February 10, 2012

MEMORANDUM TO: William H. Ruland, Director
Division of Safety Systems
Office of Nuclear Reactor Regulation

FROM: Paul M. Clifford, Senior Technical Advisor ***/RA/***
Division of Safety Systems
Office of Nuclear Reactor Regulation

SUBJECT: ECCS PERFORMANCE SAFETY ASSESSMENT AND AUDIT REPORT

This memorandum is the non-proprietary, publically available version of the original ECCS safety assessment and audit report dated September 27, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11262A017).

In response to the research findings in Research Information Letter (RIL) 0801, "Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46," (ADAMS Accession number, ML081350225), the staff performed a preliminary safety assessment of currently operating reactors (ADAMS Accession number ML081620302 Proprietary, ML090340073 Non-Proprietary). This assessment found that, due to measured cladding performance under loss-of-coolant accident (LOCA) conditions, realistic fuel rod power history, and current analytical conservatisms, sufficient safety margin exists for operating reactors. Therefore, the NRC staff determined that immediate regulatory action was not required, and that changes to the emergency core cooling system (ECCS) acceptance criteria to account for these new findings can reasonably be addressed through the rulemaking process.

Recognizing that finalization and implementation of the new ECCS requirements would take several years, the staff decided that a more detailed safety assessment was necessary. The enclosed ECCS performance safety assessment confirms, on a plant-specific basis, the safe operation of the U.S. commercial nuclear fleet. The enclosed report also captures the staff's audit report of the PWR Owners Group (ADAMS Accession number ML11139A3090) and BWR Owners Group (ADAMS Accession number ML1119501390) ECCS margin assessment reports.

Enclosure:
As stated

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ECCS Performance Safety Assessment

1. Scope and Purpose

In response to the research findings in Research Information Letter (RIL) 0801, "Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46," (ADAMS Accession number, ML081350225), the NRC performed a preliminary safety assessment of currently operating reactors (ADAMS Accession number ML081620302 Proprietary, ML090340073 Non-Proprietary). This assessment found that, due to measured cladding performance under loss-of-coolant accident (LOCA) conditions, realistic fuel rod power history, and current analytical conservatisms, sufficient safety margin exists for operating reactors. Therefore, the NRC staff determined that immediate regulatory action was not required, and that changes to the emergency core cooling system (ECCS) acceptance criteria to account for these new findings can reasonably be addressed through the rulemaking process.

Recognizing that finalization and implementation of the new ECCS requirements would take several years, the staff decided that a more detailed safety assessment was necessary. The purpose of this ECCS performance safety assessment is to confirm, on a plant-specific basis, the safe operation of the U.S. commercial nuclear fleet. This report also recommends a strategy for confirming continued safe operation in the interim until all plants are in compliance with the new 10 CFR 50.46c.

2. Background - Research Findings

In the existing regulations, the preservation of cladding ductility, via compliance with regulatory criteria on peak cladding temperature (PCT) (§ 50.46(b)(1)) and maximum local cladding oxidation (MLO) (§ 50.46(b)(2)), provides a level of assurance that fuel cladding will not experience gross failure and that the fuel rods will remain within their coolable lattice arrays. The recent LOCA research program identified new cladding embrittlement mechanisms which demonstrated that the current combination of peak cladding temperature (2200°F (1204°C)) and local cladding oxidation (17 percent equivalent cladding reacted (ECR)) criteria may not always ensure post quench ductility (PQD). The impact of these research findings on cladding ductility is addressed below.

It is important to recognize that loss of cladding ductility is the result of oxygen diffusion into the base metal and is not directly related to the growth of a zirconium dioxide layer on the cladding outside diameter. Under 10 CFR 50.46, the peak local oxidation limit is used as a surrogate to limit the fuel's time-at-temperature and the associated oxygen diffusion. The recent LOCA research program used the Cathcart-Pawel (CP) weight gain correlation to integrate time-at-temperature at which ductility was lost (nil ductility). This surrogate approach is possible because both oxidation and diffusion share a strong temperature dependency.

<u>Hydrogen-Enhanced Beta-Layer Embrittlement</u>

Section 1.4 of NUREG/CR-6967, "Cladding Embrittlement during Postulated Loss-of-Coolant Accidents" (ADAMS Accession No. ML082130389), explains that oxygen diffusion into the base metal under LOCA conditions promotes a reduction in the size (referred to as beta-layer thinning) and ductility (referred to as beta-layer embrittlement) of the metallurgical structure within the cladding that provides its overall ductility. The presence of hydrogen within the

cladding enhances this embrittlement process. During normal operation, the cladding metal absorbs some hydrogen from the corrosion process. When that cladding is exposed to high-temperature LOCA conditions, the elevated hydrogen levels increase the solubility of oxygen in the beta phase and the rate of diffusion of oxygen into the beta phase. Thus, even for LOCA temperatures below 1,204 degrees C (2,200 degrees F), embrittlement can occur for time periods corresponding to less than 17-percent oxidation in corroded cladding with significant hydrogen pickup.

