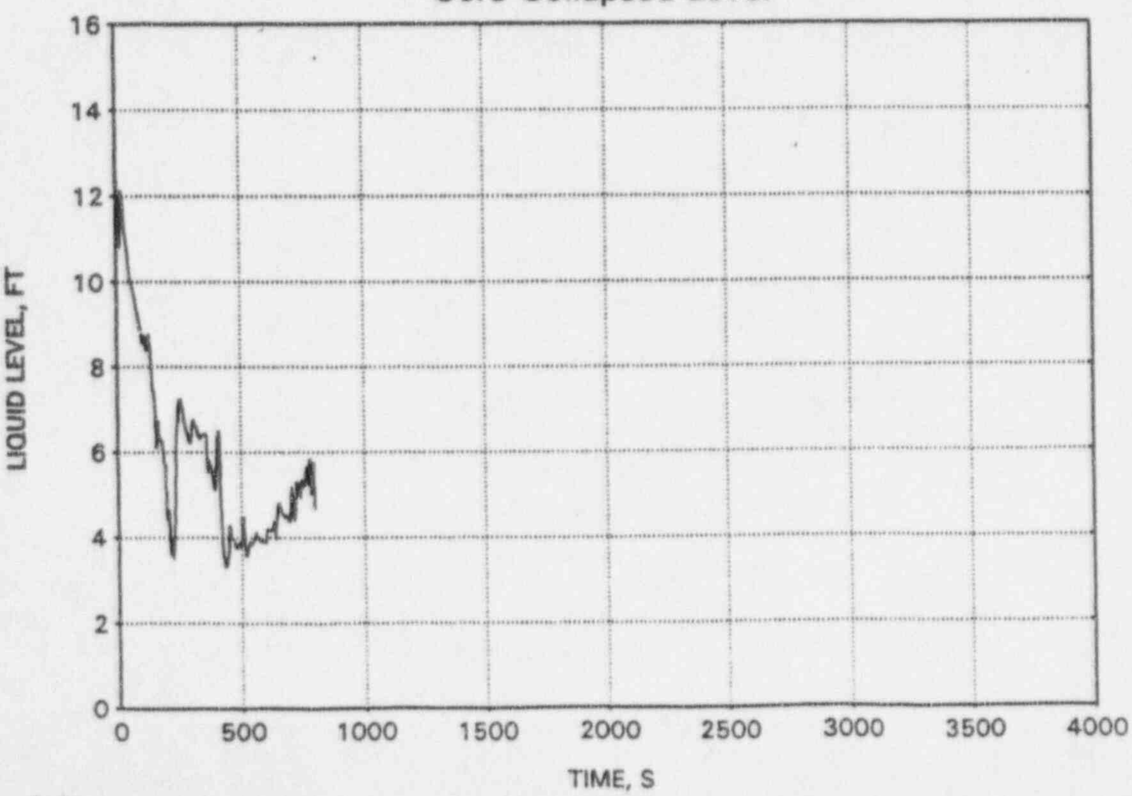


Figure 5.9.-23 5-inch Pump Discharge Break
Hot Rod Clad Temperature

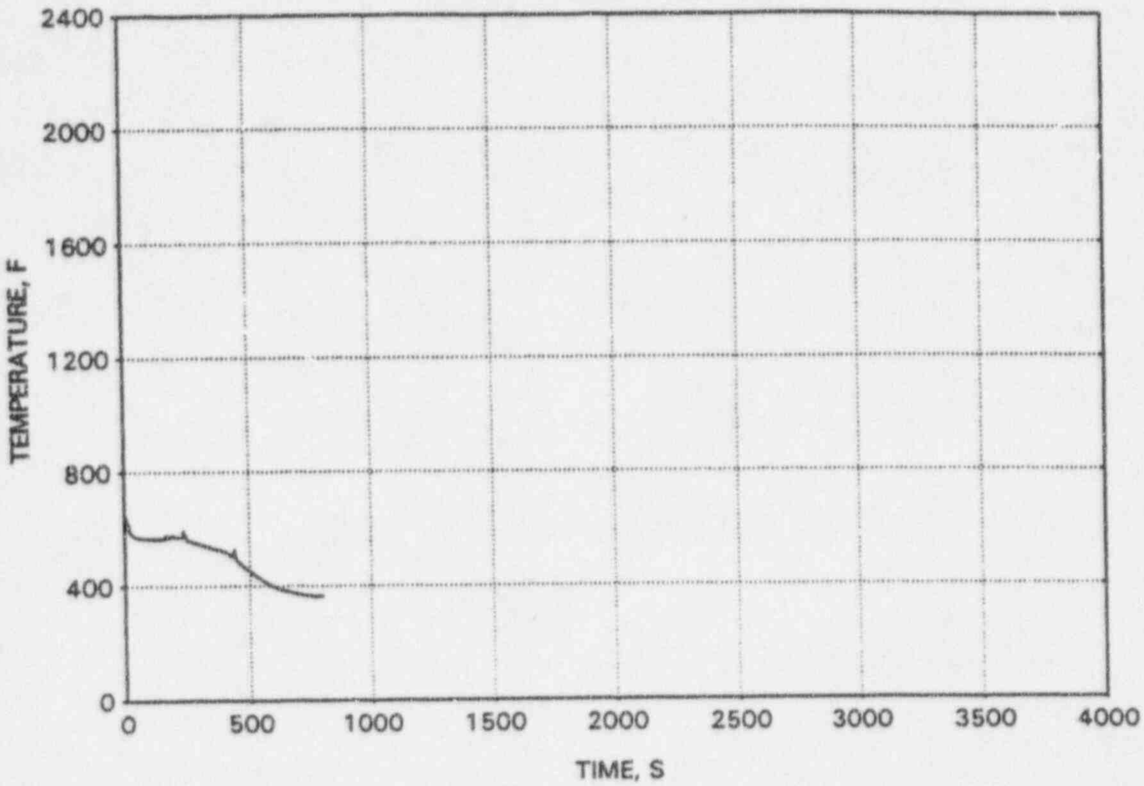


Figure 5.9-24 CCI Line Break
Primary System Pressure

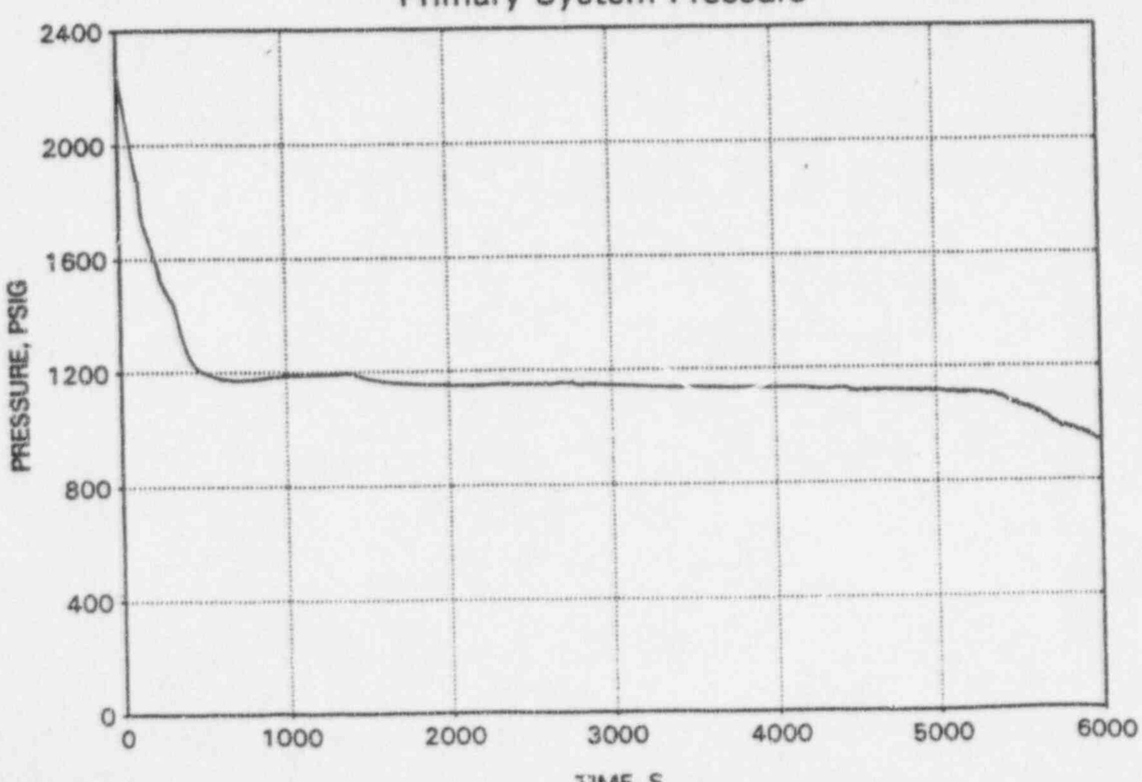


Figure 5.9-25 CCI Line Break
Leak Flow Rate

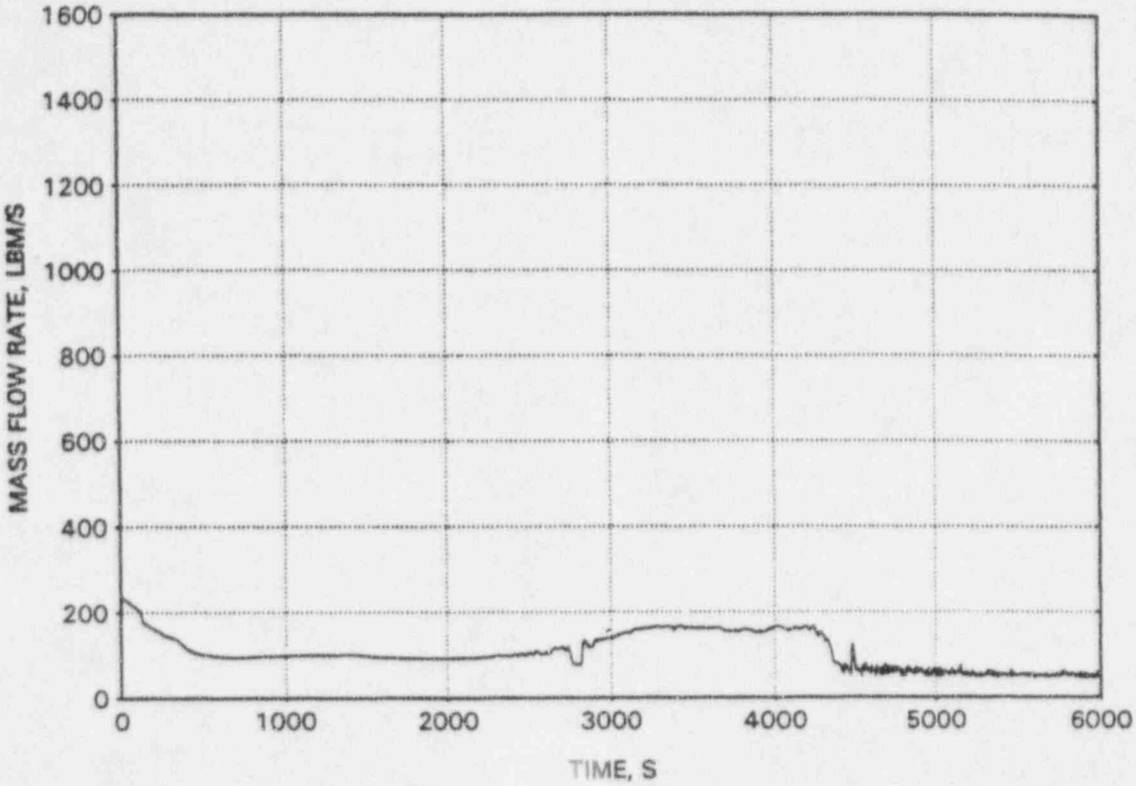


Figure 5.9-26 CCI Line Break
Pump Suction Loop Seal Levels

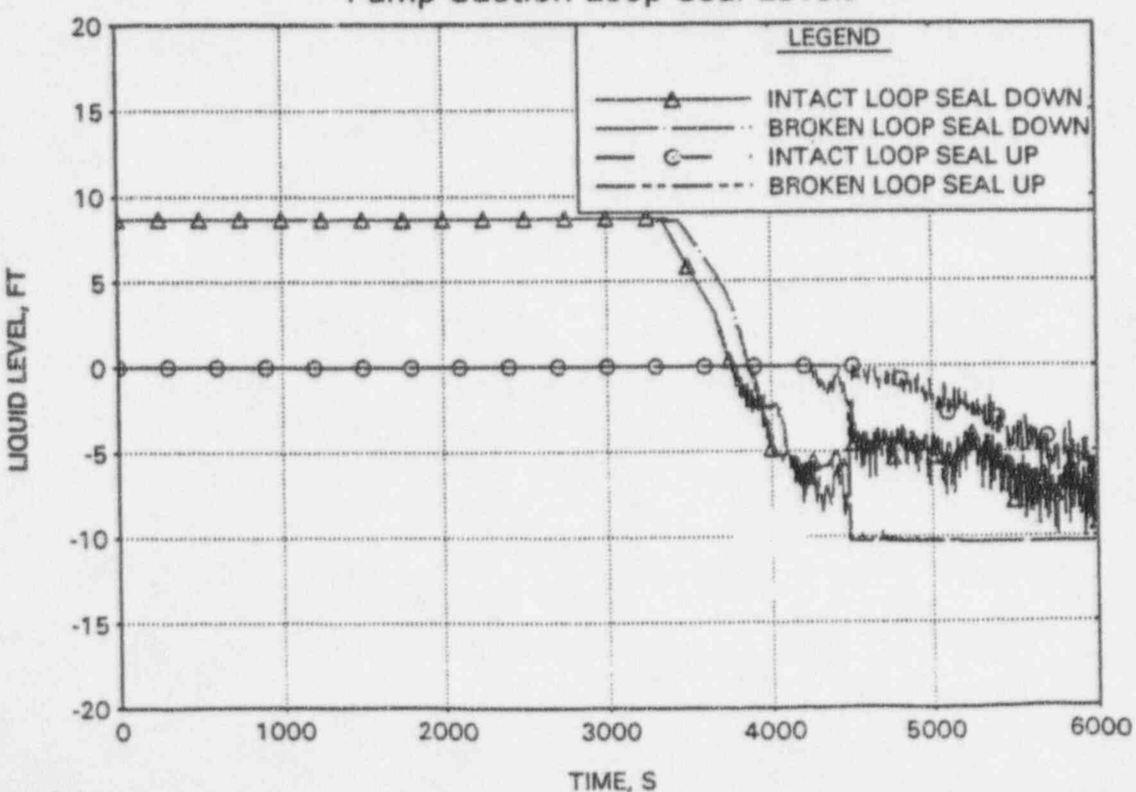


Figure 5.9-27 CCI Line Break
Core Collapsed Level

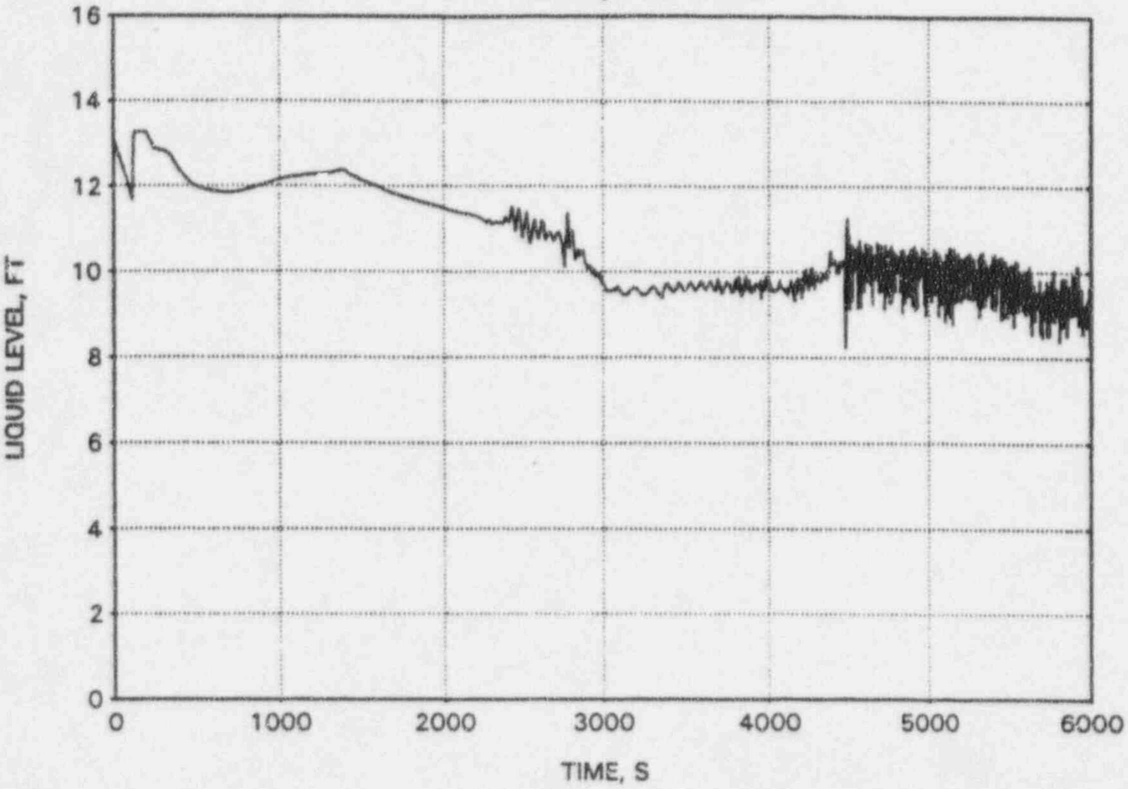
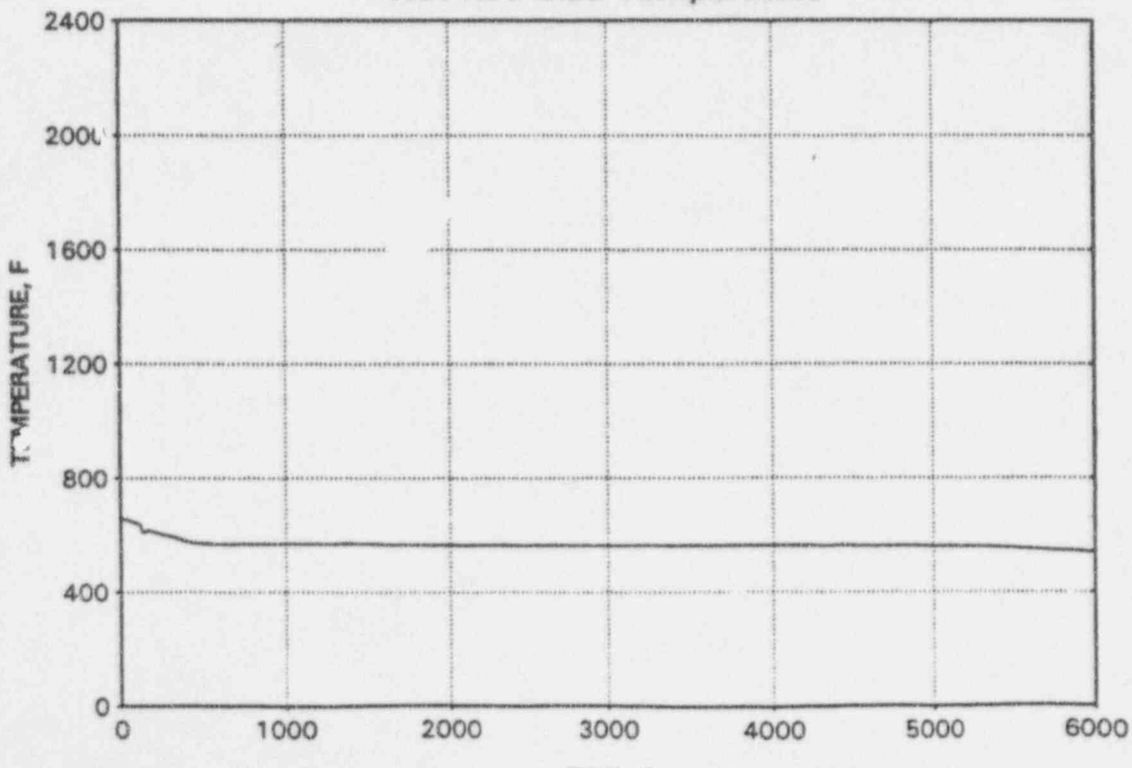


Figure 5.9-28 CCI Line Break
Hot Rod Clad Temperature



5.12 References

5-1 BAW-10168P Revision 02, BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, October 1992.

5-2 BAW-10168P Revision 03, BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, November 1993.

5-3 BAW-10164P Revision 03, RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, October 1992.

5-4 BAW-10171P Revision 02, REFLOD3B - Model for Multinode Core Reflooding Analysis, January 1989.

5-5 BAW-10166P Revision 04, BEACH - A Computer Program for Reflood Heat Transfer During LOCA, October 1992.

5-6 BAW-10162P, TACO3 - Fuel Pin Thermal Analysis Code, October 1989.

5-7 BAW-10092P, CRAFT2 - FORTRAN Program for Digital Simulation of a Multinode Reactor Plant During Loss-of-Coolant, April 1997.

5-8 BAW-10174P Revision 1, Mark-BW Reload LOCA Analysis for the Catawba and McGuire Units, September 1992.

5-9 V. H. Ransom et al., RELAP5/MOD2 Code Manual, Volumes 1 and 2, NUREG/CR-4312, EGG-2396, 8/85.

5-10 BAW-10177P, Mark-BW Reload LOCA Analysis for the Trojan Plant, October 1990.

5-11 BAW-10184P, GDTACO - Urania Gadolinia Fuel Pin Thermal Analysis Code, February 1995.

5-12 BAW-10172P, Mark-BW Mechanical Design Report, July 1988.

2. The proposed TSs deviate from the approved Standard Technical Specifications (STSs) for Westinghouse plants (NUREG-1431, Revision 1) with regard to removing items from the TSs and relocating them to the Core Operating Limits Report (COLR). Provide justification for this deviation from the Approved STSs for the quadrant power tilt ratio (QPTR) and the f_1 (ΔI) and f_2 (ΔI) inputs to the OTΔT and OPΔT reactor protection system, considering the Nuclear Steam Supply System (NSSS) and nuclear instrumentation remain Westinghouse designs. Additionally, justify the use of a QPTR limit of 1.03 which is less conservative than the formerly used 1.02.

Response

