Florida Power & Light, 9760 S.W. 344 St. Homestead, FL 33035

FPL

December 31, 2011

L-2011-561
10 CFR 50.90
10 CFR 2.390

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Thermal Conductivity Degradation

Reference:

(1) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request for Extended Power Uprate (LAR 205)," (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010.

By letter L-2010-113 dated October 21, 2010 (Reference 1), Florida Power and Light Company (FPL) requested to amend Renewed Facility Operating Licenses DPR-31 and DPR-41 and revise the Turkey Point Units 3 and 4 Technical Specifications (TS). The proposed amendment will increase each unit's licensed core power level from 2300 megawatts thermal (MWt) to 2644 MWt and revise the Renewed Facility Operating Licenses and TS to support operation at this increased core thermal power level. This represents an approximate increase of 15% and is therefore considered an extended power uprate (EPU).

As a result of recent information presented to the NRC on December 6, 2011, FPL was asked to address the impact of Thermal Conductivity Degradation (TCD) on the PTN EPU safety analyses. FPL's response to the NRC's request is presented in Attachments 1 and 2 to this letter.

Attachment 3 contains the application for withholding the proprietary information contained in Attachment 2 from public disclosure. As Attachment 2 contains information proprietary to Westinghouse Electric Company, LLC (Westinghouse), it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of §2.390 of the Commission's regulations. Accordingly, it is respectfully requested that information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of items in the response to the RAI questions in Attachment 2 of this letter or the supporting Westinghouse affidavit should reference CAW-11-3340 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, PA 16066.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-113 (Reference 1) or PTN Technical Specifications.

A001
NRR

This submittal contains no new commitments and no revisions to existing commitments.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the State Designee of Florida.

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 31, 2011.

Very truly yours,

E. W. McCaffrey for M. Kiley

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachments (3)

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Resident Inspector, Turkey Point Nuclear Plant
Mr. W. A. Passetti, Florida Department of Health (without Attachment 2)

Turkey Point Units 3 and 4

RESPONSE TO NRC SRXB RAI REGARDING EPU LAR NO. 205
AND THERMAL CONDUCTIVITY DEGRADATION

ATTACHMENT 1

Response to Request for Additional Information

1.0 RAI Introduction

The following information is provided by Florida Power and Light Company (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support License Amendment Request (LAR) No. 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL letter L-2010-113 on October 21, 2010 (Reference 1).

On October 8, 2009, the NRC issued NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," which noted that irradiation damage and the progressive buildup of fission products in the fuel pellets result in reduced thermal conductivity of the pellets. Data was collected from an instrumented assembly at the Halden ultra-high-burnup experiment during the 1990s which indicated steady degradation in the thermal conductivity of uranium fuel pellets with increasing exposure. This data indicated a degradation of approximately 5 to 7 percent for every 10 gigawatt-days per metric tonne of exposure. The NRC expressed concern that some vendors might still be using codes for safety analyses that do not account for this phenomenon and therefore may produce non-conservative results. As a result of recent information presented to the NRC on December 6, 2011, FPL was asked to address the impact of Thermal Conductivity Degradation (TCD) on the PTN EPU safety analyses. FPL's response to the NRC's request for additional information is presented in this non-proprietary attachment (Attachment 1) and in the following proprietary attachment (Attachment 2).

The affidavit that sets forth the basis for which the information may be withheld from public disclosure by the NRC in accordance with 10 CFR 2.390 is contained in Attachment 3. Proprietary information is contained within brackets and the basis for claiming the information as proprietary is indicated by means of lower case letters (a) - (f) located as a superscript immediately following the brackets enclosing each item of information identified as proprietary. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) - (4)(ii)(f) of the affidavit accompanying this submittal pursuant to 10 CFR 2.390(b)(1). In this attachment, the proprietary information has been deleted and only the brackets remain.

2.0 Summary of TCD Issue and Areas of Review

The currently approved Westinghouse PAD 4.0 fuel performance code does not account for any thermal conductivity degradation (TCD) as burnup increases in the model. The TCD impact on fuel performance predictions is a generic issue which is being addressed through a formal revision to the PAD code, but this will not be completed until a later date. However, in order to demonstrate the acceptability of the Turkey Point Units 3 and 4 EPU accounting for the effects of TCD on fuel performance predictions, additional analyses were performed with an updated Turkey Point PAD model with TCD effects. The PAD model used in these assessments was changed to reflect available data from the Halden test reactor data (See Section 3 for more information).

With the updated Turkey Point PAD model with TCD effects new fuel temperatures were calculated as a function of burnup for the EPU. These temperatures were provided as input to all fuel and safety analyses which utilize them.

In addition to fuel temperatures, some safety analyses use the core stored energy (CSE) as an input. CSE was re-calculated with the revised fuel temperatures accounting for TCD. CSE, which is a core average value, applicable to all burnup and power ranges, was determined to be slightly lower than the non-TCD CSE. The reduction occurred because more accurate burnup and power weighting was accounted for in order to offset the effects of TCD. Refer to Section 5 for additional discussion related to the calculated CSE value.

A summary of the modeling changes made to the PAD 4.0 code to address TCD and the benchmarking performed to the recent Halden test results are provided in Section 3 of the attached RAI responses. The impacts on the potentially affected analyses relative to the information presented in the Turkey Point EPU License Amendment Request (LAR) are summarized in Sections 4 and 5 below. Section 4 covers those areas where the change in fuel temperatures and CSE were determined to have minimal or no impact on previously calculated EPU results. A discussion of why there is minimal or no impact is given for each analysis. Section 5 covers those areas where the changes were found to have some impact on the EPU results presented in the LAR. For these areas, detailed discussions of the changes to the inputs, methodology, and results presented in the LAR are provided.

In order to gain the necessary margin for the best-estimate large break loss of coolant (BELOCA) analysis to offset the TCD impact, it was necessary to make some adjustments to other input assumptions. The key change made was a reduction in the core peaking factors. The other input changes made to plant operating ranges made for the purpose of the BELOCA analysis are described in the BELOCA section discussion.

3.0 PAD 4.0 Changes Made to Address TCD

The Nuclear Regulatory Commission (NRC) approved Performance and Design (PAD) 4.0 code, with NRC-approved models (Reference 4) for in-reactor behavior is used in this assessment. PAD 4.0 is a best-estimate fuel rod performance model, and in most cases, the design criterion evaluations are based on a best-estimate plus uncertainties approach. A statistical convolution of individual uncertainties due to design model uncertainties and fabrication dimensional tolerances is used. As-built dimensional uncertainties for some inputs, such as fuel pellet diameter, can be used in lieu of the fabrication uncertainties.

The licensed PAD 4.0 fuel performance models do not address the impact of TCD. However, TCD can be incorporated into the code using available input options. Fuel thermal conductivity is modeled in PAD using the following model form:

a.c

a,c

Figure 3-1: Comparison of Measured Minus Predicted Fuel Temperature as a Function of Burnup for PAD 4.0 and PAD 4.0 TCD

[

]a,c

4.0 EPU Analyses and Evaluations where TCD has Minimal or No Impact

The following licensing report (LR) sections (Reference 1 Attachment 4) were reviewed for potential impact from TCD and it was determined that the inclusion of TCD has minimal or no impact on the analysis results. The assessments for each area are summarized below.

a. Containment Pressure for LBLOCA (LR Section 2.6.3.1)

Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

There is no adverse impact on the long-term containment response to the design basis large break loss-of-coolant accident (LOCA) when the effects of TCD are considered for the fuel for Turkey Point Units 3 and 4 for the EPU. There is no adverse impact because the long-term LOCA mass and energy releases incorporate a core stored energy that is conservatively high such that it bounds the calculated stored energy when accounting for the effects of TCD.

b. Containment Pressure for MSLB (LR Section 2.6.3.2)

Mass and Energy Release Analysis for Secondary System Pipe Ruptures

Explicitly modeling the effects of TCD would not change the conservatism of the steamline break (SLB) mass and energy release inside containment analysis, the steamline break containment integrity analysis or the steam release for dose analysis. The SLB mass and energy release analysis is not sensitive to the fuel-to-coolant heat transfer coefficient input that could change due to higher fuel temperatures. The fuel temperatures do not have a first order impact on the SLB mass and energy release analysis. The dominant fuel-related input in the SLB mass and energy releases is the moderator density reactivity feedback that causes a post-trip return to power. Therefore, the steamline break containment integrity analysis would not be affected. The steam release for dose analysis used a generic core stored energy value that is more conservative than the Turkey Point specific core stored energy when calculated accounting for the effects of TCD. Therefore, the EPU steam release for dose analysis remains bounding for use as input to the subsequent dose consequence analyses. The results of the SLB mass and energy release analysis are documented in LR Section 2.6.3.2 (Reference 1) and the results of the associated containment response can be found in LR Section 2.6.1 (Reference 1).

c. Core Physics/Design (LR Section 2.8.2)

Nuclear Design

The Westinghouse PHOENIX-P/ANC nuclear design system is the primary nuclear code system for nuclear design and safety analyses across the Westinghouse reactor fleet, both domestically and abroad. As discussed in References 2 and 3, the system continues to show good performance in currently operating reactors with a broad range of design and operating characteristics. The benchmark data for the PHOENIX-P/ANC nuclear design system discussed in References 2 and 3 inherently included any affects of TCD. Therefore, the nuclear design PHOENIX-ANC nuclear design system was determined to remain valid and appropriate for nuclear design calculations. Accordingly, all nuclear design calculations previously performed to support the EPU design and safety analyses remain valid.

However, some limits have been revised as a result of TCD. Specifically, the power-to-melt kW/ft limits has been revised to be burnup dependent, the differential rod worth (reactivity insertion rate) limit has been reduced and the nuclear enthalpy rise hot channel factor (FΔH) and the full-power heat flux hot channel factor (F_Q) were reduced. As a result, these revised limits were confirmed to be satisfied for EPU cores for the affected non-LOCA accident analyses and Large Break LOCA.

HFP and HZP Steamline Break

The HZP and HFP Steam-line Break kW/ft are re-assessed. The analyses show that some EPU core designs met the revised limits and other EPU core designs did not meet the limits. This demonstrates that the core can be designed to meet the more restrictive limits. However, there are only limited designs that can accommodate the revised limits. In conclusion, the revised burnup specific kW/ft limits, which included the effect of TCD, can be met for EPU cores. This limit is confirmed on a reload specific basis as part of the Reload Safety Analyses Checklist (RSAC) process.

RCCA Drop

Because the design feature of automatic RCCA withdrawal has been disabled at Turkey Point, there is no possibility of a power overshoot during the RCCA Drop event. Therefore there is significant margin to the revised kW/ft limit for the RCCA Drop accident. In conclusion, the revised burnup specific kW/ft limits, which included the effect of TCD, can be met for EPU cores. This limit is confirmed on a reload specific basis as part of the RSAC process.

Rod Withdrawal at Power

In a review of the EPU analyses for the RWAP event, the maximum calculated differential rod worth (reactivity insertion rate) has significant margin to the revised minimum limit. In conclusion, the revised differential rod worth limit, which included the effect of TCD, can be met the EPU cores. This limit is confirmed on a reload specific basis as part of the RSAC process.

Peaking Factors (FΔH and F_Q) Reduction for LBLOCA

In a review of all EPU design and analyses, it shows that some EPU core designs with higher number of feed assembly met the reduced FΔH and F_Q limits and other EPU core designs with lower number feed assembly did not meet the limits. This demonstrates that the core can be designed to meet the more restrictive limits. However, there are only limited designs that can accommodate the revised limits. In conclusion, the reduced peaking factor limits, which included the effect of TCD, can be met for EPU cores. These limits are confirmed on a reload specific basis as part of the RSAC process.

Conclusion

The above TCD impact assessment demonstrates that the limiting burnup dependent fuel melt kW/ft and reduced peaking factors criterion could reduce the flexibility of core design and plant operation. However, it was also concluded that Turkey Point cores can be designed and safety limits remain valid for EPU conditions when accounting for the effects of TCD.

d. Fuel Thermal Hydraulic Design (LR Section 2.8.3)

Thermal and Hydraulic Design

A review has been conducted of the licensing amendment request for Turkey Point Units 3 and 4 Section 2.8.3 (Reference 1) to evaluate the impact of TCD on the conclusions contained therein. The following addresses all items contained within Fuel Thermal-Hydraulic Design Scope.

In a steady state VIPRE model for departure from nucleate boiling (DNB) safety analyses, the fuel rods are modeled as "dummy" rods to define the heat flux boundary. A "dummy" rod does not model the conduction heat transfer within the fuel rods, but instead, specifies heat input directly into the coolant. Thus, TCD has no direct impact on these models. This steady state type of model is used in most DNB analyses. However, effects of fuel stored energy and rod conduction heat transfer are significant in fast transients such as complete loss of flow and locked rotor (LR) events; therefore, a transient VIPRE model is used for those two analyses. In a transient VIPRE model, fuel rods can be modeled using the option of conducting rod. The conducting rod option allows the user to specify a number of radial nodes in the pellet, and a gap conductance model for the heat transfer between the pellet and the inner surface of the cladding. The initial gap conductance values in the transient VIPRE model are based on conservative maximum fuel temperatures (including uncertainties) from the PAD 4.0 code. The PAD 4.0 fuel pellet surface temperature is used to determine the VIPRE gap conductance at each axial node. Only the transient VIPRE models are affected by the change in fuel temperatures due to TCD given that there is no change in state points for both the steady state VIPRE models and transient VIPRE models as documented by the Non-LOCA analysis.

An assessment of the impact of TCD on the minimum calculated DNBR values have shown that the results reported in the licensing report remain conservative with no impact. Peak Clad Temperature (PCT) calculations for the LR event show an increase in PCT of 66.1°F from the value listed in the licensing report (Reference 1) to 1890.1°F and there remains significant margin to the PCT limit of 2375°F. The maximum zirconium-water reaction at the core hot spot increased by 0.06% from the value listed in Reference 1 to 0.46% and there remains significant margin to the limit of 16%. The LR PCT results above are obtained by applying the maximum PAD 4.0 fuel temperatures over all burnups and the maximum peaking factors ($F_{\Delta H}$ = 1.65 and F_q=2.4) to the analysis. It has been determined that there is no impact to the conclusions contained in Section 2.8.3 "Thermal and Hydraulic Design" due to TCD effects.

e. Non-LOCA (LR Section 2.8.5)

Non-LOCA Analyses (with Minimal or no Impact)

The following dispositions the impact of TCD on the Turkey Point non-LOCA EPU Analyses (LR section numbers are indicated):

Events Modeling Minimum Fuel Temperatures

The following EPU analyses model minimum fuel temperatures as they have shown to be more conservative when modeling maximum fuel to reactor coolant heat transfer. Therefore, the following EPU analyses are not impacted since they model minimum fuel temperatures for fuel to coolant heat transfer inputs, which are not negatively impacted by TCD, and do not require detailed state point evaluations to demonstrate that limits are met:

2.8.5.1.1.2.3	Increase in Steam Flow / Excessive Load Increase (ELI)
2.8.5.4.2	Uncontrolled RCCA Bank Withdrawal at Power (RWAP) Minimum DNB Case
2.8.5.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve (Reactor Coolant System (RCS) Depressurization)
2.8.5.7	Anticipated Transients Without Scram (ATWS)

Departure from Nucleate Boiling Ratio (DNBR) Events Modeling Minimum Fuel Temperatures

In addition to modeling minimum fuel temperatures, the following events are analyzed for minimum DNBR concerns via transient state point analysis, which means more detailed Thermal-Hydraulic calculations are performed using results from the LOFTRAN or RETRAN computer codes as input. Since minimum fuel temperatures, which are not affected by TCD, are used in the associated LOFTRAN and RETRAN runs, the predicted transient state point results are not affected by TCD. Using those unchanged state points as the basis, a subsequent Thermal-Hydraulic evaluation confirmed that the results currently reported in the subject LR sections remain valid:

2.8.5.1.1.2.2	Increase in Feedwater Flow/Feedwater Malfunction (FWM) at HZP
2.8.5.4.3	RCCA Misalignment (Dropped Rod)

Events Modeling Maximum Fuel Temperatures

The following EPU analyses model maximum fuel temperatures as they have shown to be more conservative when modeling the associated minimum fuel to reactor coolant heat transfer. In addition to modeling maximum fuel temperatures for fuel to coolant heat transfer inputs, which are impacted by TCD, minimum reactivity feedback (beginning of cycle (BOC) conditions are also modeled. The RETRAN fuel temperature calibration for maximum fuel temperatures performed for the Turkey Point EPU was conservative and resulted in initial fuel temperatures that are higher than those predicted by PAD when not taking into account the effects of TCD. This offsets the majority of the temperature increase resulting from TCD. The remaining effect is offset by another conservatism in the method, which is that the entire core (all fuel rods regardless of their burnup) is initialized at the maximum fuel temperatures at the beginning of the transient. In reality, the core is comprised of both fresh and burnt assemblies, which are at various burnup and power levels. TCD will affect the assemblies with significant burnup, but those assemblies are typically at lower power levels since the overall core power is dominated by the fresh fuel. The initial RETRAN average fuel temperature, which is based on the assumption that the entire core is at the maximum fuel temperatures, will bound the actual core average temperatures when the individual assembly powers and burnups are considered.

Therefore, the effect of TCD on the initial fuel temperatures and fuel thermal conductivity can be accommodated within existing conservatisms in the analysis method. Additionally, for those listed events that model decay, it is noted that TCD does not affect the predictions of the decay heat model.

- 2.8.5.2.1 Loss of External Electrical Load / Turbine Trip (LOL/TT) and Loss of Condenser Vacuum
- 2.8.5.2.2 Loss of Non-Emergency AC Power to Station Auxiliaries (LOAC)
- 2.8.5.2.3 Loss of Normal Feedwater Flow (LONF)
- 2.8.5.2.4 Feedwater System Pipe Break (FLB)

Specific Event Considerations

Decrease in Feedwater Temperature/Enthalpy at Hot Full Power (LR 2.8.5.1.1.2.1)

The Decrease in Feedwater Temperature/Enthalpy analysis for the EPU is not impacted by TCD since reactor trip is not required for event mitigation and is bounded by the ELI event. Additionally, this event does not explicitly model fuel temperatures as it is not analyzed with the RETRAN computer code.

Inadvertent Opening of a Steam Generator Relief or Safety Valve (LR 2.8.5.1.1.2.4)

This event models minimum fuel temperatures. However, this event is bounded by the Steam System Piping Failure accident, as demonstrated by an explicit analysis Westinghouse performed in response to an NRC Request for Additional Information. Consideration of TCD does not invalidate the results and conclusions of the credible break analysis; therefore this event continues to be bounded by the Steam System Piping Failure event.

Steam System Piping Failure / Hot Zero Power Steamline Break (LR 2.8.5.1.2.2.1)

The Hot Zero Power Steamline Break (HZP SLB) analysis models minimum fuel temperatures (not impacted by TCD) to define the fuel to coolant heat transfer inputs. Therefore, the HZP SLB analysis with respect to transient core response and resulting state point conditions does not change when considering the effects of TCD. A subsequent departure from nucleate boiling ratio (DNBR) evaluation confirmed that the calculated value and safety analysis limit (SAL) are not impacted when considering the effects of TCD since the transient core response state point conditions do not change.