Oxygen Ingress from the Inside Diameter of the Cladding

Section 1.4.6 of NUREG/CR-6967 explains that oxygen sources may be present on the inner surface of irradiated cladding because of gas-phase uranium trioxide transport before gap closure, fuel-cladding-bond formation (uranium dioxide in solid solution with zirconium dioxide), and the fuel bonded to this layer. Under LOCA conditions, this available oxygen may diffuse into the base metal of the cladding, effectively reducing the integral time-at-temperature to nil ductility.

Except within the burst node, current LOCA methods do not account for inner surface oxygen diffusion and its effects on integral time-at-temperature (ITT) limits. For high-burnup fuel rods, ITT calculations (local oxidation ECR) must consider oxygen diffusion from both the outside and inside surfaces.

Breakaway Oxidation

Section 1.4.5 of NUREG/CR-6967 explains that zirconium dioxide can exist in several crystallographic forms (allotropes). The normal tetragonal oxide that develops under LOCA conditions is dense, adherent, and protective with respect to hydrogen pickup. However, there are conditions that promote a transformation to the monoclinic phase (i.e., the phase that is grown during normal operation), which is neither fully dense nor protective. The tetragonal-to-monoclinic transformation is an instability that initiates at local regions of the metal-oxide interface and grows rapidly throughout the oxide layer. Because this transformation results in an increase in oxidation rate, it is referred to as breakaway oxidation. Along with this increase in oxidation rate resulting from cracks in the monoclinic oxide, significant hydrogen pickup occurs. Hydrogen that enters in this manner during a LOCA transient promotes rapid embrittlement of the cladding.

Table 2-1 lists the measured time to the onset of breakaway oxidation for several domestic alloys. Note that all zirconium alloys will eventually experience breakaway oxide phase transformation. The composition of the cladding alloy and the surface finish influence the timing of this phenomenon. Any fuel rod that experiences breakaway oxidation during a postulated LOCA will rapidly become brittle and more susceptible to gross failure, thereby not satisfying the GDC 35 requirements for coolable core geometry.

Table 2-1: NRC Breakaway Test Results

Alloy	Measured Breakaway Time
Zircaloy-2	>5,000 seconds
Zircaloy-4	5,000 seconds
ZIRLO^{TM}	3,000* seconds
M5	>5,000 seconds

3. ECCS Performance Safety Assessment

In summary, the PWROG and BWROG ECCS margin assessment reports provided the following information and analyses:

1. Starting with the staff's "Elements of Prospective Information Request" handout from a public workshop on April 29, 2010 (ADAMS ML1013004901 and ML1011800120), developed revised analytical limits for PQD and breakaway oxidation for which to judge ECCS performance.
    a. Alloy-specific corrosion and hydrogen uptake properties were used to convert allowable CP-ECR to a function of fuel rod exposure.
2. Collected and tabulated existing LOCA analysis-of-record (AOR) results for all operating plants.
3. Grouped plants based on margin to revised analytical limits, plant design, cladding alloy, and/or evaluation model.
4. Identified plants which satisfy revised analytical limits with no adjustments.
5. For remaining plants, identified conservatisms with regard to assumptions, analysis inputs, methodology, risk assessments and relevant research data that represent inherent margin.
    a. Quantified and justified any analytical credit.
    b. Where necessary, performed new LOCA analyses.
6. Documented ECCS margin assessment for the limiting plant within each group.

As part of the audit of the Owner's Group reports, the staff completed the following activities:

1. Confirmed that the revised PQD and breakaway analytical limits were in accordance with the research findings and that alloy-specific corrosion and hydrogen uptake models were accurate and supported by data.
2. Evaluated the quantification, justification, and application of analytical credits.
3. Reviewed a sampling of the new LOCA calculations and identified any changes to existing, approved models and methods. All of the new LOCA calculations were

* Westinghouse has performed independent breakaway oxidation tests and reported higher breakaway times. See Section 2.1 of the audit report for further information.

performed and documented in accordance with the fuel vendor's 10 CFR 50 Appendix B quality assurance program.

4. Compiled plant-specific data and evaluated each individual plant with respect to margin to the revised analytical limits.