The parameters relocated to the COLR associated with $f_1(\Delta I)$ and $f_2(\Delta I)$ are the positive and negative breakpoints for the deadband (the ΔI limits) and the slopes for reduction of the ΔT trip setpoints for each percent that the magnitude of ΔI exceeds the breakpoints. These parameters are determined on a cycle-specific basis, and their allowable magnitudes vary based on the specific fuel cycle design, as determined by the FCF cycle-specific three-dimensional core power distribution analysis. The methodology for determining the $f_1(\Delta I)$ and $f_2(\Delta I)$ limits is described in the approved topical report BAW-10163P-A. Therefore, approved methodology for calculation of these parameters on a cycle-specific basis is in place. Thus even though the NSSS and nuclear instrumentation remain Westinghouse designs, the cycle specific variation of the f(ΔI) limits and FCF methodology justify relocation of these parameters to the COLR.

FCF's approved methodology applies an allowance for increased peaking due to radial tilts in the cycle-specific design power distribution analysis that sets the core safety and operating limits, as described in section 4 of BAW-10163P-A. [ ] In addition, should the QPTR limit specified in the COLR be exceeded, the technical specification action to reduce thermal power is based on the amount of tilt in excess of the COLR limit (Specification 3.2.4).

3. Provide additional basis for not including uncertainties if the $F^N_{\Delta H}(XY)$ and $F^N_Q(XY)$ in the footnotes of TS SR 4.2.2.2 and 4.2.3.2.

Response

SR 4.2.2.2 and 4.2.3.2 specify the precalculated limit quantities that will be used for comparison of measured peaking factors. The pre-calculated limit values are determined in accordance with the methodology described in topical report BAW-10163P-A. Section 6 of the topical illustrates how the allowable limit values include consideration of the applicable calculational and measurement uncertainties. Therefore, since the allowable limits are reduced to accommodate uncertainties, multiplication of the measured peaking factors by the uncertainties is not necessary prior to making the comparison to the limit during peaking factor surveillance.

The peaking factor surveillance process is automated. A plant and cycle-specific data base of limit values is provided for the core monitoring software. The limit values provided in the data base include the applicable uncertainties so that external application of uncertainties is not needed.

4. Why is F^N_Q (XY) not reduced by 2% over what is specified in the COLR as the approved BAW-10163 prescribes in TS 4.2.2.2.C..4.e.1?