However, as a result of TCD, the peak kW/ft SAL has been revised to be a function of burnup. A peak kW/ft limit evaluation has confirmed that the burnup dependent peak kW/ft limits have been met for bounding EPU core designs when considering the effects of TCD. This limit (see Figure 4-1) is confirmed on a reload specific basis as part of the RSAC process.

Steam System Piping Failure / Hot Full Power Steamline Break (LR 2.8.5.1.2.2.2)

The Hot Full Power Steamline Break (HFP SLB) analysis models minimum fuel temperatures (not impacted by TCD) to define the fuel to coolant heat transfer inputs. Therefore, the HFP SLB analysis with respect to transient core response and resulting state point conditions does not change when considering the effects of TCD. A subsequent DNBR evaluation confirmed that the calculated value and SAL are not impacted when considering the effects of TCD since the transient core response state point conditions do not change.

However, as a result of TCD, the peak kW/ft limit has been revised to be a function of burnup. A peak kW/ft limit evaluation has confirmed that the burnup dependent peak kW/ft limits have been met for bounding EPU core designs when considering the effects of TCD. This limit (see Figure 4-1) is confirmed on a reload specific basis as part of the RSAC process.

Loss of Forced Reactor Coolant Flow (LOF) (LR 2.8.5.3.1)

See LR Section 2.8.5.3.2 discussion below.

Reactor Coolant Pump Shaft Seizure (Locked Rotor) / Reactor Coolant Pump Shaft Break (LR/SB) Rods-in-DNB Case (LR 2.8.5.3.2)

The LOF and LR/SB rods-in-DNB analyses model minimum fuel temperatures and are not impacted by the effects of TCD as stated above. Therefore, the transient core response and resulting state point conditions do not change when considering the effects of TCD. A subsequent state point evaluation of the LOF and LR/SB events concluded that the minimum DNBR results reported for the EPU remain bounding.

Reactor Coolant Pump Shaft Seizure (Locked Rotor) / Reactor Coolant Pump Shaft Break (LR/SB) Peak Clad Temperature (PCT) (LR 2.8.5.3.2)

The LR/SB PCT analysis in RETRAN models maximum fuel temperatures and BOC (minimum) reactivity feedback conditions. As discussed above with respect to maximum fuel temperatures, the RETRAN fuel temperature model remains applicable when accounting for the effects for TCD. Therefore, the transient response and resulting state point conditions do not change when considering the effects of TCD. A subsequent state point analysis of PCT calculations for the locked rotor event shows only a small increase in the maximum cladding temperature from 1824°F to 1891°F, and there remains significant margin to the PCT limit of 2735°F.

Uncontrolled Rod Control Cluster Assembly (RCCA) Bank Withdrawal from a Subcritical or Low Power Startup Condition (RWFS) (LR 2.8.5.4.1)

This event is analyzed with maximum fuel temperatures at BOC conditions using the TWINKLE spatial kinetics computer code to predict the core average power transient and the FACTRAN computer code to predict the response of the limiting fuel rod. To assess the potential impact of TCD on the predicted results, analysis has been performed in which the thermal conductivity in the FACTRAN code is conservatively modeled to reflect the impact of TCD at the maximum design burnup of 62,000 MWD/MTU. The TWINKLE input for this analysis is not changed from that previously used to analyze RWFS for the EPU.

Results for this RWFS analysis show that the inclusion of reduced fuel conductivity from TCD does not adversely impact the results for the EPU. Therefore, the limiting RWFS state point conditions predicted for the EPU remain bounding. These state point conditions are used in the reload process to confirm on a cycle specific basis that the DNB design basis is met. The reduced fuel conductivity caused by TCD does affect the heat transfer from the fuel; however, the maximum fuel temperatures calculated during the transient were negligibly impacted and very substantial margins remain to the onset of fuel melting (margin of >2000°F). It is also noted that a conservatively low melting temperature (corresponding to high burnup) is used for UO_2.

Based on the results reported above, RWFS can be considered as an event that will not be significantly impacted by TCD. As stated above, the predicted maximum fuel temperature for the RWFS event has negligibly changed; therefore, there are no changes to report in Table 4-1.

Uncontrolled RCCA Bank Withdrawal at Power (RWAP) RCS Overpressure Cases (LR 2.8.5.4.2)

The only LOFTRAN analysis that is impacted by the increases in the maximum fuel temperatures due to TCD is the RCS overpressure analysis for the RWAP accident. Calculations incorporating the impact of TCD have confirmed that the maximum RCS pressure for the EPU RWAP analysis is impacted by the increases in the maximum fuel rod temperatures, with an increase in the predicted peak RCS pressure of ~3 psi. Even with this increase, the peak pressure remains below the 2748.5 psia limit. However, it has also been determined that the existing analysis contains enough conservatism to offset the impact of TCD on the maximum fuel rod temperatures. Specifically, a small change in the maximum analyzed reactivity insertion rate from 29 pcm/sec to 28 pcm/sec produces results that continue to be bounded by the reported peak pressure value for the EPU RWAP analysis of 2741 psia. It has been determined that existing margins in the associated core design calculations can accommodate such a reduction in the maximum reactivity insertion rate. This maximum reactivity insertion rate is confirmed on a reload specific basis as part of the RSAC process.

Startup of an Inactive Loop at an Incorrect Temperature (SUIL) (LR 2.8.5.4.4)

The SUIL event is not impacted by TCD since Turkey Point is not licensed for N-1 loop operation.

Chemical and Volume Control System Malfunction (LR 2.8.5.4.5)

The Boron Dilution analysis for the EPU is not impacted since the RSAC critical boron concentrations modeled in the analysis for any operating conditions are not changed because of TCD.

Revised Non-LOCA Analysis Limits and Results

Table 4-1 below corresponds to changes applicable to the non-LOCA analysis limits table (Table 2.8.5.0-1 of the Licensing Report) as a result of the assessing the impact of TCD with respect to the non-LOCA safety analyses. Note that only items that have changed in Table 2.8.5.0-1 are identified.

Table 4-1
Non-LOCA Analysis Limits and Analysis Results

UFSAR Section	Event Description	Affected Parameter	Analysis Result	
			Analysis Limit	Limiting Case
14.1.4	RCCA Drop	Peak Linear Heat Generation (kW/ft)	See Figure 4-1[1]	Less than the limits defined in Figure 4-1
14.1.9	Locked Rotor	Peak Clad Temperature, °F	2375	1891
14.2.5	Rupture of a Steam Pipe – Zero Power (Core response only)	Peak Linear Heat Generation (kW/ft)	See Figure 4-1[1]	Less than the limits defined in Figure 4-1
14.2.5	Rupture of a Steam Pipe – Full Power (Core response only)	Peak Linear Heat Generation (kW/ft)	See Figure 4-1[1]	Less than the limits defined in Figure 4-1

1. The peak linear heat rate limit is determined as a function of fuel burnup and future core designs will ensure these limits are met on a reload basis.

Figure 4-1

Peak kW/ft Limits with Respect to Burnup
(Debris Resistant Fuel Assemblies (DRFA) and Upgrade Fuel)

a,c

f. Steam Generator Tube Rupture (SGTR) (LR Section 2.8.5.6.2)

Steam Generator Tube Rupture

The effects of TCD were assessed for the Turkey Point EPU Steam Generator Tube Rupture (SGTR) analyses. There are three SGTR analyses that support the Turkey Point EPU. The licensing basis analysis to determine the input to the doses is a conservative mass and energy balance to determine the break flow and steam releases following an SGTR and assumes break flow termination at 30 minutes. The licensing basis analysis does not model a plant-specific or fuel-specific maximum fuel temperature, but assumes a conservatively high fuel temperature, which bounds the effects of TCD. Thus, the licensing basis analysis remains valid.

To demonstrate that the licensing basis analysis provides conservative releases for input to the dose analyses, a more realistic "confirmatory" analysis is performed modeling operator actions and considering that break flow may extend beyond 30 minutes. For the confirmatory analysis, a high fuel temperature is conservative as it increases steam releases. The effects of TCD were evaluated and it was determined that although the calculated releases would increase, the results reported in Table 2.8.5.6.2-5 would not change since sufficient conservatism was added to the originally calculated values. Thus, the reported results of the confirmatory calculation and the conclusion that the licensing basis analysis is conservative remain valid.

Finally, to demonstrate that overfill of the ruptured steam generator does not occur, a more realistic analysis is performed modeling operator actions and considering that break flow may extend beyond 30 minutes. For the margin to overfill analysis, a lower fuel temperature is conservative as it decreases steam releases, and so is not impacted by the effects of TCD.

The Turkey Point EPU SGTR analyses remain valid when considering the effects of TCD. Thus, the SGTR inputs to the dose analyses are not impacted by the effects of TCD.

g. Small Break Loss-of-Coolant Accident (SBLOCA) (LR Section 2.8.5.6.3.3)

Small Break SBLOCA

The input changes made as a result of the fuel TCD impact on the large-break LOCA (LBLOCA) EPU analysis have a negligible or beneficial impact on the small break LOCA (SBLOCA) analysis; as such, the results of the SBLOCA EPU analysis remain bounding.

The affects of fuel TCD were assessed for the SBLOCA EPU analysis. TCD has the effect of increasing fuel rod temperatures and rod internal pressures later in burnup. A small break LOCA transient is a relatively slow progressing transient and is characterized by a top-down draining of the reactor coolant system; therefore, the transient results are negligibly affected by increased fuel rod temperatures as the stored energy associated with the higher temperatures is removed prior to core uncovery. Due to the low BOL PCT of 1231°F, burnup calculations to capture variations in rod internal pressure are unnecessary as any associated clad burst temperature spike calculated at an increased burnup is anticipated to be less than that calculated at BOL. Therefore, the Turkey Point EPU SBLOCA analysis is deemed to be negligibly impacted by TCD.

h. Long Term Cooling (LR Section 2.8.5.6.3.4)

Post-LOCA Subcriticality and Long Term Cooling

The primary impact of TCD in the fuel potentially important to Long Term Cooling (LTC) is an increase in initial fuel pellet temperature. This in turn leads to a higher amount of stored energy at the initiation of the LOCA event. Initial stored energy is not important to LTC evaluations as these evaluations only consider decay heat removal during the sump recirculation phase of emergency core cooling system (ECCS) operation. The increased stored energy in the fuel due to higher fuel pellet temperature is a short term effect that does not persist into the LTC phase of ECCS performance evaluations; therefore, the heat source remains limited to decay heat for LTC evaluations. Consequential impacts of higher fuel pellet temperature such as higher fuel rod internal pressure also have no impact on LTC evaluations as fuel cladding temperatures are maintained well below the threshold for cladding rupture such that cladding burst and blockage does not occur during LTC. Based on the above, it is shown that no additional LTC analysis is required to assess TCD for the Turkey Point Units 3 and 4 EPU.

The new BELOCA input values that are also used in the Subcriticality and LTC analysis have been verified by FPL and Westinghouse to remain within the range assumed in the originally submitted analyses (Reference 1), thus ensuring the integrity of the NRC Staff's review of this analysis.

i. Radiological Consequences (LR Section 2.9.2)

The radiological dose consequences of the design basis accidents presented in Chapter 14 of the PTN UFSAR were evaluated for EPU conditions as part of the Alternative Source Term License Amendment Request No. 196 which was approved with the issuance of Amendments 244 and 240 for Units 3 and 4 on June 23, 2011. TCD does not impact the inputs used in the radiological dose consequence analyses or impact the methodology or assumptions used in the analyses. Therefore, TCD does not affect the results of the radiological dose consequence analyses.

5.0 EPU Analyses and Evaluations Impacted by TCD

a. Fuel Rod Mechanical Design (LR Section 2.8.1)

Fuel System Design

Input Parameters, Assumptions and Acceptance Criteria

The assessment presented herein is based on the Turkey Point (PTN) analyses used to confirm the transition and equilibrium EPU cycles that were previously presented as part of the licensing report. Included in these inputs are appropriate design features for both 15x15 debris resistant fuel assembly (DRFA) and Upgrade fuel.

With the exception of the clad stress criterion, the NRC-approved design criteria methods (References 5 and 6) remain the same as the original licensing report. The clad stress criterion was altered to the NRC approved American Society of Mechanical Engineers (ASME) stress criterion (Reference 7).

The criteria pertinent to the fuel rod design were:

Rod Internal Pressure

The internal pressure of the lead fuel rod in the reactor will be limited to a value lower than one that could cause the diametral gap to increase due to outward clad creep during steady-state operation and extensive departure from nucleate boiling (DNB) propagation to occur.

Clad Stress

Maximum cladding stress intensities excluding pellet-cladding interaction (PCI) induced stress will be evaluated using ASME pressure vessel guidelines. Cladding corrosion is accounted for as a loss of load carrying material. Stresses are combined to calculate a maximum stress intensity, which is then compared to criteria based on the ASME code.

Clad Strain

The design limit for clad strain during steady-state operation ensures that the total plastic tensile creep strain due to uniform clad creep and uniform cylindrical fuel pellet expansion associated with fuel swelling and thermal expansion is less than 1% from the unirradiated condition. The design limit for fuel rod clad strain during Condition II events ensures that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than 1% from the pre-transient value.

Clad Oxidation and Hydriding

The design criteria related to clad corrosion require that the Zircaloy-4/ZIRLO®[1] clad metal-oxide interface temperature is maintained below specified limits to prevent a condition of accelerated oxidation, which would lead to clad failure. The calculated clad temperature (metal-oxide interface temperature) will be less than [ ]^{a,c}°F for ZIRLO clad during steady-state operation. For Condition II transients, the calculated clad temperature will not exceed [ ]^{a,c}°F for ZIRLO clad.

The best-estimate hydrogen pickup level in ZIRLO fuel rod cladding and structural components is less than or equal to the [ ]^{a,c}ppm limit on a volume-averaged basis at end of life (EOL).

Fuel Temperature

For Condition I and II events, the reactor protection system is designed to ensure that the fuel centerline temperature does not exceed the fuel melt temperature criterion. The intent of this criterion is to avoid a condition of gross fuel melting that can result in severe duty on the clad. The concern here is based on the large volume increase associated with the phase change in the fuel and the potential for loss of clad integrity as a result of molten fuel/clad interaction.

Clad Fatigue

The fuel rod design criterion for clad fatigue requires that, for a given strain range, the number of strain fatigue cycles is less than that required for failure, with factors of safety

[1] ZIRLO® High Performance Fuel Cladding Material is a registered trademark of Westinghouse Electric Company LLC in the United States and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited.

of 2.0 on the stress amplitude and 20.0 on the number of cycles. This criterion addresses concerns about the cumulative effect of short-term cyclic clad stress and strain resulting from daily load follow operation.

Clad Flattening

The clad flattening criterion prevents fuel rod failures due to long-term creep collapse of the fuel rod clad into axial gaps formed within the fuel stack. Current fuel rod designs employing fuel with improved in-pile stability provides adequate assurance that axial gaps large enough to allow clad flattening will not form within the fuel stack.

Fuel Rod Axial Growth

This criterion ensures that there is sufficient axial space to accommodate the maximum expected fuel rod growth. Fuel rods are designed with adequate clearance between the fuel rod and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly skeleton to preclude interference of these members.

Plenum Clad Support

This criterion ensures that the fuel clad in the plenum region of the fuel rod will not collapse during normal operating conditions, nor distort so as to degrade fuel rod performance.

Clad Free-Standing

The clad free-standing criterion requires that the clad is short-term, free-standing at beginning of life (BOL), at power, and during hot hydrostatic testing. This criterion precludes the instantaneous collapse of the clad onto the fuel pellet caused by the pressure differential that exists across the clad wall.

Fuel Rod End Plug Weld Integrity

The fuel rod end plug weld shall maintain its integrity during Condition I and II events and shall not contribute to any additional fuel failures above those already considered for Condition III and IV events. The intent of this criterion is to assure that fuel rod failures will not occur due to the tensile pressure differential loads which can exist across the weld.

Description of Analyses and Evaluations

Rod Internal Pressure

The rod internal pressure (RIP) no gap reopening criterion for the PTN fuel rods has been evaluated at EPU conditions by modeling the gas inventories, gas temperature, thermal conductivity degradation and rod internal volumes throughout the life of the limiting rod. An assessment of this criterion using EPU condition and standard procedures showed that this criterion was met considering the effects of TCD.

As part of a previous request for additional information (RAI), the margin to the RIP no gap reopening criterion was requested. The calculated margin to the RIP no gap reopening criterion limit with respect to TCD was [ ]^{a,c} psi.

The second part of the rod internal pressure design basis precludes extensive DNB propagation and associated fuel failure. An assessment of this criterion using EPU conditions and standard procedures showed that this criterion was met considering the effects of TCD.

Clad Stress

Originally, as part of the PTN evaluation, the clad stress limits were met using the stress criterion in Reference 5. As part of the assessment of TCD, it was found that the effects of TCD would cause this original limit to be violated. The clad stress criterion in Reference 5 is conservative compared to the requirements of the NRC Standard Review Plan (SRP) 4.2, which requires a transient strain limit of 1% strain. The Reference 5 clad stress criterion is also conservatively based on yield stress, whereas the failure mode is more realistically a function of ultimate strength. NRC approval was provided in Reference 7 for a revised stress criterion, which is based on ultimate strength rather than yield strength. The assessment conducted using the revised clad stress criterion showed that this criterion is met.

Clad Strain

An assessment of transient and steady-state clad strain criteria using EPU conditions and standard procedures showed that the criteria were met considering the effects of TCD.

Clad Oxidation and Hydriding

Cladding metal surface temperatures, which are the principal component in corrosion calculations, are calculated based on heat flux and coolant conditions, and therefore are not impacted by conditions inside the fuel rod cladding. The criteria remain met with respect to TCD.

Fuel Temperature

Fuel temperatures have been calculated as a function of local power and burnup with the effects of TCD included. The fuel surface and average temperatures with associated rod internal pressure are provided to transient analysis and LOCA groups for accident analysis of the Upgrade and DRFA fuel designs. The fuel centerline temperatures are used to show that fuel melt will not occur. For Upgrade and DRFA designs, the local linear power that precludes fuel centerline melting is provided in Table 5-1.

Clad Fatigue

An assessment of the fatigue criterion using EPU condition and standard procedures showed that this criterion was met considering the effects of TCD.

Clad Flattening

The NRC has approved WCAP-13589-A (Reference 6), which provided data to confirm that significant axial gaps in the fuel column due to densification (and therefore clad flattening) will not occur in current Westinghouse fuel designs. The data included in this topical inherently includes any effects of TCD. Therefore, the criteria remain met with respect to TCD.

Fuel Rod Axial Growth

The cladding growth model is based on in-reactor data, which inherently includes the effects of TCD in development of the model. TCD is inherently included in the in-reactor database of rod growth measurements. Thus, the criteria remain met with respect to TCD.

Plenum Clad Support

The helical coil spring used in the Upgrade and DRFA fuel designs for the PTN EPU has been shown to provide enough support to prevent potential clad collapse. This analysis is not impacted by TCD.

Clad Free-Standing

Because clad free-standing analyses are not impacted by fuel temperatures, the effects of TCD have no impact on this criterion. Thus, the criteria remain met with respect to TCD.