For completeness, the audit report is attached. Due to the amount of information collected and the size of the resulting table, the compilation of plant-specific ECCS margin (referred to as ECCS Margin Database) is not depicted within this document, but instead available in ADAMS ML11262A010. For each plant, the ECCS Margin Database provides the following information:

- Plant design
- Licensed power
- Fuel vendor
- Fuel rod cladding alloy
- Evaluation model
- AOR results (calculated PCT, MLO, and time above 800°C)
- Plant grouping
- Margin to PQD analytical limit
- Margin to breakaway oxidation analytical limit
- Identify analytical credit(s)

Examination of the ECCS Margin Database reveals that the majority of plants needed no adjustments to show a positive margin to the revised analytical limits. In summary:

<u>Revised PQD Analytical Limit:</u>

- 65 of 104 plants (63% of entire operating fleet) needed no adjustment or new calculations.
    - 27 of 35 BWRs (77% of BWR fleet)
    - 38 of 69 PWRs (55% of PWR fleet).
- Remaining 8 BWRs performed new LOCA calculations which credit COLR Thermal-Mechanical Operating Limits (TMOL) reduced rod power at higher burnup to satisfy new analytical limits.
- Remaining 31 PWRs either performed new LOCA calculations or identified credits to satisfy new analytical limits.
    - 9 PWRs performed new LOCA calculations which credit diminished fuel rod power at higher burnup.
    - 11 PWRs credit transition to improved evaluation models (e.g., ASTRUM LBLOCA or ANS 1979+2σ decay heat SBLOCA).
    - 4 PWRs credit improved statistics in ASTRUM methods.
    - 7 PWRs credited multiple items.

<u>New Breakaway Oxidation Analytical Limit:</u>

- All 104 plants needed no adjustments or new calculations.
    - Only 1 plant had a time duration above 2000 seconds.

Figure 3-1 illustrates the distribution of calculated ECR from the UFSAR AORs. Note that for PWRs, the maximum calculated ECR between LBLOCA and SBLOCA is shown. Figure 3-2 illustrates the distribution of calculated time above 800°C

Figure 3-1: Survey of Calculated Maximum Local Oxidation

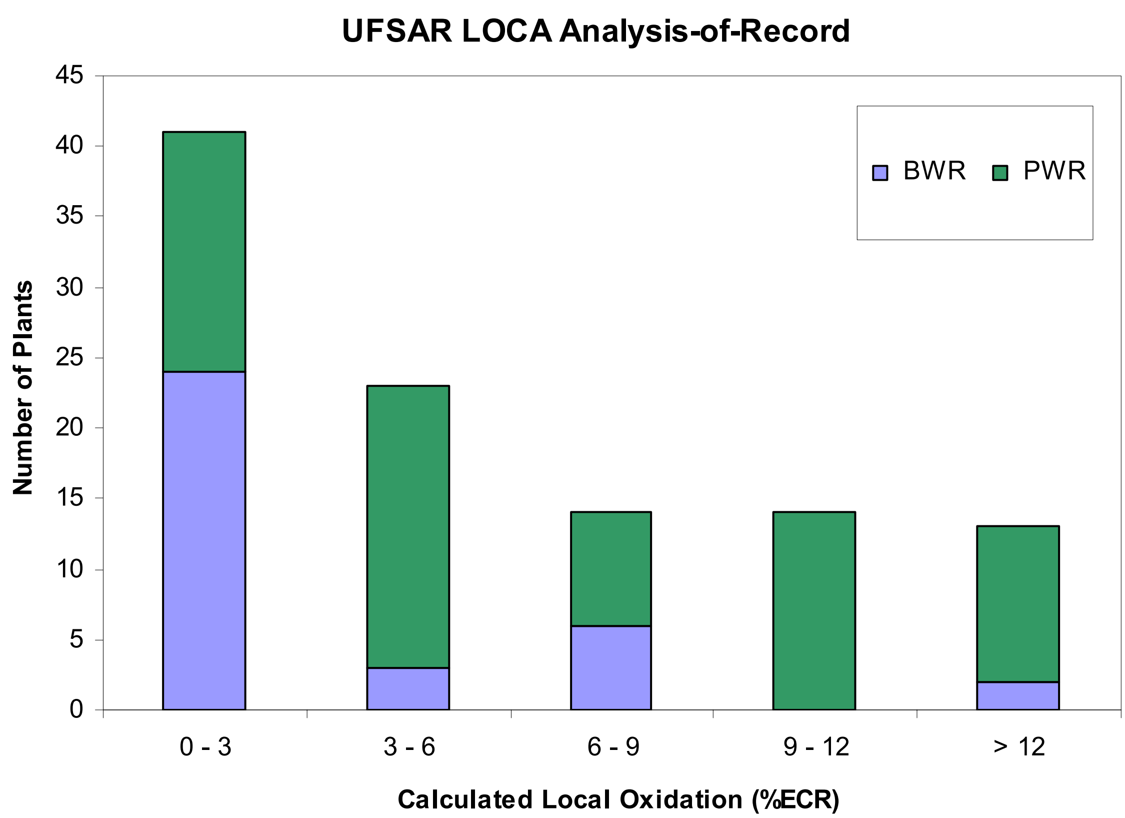
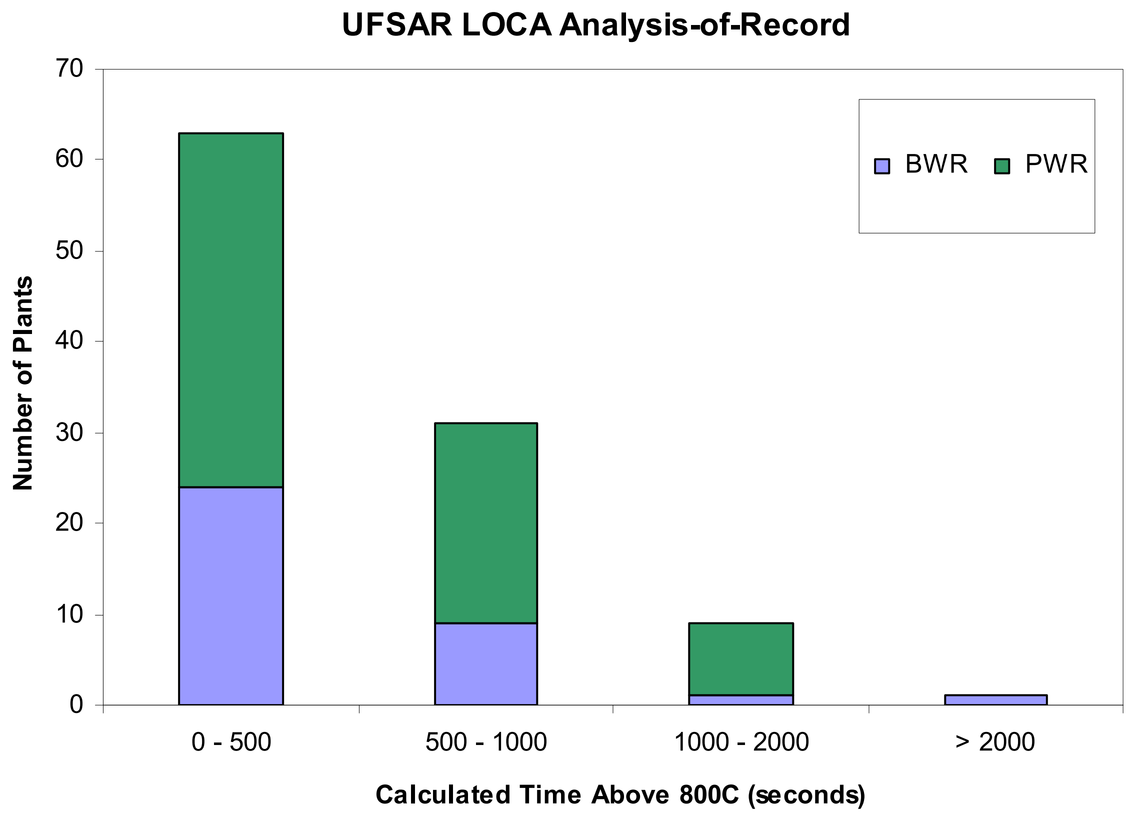