Response

[ ] Surveillance requirement 4.2.2.2.c.4.e.1 of the sample technical specifications provided in Appendix A of BAW-10163P-A accomplished this by requiring that the measured peak be increased by 2% (i.e. a reduction in measured margin) when two measurements extrapolated to 31 EFPD beyond the most recent measurement yield $F_Q^M(X,Y,Z)$ to be greater than the expected value [BQNOM(X,Y,Z)]. For Sequoyah, the constant (2%) factor is replaced by a cycle-specific parameter and relocated to the COLR. [ ] Therefore, the surveillance for the Sequoyah technical specifications was revised from that in BAW-10163P-A to state the "appropriate factor specified in the COLR" instead of "2%."

5. Explain why F_Q and $F_{\Delta H}$ are not verified each time the excore QPTR is verified with the incore detectors as the approved BAW-10163 methodology prescribes in TS SR 4.2.3.3 and 4.2.2.2.c.4.e.

Response

The methodology provided in BAW-10163P-A was developed to define limits that preserve fuel design criteria related to power peaking and reactivity for Westinghouse PWRs utilizing Mark-BW fuel. The sample technical specifications provided in Appendix A of BAW-10163P-A were created by incorporating these limits into the framework of the Westinghouse Standard Technical Specifications.

The Westinghouse Standard Technical Specifications included the requirement to verify F_Q and $F_{\Delta H}$ whenever the quadrant power tilt ratio (QPTR) indicated by the excore detectors is calibrated. This requirement is not part of the limits required to preserve the fuel design criteria as presented in BAW-10163P-A. It exists in the sample technical specifications of BAW-10163P-A because it was originally present in the Westinghouse Standard Technical Specifications.

However, the current Sequoyah Nuclear Plant Technical Specifications for power peaking factors (Specifications 3/4.2.2 and 3/4.2.3) do not contain the requirement to verify the peaking factors whenever the QPTR indicated by the excore detectors is calibrated. This requirement was not included in the technical specification change package for FCF fuel because its introduction is not specifically related to the fuel design change and would have imposed a separate requirement that does not currently exist for the Sequoyah units.

6. Justify the assumption that 15% of the steam generator tubes have been plugged. What effect will that have on the LOCA analysis results if more or less tubes are plugged in the broken or unbroken loops? (see p. 5-6) Additionally, the non-LOCA analysis assumes that 20% of the tubes are plugged. Justify why this is limiting for all transients analyzed.

Response:

Transients were analyzed for a maximum tube plugging of 15% in all generators. The 20% tube plugging for non-LOCA transients on page 6-4 is incorrect. The correct value is 15%. In general, 15% steam generator tube plugging is used as a design limit for the reload transients. The use of an upper-bound tube plugging limit is conservative in the majority of cases because of its adverse effects on core cooling and primary heat removal.

Core cooling is adversely affected by the lower RCS flows associated with increasing tube plugging. The margin to DNB is reduced. Primary heat removal interruption is exaggerated by using an upper-bound plugging limit. Secondary pressure is also reduced by maximizing tube plugging. Following turbine trip, the secondary pressure increases from its lower value to the steam line safety valve setpoint. The primary heat sink temperature changes relative to the steam pressure increase. The pressure transient and, therefore, the sink temperature band are enlarged by maximized tube plugging and the effect of turbine trip on primary heat removal is conservatively simulated.

For large break LOCA, the primary impact of steam generator tube plugging on PCT occurs during the core reflood phase of the transient. The loop pressure drop is a function of steam flow and tube flow area. The core flooding rate decreases as the tube plugging increases due to increased resistance to steam flow through the steam generators, the steam binding effect. Therefore, assuming the maximum plugging in all generators produces the minimum core flooding rate. During reflood an additional 5% plugging in all generators is included in the calculation to account for seismically-induced tube collapse.

The steam line break analysis is an exception to the use of an upper-bound steam generator tube plugging limit. For the steam line break transient, tube plugging is conservatively neglected to maximize heat transfer across the steam generator tubes. Both flow and heat transfer surfaces are adjusted commensurate with the assumption of 0% steam generator tube plugging. Increasing the RCS cooldown in this manner results in a maximum positive reactivity insertion and return to power; thereby, maximizing the severity of the event.

7: The loss-of-coolant accident (LOCA) analysis assumes that the reactor coolant system (RCS) flow is 348,000 gpm; however, the TS Figure 3.2-1 allows flow down to 342,000 gpm if power is derated. Show that the deration is sufficient to assure that no limits are exceeded. Additionally, the TS minimum RCS flow is being reduced with this submittal and the analysis on the new Framatome fuel is performed using the lower flowrate. However, TVA is relying on the current Westinghouse analysis to show no limits are exceeded for the Westinghouse fuel inserts. Justify the use of the current analysis when the TS minimum flowrate is going to be lower than was assumed in this analysis.

Additionally, in section 7.3.2, what is the basis for the equation reducing $F_{\Delta H}$ with reactor power?

Response

The loss-of-coolant accident (LOCA) was analyzed using the minimum thermal design flow rate of 348,000 gpm identified in Section 5.2 of BAW-10220. TS Figure 3.2-1 provides reduced power operation flexibility in the event that the measured flow rate, less flow measurement uncertainty, is less than the minimum thermal design flow rate of 348,000 gpm. The power to flow trade off allowance is based on maintaining the margin to the DNB safety limit at the reduced flow condition at a level that is greater than or equal to the margin at the full flow condition. The reduced flow condition is advantageous with respect to LOCA analysis results. The effective core peaking limit increases in direct proportion to reduced power. A LOCA "hot pin" initialization case at 95% flow and equivalent peaking was conducted and concluded that there was no effect of fuel pin temperature on flow reduction in this range of flow. The hot pin initial temperature in lowered flow and power LOCA transient analyses would, therefore, be equivalent to that currently analyzed. During LOCA, it is the dynamics of the average core and not the hot pin that dictates the progression of the transient. Initial core average fluid temperature would effectively be lowered, reflecting the imbalance between RCS flow and power reduction. Experience indicates that, all other things being equal, initially lower coolant temperatures produce a more subdued RCS core coolant flow transient during blowdown and slightly hotter pin temperatures at the end of blowdown. However, during reflood, the lower core power level would more than make up for any blowdown penalty. Lower core power results in lower fuel decay heat and the core would be quenched more rapidly. The net effect of a 2-for-1 trade off in power relative to flow would be lower large break LOCA peak clad temperatures.

The flow/power trade-off would also be beneficial in the progression of a small break LOCA. Small breaks are longer term transients and initial core fluid temperatures have little effect

on their progression. With lower powers and lower decay heat rates, lower depths of core uncovery would be predicted for small breaks and the peak clad temperature would be effectively reduced.

Therefore, the power to flow trade-off defined by TS Figure 3.2-1, would result in lower peak clad temperature predictions for LOCA.

The present Westinghouse fuel LOCA analysis was reviewed by Westinghouse with respect to a reduction in reactor coolant system (RCS) thermal design flow from 362,000 gpm to 348,000 gpm. The Westinghouse review concluded that the reduction in thermal design flow is inconsequential since the core flow is immediately dominated by the influence of the break during the blowdown phase of the LB LOCA transient. The influence of the break is such that the core experiences a full flow reversal very early in the transient, with lower core fluid exiting the break via the downcomer path to the cold leg break location. During the LB LOCA, the RCS is almost completely voided during the blowdown phase of the transient. The reduction in thermal design flow results in an RCS Thot increase of approximately 1°F and Tcold reduction of approximately 1°F. At the lower Tcold condition, the break flow will be slightly greater initially due to the slightly higher liquid density in the cold leg. This effect will be offset by decreased break flow when the higher Thot fluid reaches the break. During the entire blowdown phase of the transient, RCS Tave dominates the blowdown out the break. Minor changes to Thot and Tcold at the same Tave have a negligible effect on the transient results. Initial RCS flow and temperature distribution have no direct affect on the subsequent refill and reflood phases of the transient. These phases are influenced primarily by the performance of the emergency core cooling system (accumulators and safety injection). Given the small changes in initial RCS flow and temperature distributions, Westinghouse concluded that the thermal design flow reduction will not result in any changes to the present LB LOCA calculated fuel peak clad temperature.

The equation $F^N_{\Delta H\ adjusted} = F^N_{\Delta H\ nominal}\,[1 + 0.3(1 - P)]$ stated in Section 7.3.2 of BAW-10220 allows an increase in the allowable radial peak at core powers levels less than 100% power. This relationship is based on an equation presented in the Sequoyah Tech Spec Bases Section 2.1.1 and in Section 2.6 of the Sequoyah 1 Cycle 9 COLR of

$$F^N_{\Delta H} = F^{RTP}{}_{\Delta H}\,[1 + PF_{\Delta H}\,(1 - P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

$F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ limit at the RATED THERMAL POWER (RTP) specified in the COLR, and

$PF_{\Delta H}$ = the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR.

FCF has verified that this relationship remains valid and provides adequate margin to the Safety Limit by determining allowable peaking limits at [ ] as part of its standard reload methodology. FCF has further shown that this relationship remains valid and applicable under the reduced power and reduced flow conditions defined by Tech Spec Figure 3.2-1.

8. Please describe the changes made to the approved Babcock & Wilcox Nuclear Technologies (BWNT) recirculating steam generator (RSG) evaluation model (described in p. 5-79) in greater detail and discuss any implications on the prior staff review and approval.

Response:

In small break spectrum analyses, ECCS injection in the broken loop is conservatively prevented from flowing into the downcomer by the use of a fictitious leak node (node 276 in Fig. 5.9-2). All broken loop ECCS injection is directed toward this volume following loop seal clearing, ensuring complete ECCS spillage. The cold, spilled, ECCS liquid interacts with steam in node 276. In many of the spectrum cases it was noted that, if the non-equilibrium option in RELAP5/MOD2 (generally used in all volumes) is used in node 276, severe local non-physical pressure oscillations occur. The oscillations effectively cause increased leak flow and system depressurization.

To alleviate leak flow oscillations, the phase equilibrium option is selected for the leak volume. The equilibrium option continuously maximizes the steam condensation by the imposition of thermal equilibrium, and significantly dampens pressure oscillations and stabilizes leak flow. The equilibrium option has no impact on the timing of loop seal clearing since, prior to clearing, little steam enters the leak volume to alter leak flow characteristics. In addition, the relatively stable leak flow will prolong core boildown during the post-loop seal clearing phase of the transients. Therefore, it is conservative to utilize the equilibrium option if severe oscillations occur.

The changes to the small break model leak node will not affect previously reviewed and approved small break LOCA results such as those reported in Appendix A of BAW-10168P, Revision 3, Volume II. These cases are characterized by high leak volume void fractions that exhibit insignificant non-equilibrium effects. The analyses presented in support of licensing RELAP5 for use in small break analyses, Appendix J of BAW-10164P, Revision 2, do not utilize a fictitious leak node and are not affected by leak node model changes.

9. Has the core down flow bypass in the baffle region been explicitly modeled in the LOCA and non-LOCA analysis? Describe how it is modeled.

Response

LOCA

Nodalization is included in the RELAP5 large break model for blowdown analysis, and in the small break model, that characterize the baffle region (pipe component 350). The baffle region discretization is similar to that of the core bypass region (pipe component 346). Representation of the baffle region is the same as that of predecessor McGuire/Catawba and Trojan RELAP5 models with the exception that flow enters the top of the region and flows down to the vessel lower plenum, reflecting the down flow design at Sequoyah. The flow through the baffle region is small and has minimal effects on either LOCA transient.

In the REFLOD3B large break model, the core baffle region is combined with the downcomer. This is conservative for reflood as the "effective" downcomer fills more slowly and more ECCS mass must be added to the downcomer before it is filled to the bottom of the cold leg nozzle.

Non-LOCA

The downcomer model for the non-LOCA analysis includes the volume of the downcomer and the core downflow bypass in the baffle region. The downcomer is divided into two azimuthal regions (control volumes 304 and 306, and control volumes 370 and 372) as shown in Figure 6.1-1 to allow characterization of asymmetric loop conditions.

10. With regard to the fuel design features changed from the approved topical report BAW-10172, describe the changes in greater detail. The submittal is unclear with regard to the bottom nozzle changes. Have these changes been approved by the staff? Describe what testing and reanalysis has been performed on assemblies with the described changes (structural, flow, CHF). Please verify that the structural analysis performed in Chapter 8 of the topical report includes the changes identified.

Response

As described in Section 3.1 of BAW-10220P, there are four fuel assembly design differences on the Mark-BW Sequoyah fuel design that differ from the design defined in the approved BAW-10172 topical. These are:

1) reduction in number of restraining guide thimbles from twelve to eight per grid,
2) attachment of the ferrules using dimples rather than resistance welding,
3) utilization of a debris-resistant bottom nozzle rather than the standard bottom nozzle, and
4) incorporation of gadolinia fuel pellets and axial blankets of either natural or low enriched UO_2 fuel pellets.