Fuel Rod End Plug Weld Integrity

An assessment of the fuel rod end plug weld integrity criterion using EPU condition and standard procedures showed that this criterion was met considering the effects of TCD.

Fuel Temperatures and Rod Internal Pressures

Fuel temperatures and associated rod internal pressures have been generated using the NRC-approved PAD code (Reference 4) and have included the effects of TCD. The integral fuel burnable absorber (IFBA) and non-IFBA fuel temperature and/or rod internal pressures were used as initial conditions for LOCA and non-LOCA transients. The linear power limit to preclude fuel centerline melt is included in Table 5-1.

In addition to the fuel rod temperatures and rod internal pressures, the core stored energy for the Upgrade and DRFA fuel has been determined for input to LOCA mass and energy release and non-LOCA steam release for dose analyses. Core stored energy is defined as the amount of energy in the fuel rods in the core above the local coolant temperatures. Typically, these values are calculated at every local power and burnup, with the worst value conservatively applied to the whole core. This approach is overly conservative. In this assessment, a conservative rod average burnup is applied to find core stored energy values for feed, once-burned and twice-burned fuel. Powers of every rod were then binned, and a weighted average of core stored energy at these power levels was calculated. This bounding weighted average of core stored energy for the entire core was then provided for input to LOCA mass and energy and non-LOCA steam release for dose analyses (see discussion provided in Sections 4a (LR 2.6.3.1) and 4b (LR 2.6.3.2), respectively). With weighted averaging and TCD considered, the core stored energy is [ ][a,c] full power seconds. The original core stored energy value without TCD was [ ][a,c] full power seconds.

Fuel System Design Results

An assessment of the impact of TCD has been performed for the EPU transition and equilibrium cycles to demonstrate that the design criteria can be satisfied for all fuel rod types in the core. TCD has impacted several design criteria including RIP, clad stress, clad strain and fatigue analyses. The RIP, clad strain and fatigue criterion remain satisfied, using the same methods and procedures used in the EPU analysis. The clad stress criterion requires a methodology change. Specifically, the implementation of the ASME clad stress criterion as approved in Reference 7 should be applied for the EPU.

In summary, all fuel rod design criteria have been evaluated for application of the Westinghouse Upgrade and DRFA fuel assembly designs in PTN at EPU conditions using the TCD model and the design limits can be met if the revised stress criterion is applied.

Table 5-1:
Power to Melt Limit

a,c

b. Non-LOCA – RCCA Ejection (LR Section 2.8.5.4.6)

Spectrum of RCCA Ejection Accidents (Rod Ejection)

Input Parameters, Assumptions and Acceptance Criteria

Almost all of the RCCA ejection input parameter descriptions currently provided in Section 2.8.5.4.6.2.2 of the EPU Licensing Report for Turkey Point Units 3 and 4 are unaffected by consideration of TCD. Specifically, the analysis performed to address TCD for the RCCA ejection accident does not require any changes to the following input parameters:

- Ejected rod worths & hot channel factors (F_Q)
- Delayed neutron fraction (β)
- Reactivity weighting factor
- Moderator and Doppler coefficients
- Coolant mass flow rates
- Trip reactivity insertion

As discussed below in the "Description of Analyses and Evaluations" section, the analysis for the RCCA ejection event including the effects of TCD does involve changing some heat transfer modeling inputs to the FACTRAN computer code. Table 5-2 (which corresponds to Table 2.8.5.4.6-1 in LR Section 2.8.5.4.6) documents the input parameters used in the RCCA ejections analysis with TCD. All of these input parameters are completely unchanged from those used previously without consideration of TCD.

Table 5-2
Input Parameters for the Limiting RCCA Ejection Analysis

	Beginning of Cycle Hot Full Power	**Beginning of Cycle** Hot Zero Power	**End of Cycle** Hot Full Power	**End of Cycle** Hot Zero Power
Initial core power level, percent of 2644 MWt	100.3%	0%	100.3%	0%
Ejected rod worth, % ΔK	0.33	0.71	0.30	0.84
Delayed neutron fraction, %	0.55	0.55	0.44	0.44
Feedback reactivity weighting	1.29	1.60	1.30	2.28
Trip reactivity, % ΔK	4.0	2.0	4.0	2.0
F_Q before rod ejection	2.4	--	2.4	--
F_Q after rod ejection	5.48	8.0	5.52	14.3
Number of operational RCPs	3	2	3	2

Consideration of TCD does not affect any of the acceptance criteria applied to the RCCA ejection analysis. Therefore, all of the acceptance criteria described in Section 2.8.5.4.6.2.2 of the EPU Licensing Report for Turkey Point Units 3 and 4 continue to apply.

Description of Analyses and Evaluations

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed using the TWINKLE spatial kinetics computer code to determine the average power generation with time including the various total core feedback effects; i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients at the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient. The FACTRAN computer code is used to perform the hot spot analysis, using the transient core average power prediction from TWINKLE as input.

The analysis performed to address TCD has made no changes to the input for the TWINKLE computer code. Therefore, the core power transients for all four rod ejection cases are modeled as being the same with TCD as without. This is conservative because in reality, the higher fuel temperatures associated with TCD would increase the Doppler defect which would reduce the magnitude of the post-ejection power excursion. All the modeling changes to address TCD for the RCCA ejection analysis are made to inputs for the FACTRAN computer code that performs the core hot spot analysis.

Two portions of the FACTRAN input files have been revised to address the impact of TCD. First, the input maximum fuel temperatures used to calibrate initial conditions in the fuel pellet for the hot full power (HFP) RCCA ejection cases have been updated to reflect the fuel temperatures with TCD. Secondly, the input to FACTRAN has been modified for all the RCCA ejection cases analyzed to model fuel conductivity that reflects the impact of TCD. The fuel conductivity inputs used are burnup dependent. Therefore, the FACTRAN inputs used for BOC and end of cycle (EOC) fuel conductivity are different. It is important to note that the subject FACTRAN input changes are completely consistent with the existing methodology for analyzing the RCCA ejection accident.

The primary impact of TCD on the RCCA ejection hot rod analysis is an increase in the initial fuel temperatures (and fuel enthalpy). The fuel enthalpy deposition during the RCCA ejection transient is minimally affected by TCD because the analysis conservatively models that DNB is occurring at the start of the event. Therefore, heat transfer out of the fuel is severely restricted and TCD-related reductions in the fuel conductivity have little impact on the total energy deposition in the fuel. The analysis covers BOC and EOC conditions such that a broad range of burnup is assumed for the limiting fuel rod; up to 48,200 MWD/MTU at EOC. This bounding range of burnup and initial Fq assures that the results presented below in Table 5-3 conservatively address the effects of TCD on the RCCA ejection analysis.

Summary of Results

Tabulated results for the RCCA ejection analysis including the effects of TCD are presented in Tables 5-3 and 5-4 (which correspond to the Analysis Results table and Table 2.8.5.4.6-2, respectively, in LR Section 2.8.5.4.6) along with Figures 5-1 through 5-4 (which correspond to the Figures 2.8.5.4.6-1 through 2.8.5.4.6-4 in LR Section 2.8.5.4.6). In all cases the maximum average fuel pellet enthalpy remains below 360 Btu/lb (200 cal/gm). The maximum average clad temperature remains below 3000°F in all cases, and the zirconium-water reaction remains below 16%. In the HFP cases, less than 10% fuel melt occurs. No fuel melt occurs in the HZP cases.

Compared to the results reported in the EPU Licensing Report, the results in Tables 5-3 and 5-4 show that inclusion of TCD had very little impact on the cases initiated from HZP conditions. For the HZP cases, the fuel temperatures and the maximum fuel enthalpy did increase due to TCD but these increases were relatively small and no safety analysis limits were approached. For the HFP cases, inclusion of TCD did increase the maximum fuel stored energy and substantially increased the predicted fuel melting. However, all safety analysis limits continued to be met for the rod ejection event with TCD.

It should be noted that the approach used here to address TCD for RCCA ejection was to maintain intact almost all of the RSAC limits that were defined by the EPU analysis and only change the fuel temperatures and fuel conductivity that are directly impacted by TCD. This approach minimizes any impact on existing reload interfaces between the non-LOCA analysis and the core design. As a result there are a number of potential margin sources available for the rod ejection analysis (particularly for the HFP cases), including the following:

- Maximum ejected rod worths – analyzed values higher than predicted reload values
- Post-ejection power peaking (F_Q) - analyzed values higher than predicted reload values
- Crediting increased Doppler defect in TWINKLE due to higher fuel temperature

Table 5-3 RCCA Ejection Results Including the Effects of TCD			
Beginning of Cycle Cases			
	BOC HFP	BOC HZP	Limit
Max. Average Fuel Pellet Enthalpy, Btu/lbm	321	157.6	360
Fuel Melt at the Hot Spot, %	8.23	0.0	10
Max. Fuel Pellet Average Temperature, °F	4076	2222	NA
Max. Fuel Centerline Temperature, °F	>4900	2562	NA
End of Cycle Cases			
	EOC HFP	EOC HZP	Limit
Max. Average Fuel Pellet Enthalpy, Btu/lbm	306.3	249.3	360
Fuel Melt at the Hot Spot, %	8.44	0.0	10
Max. Fuel Pellet Average Temperature, °F	3920	3296	NA
Max. Fuel Centerline Temperature, °F	>4800	3691	NA

Table 5-4 Time Sequence of Events – RCCA Ejection		
Event	**Time (sec)**	
	BOC HFP	EOC HFP
Initiation of Rod Ejection	0.0	0.0
Power Range High Neutron Flux Setpoint Reached	0.04	0.04
Peak Nuclear Power Occurs	0.13	0.13
Rods Begin to Fall	0.54	0.54
Peak Fuel Average Temperature Occurs	2.14	2.09
Peak Clad Temperature Occurs	2.24	2.25
	BOC HZP	EOC HZP
Initiation of Rod Ejection	0.0	0.0
Power Range High Neutron Flux Setpoint Reached	0.30	0.17
Peak Nuclear Power Occurs	0.36	0.21
Rods Begin to Fall	0.80	0.67
Peak Clad Temperature Occurs	2.29	1.37
Peak Fuel Average Temperature Occurs	2.35	1.62