Figure 3-2: Survey of Calculated Time Above 800°C



3.1 Safety Assessment of New Reactors

The NRC staff's initial safety assessment for RIL-0801 and the subsequent ECCS margin assessment reports prepared by the industry focused on the current U.S. operating fleet of PWRs and BWRs. During the audit, the staff inquired about the anticipated future startup of Watts Bar Unit 2 (Westinghouse 4-Loop PWR, 3411 MWt) and Bellefonte Units 1 and 2 (B&W PWR, 3763 MWt). As documented in the attached audit report, the Watts Bar Unit 2 LOCA analyses have been completed and demonstrate sufficient margin to the revised PQD and breakaway analytical limits. The Bellefonte Cycle 1 safety analyses have not been completed. However, based on anticipated ECCS performance (relative to similar plants) and the use of M5 alloy cladding, these units are not expected to be susceptible to either PQD or breakaway oxidation. The NRC staff will confirm that the Bellefonte plants will begin operation with sufficient margin to the revised analytical limits once the Cycle 1 LOCA analyses are complete.

The NRC is currently reviewing, or has approved, various new standardized reactor designs. Although outside the scope of the NRC's original safety assessment and the industry's ECCS margin reports, it is important to address the future startup and operation of these new designs. Table 3-1 lists the predicted PCT and MLO for four of the standard designs. The information provided in the table has been taken from the Design Certification Documents submitted for each design and has undergone no adjustments or credits as used in some cases for the operating fleet. Examination of Table 3-1 reveals significant margin to the research data and likely analytical limits for their modern cladding alloys (PQD = 2200 F, greater than 10%ECR, breakaway < 5,000 seconds above 800ºC).