The axial positions of the intermediate spacer grids on the Mark-BW for Sequoyah are maintained by the use of ferrules that are attached to [ ] guide thimbles. The short ferrules, attached to the guide thimbles, provide direct axial interference with the spacer grid interior strips around the guide thimbles to inhibit the grids from shifting above allowable positions. The ferrules are attached at positions that allow the spacer grid to "float" over a short axial distance. Refinements on the ferrule positioning requirements and manufacturing tolerances have permitted the reduction of necessary ferrules from [ ] per grid to [ ] per grid.

The ferrules were previously attached to the guide thimbles using a resistance weld as described in BAW-10172P. However, FCF has developed an improved attachment connection using a dimpling of the guide thimble into the surrounding ferrule produced from a force applied to the guide thimble inner wall. This connection has the improved performance characteristic of a softer interface that distributes the axial loads into the restraining guide thimbles and compensate for tolerances on ferrule positioning.

The bottom nozzle on the Mark-BW for Sequoyah contains small circular flow holes that are more effective for filtering debris from the coolant than the large flow slots on the earlier Mark-BW design described in BAW-10172P. The debris bottom nozzle design

is the same as described as an alternative Mark-BW bottom nozzle design in BAW-10172P. FCF selected the debris-resistant bottom nozzle design after performing debris trapping effectiveness tests to verify the adequacy of the flow hole size and performing full-scale pressure drop tests to verify the acceptablility of the pressure drop impact. The small pressure drop increase associated with the debris-resistant bottom nozzle has been incorporated into the analyses dependent on the hydraulic characteristics of the fuel design.

Gadolinia fuel pellets and axial blankets are being utilized in the Mark-BW for Sequoyah to provide desired operational flexibility and improved economics. Both of these features have proven records of acceptable in reactor service in other FCF supplied cores.

All of the aforementioned fuel assembly design changes have been developed in the last decade, incorporated into the Mark-BW product line, and have accumulated extensive reactor performance experience. As each design change was developed and implemented, FCF followed previously approved analysis methods to verify the change was acceptable. However, FCF did not submit a request for NRC approval for each intermediate design change as product upgrades or minor evolutionary changes to the fuel assembly design can be expected in any of the vendor fuel assembly design. BAW-10220P is the first submittal that collectively identifies all the design changes.

The incorporation of the debris-resistant bottom nozzle was the only change that required full-scale pressure drop testing to quantify the impact on the fuel assembly pressure drop. Tests performed in a cold water flow loop have shown the design change increases the fuel assembly pressure drop by [] This increase has been factored in the reload analyses for the Mark-BW. The hydraulic impact of the changes in the number of restraining guide thimbles and the type of ferrule attachment is small and has been determined analytically based on relationships benchmarked to pressure drop tests. None of the design changes have required the need to perform additional critical heat flux (CHF) tests.

A detailed finite element model for the anti-debris bottom nozzle was developed for use in the structural analysis. The finite element model was benchmarked against bottom nozzle strain data obtained from the fuel assembly drop test (FADT). Based on the benchmark of analysis results to FADT data, it was assured that the anti-debris bottom finite element model provides a close representation of the anti-debris bottom nozzle design. Further, it was determined that the maximum stresses determined from this model would be a good representation of maximum stresses expected to experience by the anti-debris bottom nozzle given the same loading.

The structural analysis performed in Chapter 8 of BAW-10220P, Rev. 0 for the Sequoyah plant specific loads includes the changes identified. The stresses in the anti-debris bottom nozzle of the Mark-BW fuel assemblies for normal operating conditions as well as for faulted conditions are reported in Tables 8.1 and 8.5 of BAW-10220P, Rev. 0 respectively. The worst case loading for normal operation is at the design flow rate plus [ ] scram load. The minimum margin of safety is [ ] for membrane plus bending stress, so that the stresses are acceptable. The minimum margin of safety for the Sequoyah LOCA plus SSE loads is [ ] for membrane plus bending stress. The allowable stresses are determined as set forth in the ASME Boiler and Pressure Vessel Code.

11. The pressurizer heaters and sprays are not modeled for the non-LOCA analysis. The results of some transients are worse if these control features function (i.e., the peak steam generator pressure can be higher if the sprays act to delay a reactor trip on high RCS pressure). The staff safety evaluation (SE) on the methodology requires consideration of the control features. Describe why these control features are not modeled.

Response

There are a total of six safety analysis events analyzed for Sequoyah reload. Each one will behave differently when the pressure control system (sprays, heaters and power operated relief valve) is assumed to function. The effects of the pressure control system are considered for each one of the transients analyzed. They are described below.

RCCA Withdrawal at Full Power

The 75 pcm/sec (maximum withdrawal rate) RCCA withdrawal case provided did not assume pressure control system operation so that a maximum primary system pressure could be calculated to assure that the RCS pressure acceptance criterion is met. However, a large number of RCCA withdrawal cases with various reactivity insertion rates were analyzed to verify that the minimum DNBR is greater than the limit value in all cases. Those analyses assumed that the pressure control system, most notably the pressurizer PORVs, controlled the primary system pressure to 2350 psig. Use of the pressure control system in these cases provided the minimum DNBR by limiting the increase in system pressure and by requiring a reactor trip on over-temperature ΔT.

In conclusion, independent analyses were performed for this event that either utilized or ignored the effects of pressurizer control to minimize the margin to the DNB and RCS pressure limits, respectively.

Loss of Electric Load

In this event, operation of the pressure control system mitigates the primary system pressure increase, providing a less conservative peak primary system pressure relative to the RCS pressure limit. Therefore, FCF analyzes the event without primary pressure control. It is true that operation of the pressure control system, specifically, the pressurizer PORVs, can delay reactor trip (by avoiding a high pressurizer pressure trip) until the over-temperature ΔT trip setpoint is reached. This can increase the peak secondary system pressure as compared with the case where the pressure control system is not assumed to function. However, in our experience, the over-temperature ΔT

trip is reached shortly after the high pressurizer pressure trip. Consequently, there is not a significant difference in secondary system peak pressures with or without primary pressure control system.

In conclusion, the results for this event are conservatively arrived at by ignoring the effects of the pressurizer pressure control system.

Loss of Forced Flow

The system response shown for this event assumes that the pressure control system does not operate. This provided the greatest peak pressure for the event. However, the core DNB response is calculated assuming that the core outlet pressure remains at the initial value throughout the event, yielding the most restrictive value for DNBR. Consequently, the results for this event are conservative, regardless of the availability of the pressure control system.

Locked Reactor Coolant Pump Rotor

The system response shown for this event assumes that the pressure control system does not operate. This provides the greatest peak pressure for this event. However, the core DNB response is calculated assuming that the core outlet pressure remains at the initial value throughout the event, yielding the most restrictive value for DNBR. Consequently, the results for this event are conservative, regardless of the availability of the pressure control system.

Main Steam Line Break

This accident results in a depressurization of the reactor coolant system. Because pressurizer heaters would act to increase the system pressure, which is non-conservative with respect to minimum DNBR, they are not modeled.

Steam Line Break Coincident With Rod Withdrawal at Power

This event is a Condition IV steam line break with a coincident withdrawal of the regulating control rod bank by the reactor control system. The event is terminated by reactor trip on a low steam line pressure SI signal or on an over-power ΔT trip. Neither of these functions are affected by the primary system pressure control system. Furthermore, the DNB response to this event is calculated assuming the core exit pressure remains at the initial value. Consequently, the calculated results are conservative regardless of the availability of primary system pressure control.

12. Discuss in greater detail the implications of no longer modeling a "hot channel" and an "average channel" for non-LOCA transient methodology (described on p. 6-4).

Response

In past applications, the full power RELAP5 model utilized in non-LOCA transient analyses were adapted from the RELAP5 LOCA model primarily because they simulate the plant at equivalent power levels. Hot channel modeling is a necessity in LOCA analysis because of the emphasis in predicting peak clad temperature in the associated hot pin with RELAP5-based computer codes. The hot channel is, however, not used in non-LOCA analysis.

The hot channel in predecessor (McGuire/Catawba and Trojan) reload transient analyses represented a single pin/channel and, relative to the rest of the core, insignificantly affects the progression of any of the non-LOCA transients. Full power non-LOCA analyses utilize the RELAP5 model to determine the general plant, or system, response to transients. The average channel inlet flow, temperatures, and system pressure response are combined for use as boundary condition information for detailed subchannel analyses. The subchannel analyses are performed independent of the RELAP5 model utilizing computer codes such as LYNXT for the determination of hot pin/channel DNB margin.

For the SQN applications, the zero-power steam line break RELAP5 model was adapted for use in the analysis of the remaining non-LOCA events. This is a deviation from past modeling efforts in support of reload. The non-LOCA model is, however, uniform for all transients and represents a simplification of past modeling approaches.

In summary, an explicit representation of the hot channel is not included in the non-LOCA RELAP5 model. This is acceptable because the hot channel contribution to non-LOCA transient progression is not noticeable. Further, the hot channel response to non-LOCA transients has not been used in the past, detailed hot pin analyses are performed by subchannel codes. The standardization of the non-LOCA RELAP5 model, finally, represents a simplification in model construction.

13. The staff SE for BAW-10169 states that the acceptance criteria for a locked RCP rotor is the 95/95 DNBR criteria; however, the analysis predicts DNB and applies the acceptance criteria for infrequent incidents (Condition III). Please correct this departure from the approved methodology.

Response

The 95/95 DNBR limit is used as the fuel failure criterion as per BAW-10169. Any pin that fails this limit is considered as failed in calculating the effects of failed pins on the radiological consequences of the event. The prediction shows less than 5 percent fuel pin failure, using this criterion. Existing dose calculations assume 10 percent failed pins. Because less than 5 percent pin failure is predicted in the calculations, the dose consequences of this event are bounded by the existing calculation. The fuel/clad temperature limits are used to demonstrate that the core stays in a coolable configuration.

14: The acceptance criteria established (pp. 2-3 and 7-1) for events of moderate frequency (condition I and II) includes a 99.9% probability that "DNB will not occur core wide." The SRP acceptance criteria requires "at least 99.9% of fuel rods in the core will not experience DNB" (SRP 4.4-3). The two acceptance criteria are not equivalent. Please correct or clarify the difference.

Response

The wording of the DNBR acceptance criterion as established in BAW-10220P (pp. 2-3 and 7-1) was a misstatement. However, FCF analysis practice is correct and consistent with SRP 4.4-3. Specifically, as stated in BAW-10170P-A "Statistical Core Design for Mixing Vane Cores" (p. 1-7), "The SDL of 1.345 (subject to core-specific verification) developed in this report [BAW-10170P-A] provides 95 percent protection at a 95 percent confidence level against hot pin DNB. The corresponding corewide protection on a pin-by-pin basis using real peaking distributions is greater than 99.9 percent." This agrees with SRP 4.4-3.

FCF has performed the core-specific verification of the SDL for the Sequoyah core and shown that the 1.345 SDL remains valid.

15. It is unclear from the submittal which of the Chapter 15 analyses were redone to support the fuel change and which ones were re-evaluated. Table 6.1-1 is not consistent with the verbiage in text on a number of examples. Table 6.1-1 only lists six transients that were reanalyzed; however, analysis results are discussed for other transients (for example, misaligned RCCA discusses results in Section 6.2.3).

    To clarify the situation, state for each Chapter 15 transient whether it was reanalyzed using the Framatome methods with acceptable results or why reanalysis is not necessary (why current analysis remains bounding with the new fuel and a lower RCS flowrate). Include both the effects of the new fuel and the lower RCS flow.

Response

Table 6.1-1 presents the non-LOCA "transient events" that were analyzed and those that were evaluated in support of fuel reload at SQN. The analyzed events include:

1. Rod Cluster Control Assembly (RCCA) Withdrawal at Full Power

2. Loss of Electric Load

3. Four Pump Coastdown

4. Main Steam Line Break (MSLB)

5. Locked Reactor Coolant Pump Rotor

6. Steam Line Break with Coincident Rod Withdrawal at Power

Several events, the misaligned RCCA included, are not transient events in the strictest sense; time-dependent hydraulics (RELAP5) calculations are not performed for these events. They may be, however, either evaluated or analyzed as static physics exercises for each cycle in support of individual core designs. A new table has been constructed - Table 15-1 - to indicate the non-LOCA events that are: (1) analyzed with RELAP5 to support the loading of Mark-BW fuel at SQN, (2) evaluated only, and (3) analyzed/evaluated as part of the core design. Note that errors contained in Table 6.1-1 (the loss of feedwater was not analyzed, the loss of coolant flow was) have been corrected in Table 15-1.