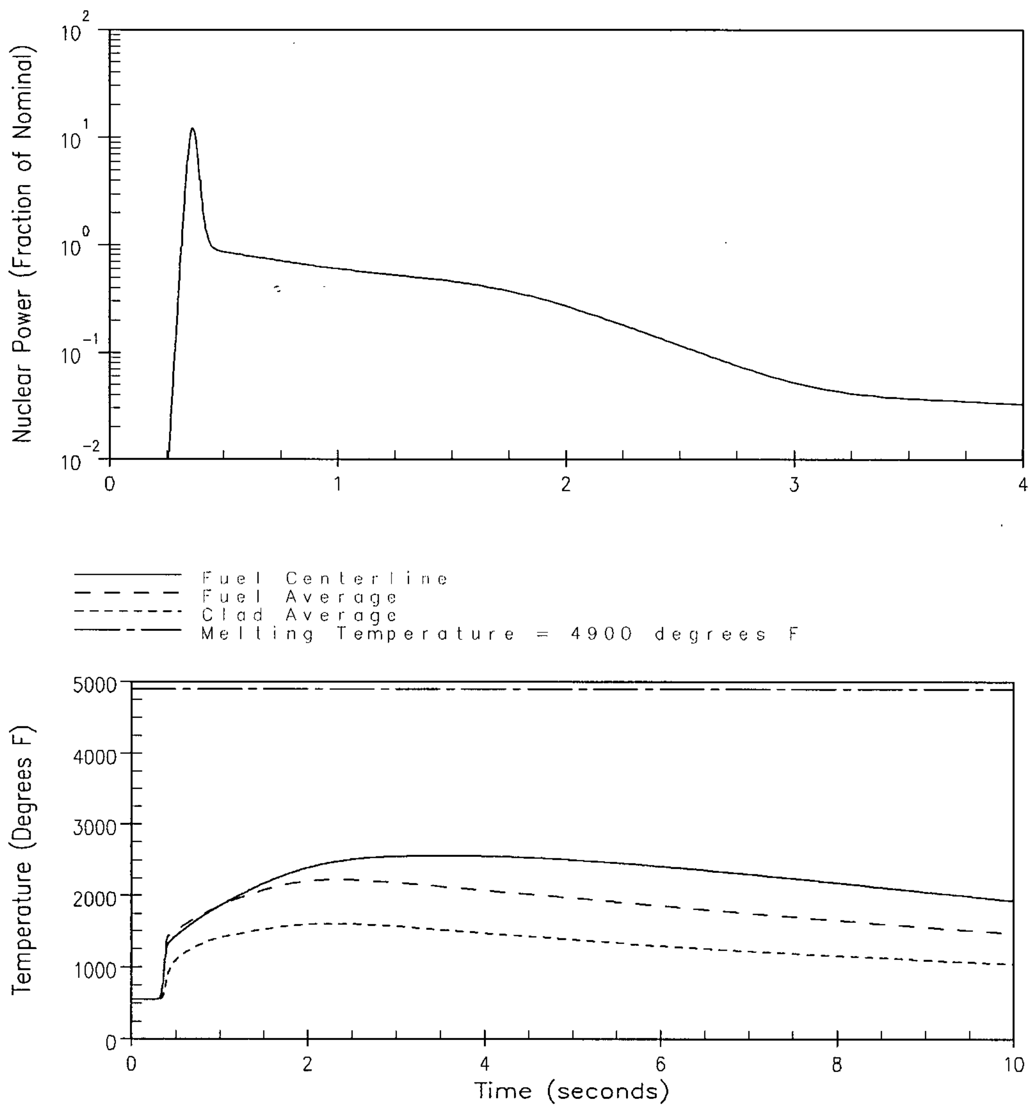

Figure 5-1
Rod Ejection – Beginning of Cycle / Hot Zero Power Case

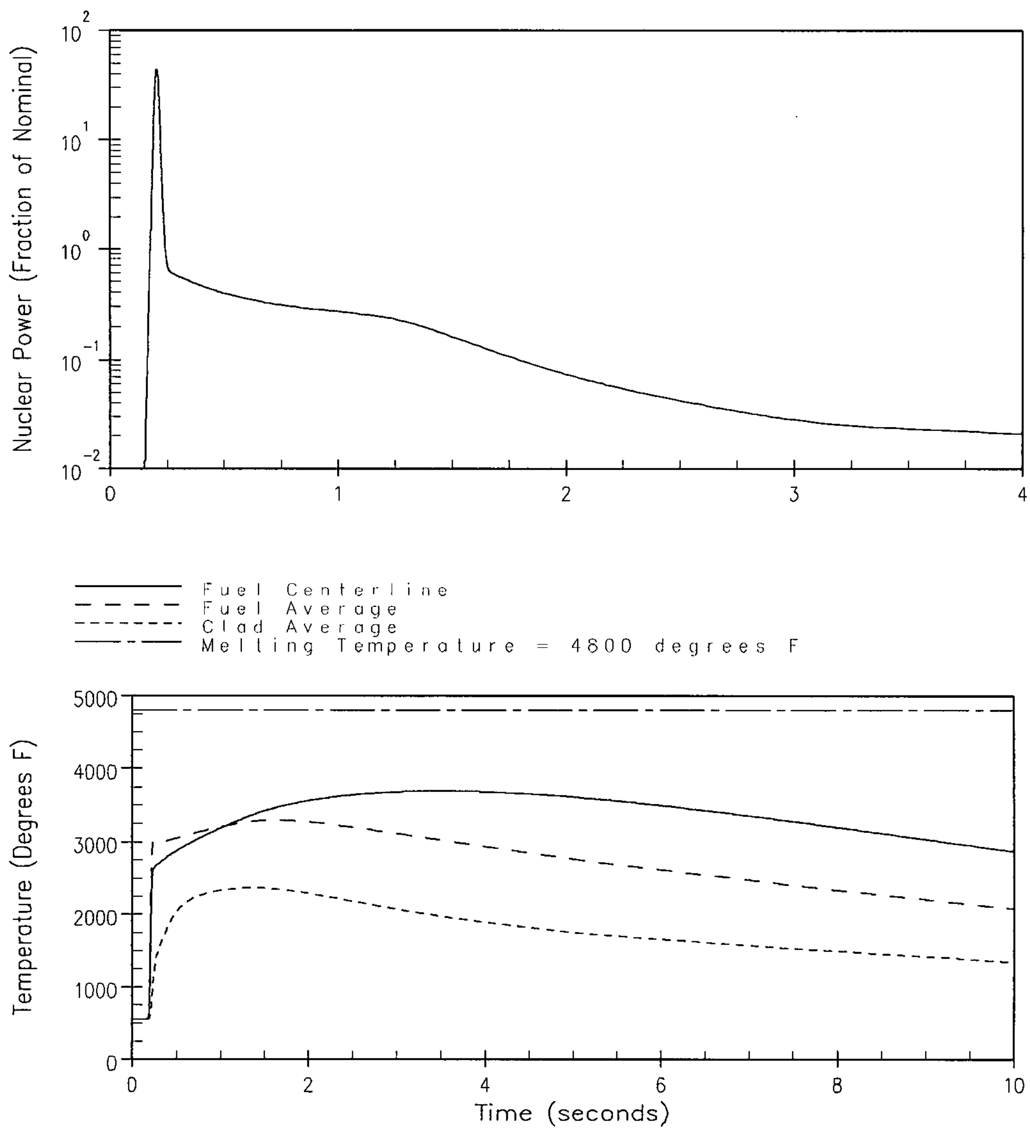

Figure 5-2
Rod Ejection – End of Cycle / Hot Zero Power Case

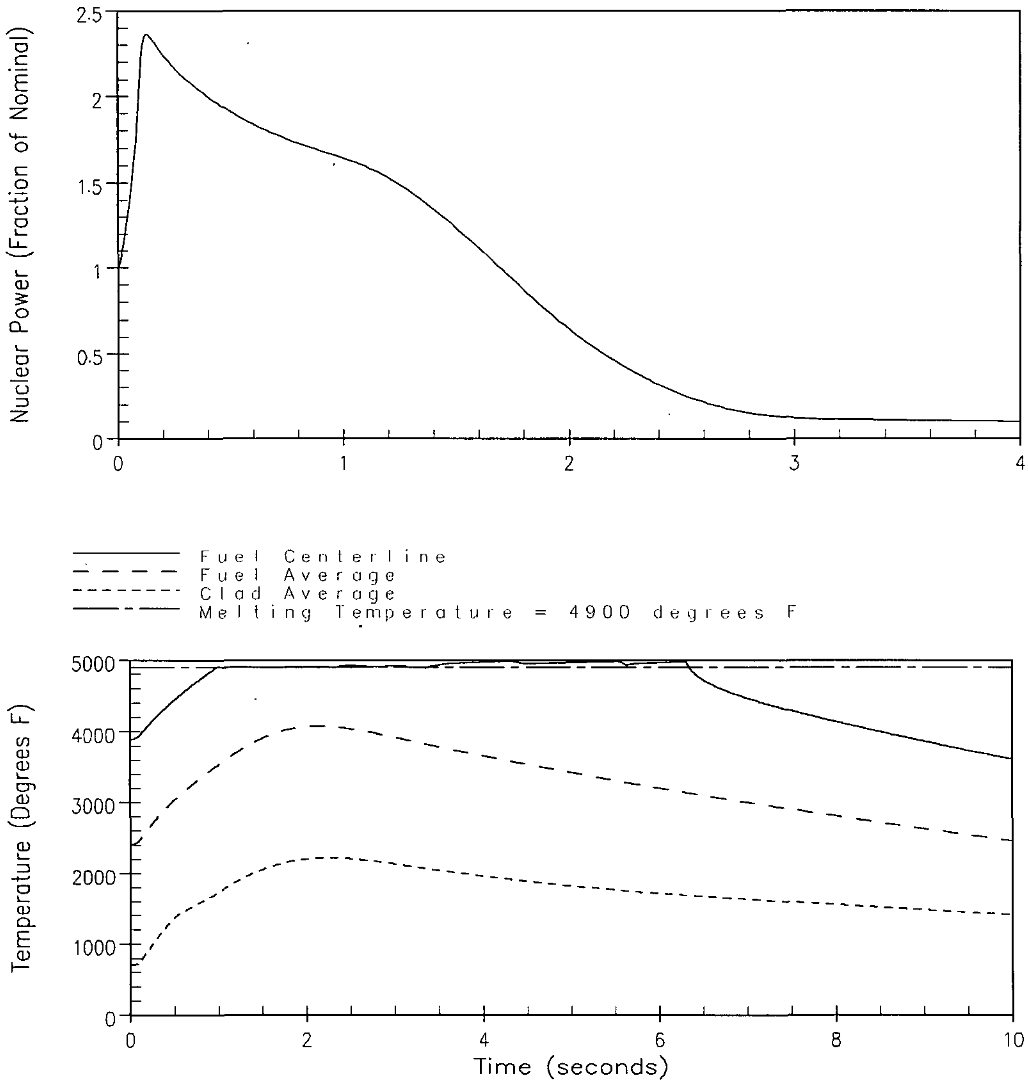

Figure 5-3
Rod Ejection – Beginning of Cycle / Hot Full Power Case

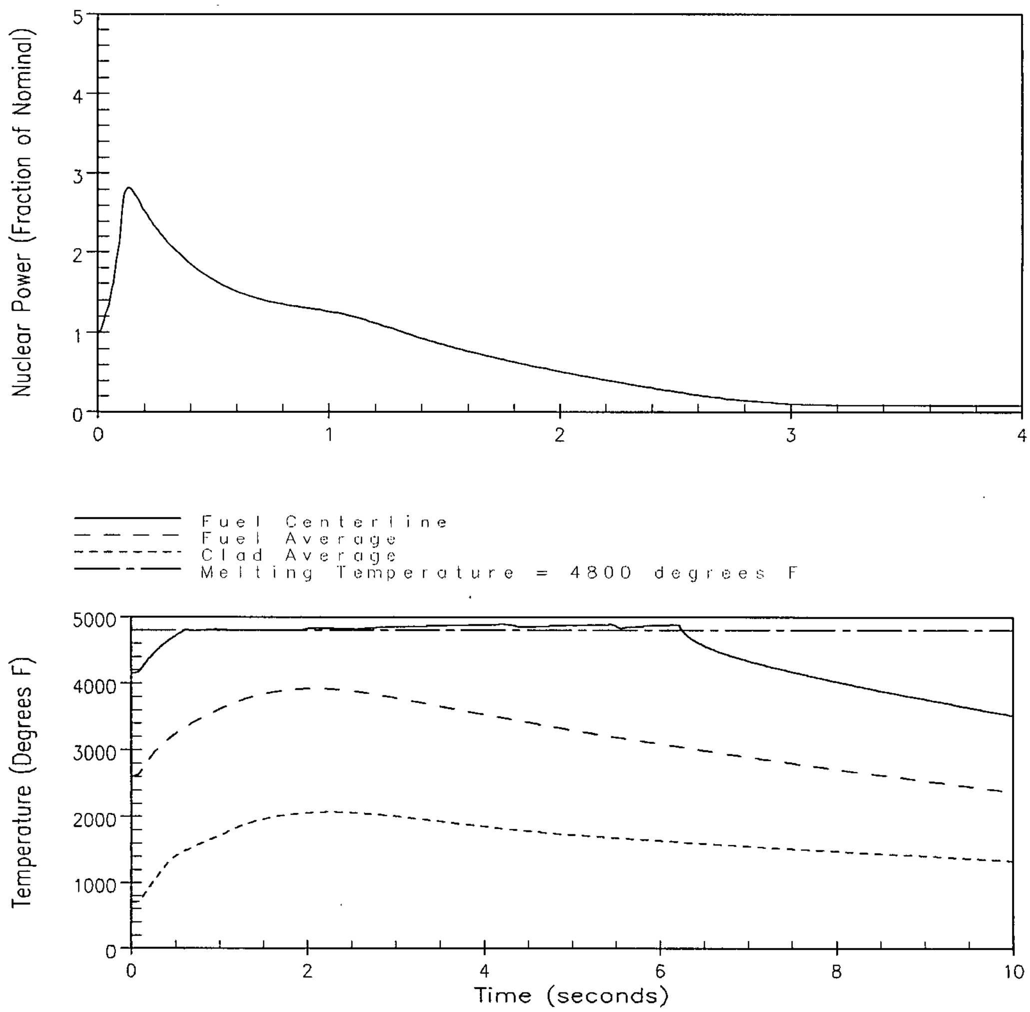

Figure 5-4
Rod Ejection – End of Cycle / Hot Full Power Case

c. Large Break Loss-of-Coolant Accident (LBLOCA) (LR Section 2.8.5.6.3.1)

This section is supplemental to Section 2.8.5.6.3.2.1 of the Florida Power and Light Extended Power Uprate License Amendment Request (Reference 1).

The NRC-approved Westinghouse Best-Estimate Loss-of-Coolant Accident (BELOCA) Automated Statistical Treatment of Uncertainty Method (ASTRUM) methodology (Reference 9) is based on the PAD 4.0 fuel performance code (Reference 4). PAD 4.0 was licensed without explicitly considering fuel thermal conductivity degradation (TCD) with burnup. Explicit modeling of TCD in the fuel performance code leads directly to increased fuel temperatures (pellet radial average temperature) as well as other fuel performance related effects beyond beginning-of-life. Since PAD provides input to the large-break LOCA analysis, this will tend to increase the stored energy at the beginning of the simulated large-break LOCA event. This in turn leads to an increase in Peak Cladding Temperature (PCT) if there is no provision to credit off-setting effects.

The Unit 3 and Unit 4 Turkey Point Nuclear Power Plant (PTN) PCT was calculated to be 2064°F, the Maximum Local Oxidation (MLO) was calculated to be 4.25% and the Core-Wide Oxidation (CWO) was calculated to be 0.43% with the plant-specific adaptation of currently licensed Westinghouse BELOCA methodology (Reference 9). The PCT limit of 2200°F is met with 136°F margin (124°F when considering transition core penalties), the MLO limit of 17% is met with 12.75% margin, and the CWO limit of 1% is met with 0.57% margin.

Fuel performance data that accounts for fuel TCD was used as input to the updated PTN BELOCA analysis. The new PAD fuel performance data was generated with an updated PTN PAD model that includes explicit modeling of TCD. Therefore, the BELOCA analysis was updated to consider the fuel TCD effects cited in NRC Information Notice 2011-21 (Reference 10).

Input Parameters, Assumptions and Acceptance Criteria

Updates to design inputs and plant operating ranges to gain large-break LOCA margin were considered in this updated analysis to show compliance with the 10 CFR 50.46 acceptance criteria while maintaining a margin of safety to the prescribed limits. The acceptance criteria and results of the updated BELOCA analysis considering TCD effects are discussed in the summary of results. The base input assumptions are provided in Licensing Report (LR) Section 2.8.5.6.3.2 (Reference 1).

In order to mitigate the impact of the increasing effect of pellet TCD with burnup, the updates to the large-break LOCA analysis utilized reduced peaking factors from those shown in LR Table 2.8.5.6.3.2-1 (Reference 1). The reduced peaking factors considered fall into two different categories:

1) A reduction of peaking factors for the hot rod and hot assembly in the 1st cycle of operation to reflect revised power peaking limits, and
2) Burndown credit for the hot rod and hot assembly in the 2nd and 3rd cycle of operation.

The reduction of peaking factors for the hot rod and hot assembly in the 1st cycle described in (1) above is used to mitigate the impact of the increased fuel temperature calculated by PAD with explicit TCD modeling. Burndown credits in the 2nd and 3rd cycle described in (2) above are used to demonstrate that analyzing the hot rod and hot assembly in the first cycle of operation is still bounding with respect to PCT and MLO. The PTN peaking factor limits supported by this updated analysis are shown in Table 5-5.

Additionally, some other analysis inputs were updated to gain large-break LOCA margin (not considered changes to the approved ASTRUM Evaluation Method (EM) (Reference 9)). The updated inputs were:

1. Accumulator Water Volume – Change minimum accumulator volume from 865 to 872 ft^3 (6470 to 6520 gals) consistent with TS Surveillance Requirement.
2. High Head Safety Injection Delay Time (with offsite power available) – Change delay time from 23 seconds to 17 seconds.
3. Vessel Average Temperature - Change the minimum full-power normal operating vessel average temperature (T_{avg}) from 570°F to 577°F.
4. Steam Generator Tube Plugging (SGTP) – Change maximum assumed SGTP from 10% to 5%.

A major plant parameter assumptions table for the updated PTN BELOCA analysis considering the effects of TCD is provided in Table 5-6.

Additionally, it is noted that the updates to the analysis are in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A (Reference 9), as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A (Reference 9) was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report appended to this topical report.

PTN and its vendor, Westinghouse Electric Company LLC, utilize a process which will ensure that LOCA analysis input values conservatively bound the as-operated plant values for those parameters. This includes accounting for transition core penalties (as applicable) on a cycle-by-cycle basis.

Description of Analysis and Evaluations

The purpose of this update to the analysis is to consider fuel performance inputs that explicitly model TCD and utilize changes to design inputs and plant operating ranges (major plant parameter assumptions table for the updated PTN BELOCA TCD analysis is provided in Table 5-6) to show compliance with the 10 CFR 50.46 acceptance criteria while maintaining a margin of safety to the prescribed limits. The updated BELOCA analysis considering the effects of TCD is supplemental to information provided in LR Section 2.8.5.6.3.2.1 (Reference 1).

Fuel performance data that accounts for fuel TCD was used as input to the updated PTN BELOCA analysis. The new PAD fuel performance data was generated with an updated PAD model that includes explicit modeling of TCD. In addition, the STAV-7.3 thermal conductivity model in WCOBRA/TRAC was used to more accurately model the fuel temperature profile.

The updated analysis also credited peaking factor burndown to address fuel in its second cycle of irradiation. Analysis of fuel in its second cycle of irradiation is beyond the first cycle considered in the approved ASTRUM EM, but was considered in the updated analysis when explicitly modeling TCD to demonstrate that analyzing the hot rod and hot assembly in the first cycle of operation is still bounding with respect to PCT and MLO. Physically, accounting for TCD leads to an increase in fuel temperature as the fuel is burned, while accounting for peaking factor burndown leads to a reduction in fuel temperature as the fuel is burned. The compensating nature of these phenomena is considered in the updated analysis in order to appropriately capture the effect of TCD in the updated PTN BELOCA analysis.

The analysis was performed updated by re-running a subset of cases from the original ASTRUM analysis (LR Section 2.8.5.6.3.2 (Reference 1)). The intent for the selection of the subset of cases was to capture the fraction of the sample that includes all cases that can potentially become the rank k=1 case of the 124 run-set after TCD effects are included. Therefore the same non-parametric order statistics singular statement of a 95th percentile at the 95-percent confidence joint probability for PCT, MLO and CWO of an ASTRUM re-analysis is ensured with high confidence for the PTN updated analysis.

[

]^{a,c}. As such, all cases from the original ASTRUM execution (Reference 1) which had a PCT within 400-500°F of the analysis result from (Reference 1) were executed. Executing the subset of runs that falls within the maximum hot rod fuel temperature increase plus additional margin is considered adequate to capture all cases that can potentially become the rank k=1 case of the 124 run-set after TCD effects are included. This subset also captures all original ASTRUM analysis runs with maximum local oxidation greater than 1%; as maximum local oxidation tends to increase with PCT, this subset is also considered adequate for assessing the effect on local oxidation.

It is noted that the confirmatory configuration and the conservatively low containment backpressure from the original ASTRUM run set were not explicitly re-evaluated considering the effects the TCD. The limiting plant configuration was considered and determined to remain the same, although the full parametric study was not performed. The conservatively low containment backpressure from the original analysis remains bounding since the core stored energy increases when explicitly modeling fuel TCD.

Summary of Results

The PTN PCT-limiting transient remains a double-ended cold leg guillotine break when considering fuel TCD and analyzing the conditions provided in Tables 5-5 and 5-6. Table 5-7 summarizes the results of the updated BELOCA analysis considering the effects of TCD. Table 5-8 provides the sequence of events for the PCT-limiting transient from the updated analysis.

The scatter plot presented in Figure 5-5 shows the impact of the effective break area on the PCT in the limiting subset of runs performed as part of the updated BELOCA analysis considering the effects of TCD. The effective break area is calculated by multiplying the discharge coefficient CD with the sampled value of the break area, normalized to the cold-leg cross sectional area. Figure 5-5 is provided because the break area is a contributor to the variation in PCT and to show that the range of limiting break areas from the ASTRUM analysis (see LR Figure 2.8.5.6.3.2-1 (Reference 1)) considered in the updated BELOCA analysis considering the effects of TCD.

Figures 5-6 and 5-7 are presented to show the limiting cladding transient for each 10 CFR 50.46 criterion evaluated in the updated BELOCA analysis considering the effects of TCD. Figure 5-6 shows the HOTSPOT predicted clad temperature transient at the PCT and MLO limiting elevation for the limiting PCT and MLO case. Figure 5-7 shows the WCOBRA/TRAC predicted peak cladding temperature for the CWO limiting transient.

Additional Plots for the Limiting PCT Transient

Figures 5-8 through 5-20 were generated using the limiting PCT case. The PCT-limiting case was chosen to illustrate a conservative representation of the response to a large break LOCA that includes the effects of fuel TCD.

Figure 5-8 is a plot of the pressurizer pressure throughout the PCT-limiting transient. Figures 5-9 and 5-10 are plots of the mass flow rate through the break (Vessel and Pump side, respectively). Figure 5-11 presents the void fraction in both the intact and broken loop pumps; the dashed curve represents the broken loop pump. Figure 5-12 is a plot of the vapor flow rate near the top of the core in the Hot Assembly.

Figure 5-13 is a plot of an intact loop accumulator injection flow. Figure 5-14 is a plot of the Safety Injection Flow into one of the intact cold legs. Figures 5-15, 5-16, 5-17 are plots of the lower plenum, downcomer, and core channels collapsed liquid levels, respectively. The reference point for the downcomer liquid level is the point at which the outside of the core barrel, if extended downward, intersects with the vessel wall. The reference point for the core collapsed liquid levels is the bottom of the active fuel.

The vessel fluid inventory throughout the transient is plotted in Figure 5-18. Figure 5-19 is a plot of the PCT for all 5 rods modeled in WCOBRA/TRAC, and Figure 5-20 is a plot of the hot rod PCT elevation versus time. Note: The PCTs in Figure 5-19 are the WCOBRA/TRAC calculated temperatures, not the HOTSPOT calculated temperatures (Figure 5-6 is HOTSPOT calculated temperatures).

The axial power distribution shown in LR Figure 2.8.5.6.3.2-17 (Reference 1) and the containment response analyzed as discussed in LR Section 2.6.6 (Reference 1) remain unchanged in the updated BELOCA analysis considering the effects of TCD.

Figures 5-6 through 5-20 show that the modeling TCD does not significantly alter the PTN transient response. The time of PCT is observed to remain shortly after the beginning of core recovery. The reactor coolant system depressurization and the initiation of the emergency core cooling equipment is not affected by the increased stored energy, and the global system response for the new limiting case is observed to be similar to the limiting case of the original ASTRUM execution.

10 CFR 50.46 Requirements

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 2093°F, the updated analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Clad Temperature less than 2200°F," is demonstrated. The result is shown in Table 5-7.

(b)(2) The maximum local cladding oxidation corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting transient MLO for the limiting case is 7.46 percent, the updated analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Maximum Local Oxidation of the cladding less than 17 percent," is demonstrated. The result is shown in Table 5-7.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A detailed CWO calculation is not needed because the outcome is always less than 0.40 percent. Therefore, the updated analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent," is demonstrated. The result is shown in Table 5-7.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The approved methodology (Reference 1) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 28 assemblies in the low-power channel; this situation is not calculated to occur for PTN (Reference 1), and this conclusion is not affected by the modeling of fuel TCD. Therefore, acceptance criterion (b)(4) is satisfied.

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at these plants to maintain long-term cooling remain unchanged due to the modeling of fuel TCD. See Long Term Cooling assessment herein for details.

It is noted that the responses to requests for additional information by the staff on LR Section 2.8.5.6.3.2 (Reference 1) either: 1) not impacted, 2) showed trends that are not invalidated by the updated BELOCA analysis considering the effects of TCD (though specific numbers may differ), or 3) are addressed herein.

It is noted that the most severe analyzed case in the ASTRUM analysis (LR Section 2.8.5.6.3.2 (Reference 1)) had an assumed hot rod burnup of 1,552 MWD/MTU (per response to RAI SRXB-1.3.33). When considering the effects of TCD, the most severe case had an assumed hot rod burnup of 26,827 MWD/MTU. The change in hot rod burnup of the most severe case analyzed is expected since fuel average temperature increases as a function of burnup when explicitly modeling TCD. The increased fuel average temperature leads to an increase in blowdown heatup, and this PCT increase is carried through refill and to the time of PCT. Figure 5-21 shows the PCT as a function of hot rod burnup in the limiting subset of runs performed as part of the updated BELOCA analysis considering the effects of TCD.

The PTN ASTRUM uncertainty attributes provided in Table 1.3.34-1 in the response to RAI SRXB-1.3.34 are provided for the updated analysis in Table 5-9. As previously discussed above, only a subset of cases from the original ASTRUM execution was executed as part of the updated analysis. It is noted that only the TAVG, VACC, FDH, and FQ values differ from Table 1.3.34-1 in the response to RAI SRXB-1.3.34. The application of the approved statistical methodology applies a random number generator that returns a value between 0 and 1 (see Section 11 of the ASTRUM Topical (Reference 9)). The value(s) between 0 and 1 are then translated into the sampled parameter value using the assigned distribution(s) (e.g. normal, uniform, actual) and the range or nominal value provided as input. Figure 5-22 illustrates the effect on updating the input range on the accumulator water volume sampled parameter. In the case of the PTN updated BELOCA analysis, the random numbers generated between 0 and 1 from the original ASTRUM execution were maintained for all sampled parameters.

An update to the analysis was performed considering fuel performance inputs that explicitly model TCD and changes to allowable peaking factors and other analysis inputs to show compliance with the current 10 CFR 50.46 acceptance criteria while maintaining a margin of safety to the prescribed limits. This updated analysis also demonstrated that assuming the hot assembly in the first cycle of operation is still bounding with respect to PCT and MLO (transient) when considering the physically off-setting effects of fuel TCD and peaking factor burndown.