Table 3-1: ECCS Performance of New Reactor Designs

Design	PCT (ºF)	ECR (%)
ESBWR	No uncovery or heatup	
AP1000	1837	2.25
EPR	1695	1.53
US-APWR	1766	3.70

4. Future Confirmation of Plant Safety

The BWROG and PWROG margin assessment reports and audit findings represent a snapshot in time. Moving forward, it is important that the safe operation of the fleet is continuously monitored and assessed.

Recognizing that planned changes to plant systems or fuel design which may impact the LOCA calculations and margin assessment would likely necessitate a license amendment request (LAR) and that unplanned changes would be captured via existing 50.46(a)(3) reporting requirements, the staff proposes the following actions to confirm plant safety in the interim until the revised rule (10 CFR 50.46c) is implemented.

- On an annual basis, the staff will update the ECCS Margin Database using the annual licensee 50.46(a)(3) reports.
- On a continuous basis, the staff will scrutinize any License Amendment Request (LAR) which necessitates a change to the LOCA analysis-of-record and may impact the existing margin assessment.
- On a continuous basis, the staff will scrutinize all 30-day significant 50.46(a)(3) reports to confirm existing margin assessment.
- As part of the annual vendor/NRC fuel update meetings, the staff will confirm that all changes which may impact the existing margin assessment have been identified and discuss future LARs which may impact the LOCA analysis-of-record.

5. Implementation Plan for 10 CFR 50.46c Rule

To date, the staff has been considering three possible implementation strategies for 10 CFR 50.46c:

1. Expedite implementation whereby all plants shall be in compliance on a specified date (e.g., 24 months from date of rule).
2. Staged implementation plan whereby plants with the least available safety margin would be required to be in compliance earliest.
3. Staged implementation plan which balances the above two goals.

Recognizing that plants with the least amount of safety margin are likely to require the most effort and calendar time to document compliance with the new requirements and that work-force limitations exist in the industry to develop new models and methods and complete plant-specific analyses, license amendment requests, and UFSAR updates, and within the NRC to review new models and methods and all of the license amendment requests, a staged implementation approach would be the most effective and efficient way to implement 10 CFR 50.46c.

Based upon the ECCS margin assessment, plants may be divided into two categories: (1) those which satisfy the new PQD and breakaway analytical limits without new analysis and (2) those which require model changes and/or new calculations. To balance work load, the second group may be further sub-divided based on available safety margin. The following implementation plan achieves the objectives of (1) expediting implementation to as many plants as soon as possible, (2) prioritizing implementation on plants with less inherent safety margin, and (3) balancing work load.

A review of the ECCS Margin Database reveals that 27 BWR plants and 38 PWR plants required no new analyses to satisfy the new analytical limits. These plants should require minimal effort to demonstrate compliance with the proposed 50.46c. As such, these 65 plants will be designated within implementation track #1.

In general, PWR plants employing realistic models have less inherent safety margin than PWR plants employing Appendix K models. As such, PWR plants currently employing realistic models and requiring new analyses to satisfy the new analytical limits will be designated within implementation track #2. The PWR plants currently using Appendix K models and requiring new analyses to satisfy the new analytical limits will be designated within implementation track #3.

BWR/2 plants exhibit lower margin to the new PQD and breakaway analytical limits than BWR/3 plants. As such, BWR/2 plants will be designated within implementation track #2; whereas, BWR/3 plants will be in implementation track #3.

At first glance it appears that track #3 (23 plants) is overloaded from a workload management perspective relative to track #2 (16 plants). However, when you consider the number of multiple unit sites which rely on a single analysis-of-record, the level of effort to prepare and review the new analysis for each track is almost even.

Table 5-1 summarizes the 50.46c implementation plan. Track 5-2 identifies the plants within each track.

Table 5-1: 50.46c Implementation Plan

Implementation Track	Basis	Anticipated Level of Effort	Number of Plants BWR	Number of Plants PWR	Compliance Demonstration
1	All plants which satisfy new requirements without new analyses or model revisions[1].	Low	27	38	No later than 24 months from effective date of rule
2	PWR plants using realistic LBLOCA models requiring new analyses[2]. BWR/2 plants.	Medium	2	14	No later than 48 months from effective date of rule
3	PWR plants using Appendix K LB and SB models requiring new analyses[3]. BWR/3 plants.	Medium - High	6	17	No later than 60 months from effective date of rule

Table Notes:

[1] Recognizes that integral time-at-temperature will need to be re-calculated with C-P correlation for consistency with Regulatory Guide PQD analytical limits and that this may necessitate changes to the model. Furthermore, an approved alloy-specific hydrogen uptake model will be required to implement the new PQD analytical limits.

[2] Comprised on plants within LBLOCA Groups 2, 4, 5, and 6 in PWROG report.

[3] Comprised on plants within LBLOCA Groups 3 and 7 and 3 plants within SBLOCA Group 2 not already included in implementation track 2 (LBLOCA realistic) in PWROG report.