Section 6 of the topical contains an event by event evaluation of FSAR chapter 15 transients in support of the fuel reload, only. The 6 transients chosen for re-analysis comprise the most limiting event in the SRP event classifications - overcooling, heatup, loss of RCS flow, and reactivity anomaly - and are re-

analyzed with FTI (Framatome Technology Inc.) methodology to demonstrate compliance of the Mark-BW fuel with the relevant success criteria for each event. Analyses in chapter 6 are a direct application of, and conform to the approved methods of, BAW-10169.

Safety analyses supporting the operation of SQN with Mark-BW fuel were performed with lower-limit flow rates of 348,000 gpm. Use of lower flows in the analyzed events is conservative, with the exception of steam line break and steam line break coincident with rod withdrawal at power (the steam line break transient was performed with zero percent steam generator tube plugging to maximize primary heat removal and the RCS flows for the event are higher as a result). Of the various acceptance criteria, the greatest negative impact of RCS flow reduction is on DNB margin. It should be noted that the limiting events with respect to RCS flow are contained in the events chosen for the support of the fuel reload as the events that are also bounding in DNB (Condition II - RCCA withdrawal at power, Condition III - four pump coastdown, and Condition IV - locked rotor). All of the limiting event successfully demonstrate adherence to the relevant acceptance criteria. FCF will, however, compile an evaluation of the Chapter 15 events that is independent of this response. The evaluation will directly demonstrate compliance with the NRC topical query regarding lowered RCS technical specification limits.

Table 15 - 1
Summary of Non-LOCA Assessment for Reload with Mark-BW Fuel

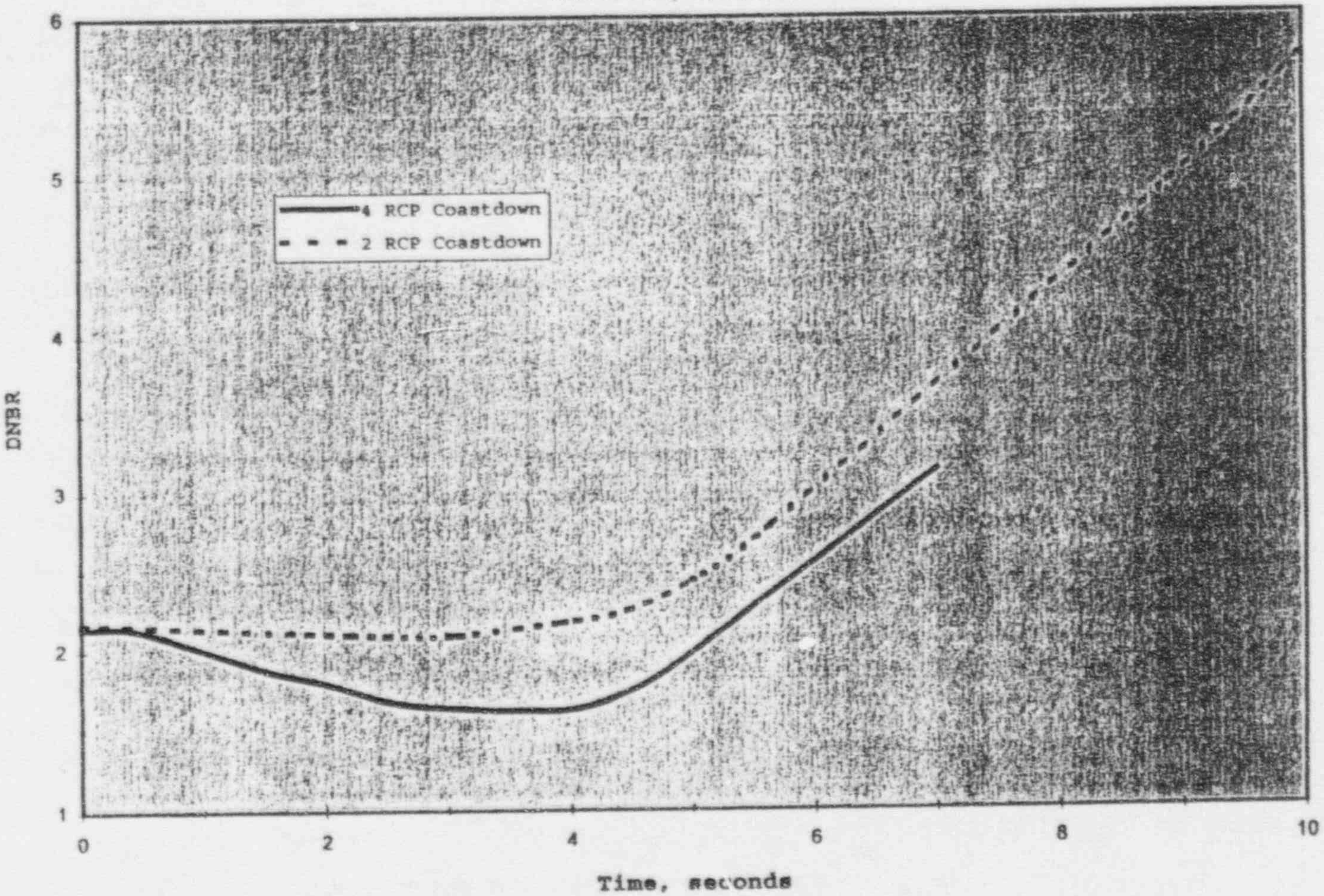
Event	Full Safety (RELAP5) Analysis	Qualitative Evaluation	Core Design Analysis
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition		x	
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power	x		
Dropped RCCA/Bank			x
Misaligned RCCA			x
Boron Dilution			x
Partial Loss of Reactor Coolant Flow		x	
Loss of External Electrical Load and/or Turbine Trip	x		
Loss of Normal Feedwater		x	
Loss of Non-Emergency AC Power to the Station Auxiliaries		x	
Excessive Heat Removal Due to Feedwater System Malfunctions		x	
Excessive Increase in Steam Flow		x	
Accidental Depressurization of the Reactor Coolant System			x
Spurious Operation of the Safety Injection System at Pcwer		x	
Inadvertent Loading of a Fuel Assembly into an Improper Position		x	
Complete Loss of Forced Reactor Coolant Flow	x		
Single Rod Cluster Control Assembly Withdrawal at Full Power			x
Rupture of a Main Steam Line	x		
Major Rupture of a Main Feedwater Pipe		x	
Steam Generator Tube Rupture		x	
Single Reactor Coolant Pump Locked Rotor	x		
Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)			x
Steam Line Break at Power With Coincident Rod Withdrawal	x		

16. There is no justification that a complete loss of flow is more limiting than a partial loss of flow as stated in the submittal. Please provide the justification.

Response

Both partial and complete loss of flow transients were analyzed in the McGuire/Catawba application report for Mark-BW reload, sections 4.3.1 and 4.3.2 of BAW-10173, respectively. The results of these analyses clearly demonstrate that because the complete loss of flow exhibits a more severe reduction in core flow as compared with the partial loss of flow event, the DNB response for the complete loss of flow is bounding (see Figure 16-1). McGuire/Catawba and SQN are similar in design (all are 4-loop Westinghouse plants) with identical power densities (Mark-BW cores rated at 3411 MWth), similar flows, etc. It is, therefore, concluded that the comparative results apply equally to SQN - the complete loss of coolant flow event bounds the partial loss of coolant flow event with respect to DNBR.

Figure 16-1 Loss of Coolant Flow DNB Response
McGuire/Catawba



17. For a number of transients (loss of flow), a delayed neutron fraction corresponding to end of life (EOL) is chosen. Describe why this is conservative.

Response

For events, such as the loss of flow event, that are posed to conservatively predict a minimum margin to DNB, the power response prior to reactor trip is maximized. One of the measures taken to maximize power response is to utilize the most limiting reactivity feedback parameters, e.g. moderator and Doppler feedback, to suit the transient even if this requires mixing beginning-of-life and end-of-life coefficients. Power response, however, is related to both reactivity and delayed neutron fraction. A minimum, EOL, delayed neutron fraction maximizes the power response to a given reactivity addition. This method of modeling reactivity feedback is used frequently in FCF safety analysis and is consistent with the approved safety analysis methodology of BAW-10169.

18. Describe how the stuck rod is modeled for each transient and identify which analyses assume a stuck rod and which ones do not.

Response

All of the events analyzed in the reload applications topical assume a stuck rod. The most reactive rod is assumed to be stuck out of the core and is reflected in both the development of the rod worth for reactor trip or in the characterization of plant shutdown margin. In addition, the steam line break analysis explicitly models localized peaking and reactivity feedback near the stuck rod in the process of predicting DNB margin.

19. The transient analysis presented does not discuss single failure assumptions in any detail. For each transient, state the limiting single failure chosen and why it is limiting.

Response

The following six events are were reanalyzed for use of FCF fuel at Sequoyah.

1. RCCA withdrawal at full power
2. Loss of electric load
3. Loss of forced flow
4. Locked rotor
5. Main SLB at subcritical condition
6. Steam line break coincident with rod withdrawal at power

Each one of these events is discussed here with respect to the single active failure assumption used.

RCCA Withdrawal at Full Power

The only active safety-grade system that mitigates this event is the reactor protection system. A single failure of the RPS will not stop it from performing its design function (i.e. insert control rods).

Loss of Electric Load

The only active safety-grade system that mitigates this event is the reactor protection system. A single failure of the RPS will not stop it from performing its design function (i.e. insert control rods). Although pressurizer safety valves are passive devices and are exempt from the single failure requirement, a single pressurizer safety valve was failed to maximize the primary system pressure for this event.

It should be noted that the auxiliary feedwater system, which is an active safety system, is required for long-term core heat removal. This system is also designed with the defense-in-depth concept, allowing for single failures of various components within the system. This system is not modeled in our analyses because it does not mitigate the system pressure or core DNB responses and because FCF fuel does not affect the ability of the system to meet its design function.

Loss of Forced Flow and Locked Reactor Coolant Pump Rotor

The only active safety-grade system that mitigates these events is the reactor protection system. A single failure of the RPS will not stop it from performing its design function (i.e. insert

control rods). Although pressurizer safety valves are passive devices and are exempt from the single failure requirement, a single pressurizer safety valve was failed to maximize the primary system pressures for these events.

Main Steam Line Break

The active safety-grade systems that mitigate this event are the reactor protection system, the steam and feedwater isolation systems, and the safety injection system. A single failure of the RPS will not stop it from performing its design function (i.e. insert control rods). Failure of a single main feedwater isolation valve to close cannot prevent isolation of feedwater to the affected steam generator because an isolation signal closes the feedwater control valves and trips the feedwater pumps. Failure of the feedwater pumps to trip cannot prevent feedwater isolation because the feedwater isolation valves will close. Consequently, the only single active failure that can affect the core response to main steam line break is the failure of a single train of safety injection. This failure minimizes the injection of boron into the reactor coolant system. This is the single active failure assumed in the FCF analysis of this event.

Steam Line Break Coincident with Rod Withdrawal at Power

This event results in an increase in power and a coincident decrease in core DNBR until the reactor trips and control rods are inserted. The only active safety-related component that mitigates this event is the reactor protection system. A single failure of the RPS will not stop it from performing its design function (i.e. insert control rods).

20. For the main steam line break analysis (p. 6-54), the volume of water between the cold leg piping and the first check valve is considered to be at 0 ppm boron concentration. The current Final Safety Analysis Report (FSAR) analysis assumes the volume of water in the piping from the RWST to the cold leg piping (a much bigger volume) is all at 0 ppm. Please describe how you can assure that the water in the piping between the RWST and the first check valve is at least 1950 ppm.

Response

The safety injection system has two functions, post-accident injection and the filling of the accumulators. In the post-accident injection mode, the system delivers borated water from the refueling water storage tank (RWST) to the RCS. In filling the accumulators, the system delivers borated water from the RWST to the accumulators. Since the boron concentration in the RWST is controlled administratively to a technical specification limit (1950 ppm), it is reasonable to assume that the suction piping of the SI system also remains at a boron concentration of 1950 ppm.

For the main steam line break analysis, it is assumed that the coolant in the safety injection line, downstream of the check valve accounts for possible dilution via the diffusion of RCS coolant back into the lines. Once this coolant is purged, boron is assumed to be injected into the RCS at a concentration of 1950 ppm. FCF is convinced that this is a reasonable method of modeling post-accident boron injection and that it reflects the normal plant configuration. Further, the limiting main steam line break does not require boron to mitigate the transient prior to the occurrence of peak power and minimum DNBR. Nominal deviations in SI/RWST boron concentrations will, therefore, not affect the results as reported. Assumptions regarding the modeling of boron injection will be clearly defined in revisions to the FSAR subsequent to Mark-BW fuel load.