Based on the results from the updated BELOCA analysis (see Table 5-7), it is concluded that PTN continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

Table 5-5
Summary of Peaking Factor Margin and Burndown Supported by PTN BELOCA Updated Analysis

Hot Rod Burnup (GWD/MTU)	FdH (with uncertainties)	FQ Transient (with uncertainties)	FQ Steady-state (without uncertainties)
0	1.6	2.3	1.9
20	1.6	2.3	1.9
30	1.6	2.3	1.9
49	1.33	1.84	1.52
60	1.33	1.84	1.52
65	1.33	1.84	1.52

* The standard BELOCA assumption of [ ]^{a,c}

Table 5-6
Major Plant Parameter Assumptions Used in the PTN Best-Estimate Large-Break LOCA Updated Analysis Considering the Effects of Thermal Conductivity Degradation

Plant Physical Description	
SG Tube Plugging	≤ 5%
Fuel Assembly Type	15x15 Upgrade Fuel with ZIRLO® or Optimized ZIRLO™ cladding, non-IFBA or IFBA, IFMs
Plant Initial Operating Conditions	
Reactor Power	≤ 2652 MWt (Bounding)
Peaking Factors	See Table 5-5
Axial Power Distribution	See LR Figure 2.8.5.6.3.2-17 (Reference 1)
Fluid Conditions	
Tavg	577.0–6.0°F ≤ Tavg ≤ 583.0 + 6.0 °F
Pressurizer Pressure	2250–53 psia ≤ PRCS ≤ 2250 + 53 psia
Thermal Design Flow	≥ 86,900 gpm/loop
Accumulator Temperature	85 °F ≤ TACC ≤ 126 °F
Accumulator Pressure	589.7 psia ≤ PACC ≤ 714.7 psia
Accumulator Water Volume	872 ft^3 ≤ VACC ≤ 920 ft^3
Accumulator Boron Concentration	≥ 2300 ppm
Accident Boundary Conditions	
Single Failure Assumptions	Loss of one ECCS train
Safety Injection Flow	Minimum
Safety Injection Temperature	34 °F ≤ TSI ≤ 105°F
High Head Safety Injection Initiation Delay Time	≤ 17 sec (with offsite power) ≤ 35 sec (without offsite power)
Low Head Safety Injection Initiation Delay Time	≤ 23 sec (with offsite power) ≤ 35 sec (without offsite power)
Containment Pressure	See LR Figure 2.6.6-1 (Reference 1)

Table 5-7
PTN Best-Estimate Large-Break LOCA Updated Analysis Considering the Effects of Thermal Conductivity Degradation – Comparison of Results to Current 10 CFR 50.46(b) Acceptance Criteria

	Result	Acceptance Criterion
95/95 PCT[1]	2093°F	< 2200°F
95/95 Transient MLO[2]	7.46%	< 17%
95/95 CWO[3]	0.39%	< 1%
Coolable Geometry	Criterion Met	Remains Coolable
Long-Term Cooling	See Long Term Cooling TCD assessment in Section 4	

Notes:
1. Peak Cladding Temperature
2. Maximum Local Oxidation, transient
3. Core-wide Oxidation

Table 5-8
Sequence of Event for the PTN Best-Estimate Large-Break LOCA Updated Analysis Considering the Effects of Thermal Conductivity Degradation Limiting PCT Transient

Event	Time After Break (sec)
Start of Transient	0
Safety Injection Signal	4.8
Accumulator Injection Begins	8
Safety Injection Begins	21.8
End of Blowdown	22
Bottom of Core Recovery	31.5
Accumulator Empty[1]	36.8
PCT Occurs	46
End of Transient	300

Notes:
1. Accumulator liquid injection ends

a.c

Table 5-9: ASTRUM Uncertainty Attributes for PTN Best-Estimate Large-Break LOCA Updated Analysis (Ranked by Updated HOTSPOT PCT)

a.c

Table 5-9: ASTRUM Uncertainty Attributes for PTN Best-Estimate Large-Break LOCA Updated Analysis (Ranked by Updated HOTSPOT PCT)

a.c

Table 5-9: ASTRUM Uncertainty Attributes for PTN Best-Estimate Large-Break LOCA Updated Analysis (Ranked by Updated HOTSPOT PCT)

Figure 5-5
PTN HOTSPOT PCT vs. Effective Break Area

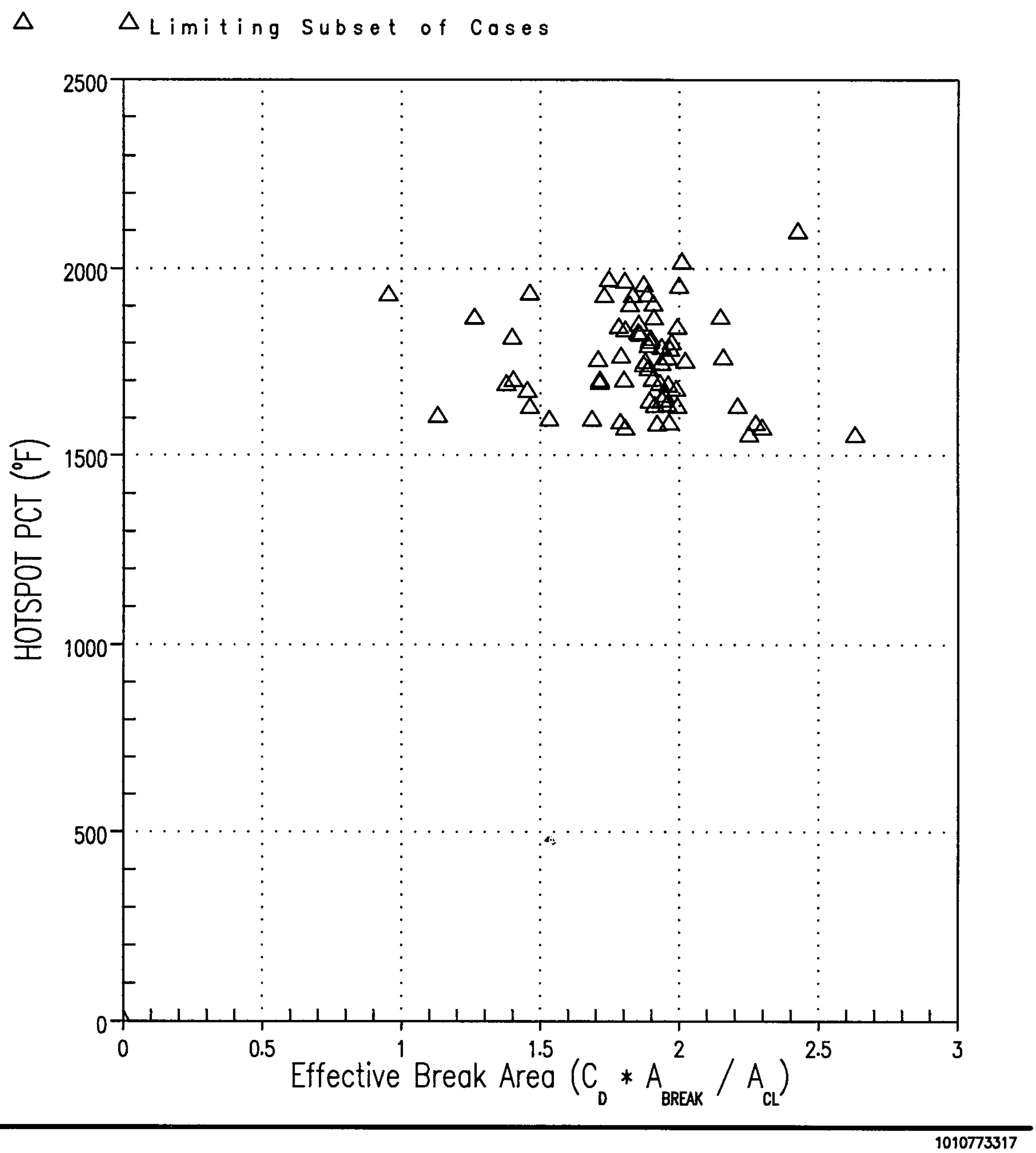

Figure 5-6
PTN HOTSPOT Clad Temperature Transient at the Limiting Elevation for the Limiting PCT and MLO Case