Table 5-2: 50.46c Implementation Track Assignments

Track	Reactor Type	Plant Name	Compliance Demonstration
1	PWR	Arkansas Nuclear One - Unit 1	No later than 24 months from effective date of rule
		Braidwood Station – Unit 1	
		Byron Station – Unit 1	
		Calvert Cliffs Nuclear Power Plant – Unit 1	
		Calvert Cliffs Nuclear Power Plant – Unit 2	
		Comanche Peak Nuclear Power Plant – Unit 1	
		Comanche Peak Nuclear Power Plant – Unit 2	
		Crystal River Nuclear Generating Plant – Unit 3	
		Davis-Besse Nuclear Power Station – Unit 1	
		Diablo Canyon Power Plant – Unit 2	
		Fort Calhoun Station – Unit 1	
		H.B. Robinson Steam Electric Plant - Unit 2	
		Indian Point Nuclear Generating – Unit 2	
		J.M. Farley Nuclear Plant – Unit 1	
		J.M. Farley Nuclear Plant – Unit 2	
		Millstone Power Station – Unit 2	
		Millstone Power Station – Unit 3	
		North Anna Power Station – Unit 1	
		North Anna Power Station – Unit 2	
		Oconee Nuclear Station – Unit 1	
		Oconee Nuclear Station – Unit 2	
		Oconee Nuclear Station – Unit 3	
		Palisades Nuclear Plant	
		Point Beach Nuclear Plant – Unit 1	

Track	Reactor Type	Plant Name	Compliance Demonstration
		Point Beach Nuclear Plant – Unit 2	
		Prairie Island Nuclear Generating Plant – Unit 1	
		Prairie Island Nuclear Generating Plant – Unit 2	
		R.E. Ginna Nuclear Power Plant	
		Saint Lucie Plant – Unit 1	
		Seabrook Station – Unit 1	
		Sequoyah Nuclear Plant – Unit 1	
		Sequoyah Nuclear Plant – Unit 2	
		Three Mile Island – Unit 1	
		Turkey Point Nuclear Generating – Unit 3	
		Turkey Point Nuclear Generating – Unit 4	
		Vogtle Electric Generating Plant – Unit 1	
		Vogtle Electric Generating Plant – Unit 2	
		Wolf Creek Generating Station – Unit 1	
	BWR	Browns Ferry Nuclear Plant – Unit 1	
		Browns Ferry Nuclear Plant – Unit 2	
		Browns Ferry Nuclear Plant – Unit 3	
		Brunswick Steam Electric Plant – Unit 1	
		Brunswick Steam Electric Plant – Unit 2	
		Clinton Power Station – Unit 1	
		Columbia Generating Station	
		Cooper Nuclear Station	
		Duane Arnold Energy Center	
		E.I. Hatch Nuclear Plant – Unit 1	
		E.I. Hatch Nuclear Plant – Unit 2	
		Fermi – Unit 2	

Track	Reactor Type	Plant Name	Compliance Demonstration
		Hope Creek Generating Station – Unit 1	
		Grand Gulf Nuclear Station – Unit 1	
		J.A. Fitzpatrick Nuclear Power Plant	
		LaSalle County Station – Unit 1	
		LaSalle County Station – Unit 2	
		Limerick Generating Station – Unit 1	
		Limerick Generating Station – Unit 2	
		Nine Mile Point Nuclear Station – Unit 2	
		Peach Bottom Atomic Power Station – Unit 2	
		Peach Bottom Atomic Power Station – Unit 3	
		Perry Nuclear Power Plant – Unit 1	
		River Bend Station – Unit 1	
		Susquehanna Steam Electric Station – Unit 1	
		Susquehanna Steam Electric Station – Unit 2	
		Vermont Yankee Nuclear Power Station	
2	PWR	Beaver Valley Power Station – Unit 1	No later than 48 months from effective date of rule
		Beaver Valley Power Station – Unit 2	
		Braidwood Station – Unit 2	
		Byron Station – Unit 2	
		Catawba Nuclear Station – Unit 1	
		Catawba Nuclear Station – Unit 2	
		D.C. Cook Nuclear Plant – Unit 1	
		D.C. Cook Nuclear Plant – Unit 2	
		Diablo Canyon Power Plant – Unit 1	
		Indian Point Nuclear Generating – Unit 3	
		Kewaunee Power Station	