21. Explain why there is no flow to the intact steam generators after 20 seconds into the main steam line break transient analysis (Figure 6.4-13). Shouldn't emergency feedwater start injecting when main feedwater is isolated?

Response

The Sequoyah Nuclear Plant design criteria for auxiliary feedwater (SQN-DC-V-13.9.8 Revision R6, revised through 3/7/96) require that the maximum flow contribution from the AFW system to containment during a main steam line rupture must not exceed 2250 GPM. A conservatively high auxiliary feedwater flow rate (2350 GPM) is assumed to be delivered only to the faulted steam generator to increase the severity of the core cooldown during the main steam line break transient. The assumption is consistent with the analysis of record (see FSAR Section 15.4.2.1.2). This is why there is no flow to the intact steam generators (SGs) after main feedwater flow is isolated after 22 seconds as per Table 6.4-1. In fact, the intact SGs become a heat source shortly after closure of the MSIVs (at 10 seconds as per the Table 6.4-1). Thus, distributing a part of the emergency flow to the intact SGs would have resulted in less severe overcooling.

22: A statistical core design (SCD) methodology is used to analyze some of the transients and used to derive some safety limits, peaking limits, and departure from nucleate boiling ratio (DNBR) limits. Describe how the non-SCD transients are used to provide input to the same limits. Also, identify the conditions for their application to this reload.

Response

The DNB-based safety limits and DNB-based peaking limits were developed using the SCD methodology as defined in BAW-10170P-A. The SCD methodology is applicable because the system and core conditions for these calculations remain within the allowable parameter ranges defined in Table 2.2 of BAW-10170P-A, as shown below.

Parameter		Units	Allowable Parameter Range
Q	Reactor Thermal Power	(% of 3411 MWt)	[ ]
W	RCS Flow	(% of Nominal Flow)	[ ]
P	System Pressure	psig	[ ]
T	Inlet Subcooling	°F	[ ]

Similarly, those transients that have predicted system and core conditions within the range of the parameters shown above were analyzed with the SCD methodology. However, those transients that have predicted system and core conditions falling outside the range of the parameters shown above are analyzed using non-SCD methods (where the parameter uncertainties are treated in a deterministic manner). The non-SCD transients include the steam line break event and the subcritical rod withdrawal event. The results of these transients are not used in establishing DNB-based safety and/or peaking limits. However, the non-SCD analyses are used to evaluate the relevant core parameters for the Sequoyah core and establish that the appropriate acceptance criteria are met. For the steam line break, it was shown that DNB does not occur and that the fuel and cladding temperatures are acceptable (Section 6.1.4 of BAW-10220) For the subcritical rod withdrawal, the non-SCD evaluation established that the reference analysis remained bounding and applicable (Section 6.2.1 of BAW-10220).

23: Is rod bow explicitly accounted for in the DNBR methodology? If so, where is it accounted for (retained margin, DNBR penalty, peaking factor adjustments)?

Response

Yes, the impact of potential fuel rod bow is explicitly accounted for in FCF's DNBR methodology.

As discussed in Section 4.1.1.7 of BAW-10172P "Mark-BW Mechanical Design Report", the Mark-BW fuel design has several features (non-rigid grid attachment and keyable grids) that make its fuel rod bow performance similar to that of other FCF fuel designs. In BAW-10186P "Extended Burnup Evaluation", FCF presented new data that extended the rod bow data base for FCF fuel to [ ] The topical report concluded that the rod bow correlations from BAW-10147PA-R1 "Fuel Rod Bowing in Babcock and Wilcox Fuel Designs" are applicable at extended burnups and apply to the Mark-BW.

BAW-10147PA-R1 has shown [ ] The average power hot channel factor, $F_{\Delta H}^{\varepsilon}$, has a value of 1.03 and is combined statistically with other uncertainties to establish the statistical design limit (SDL) DNBR used with the statistical core design method (discussed in Section 7.2 of BAW-10220). For non-SCD analyses, the average power hot channel factor, which includes the rod bow effect, is incorporated into the LYNXT model as a direct multiplier on the hot pin average power.

24: Why is the RCS pressure used for DNBR purposes (Table 7.1-2) chosen to be 2280 psia when the nominal RCS pressure is 2235 psig?

Response

The RCS pressure quoted for DNB analyses is the pressure that sets the boundary condition for the analyses, that being the core exit pressure. A 2280 psia core exit pressure used for core DNBR predictions (Table 7.1-2) is based on a nominal RCS pressure of 2235 psig (at the pressurizer) with adjustments for gage pressure and pressure drop from the pressurizer to the core exit.

Core Exit Pressure	=	Nominal RCS Pressure	+	15 psi adjustment to gage	+	30 psi pressure drop from the pressurizer to the core exit
2280 psia	=	2235 psig	+	15 psi	+	30 psi

25: Please provide a justification for the Sequoyah-specific uncertainties chosen to calculate the statistical design limit.

Response

The Sequoyah-specific uncertainty allowances incorporated into the determination of the statistical design limit (SDL) for core power, RCS flow rate, RCS pressure, inlet temperature, and measured $F_{\Delta H}^{N}$ are justified as follows:

Parameter	Uncertainty	Basis
Core Power	[]	This value represents the maximum allowance for calorimetric measurement uncertainty and is consistent with the maximum steady state uncertainties outlined in Sequoyah FSAR Section 15.1.2.2
Core Flow	[]	This alue represents a bounding "generic uncertainty" for RCS flow measurement which is consistent with Technical Specification 3/4.2.5 (Table 3.2-1). The value is conservative with respect to actual flow measurement techniques used to verify minimum RCS flow values which have 2.4% uncertainty (i.e. elbow tap data normalization to baseline calorimetric flow measurements).
Core Pressure	[]	This value represents the maximum allowance for steady state system pressure fluctuations and instrument measurement errors. It is consistent with the maximum steady state uncertainties outlined in Sequoyah FSAR Section 15.1.2.2.
Core Inlet Temperature	[]	This value represents the maximum allowance for deadband and instrument measurement errors. It is consistent with the maximum steady state uncertainties outlined in Sequoyah FSAR Section 15.1.2.2.

Measured $F_{\Delta H}^{N}$	[]	This value represents a bounding uncertainty for measurement of the enthalpy hot channel factor using the incore neutron detectors and is consistent with the uncertainty required by Technical Specification 4.2.3.2.c.

26: The critical heat flux correlation used for non-LOCA applications is approved for both the Framatome and Westinghouse fuel; however, it has not been approved for mixed core applications. Provide an appropriate penalty to the limiting rod that will bound the misapplication of the critical heat flux correlation. Additionally, the analysis provided does not include the Westinghouse standard fuel in the mixed core penalty. Provide a justification why this is not accounted for.

Response

The BWCMV CHF correlation is approved for application to homogenous cores of Mark-BW or Westinghouse VANTAGE 5H fuel. In addition, FCF has demonstrated that [ ] This conclusion is based on the fact that the BWCMV CHF performance level is applicable to the VANTAGE 5H fuel design and that the hydraulic effects of transition cores are conservatively evaluated in the method used to predict steady-state and transient DNBRs. Specific mixed core analysis results that support this conclusion were submitted by letter to the NRC on 1/10/97 [1] and are discussed in greater detail in the response to Question 28.

The presence of Westinghouse standard fuel assembly reinserts in Sequoyah transition cores is not significant because: 1) the number of standard fuel assemblies planned for use is relatively low, and 2) the Westinghouse standard fuel assemblies will have relatively low-powered operation as reinserts and will therefore be non-limiting. Also, the mixed core penalty analysis does not include the Westinghouse standard fuel because the effect of these assemblies is bounded by the effect calculated for the transition from VANTAGE 5H to Mark-BW fuel. The transition penalty applicable to the transition from Westinghouse standard fuel to the Mark-BW was shown in BAW-10178 to be [ ] This penalty was calculated with the same mixed core method and modeling technique as used to calculate the corresponding penalty for transition from VANTAGE 5H to Mark-BW. In both cases the penalty was determined for the bounding case, [ ]

1. Letter, J. H. Taylor, Framatome Technologies to USNRC, Thermal-Hydraulic Methods for the Transition from Vantage 5H to Mark-BW Fuel at TVA's Sequoyah Plant, JHT/97-2, January 10, 1997

27: Describe background information and the bases for those studies related to the Trojan Plant which concluded that a [ ] transition core DNBR penalty should be applied to the Mark-BW when it is being inserted into a Westinghouse standard core with respect to the hydraulic compatibility of the Mark-BW fuel design with the Westinghouse standard design. Identify the similarity or difference in relation to the transition core DNB penalty between the Sequoyah and Trojan reloads.

Response

The analysis methods used to determine and apply the transition core penalty for the introduction of the Mark-BW at Trojan are the same methods used in the analyses supporting the transition to Mark-BW fuel at Sequoyah. The principle followed in the mixed core analyses was to determine the minimum DNBR performance for the core by examining all bounding mixed core combinations and using the most limiting result to establish the transition core penalty. The bounding combinations included:

Configuration Number	Configuration Description
1	Full Core Resident Fuel Design
2	[ ]
3	[ ]
4	Full Core of Mark-BW Fuel

A [] transition core penalty was applied to the Mark-BW at Trojan. The penalty was determined by examining the DNBR differences between Configurations #2 and #4 for statepoints that set the DNB-based safety limits and operating limits as well as the limiting DNB transient. Although the actual transition core penalty would have been smaller than [] if the DNB reload analyses had considered the actual cycle-specific core distribution of fuel designs with a configuration somewhere between Configurations #2 and #3, the conservative application of the [] transition core penalty assured DNB protection.

In the same manner, the [] transition core penalty (applied to the Mark-BW results using Configuration #4) was determined for the Mark-BW introduction at Sequoyah. Specific mixed core analysis results that support the [] transition core penalty were submitted by letter to the NRC on 1/10/97 [1] and are discussed in greater detail in the response to Question 28.

1. Letter, J. H. Taylor, Framatome Technologies to USNRC, Thermal-Hydraulic Methods for the Transition from Vantage 5H to Mark-BW Fuel at TVA's Sequoyah Plant, JHT/97-2, January 10, 1997

28: Provide the final conservatively bounding mixed core configuration for SQN mixed core DNBR analysis and the transition penalty based on assuming that the center hot assembly is either a single Mark-BW fuel assembly in a core of the VANTAGE 5H or a single VANTAGE 5H in a Mark-BW core. Also, provide the result of the DNBR analysis using plant- and cycle-specific core loading configuration and the same limiting power distribution input in the above analyses. Show that the VANTAGE 5H to Mark-BW design peak difference will offset any transition core effects on the VANTAGE 5H and provide the description of the retained thermal margin in relation to the transition core penalty.

Response

The method of determining the DNB-based safety limits and peaking limits for the mixed core situation at Sequoyah is described in detail in the discussion of Sequoyah transition core thermal-hydraulic analysis methods submitted by letter to the NRC on 1/10/97 [1]. FCF determined the DNB-based safety limits and peaking limits by examining the performance of the resident VANTAGE 5H and the Mark-BW in the four configurations defined in the response to Question 27:

Configuration Number	Configuration Description
1	Full Core Resident Fuel Design
2	[ ]
3	[ ]
4	Full Core of Mark-BW Fuel

One ground rule established for the thermal-hydraulic DNB analyses was that the existing reactor core safety limits would be maintained for Sequoyah. Therefore, the DNB assessment was composed of statepoint evaluations which demonstrated that the minimum DNBR remains above the thermal design limit (TDL) of 1.50 using SCD methodology. The DNB-based reactor core safety limits were evaluated using all the above configurations. Results showed [ ] thermal margin to the TDL for the existing DNB-based safety limits.

The VANTAGE 5H DNB-based peaking limits were developed using Configurations #1 (full VANTAGE 5H core) and #3 [ ] Final peaking limits for the resident fuel design were established by calculating allowable peaks with both of these configurations and then selecting the most limiting values as the final limit. The Mark-BW DNB-based peaking limits were developed using Configuration #4 (full Mark-

BW core). The transition core penalty for the Mark-BW was developed by evaluating the DNBR difference between this model and [ ] core model (Configuration #2). The [] transition core penalty for the Mark-BW bounds the most limiting of the DNBR differences determined by that comparison.