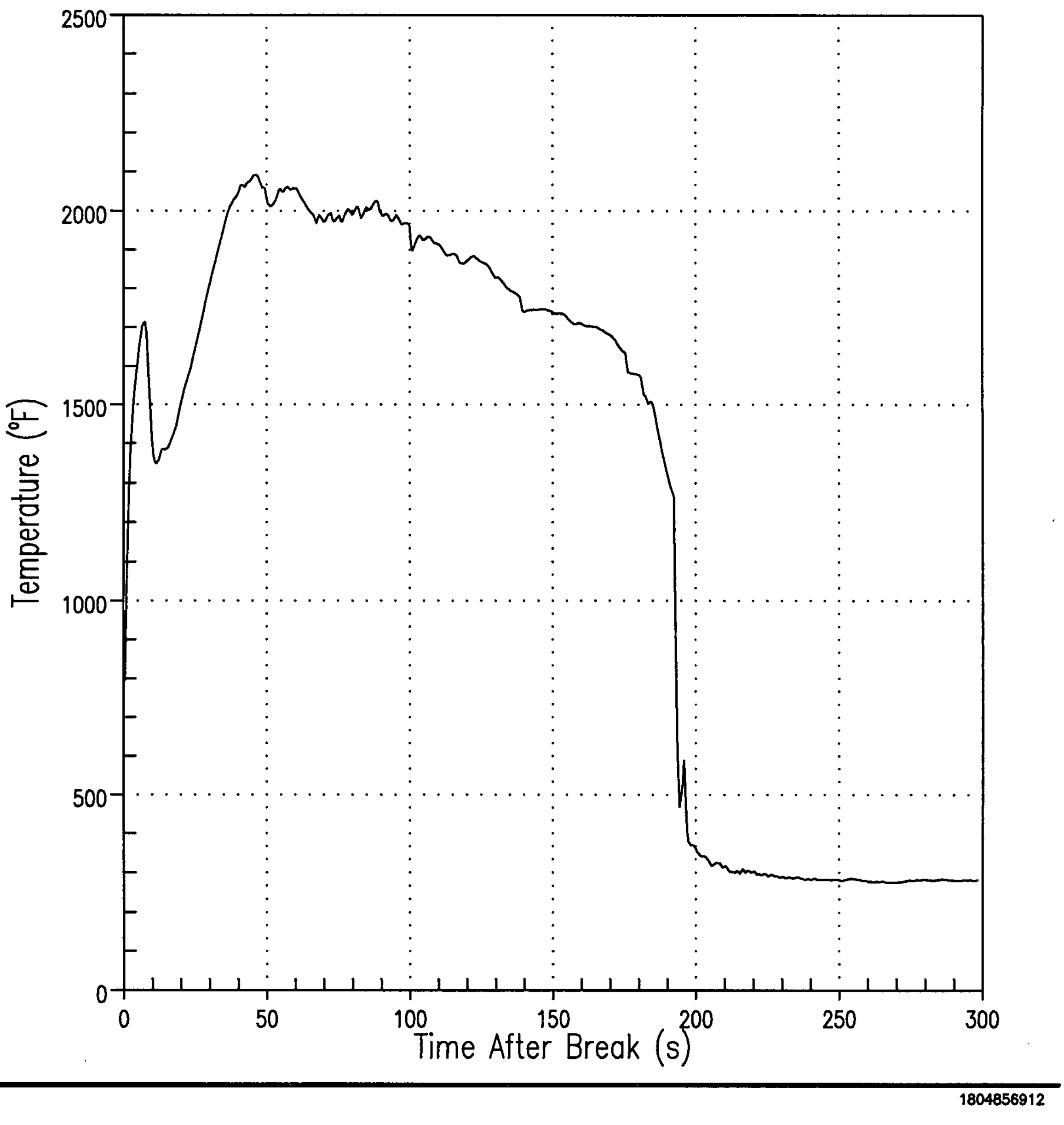

Figure 5-7
PTN WCOBRA/TRAC Hot Rod PCT Transient for Limiting CWO Case

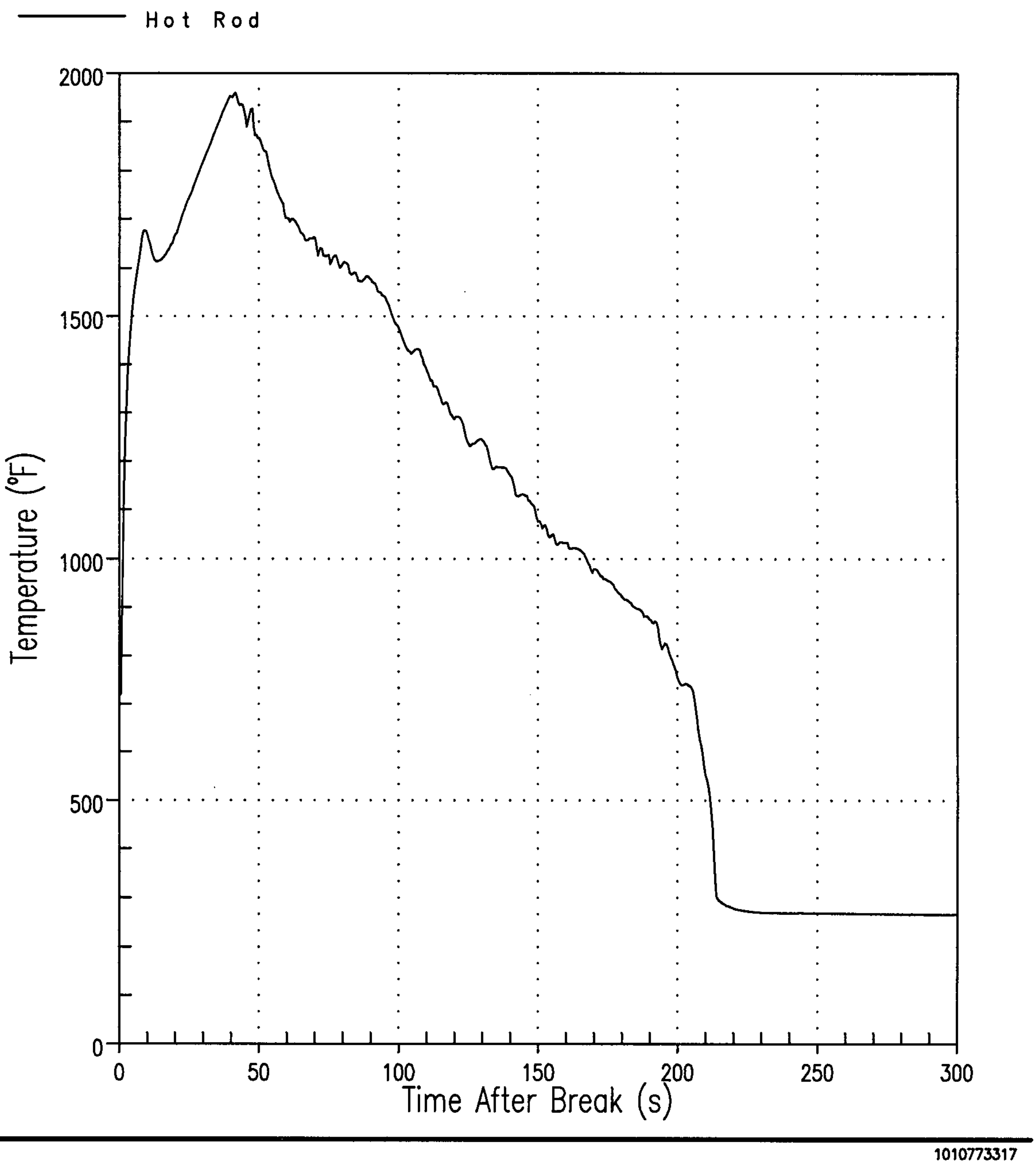

Figure 5-8
PTN Limiting PCT Case Pressurizer Pressure

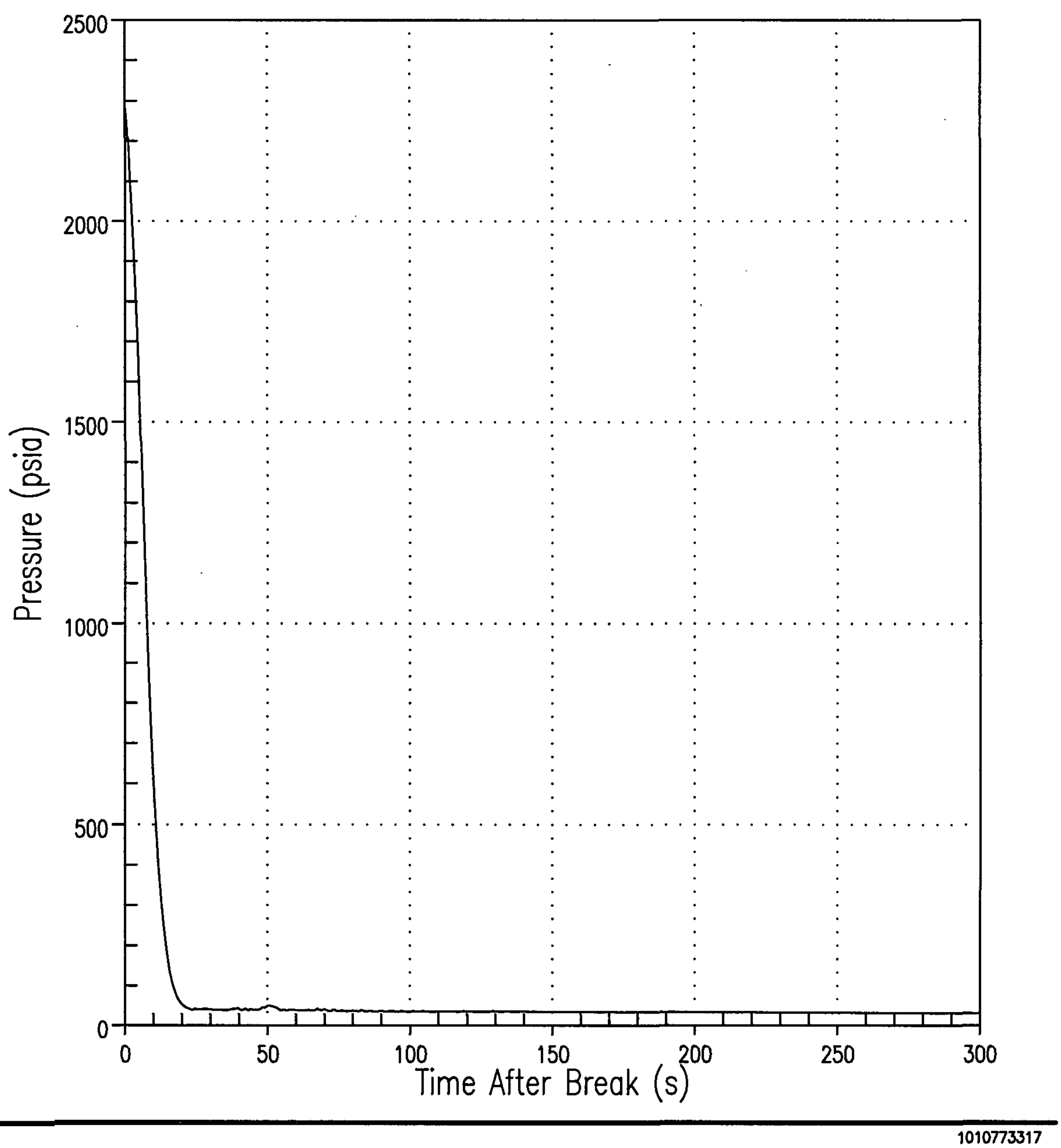

Figure 5-9
PTN Limiting PCT Case Vessel Side Break Flow

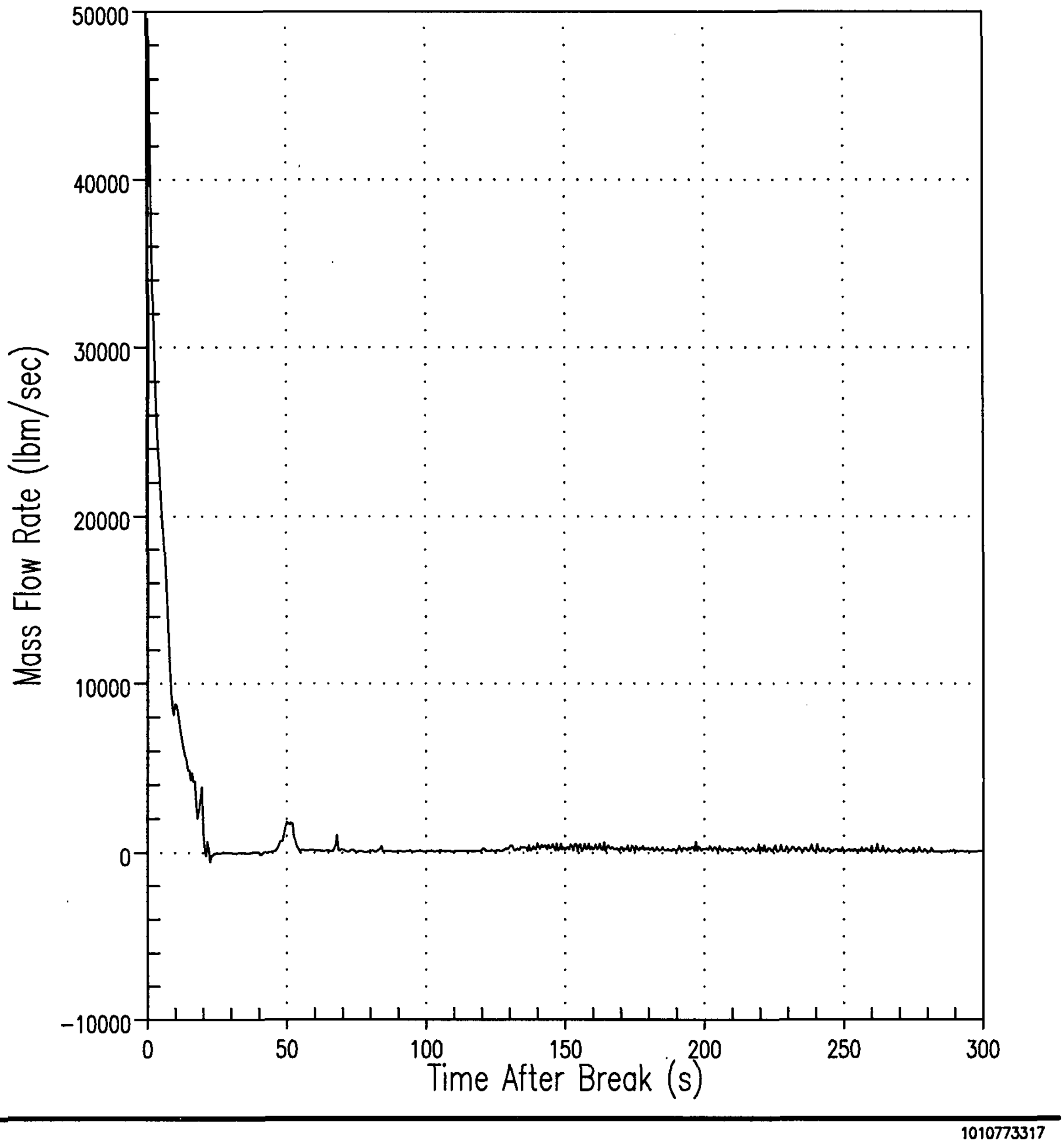

Figure 5-10
PTN Limiting PCT Case Pump Side Break Flow

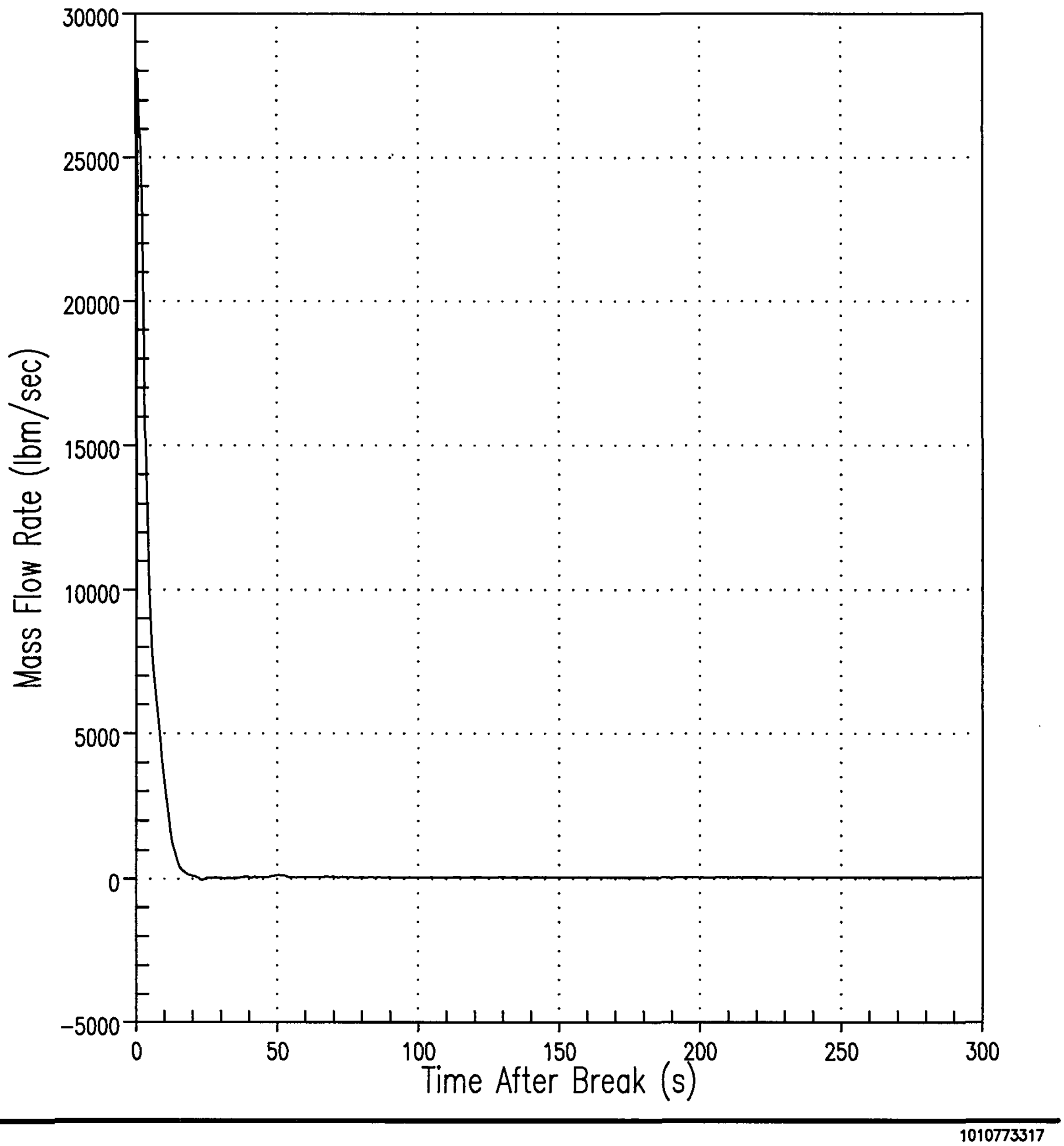

Figure 5-11
PTN Limiting PCT Case Broken and Intact Loop Pump Void Fractions

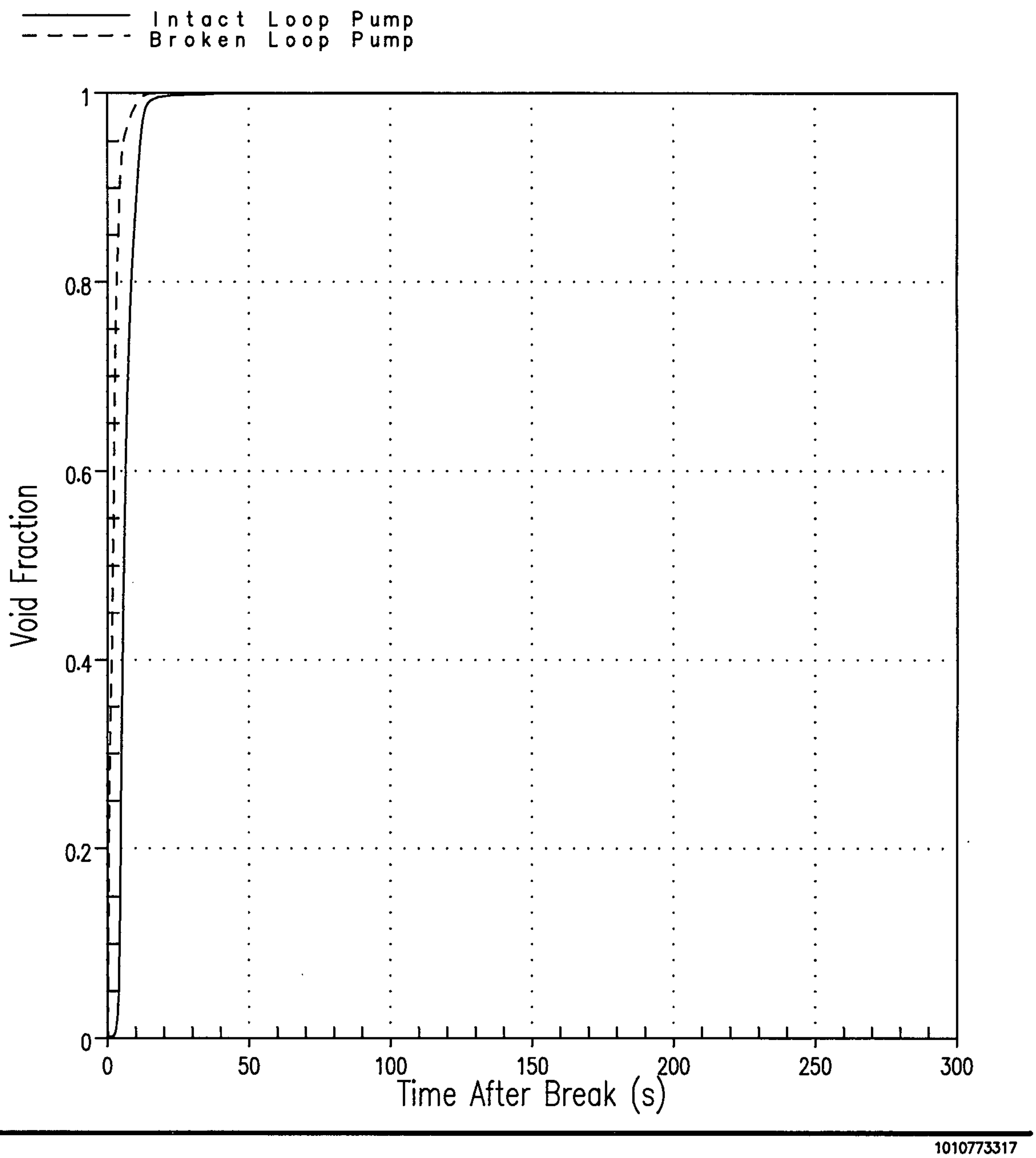

Figure 5-12
PTN Limiting PCT Case Core Vapor Flow near the Top of the Core for the Hot Assembly Channel