Track	Reactor Type	Plant Name	Compliance Demonstration
		McGuire Nuclear Station – Unit 1	
		McGuire Nuclear Station – Unit 2	
		Watts Bar Nuclear Plant – Unit 1	
	BWR	Nine Mile Point Nuclear Station – Unit 1	
		Oyster Creek Nuclear Generating Station	
3	PWR	Arkansas Nuclear One - Unit 2	No later than 60 months from effective date of rule
		Callaway Plant – Unit 1	
		Palo Verde Nuclear Generating Station – Unit 1	
		Palo Verde Nuclear Generating Station – Unit 2	
		Palo Verde Nuclear Generating Station – Unit 2	
		Saint Lucie Plant – Unit 2	
		Salem Nuclear Generating Station – Unit 1	
		Salem Nuclear Generating Station – Unit 2	
		San Onfre Nuclear Generating Station – Unit 2	
		San Onfre Nuclear Generating Station – Unit 3	
		Shearon Harris Nuclear Power Plant – Unit 1	
		South Texas Project – Unit 1	
		South Texas Project – Unit 2	
		Surry Power Plant – Unit 1	
		Surry Power Plant – Unit 2	
		V.C. Summer Nuclear Station – Unit 1	
		Waterford Steam Electric Station – Unit 3	
	BWR	Dresden Nuclear Power Station – Unit 2	
		Dresden Nuclear Power Station – Unit 3	
		Monticello Nuclear Generating Plant – Unit 1	
		Pilgrim Nuclear Power Station	

Track	Reactor Type	Plant Name	Compliance Demonstration
		Quad Cities Nuclear Power Station, Unit 1 Quad Cities Nuclear Power Station, Unit 2	

6. Conclusion

The LOCA research program identified new cladding embrittlement mechanisms which demonstrated that the current combination of peak cladding temperature (2200°F (1204°C)) and local cladding oxidation (17 percent equivalent cladding reacted (ECR)) criteria may not always ensure fuel rod cladding post quench ductility. Based upon this research, new analytical limits for PQD and breakaway oxidation were developed along with new analytical requirements for cladding inside-diameter oxygen ingress. As an alternative to a proposed Generic Letter, the BWROG and PWROG completed an ECCS margin assessment which concluded that, through the use of existing conservatisms and analytical credits, each operating reactor has positive margin to the new analytical limits.

The NRC staff conducted an audit of the ECCS margin assessment and its underlying Westinghouse, AREVA, and GEH engineering calculations (See attached audit report). As part of the audit, the staff evaluated the quantification, justification, and application of analytical credits and reviewed new LOCA calculations. In addition, the staff collected plant-specific information and compiled this information into the ECCS Margin Database (ADAMS ML11262A010).

Examination of the ECCS Margin Database reveals that the majority of plants needed no adjustments to show a positive margin to the revised analytical limits. For the remaining plants, the staff audited the supporting calculations and found that the applied credits and new LOCA calculations were sufficient to demonstrate positive margin to the new analytical limits. Therefore, the staff concludes that the entire U.S. commercial nuclear fleet continues to operate in a safe manner relative to the new research findings. The audit also addressed the future start-up of Watts Bar Unit 2 and Bellefonte Units 1 and 2.

Recognizing that finalization and implementation of the new ECCS requirements would take several years, Section 4 recommends a strategy for confirming continued safe operation in the interim until all plants are in compliance with the new 10 CFR 50.46c.

Based upon the plant-specific information and relative margin to the new analytical limits, Section 5 recommends an implementation plan for 10 CFR 50.46c. Aside from the assurance provided by the ECCS margin assessment which confirms, on a plant-specific basis, the safe operation of the U.S. commercial nuclear fleet, industry trends are in a positive direction with respect to continued safe operation in light of the new requirements. Plants continue to migrate away from zirconium alloys susceptible to hydrogen-enhanced embrittlement. Only 4 plants currently load fresh fuel batches with Zircaloy-4 cladding and these plants have plans to transition to fuel batches with M5 alloy cladding in the next few years. Many other plants have plans to transition from ZIRLO™ to Optimized ZIRLO™ cladding. BWR Zircaloy-2 cladding alloys have also seen significant improvements in corrosion resistance.

Based upon the information provided in the PWROG and BWROG margin assessment reports and the information collected during the staff audits of the Westinghouse, AREVA, and GEH engineering calculations, the staff's audit report (attached) concluded that sufficient plant-specific information has been documented to complete this safety assessment. Therefore, no further regulatory action is required and the draft GL entitled, "Potential Embrittlement of Fuel Rods During Postulated Loss-of-Coolant Accidents" (ADAMS ML102650015), will not be issued.

Audit Report for PWROG and BWROG ECCS Performance Margin Assessment

1. Scope and Purpose

In response to recent research findings which demonstrate that the current performance requirements for emergency core cooling systems (ECCSs), specified in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.46(b), may not always be sufficient to ensure an acceptable level of fuel rod cladding post quench ductility (PQD) following a postulated loss of coolant accident (LOCA), the NRC drafted a generic letter (GL) entitled, "Potential Embrittlement of Fuel Rods During Postulated Loss-of-Coolant Accidents" (ADAMS ML102650015). The information requested in the proposed GL would provide the NRC a basis for confirming an earlier, preliminary assessment and provide the NRC a basis for confirming continued safe operations pending completion of the regulatory framework.