DNBR calculations using plant- and cycle-specific core loading configurations with the design radial and axial peaking imposed on the limiting hot bundles of the respective fuel designs have also been performed. Those analyses utilized the [ ] model depicted in Figure 3 of the 1/10/97 letter that defined mixed core methods. As noted below several different configurations were analyzed with the emphasis being on isolating the DNBR difference between the full core models, the limiting design models, and the cycle specific core loading models. For the cycle specific core loading models, all three assembly types that will be in core (MK-BW, V5H and W STD) were modeled in their actual locations. Results are summarized as follows:

Cases with Design Power Distribution Hot Bundle at [ ]		
	BWCMV MDNBR	% Difference (w.r.t. Full Core)
Full Core MK-BW	[ ]	
[ ]	[ ]	[ ]
Full Core V5H	[ ]	
[ ]	[ ]	[ ]

Cases with Cycle Specific Power Distribution Scaled to Design Peak For MK-BW Cases: Hot Bundle at [ ] For V5H Cases: Hot Bundle at [ ]		
	BWCMV MDNBR	% Difference (w.r.t. Full Core)
Full Core MK-BW	[ ]	
Cycle Specific Core Loading - Hot MK-BW	[ ]	[ ]
Full Core V5H	[ ]	
Cycle Specific Core Loading - Hot V5H	[ ]	[ ]

The fact that the DNBR difference for the full core to single assembly comparison is greater that the DNBR difference for the full core to cycle specific comparison shows that the [ ] mixed core models conservatively represent the cycle specific core loading pattern and that the method FCF has used to develop core peaking limits and the transition penalty for the initial Sequoyah transition core is bounding and conservative. For future reload cores, FCF will utilize cycle-specific core loading patterns and power distributions to reduce the mixed core penalty as the number of Mark-BW assemblies increases and the number of resident Westinghouse assemblies decreases.

The design radial peaks for the VANTAGE 5H and Mark-BW are 1.62 and 1.70, respectively. By maintaining the VANTAGE 5H fuel at its current design peak, it is ensured that the resident fuel will retain its current safety margins and that the safety margins for the Mark-BW and VANTAGE 5H are similar. However, this difference in design radial peaks is not required to support the absence of a VANTAGE 5H fuel design transition core penalty, since the method of evaluating the VANTAGE 5H performance in bounding core configurations yielded DNBR results that require [ ]transition core penalty on the resident fuel.

The retained thermal margin for the Sequoyah DNB analysis resides in the thermal design limit (TDL). The TDL (1.50) is 10% higher than the 1.345 statistical design limit (SDL) using the BWCMV CHF correlation.

Retained Margin in the Thermal Design Limit	[]
Transition Core Penalty for the Mark-BW	[]
Balance of Retained Thermal Margin for the Mark-BW	[]

Similarly, the VANTAGE 5H fuel design has an equivalent [] retained thermal margin:

Retained Margin in the Thermal Design Limit	[]
Transition Core Penalty for the Vantage 5H	[]
Rod Bow Penalty for the Vantage 5H	[]
Balance of Retained Thermal Margin for the Vantage 5H	[]

1. Letter, J. H. Taylor, Framatome Technologies to USNRC, Thermal-Hydraulic Methods for the Transition from Vantage 5H to Mark-BW Fuel at TVA's Sequoyah Plant, JHT/97-2, January 10, 1997

29: Provide clarification of the limited use of Westinghouse standard reinserts in a SQN transition core application and provide justification that the transition penalty will bound the SQN application.

Response

The last full batch of Westinghouse standard fuel assemblies was discharged from the Sequoyah cores in 1994. Since that time, the Westinghouse standard fuel assemblies have been reinserted in limited numbers as an economic measure and as a means to reduce peaking next to adjacent fresh fuel. Since the standard fuel assemblies available to TVA have undergone earlier irradiation, they can only be used in limited numbers because of their reduced energy potential. For the first Mark-BW transition cycle at Sequoyah 1 there will be ten Westinghouse standards assemblies in core. TVA expects to continue using the Westinghouse standard fuel assemblies in this limited fashion for future cycles.

FCF has shown based on hydraulic testing that the standard fuel assembly has a pressure drop [ ] than either the resident VANTAGE 5H or Mark-BW design. Therefore, the standard fuel assemblies will receive, on the average, slightly more flow than the other two designs in the mixed core. Combining this effect with the fact that the standard fuel assemblies will be used in low power locations in the core clearly demonstrates that the standard fuel assemblies will have appreciably more thermal margin than either the VANTAGE 5H or the Mark-BW.

As discussed in BAW-10178 (the transition core report for the Westinghouse standard to Mark-BW transition at the Trojan plant), because the Westinghouse standard has [ ] than the Mark-BW there is no need for a transition core penalty on the Westinghouse standard. However, that report established a [] penalty for the Mark-BW during that transition. As outlined in the response to Question 26, that [] penalty is bounded by the [] penalty being imposed on the Mark-BW in the Sequoyah transition core.

30: Provide the bases for obtaining 2% of an increase in lift force for the limiting transition core configuration (one VANTAGE 5H in a Mark-BW core). Also, provide the data for lateral crossflow velocities for the mixed core configuration and an acceptable criterion for lateral crossflow.

Response

The basis for the 2% increase in lift force is the increase in pressure drop seen when the total unrecoverable pressure drop for the single VANTAGE 5H assembly in the Mark-BW core is compared to the total unrecoverable pressure drop for the VANTAGE 5H assembly in the same core location in the full VANTAGE 5H core. [ ] As FCF demonstrated with detailed mixed core hydraulic evaluation results submitted to the NRC by letter dated 1/10/97 [1], the predominate hydraulic difference between the two assembly types is the pressure drop across the grids. [ ] This flow diversion results in the small increase seen in the VANTAGE 5H lift force in the mixed core.

Plots of crossflow velocities for the limiting mixed core configurations were submitted to the NRC in the 1/10/97 letter. Those plots showed that the maximum crossflow velocity in the mixed core is less than [] and the corresponding span average crossflow velocity is less than [] The design acceptance criterion for crossflow, to preclude any adverse flow induced vibration effects, is that the average crossflow velocity across an axial span between two spacer grids must be less than [] Therefore, it can be seen that the lateral crossflow velocities for the mixed core configuration are well within the acceptance criterion.

1. Letter, J. H. Taylor, Framatome Technologies to USNRC, Thermal-Hydraulic Methods for the Transition from Vantage 5H to Mark-BW Fuel at TVA's Sequoyah Plant, JHT/97-2, January 10, 1997

31: Form loss coefficients for the fuel subcomponents were determined using measured pressure drops. A LYNXT hydraulic model using those form loss coefficients showed that the total pressure drop of the Westinghouse VANTAGE 5H design is approximately [ ] than that of the Mark-BW and the Westinghouse standard fuel assembly is approximately [ ] in the pressure drop than the current Mark-BW. Provide the detailed analysis with respect to the overall impact on the mixed core DNBR analysis based on these [ ] pressure drops. Also, describe how the Figure 3.2 is generated and its application to the mixed core DNBR calculation if flow is much greater than 383,000 gpm.

Response

The VANTAGE 5H fuel design exhibits about a [] greater total fuel assembly pressure drop than the Mark-BW. However, as was demonstrated through the detailed mixed core calculations submitted to the NRC by letter dated 1/10/97 [1], the mixed core DNBR prediction is influenced more by the axial distribution of the hydraulic differences than absolute difference in total assembly pressure drop.

Results from the hydraulic testing of the VANTAGE 5H fuel design showed that [ ] The pressure drop difference in the lower part of the assembly is further exaggerated by the fact that the first intermediate grid on the Mark-BW is a non-mixing grid while the VANTAGE 5H has a mixing grid at that location. However, as Figure 3-2 of BAW-10220 shows, as elevation increases, []

The impact of these component hydraulic differences is illustrated by the hot channel mass velocity plots included with the 1/10/97 letter. Those plots show that in the mixed core [ ] (As discussed in the response to questions 26 through 28, a bounding [] penalty has been applied to the first transition core).

Similar behavior was demonstrated for the transition from Westinghouse standard fuel to the Mark-BW at the Trojan Plant in BAW-10178. For that transition, [ ] For DNBR analyses in mixed cores the grids are the most important flow resistance and therefore the results of the two analyses (when bounding cases are selected) are quite similar. The bounding transition core penalty, applied to the Mark-BW was [] for the transition from Westinghouse standard to Mark-BW.

Figure 3-2 was developed to demonstrate the pressure drop differences between the resident VANTAGE 5H fuel design and the Mark-BW design at nominal plant operating conditions (i.e. the best estimate flow rate). The figure is generated by analyzing a homogenous core of each fuel type and plotting the corresponding total core pressure drop versus core height for the two cases on a single figure. Pressure drop conditions have been calculated for higher flow rates (greater than 383,000 gpm) for hydraulic

lift analyses. For those cases, the trend between the two fuel designs' axial pressure drop profiles remains the same but at elevated pressure drop magnitudes. As presented, Figure 3-2 is not a direct reflection of conditions evaluated in the transition DNB analyses, since those analyses are performed at the thermal design flow of 360,100 gpm. However, the flow diversion trends discussed earlier in this response are valid regardless of flow rate.

1. Letter, J. H. Taylor, Framatome Technologies to USNRC, Thermal-Hydraulic Methods for the Transition from Vantage 5H to Mark-BW Fuel at TVA's Sequoyah Plant, JHT/97-2, January 10, 1997

32. The horizontal seismic and LOCA structural loads were calculated for the mixed core for Mark-BW fuel and Westinghouse Standard fuel. Why was Westinghouse V-5H not used? Additionally, explain how the mixed core calculation was performed to assure conservative results.

Response

Framatome Cogema Fuels (FCF) has calculated the horizontal seismic and LOCA loads for the mixed core for Mark-BW and Westinghouse Vantage 5H fuel assemblies as described on page 8-6 of BAW-10220P, Rev. 0. The results of the calculations were compared with the seismic and LOCA loads results of the full Mark-BW core configuration. The resulting changes in spacer grid impact loads are minor [ ] as stated on page 8-6 of BAW-10220P, Rev 0. The spacer grid impact loads for all the faulted conditions are within the elastic load limit. Therefore, the requirement of a core coolable geometry is met.

The core configurations as indicated below and shown in Figure 32-1 were analyzed for a 5-assembly model. These configurations are based on providing combinations of W Vantage 5H and FCF mark-BW fuel assemblies in mixed cores.

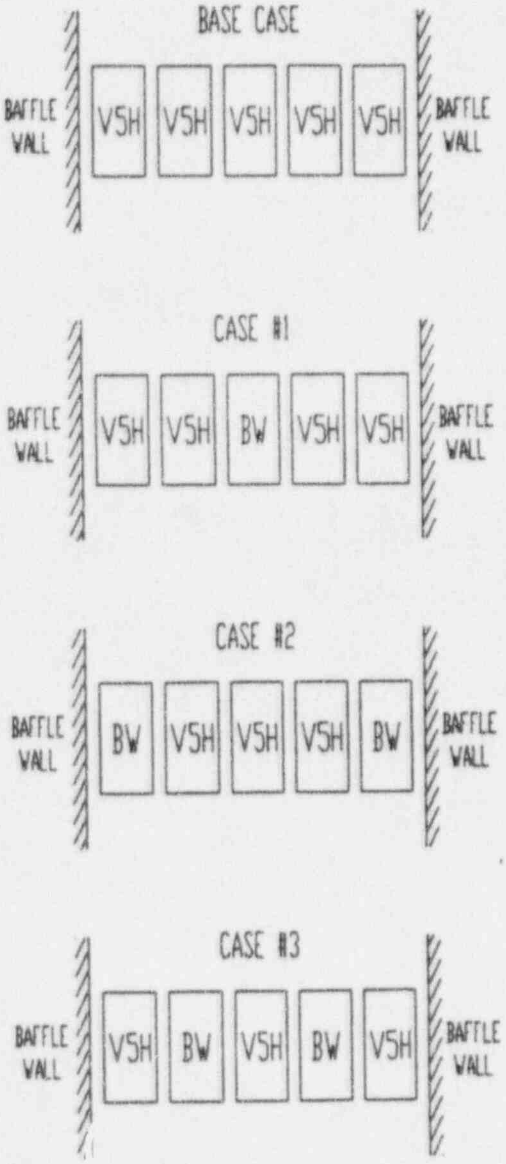
V5H	V5H	V5H	V5H	V5H	-	Base Case
V5H	V5H	BW	V5H	V5H	-	Core Conf I
BW	V5H	V5H	V5H	BW	-	Core Conf II
V5H	BW	V5H	BW	V5H	-	Core Conf III

The all-V5H case is given because that establishes a base line loading for V5H assemblies. A goal of the mixed core analysis is to demonstrate the adequacy of the FCF fuel assemblies when mixed with W resident fuel assemblies.

Based on previous calculations as discussed in Section 4.2 of BAW-10133P, Rev.1 (NRC approved topical), the five fuel assembly model gave the highest impact load, and as the number of FAs increased beyond five, the load decreased. For the Sequoyah plant, five FAs do not exist in any row of the core. The minimum number of fuel assemblies that exist in one row is seven. However, for additional conservatism, the five-FA row was selected for the Sequoyah plant.