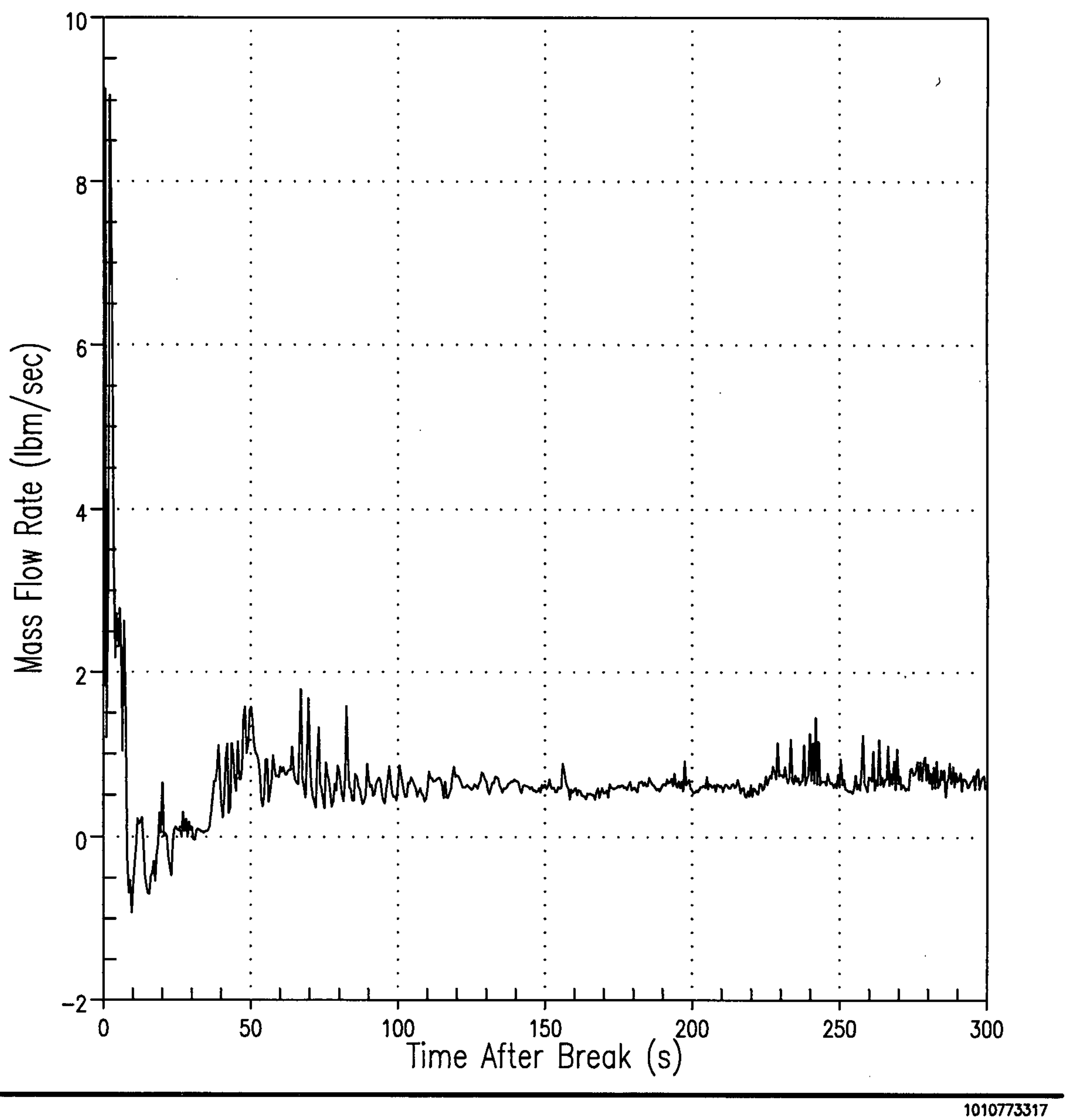

Figure 5-13
PTN Limiting PCT Case Intact Loop Accumulator Flow

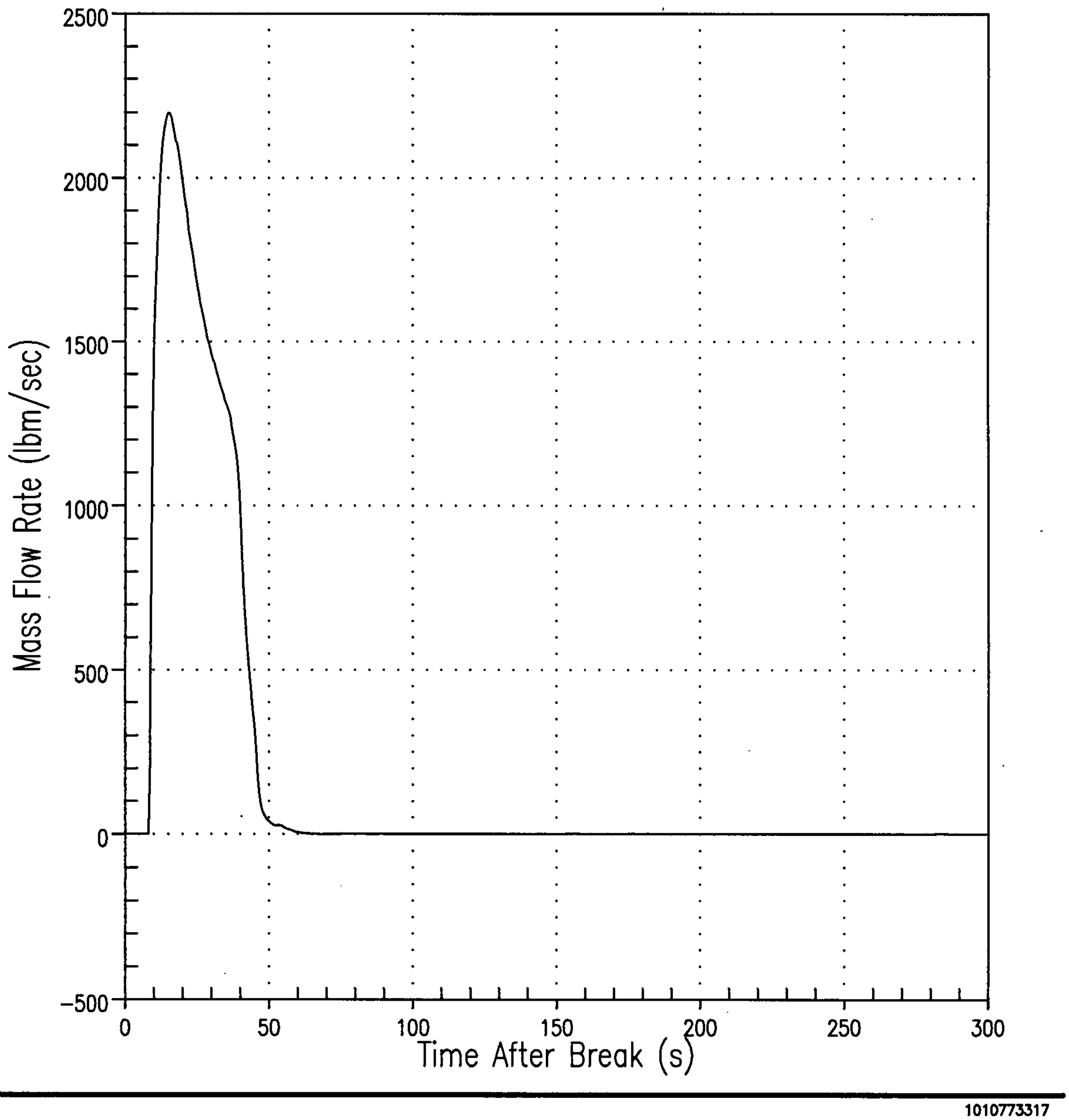

Figure 5-14
PTN Limiting PCT Case Intact Loop Safety Injection Flow

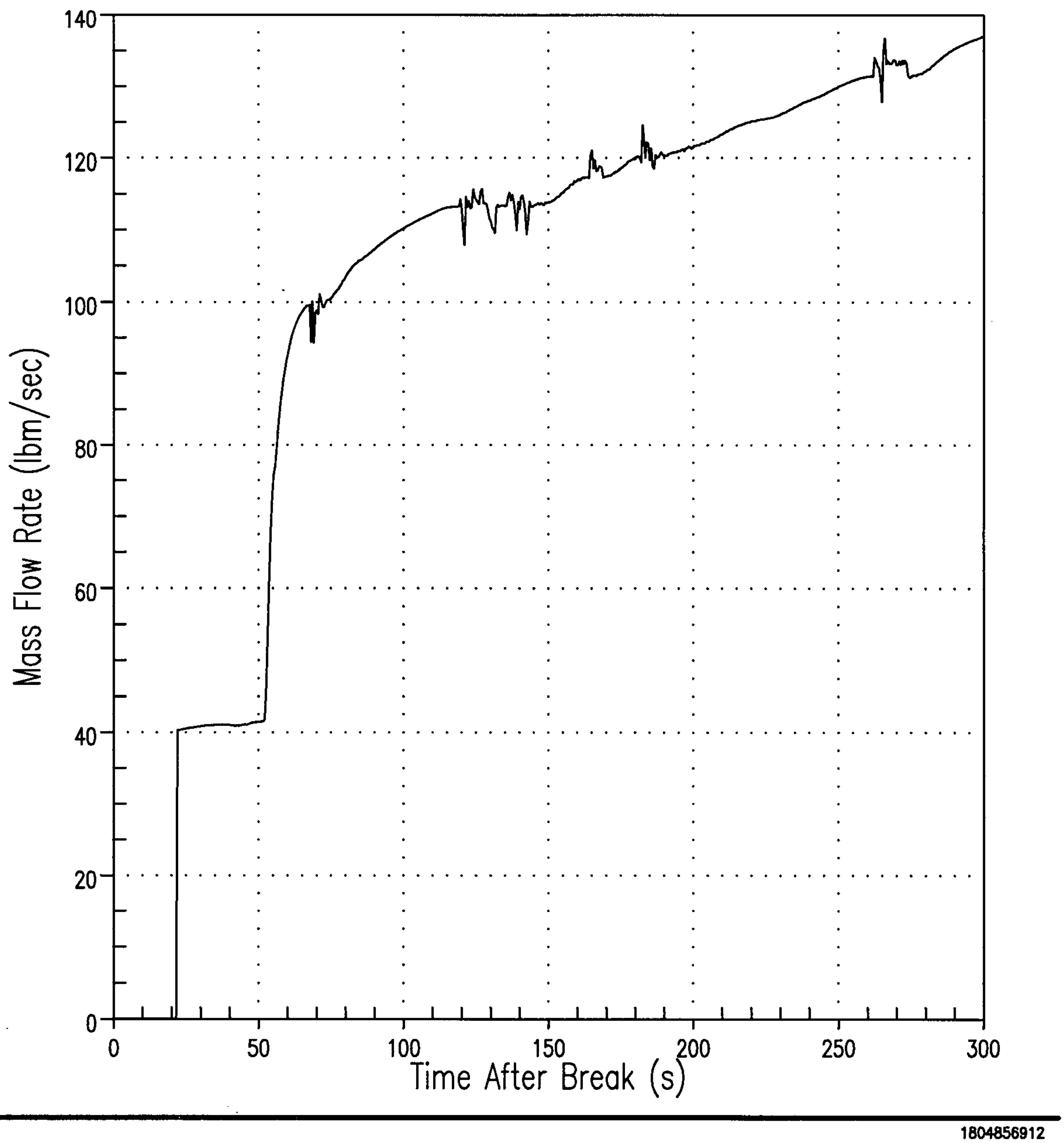

Figure 5-15
PTN Limiting PCT Case Lower Plenum Collapsed Liquid Level

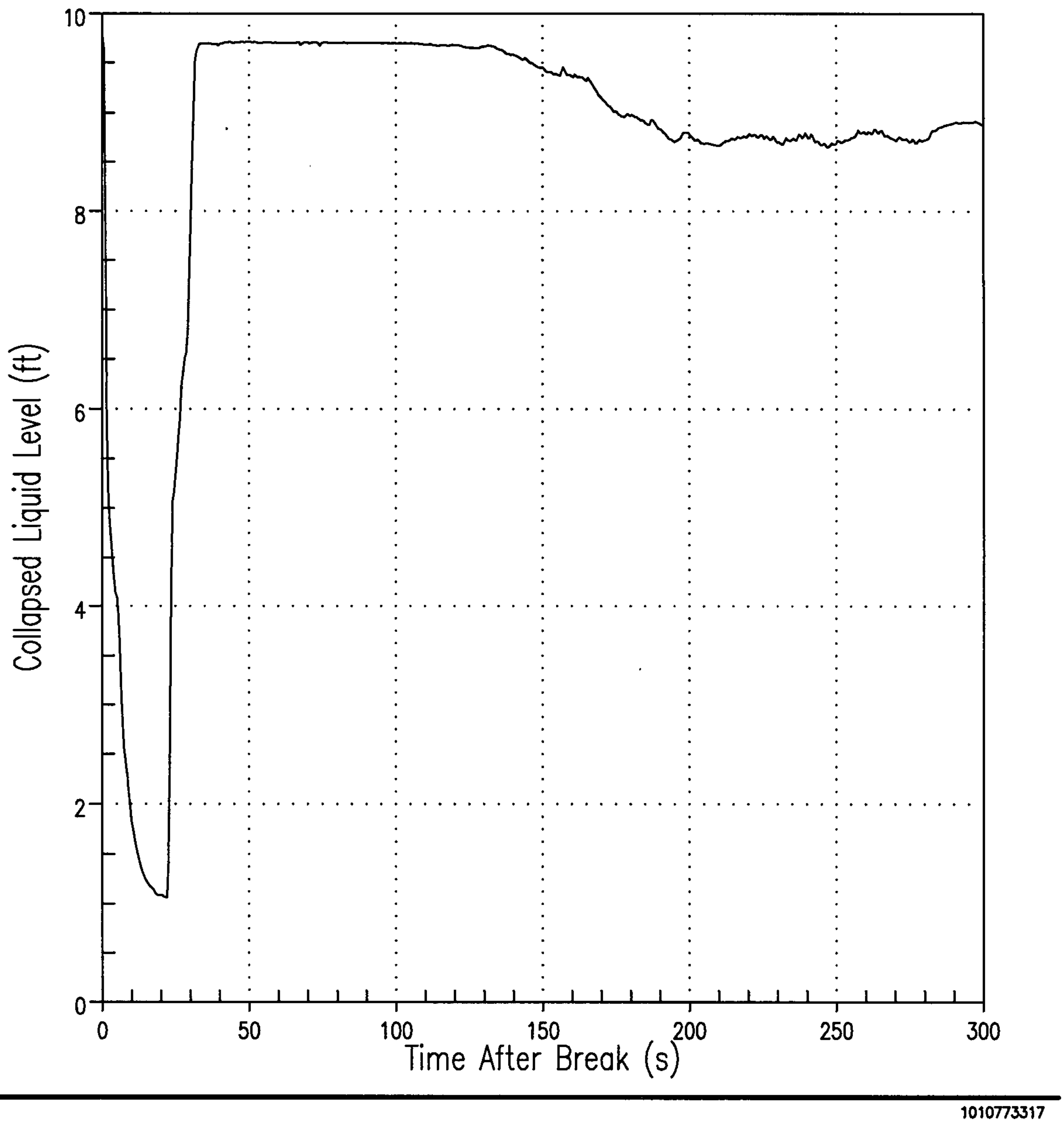

Figure 5-16
PTN Limiting PCT Case Average Downcomer Collapsed Liquid Level

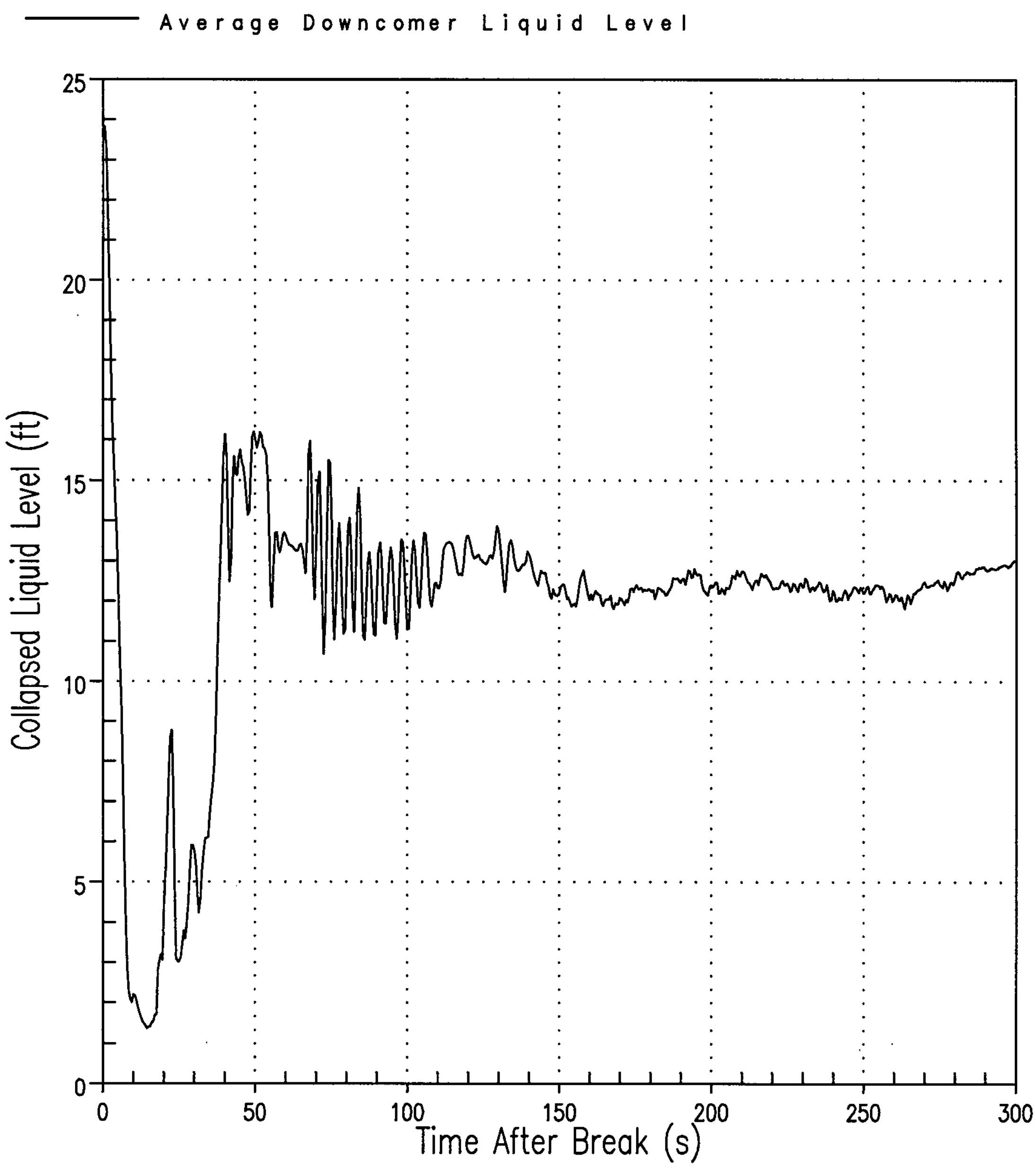

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Figure 5-17
PTN Limiting PCT Case Core Channels Collapsed Liquid Levels

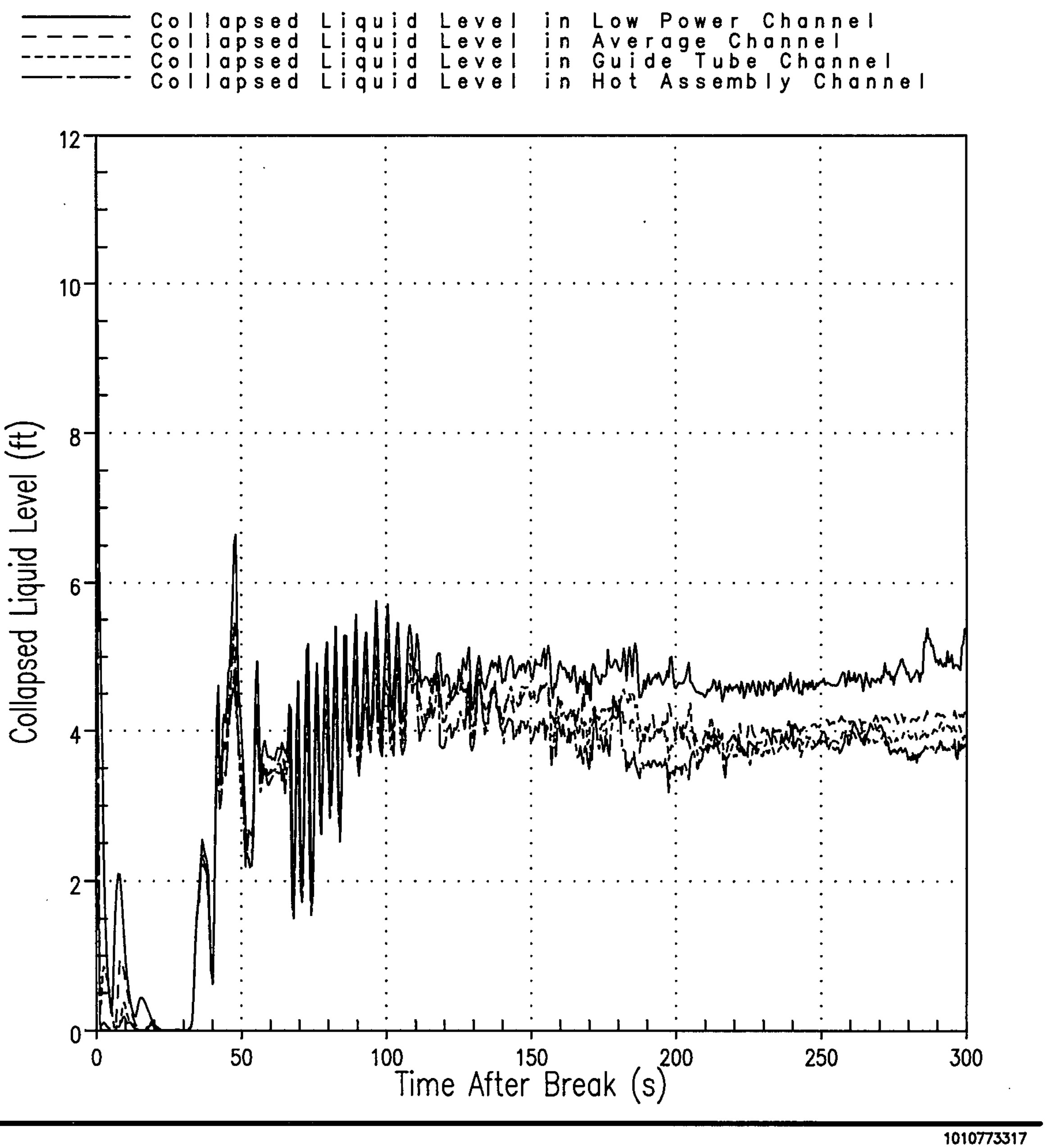


Figure 5-18
PTN Limiting PCT Case Vessel Fluid Mass

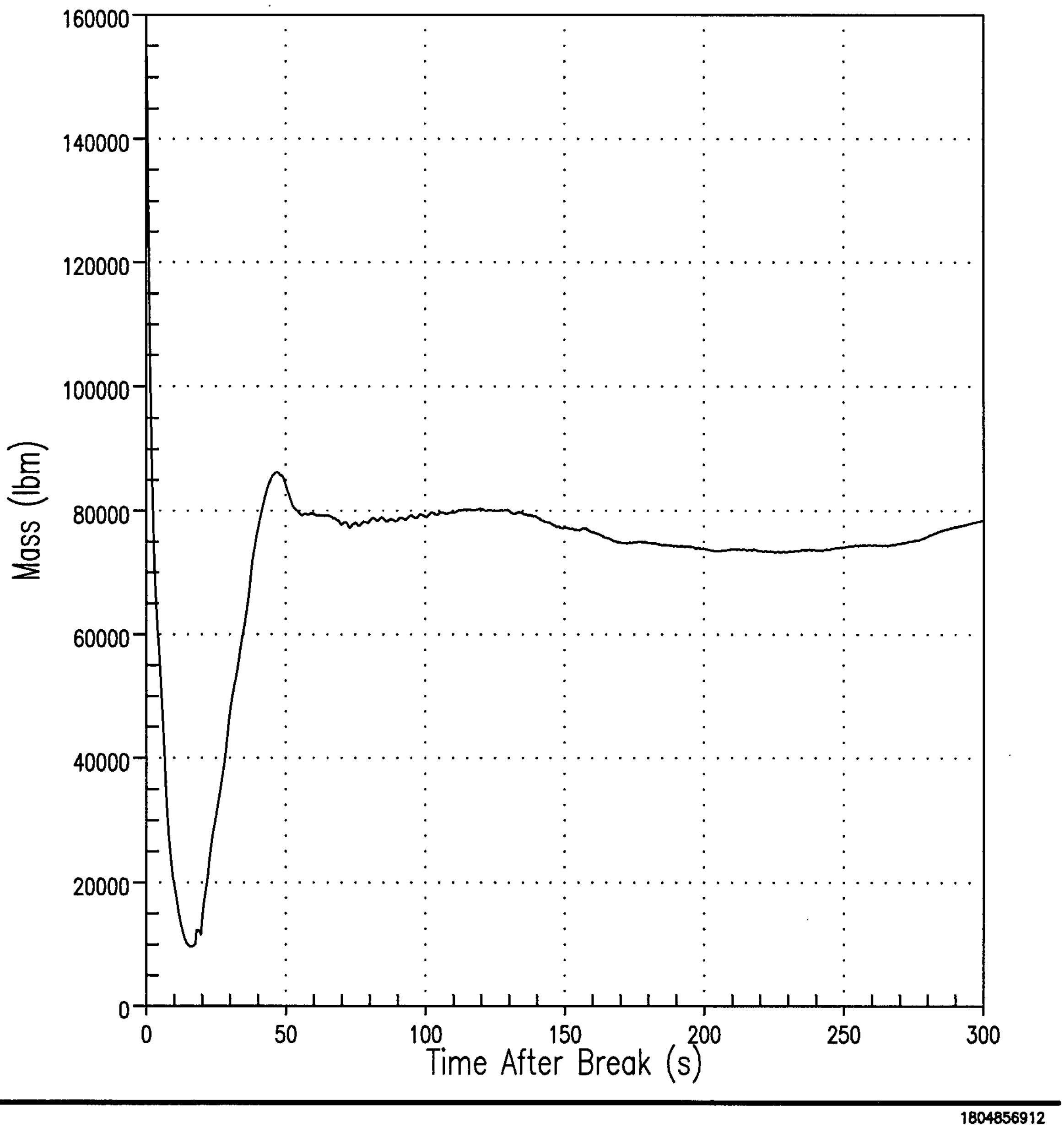


Figure 5-19
PTN Peak Cladding Temperature for all 5 Rods for Limiting PCT Case

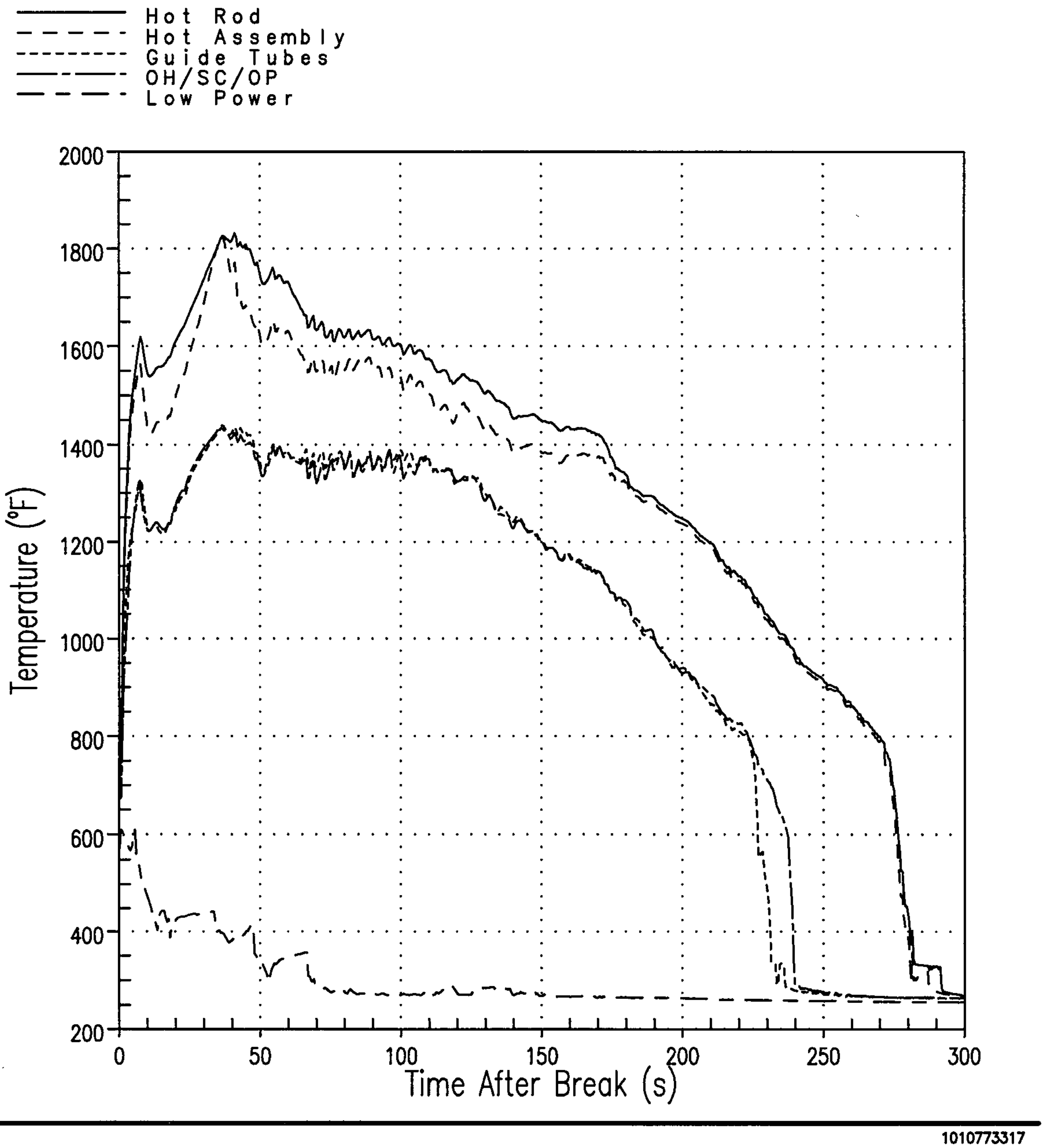


Figure 5-20
PTN PCT Location for Limiting PCT Case

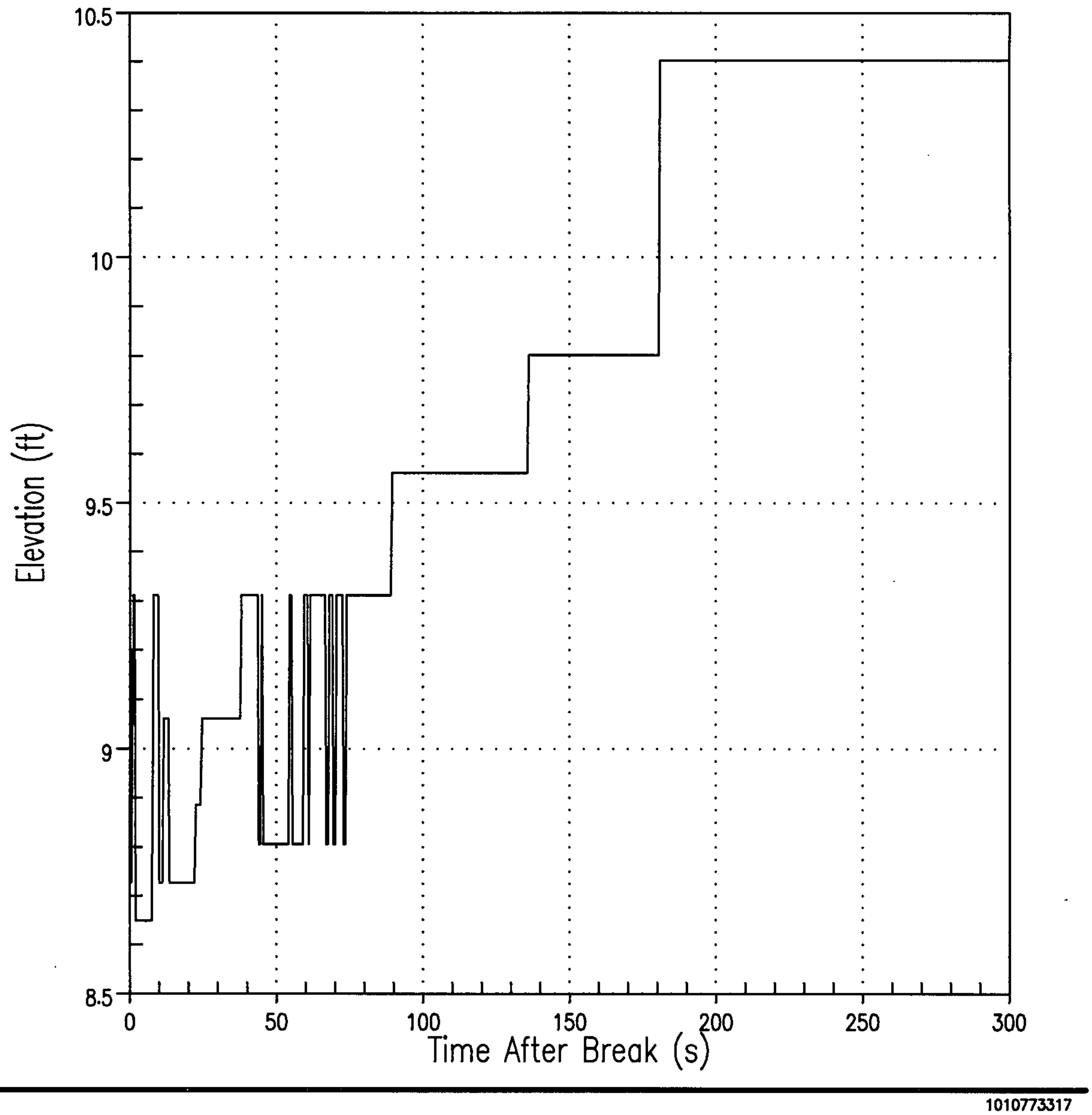


Figure 5-21
PTN HOTSPOT PCT vs. Hot Rod Burnup

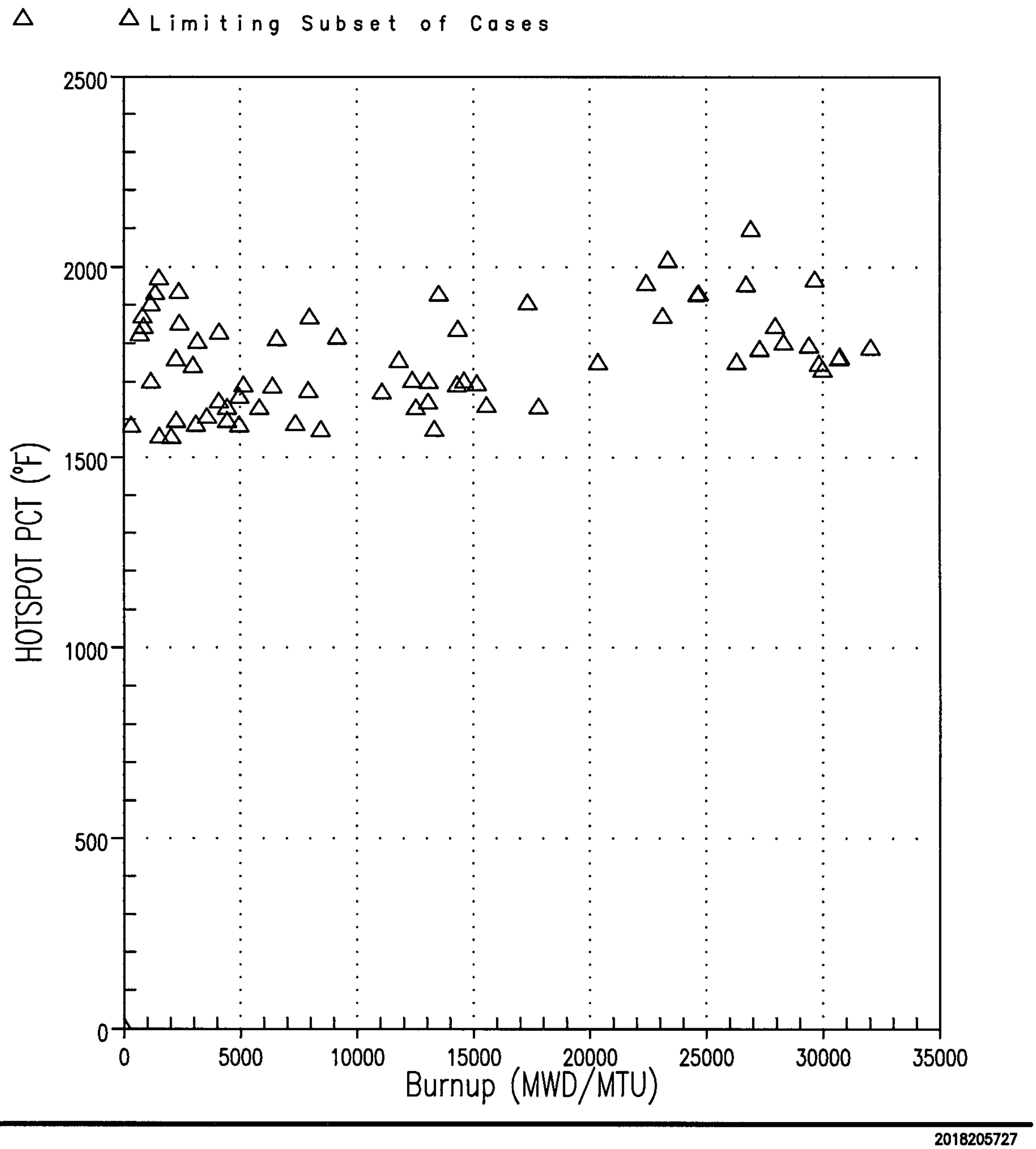
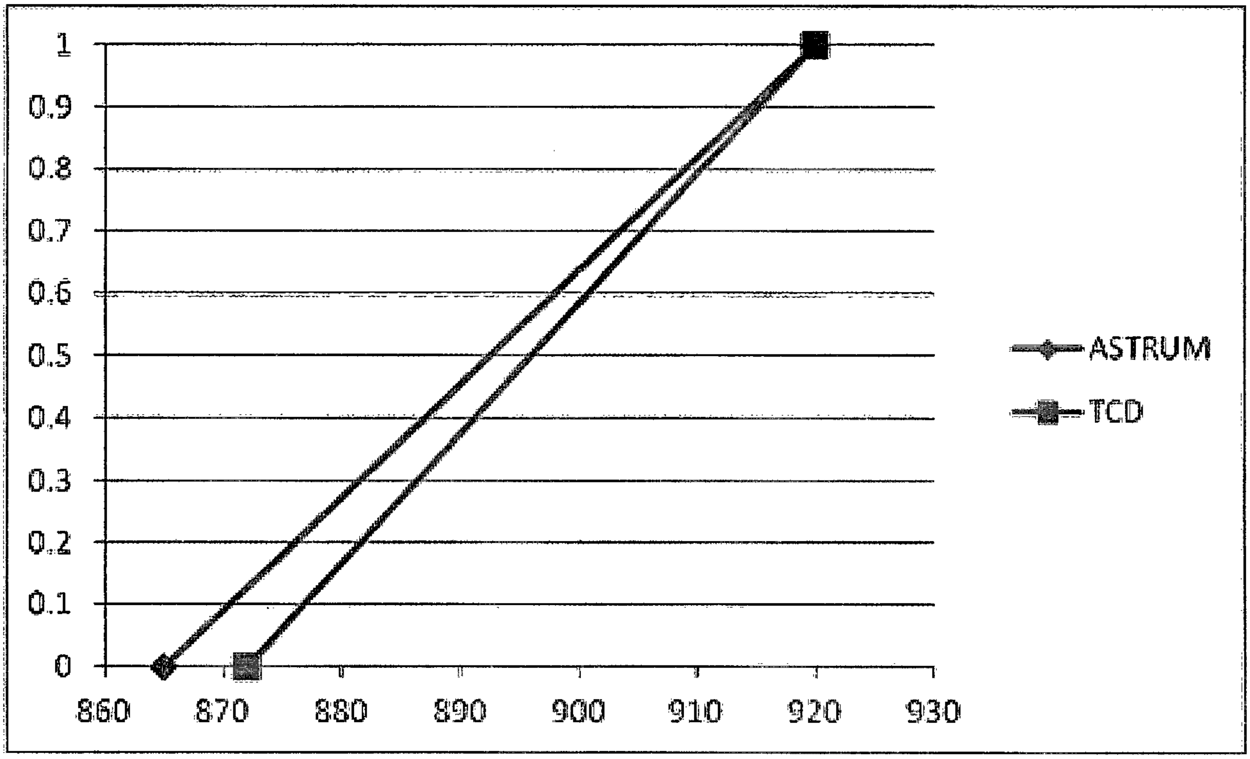


Figure 5-22
Illustration of Input Range Effect on Accumulator Water Volume Sampled Parameter



6.0 Conclusions

The Turkey Point EPU analyses have been revised to incorporate the effect of TCD on the fuel performance predictions as burnup increases during the cycle. Several other changes to plant conditions were made to gain back margin lost due to the negative impact of TCD on the BELOCA analysis. As a result, the maximum large break LOCA PCT increased from 2064°F to 2093°F, but still remains below the 2200°F acceptance criterion, with adequate margin. Some other analytical results did increase from the original EPU results, but there remains margin to the limits in all cases. This includes the peak linear heat rate limit (kw/ft), for which a burndown-dependent limit is being established, and the maximum PCT for the locked rotor event. This PCT was increased from 1824 to 1891°F, still well below the limit of 2375°F. The RCCA ejection results were also affected to some degree, but there still remains sufficient margin to all the safety analysis limits.

7.0 References

1. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request for Extended Power Uprate (LAR 205)," Accession No. ML103560169, October 21, 2010.
2. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-100), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Nuclear Performance and Code Review Issues," May 18, 2011.
3. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-226), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Nuclear Performance and Code Review Issues," July 7, 2011.
4. WCAP-15063-P-A, Revision 1 with Errata, Foster J. P., et al., Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), July 2000. (Westinghouse Proprietary Class 2).
5. WCAP-10125-P-A, Davidson, S. L., et al., Extended Burnup Evaluation of Westinghouse Fuel, December 1985. (Westinghouse Proprietary Class 2).
6. WCAP-13589-A, Kersting, P. J., et al., Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel, March 1995.
7. WCAP-10125-P-A, Addendum 1-A, Revision 1-A (Proprietary), Bahr, K. E., Extended Burnup Evaluation of Westinghouse Fuel, Revision to Design Criteria, May 2005.
8. WCAP-15836-P-A (Proprietary), Harris, W. R., et al., Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1, April 2006.
9. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005. (Westinghouse Proprietary Class 2).
10. NRC Information Notice 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," December 13, 2011.

Turkey Point Units 3 and 4

RESPONSE TO NRC SRXB RAI REGARDING EPU LAR NO. 205
AND THERMAL CONDUCTIVITY DEGRADATION

ATTACHMENT 3

Westinghouse Affidavit CAW-11-3340 for Attachment 2
December 30, 2011

This coversheet plus 8 pages

Westinghouse

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com
Proj letter: FPL-11-322

CAW-11-3340

December 30, 2011

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: FPL-11-322, P-Attachment, "Turkey Point Units 3 and 4 – Impact of Fuel Thermal Conductivity Degradation on Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 (TAC Nos. ME 4907 and ME 4908)" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-11-3340 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Florida Power and Light.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-11-3340, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 30 day of December 2011

Notary Public

KRISTIN JOY SCHADE
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Kristin Joy Schade, Notary Public
North Huntingdon Twp., Westmoreland County
My Commission Expires March 11, 2013
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in FPL-11-322, P-Attachment, "Turkey Point Units 3 and 4 – Impact of Fuel Thermal Conductivity Degradation on Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 (TAC Nos. ME 4907 and ME 4908)" (Proprietary), for submittal to the Commission, being transmitted by Florid Power and Light letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for use by Turkey Point Units 3 and 4 is expected to be applicable for other licensee submittals in response to certain NRC requirements for Extended Power Uprate (EPU) submittals and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Provide input to the U.S. Nuclear Regulatory Commission for review of the Turkey Point Extended Power Uprate (EPU) submittals.

(b) Provide results/limits for fuel thermal conductivity degradation issue.

(c) Provide licensing support for customer submittal.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of this information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the technology to its customer in the licensing process.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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