Figure 32-1

Vantage 5H/Mark-BW Mixed Core Configuration



LEGEND:

V5H - WESTINGHOUSE VANTAGE 5H FUEL ASSEMBLY

BW - FCF ADVANCED BW FUEL ASSEMBLY

33. Is a full core analysis performed for the combined LOCA/Safe Shutdown Earthquake (SSE) loads for mixed core applications? Compare the critical loads (crushing loads) for the three types of fuel

Response

An analysis described in the response to above Question 32 demonstrates the adequacy of the FCF Mark-BW fuel assemblies when mixed with W Vantage 5H fuel assemblies. For W Standard fuel assemblies, a mixed core bounding analysis as discussed on page 8-6 of BAW- 10220P, Rev. 0 was performed for a mixed W and FCF fueled core. The resulting changes in spacer grid impact loads as discussed on page 8-7 of BAW 10220P, Rev. 0 are minor [ ] and well within the spacer grid elastic load limit (crushing load). Hence, the requirement of a core coolable geometry is met for all combinations of Westinghouse (Vantage 5H and Standard) and FCF Mark-BW fuel assemblies.

The elastic load limit for the Mark-BW zircaloy intermediate spacer grid at 600^0F is [ ] lbs. This value was determined through testing. The buckling strength (crushing strength) of the Westinghouse Vantage 5H and Standard spacer grids were determined through a comparison of basic grid geometry and the use of test data. The calculated crushing strength of the Westinghouse Vantage 5H and Standard spacer grids at 600^0F are:

P_{V5H} = [ ] lbs.

$P_{Standard}$ = [ ] lbs.

The load for the Seismic plus LOCA for the worst case mixed core configuration is [ ] lbs. This load is less than the elastic load limit of the Westinghouse Vantage 5H, Standard and the Mark-BW intermediate spacer grids.

34. The methodology approved for the mixed core structural analysis is contained in BAW-10133. There is no reference to this methodology in the submittal. Please verify that this methodology was used.

Response

A reference is made to the NRC approved FCF Mark-BW 17X17 fuel assembly topical report BAW-10172P. This topical report provides the structural evaluations specific to the Mark-BW fuel assembly. However, this topical report refers to BAW-10133P, Rev.1.(NRC approved) for the methodology used for the fuel assembly seismic and LOCA mechanical response analysis.

35. On p. 8-7 the stated design criteria (with a reference to the Standard Review Plan [SRP]) for the LOCA combined with the SSE does not include control rod insertability; however the SRP does require control rod insertability for this event. Correct the criteria and verify that control rod insertability is maintained for the combined loads.

Response

The acceptance criteria of the Standard Review Plan (NUREG 0800), Section 4.2, states that loads from the worst-case LOCA that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load. The FCF design basis LOCA does not require control rod insertion. Control rod insertion is only required for small break LOCA.

The FCF-supplied fuel assemblies will not inhibit control rod insertability for the design basis combined seismic and LOCA loads, as the spacer grid impact loads for all the faulted conditions and core configurations are within the spacer grid crush load limit. This assures that the FCF- supplied assemblies will not inhibit control rod insertability for small break LOCA, because the design basis combined seismic and LOCA loads are substantially higher than those that would result from a small break LOCA. FCF has conservatively complied with the SRP acceptance criteria.

36: The submittal states that the target burnups for SQN are 62,000 MWD/mtU for the peak rod; however, the safety evaluation for the Mark-BW fuel only approves the fuel up to burnups of 60,000 MWD/mtU for the peak rod. Verify that the peak burnups will not exceed approved values and describe how each of the other limitations contained in the Safety Evaluation for BAW-10172 (Section 6.0 Conclusions) are met.

Response

Peak fuel assembly and fuel rod burnups will not exceed the approved values, as defined in BAW-10172 or in later documents (such as BAW-10186, currently under review). Other limitations in the SER for BAW-10172 (Section 6.0 Conclusions) are as follows:

> "Those licensees that use the Mark-BW fuel design for reload applications are required to submit the following plant-specific analyses: rod pressure, cladding collapse, DNB analysis, and fuel melting. In addition, DNB analyses of mixed cores containing Mark-BW and Westinghouse fuel designs must be performed using an approved mixed core methodology."

The requested plant-specific analyses have been performed with approved methods, as discussed in Sections 7.6 (Mixed Core Analysis) and 7.7 (Fuel Thermal Performance Analysis). Detailed results of the plant-specific mixed-core DNB analyses have been provided by letter submitted to the NRC dated 1/10/97.

Fuel thermal performance analyses that predict fuel rod temperature and internal pressure conditions during core operation have been performed for the UO_2 rods with TACO3 (BAW-10162P-A) and the Gadolinia fuel rods with GDTACO (BAW-10184P-A). These analyses are used to determine the centerline melt limit and maximum fuel rod burnup limit based on the fuel rod internal pressure. The TACO3 code and internal gas pressure analysis methodology have been applied to fuel rod operation with pressure greater than reactor coolant system pressure as described in the NRC-approved topical report BAW-10183P-A . The results of the plant specific analyses are presented below for both the uranium dioxide and urania-gadolinia fuel rods.

The cladding collapse analysis that predicts fuel cladding creep collapse has been performed for the UO_2 and gadolinia rods with CROV (BAW-10084P, rev. 3). This analysis is used to show that cladding creep collapse will not occur during the life of the fuel.

Fuel Rod Internal Pin Pressure

The criteria for fuel rod internal pressure, stated in the safety evaluation of BAW-10183P-A, requires "the rod internal pressure for a limited number of rods will be permitted to exceed the RCS pressure, but will not exceed the smaller of the following: the

proprietary limit above RCS pressure [], or that pressure which would cause cladding liftoff to occur at significant LHGRs. The number of rods which will exceed the RCS pressure is limited by a core protection criterion which declares that the number of such rods, which also experience DNB in overpower transients, shall not exceed 0.01% of the rods in the core."

The predicted fuel rod internal pin pressure determined with the approved TACO3 methods shows that both the UO_2 and gadolinia fuel rod designs are acceptable to a burnup of [ ] MWd/mtU.

Mark-BW Fuel Rod Design	Maximum Predicted Internal Pressure (psia)	Burnup (MWd/mtU)
UO_2	[ ]	[ ]
Urania-Gadolinia	[ ]	[ ]

Fuel Melt Limit

The beginning-of-life (BOL) linear heat rate to melt limit bounds the time-in-life predictions. The Melt Limit is calculated at various burnups using the approved TACO3 methodology. The BOL limit, which bounds all of the calculated values, has been applied over the entire burnup range.

Fuel Rod	Melt Limit (kW/ft)	Burnup (MWd/mtU)
UO_2	[ ]	BOL
Urania-Gadolinia	[ ]	BOL

Cladding Creep Collapse

The cladding creep collapse criterion in BAW-10084P-A, rev. 3 requires that "creep collapse will not occur during the life of the fuel." Creep collapse of the UO_2 and gadolinia fuel rods, calculated with the approved CROV methods, will not occur within a burnup of [ ] MWd/mtU which bounds the design fuel rod burnup [ ] in the Sequoyah core.

ENCLOSURE 3

NRC RAI ON TS 96-01

DATED

JANUARY 8, 1997

APPLICABILITY FOR WITHHOLDING

AND

AFFIDAVIT

AFFIDAVIT OF JAMES H. TAYLOR

A. My name is James H. Taylor. I am Manager of Licensing Services for Framatome Technologies, Inc. (FTI). Framatome Cogema Fuels is administratively responsible to Framatome Technologies, Inc. Therefore, I am authorized to execute this Affidavit.

B. I am familiar with the criteria applied by FTI to determine whether certain information of FTI is proprietary and I am familiar with the procedures established within FTI to ensure the proper application of these criteria.

C. In determining whether an FTI document is to be classified as proprietary information, an initial determination is made by the Unit Manager, who is responsible for originating the document, as to whether it falls within the criteria set forth in Paragraph D hereof. If the information falls within any one of these criteria, it is classified as proprietary by the originating Unit Manager. This initial determination is reviewed by the cognizant Section Manager. If the document is designated as proprietary, it is reviewed again by Licensing personnel and other management within FTI as designated by the Manager of Licensing Services to assure that the regulatory requirements of 10 CFR Section 2.790 are met.

D. The following information is provided to demonstrate that the provisions of 10 CFR Section 2.790 of the Commission's regulations have been considered:

(i) The information has been held in confidence by FTI. Copies of the document are clearly identified as proprietary. In addition, whenever FTI transmits the information to a customer, customer's agent, potential customer or regulatory agency, the transmittal requests the recipient to hold the information as proprietary. Also, in order to strictly limit any potential or actual customer's use of proprietary information, the substance of the following provision is included in all agreements entered into by FTI, and an equivalent version of the proprietary provision is included in all of FTI's proposals:

"Any proprietary information concerning Company's or its Supplier's products or manufacturing processes which is so designated by Company or its Suppliers and disclosed to Purchaser incident to the performance of such contract shall remain the property of Company or its Suppliers and is disclosed in confidence, and Purchaser shall not publish or otherwise disclose it to others without the written approval of Company, and no rights, implied or otherwise, are granted to produce or have produced any products or to practice or cause to be practiced any manufacturing processes covered thereby.

Notwithstanding the above, Purchaser may provide the NRC or any other regulatory agency with any such proprietary information as the NRC or such other agency may require; provided, however, that Purchaser shall first give Company written notice of such proposed disclosure and Company shall have the right to amend such proprietary information so as to make it non-proprietary. In the event that Company cannot amend such proprietary information, Purchaser shall, prior to disclosing such information, use its best efforts to obtain a commitment from NRC or such other agency to have such information withheld from public inspection.

Company shall be given the right to participate in pursuit of such confidential treatment."

(ii) The following criteria are customarily applied by FTI in a rational decision process to determine whether the information should be classified as proprietary. Information may be classified as proprietary if one or more of the following criteria are met:

a. Information reveals cost or price information, commercial strategies, production capabilities, or budget levels of FTI, its customers or suppliers.

b. The information reveals data or material concerning FTI research or development plans or programs of present or potential competitive advantage to FTI.

c. The use of the information by a competitor would decrease his expenditures, in time or resources, in designing, producing or marketing a similar product.

d. The information consists of test data or other similar data concerning a process, method or component, the application of which results in a competitive advantage to FTI.

e. The information reveals special aspects of a process, method, component or the like, the exclusive use of which results in a competitive advantage to FTI.

f. The information contains ideas for which patent protection may be sought.

The document(s) listed on Exhibit "A", which is attached hereto and made a part hereof, has been evaluated in accordance with normal FTI procedures with respect to classification and has been found to contain information which falls within one or

more of the criteria enumerated above. Exhibit "B", which is attached hereto and made a part hereof, specifically identifies the criteria applicable to the document(s) listed in Exhibit "A".

(iii) The document(s) listed in Exhibit "A", which has been made available to the United States Nuclear Regulatory Commission was made available in confidence with a request that the document(s) and the information contained therein be withheld from public disclosure.

(iv) The information is not available in the open literature and to the best of our knowledge is not known by Combustion Engineering, EXXON, General Electric, Westinghouse or other current or potential domestic or foreign competitors of Framatome Technologies, Inc.

(v) Specific information with regard to whether public disclosure of the information is likely to cause harm to the competitive position of FTI, taking into account the value of the information to FTI; the amount of effort or money expended by FTI developing the information; and the ease or difficulty with which the information could be properly duplicated by others is given in Exhibit "B".

E. I have personally reviewed the document(s) listed on Exhibit "A" and have found that it is considered proprietary by FTI because it contains information which falls within one or more of thecriteria enumerated in Paragraph D, and it is information which is customarily held in confidence and protected as proprietary information by FTI. This report comprises information

AFFIDAVIT OF JAMES H. TAYLOR (Cont'd.)

utilized by FTI in its business which afford FTI an opportunity to obtain a competitive advantage over those who may wish to know or use the information contained in the document(s).

James H Taylor

JAMES H. TAYLOR

State of Virginia)

) SS. Lynchburg

City of Lynchburg)

James H. Taylor, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

James H Taylor

JAMES H. TAYLOR

Subscribed and sworn before me
this 31st day of January 1997.

Danita R. Kidd

Notary Public in and for the City
of Lynchburg, State of Virginia.

My Commission Expires 12/31/2000

EXHIBITS A & B

EXHIBIT A

Responses to Request for Additional Information - Technical Specification Change Request 96-01 on Conversion to Cogema Fuel - Sequoyah Nuclear Plants Units 1 and 2 (TAC Nos. M95144 and 95145)

EXHIBIT B

The above listed document contains information which is considered Proprietary in accordance with Criteria b, c and d of the attached affidavit.