As an alternative to the GL, the industry has undertaken a project to voluntarily deliver the information being sought in the GL. By letter dated May 4, 2011, the Nuclear Energy Institute (NEI) transmitted the Pressurized Water Reactor Owner's Group (PWROG) emergency core cooling system margin assessment report (ADAMS ML11139A309). The Boiling Water Reactor Owner's Group (BWROG) report was provided through NEI in a letter dated June 14, 2011 (ADAMS ML11166A156). The PWROG and BWROG reports summarize the ECCS performance margin assessment performed to compare current margins of safety relative to the research data.

The purpose of this audit is to review the underlying Westinghouse, AREVA, and General Electric Hitachi Nuclear Energy (GEH) engineering calculations and collect necessary plant-specific information to confirm safe operation. As part of this audit, the staff will determine whether the industry sponsored alternative has provided the necessary information being sought by the GL.

A list of participants in the audit is provided in Table 1-1.

Table 1-1: List of Attendees

Name	**Affiliation**
Paul Clifford	NRC – NRR
Harold Scott	NRC – RES
Ralph Landry	NRC – NRO
Len Ward	NRC – NRR
Anthony Attard	NRC – NRR
Bert Dunn	AREVA
Mireille Cortes	AREVA
John Blaisdell (telecon)	Westinghouse
Jeffrey Kobelak	Westinghouse
John Ghergurovich	Westinghouse

ENCLOSURE

Ann Marie DiLullo	Westinghouse
Naugab Lee	Westinghouse
Mitch Nissley (telecon)	Westinghouse
Mike Volodzko	Westinghouse
Kurshad Muftuoglu	GEH

2. Discussion

In preparation of the audit, the staff issued an audit plan (ADAMS ML11165A2381) which defined the scope of the audit including 18 focus topics. These topics were based on the staff's review of the PWROG and BWROG reports.

Due to the proprietary nature of ECCS models, a separate audit was conducted on the following dates at the local offices for each fuel vendor:

GEH – July 7, 2011

AREVA – July 14, 2011

Westinghouse – July 19, 2011

During the audit, the following vendor engineering calculations were reviewed. All of the evaluations and loss-of-coolant accident (LOCA) calculations were performed and documented in accordance with the fuel vendor's 10 CFR 50 Appendix B quality assurance program.

- GEH Record DRF 0000-0132-1436, Revision 0 (BWR/2)
- AREVA Calculation 51-9156952-000 (Westinghouse and CE PWRs)
- AREVA Calculation 51-9153391-000 (B&W PWRs)
- Westinghouse Calculation CN-LAM-11-4 (CE PWRs)
- Westinghouse Calculation CN-LIS-11-6 (W PWRs, LBLOCA)
- Westinghouse Calculation CN-LIS-11-8 (W PWRs, SBLOCA)
- Westinghouse Calculation CN-LAM-11-16 (BWR/3)

Section 2.1 of this audit report describes the analytical limits employed in the BWROG and PWROG reports. Section 2.2 provides the staff's assessment of the margin credits. Section 2.3 documents the plant-specific margin assessment.

2.1 Revised Analytical Limits

Starting with the staff's "Elements of Prospective Information Request" handout from a public workshop on April 29, 2010 (ADAMS ML1013004901 and ML1011800120), the industry developed revised criteria to judge ECCS performance. Allowable equivalent cladding reacted (ECR) as a function of cladding hydrogen content is shown in Figure 1 of the BWROG and PWROG reports. The BWROG criterion is identical to the staff's proposed criteria, beginning at 18% ECR then dropping to 6% ECR at 400 wppm and ending at 4% ECR at 600 wppm. The PWROG criterion is slightly different, beginning at 17%ECR. The industry's allowable ECR conservatively reflects the measured ductile-to-brittle transition and proposed analytical limit in DG-1263 and therefore is acceptable.

The BWROG evaluation utilized the best-estimate FRAPCON-3.4 Zircaloy-2 hydrogen uptake model to convert the allowable ECR to a function of fuel burnup. FRAPCON-3.4 has been tuned against a large database of hydrogen measurements and provides an acceptable best-estimate prediction of maximum fuel rod cladding hydrogen content for modern BWR cladding. At end-of-life exposure (62 gigawatt days / metric ton of uranium (GWd/MTU) rod average), the allowable ECR equals approximately 12.6% ECR based upon a cladding hydrogen content of 160 weight parts per million (wppm). For the purpose of this safety assessment, use of a best-estimate hydrogen model for predicting maximum fuel rod hydrogen content as a function of fuel rod exposure is acceptable. For future compliance demonstration, the staff may require an approved, upper-bound model (e.g., peak nodal circumferential average hydrogen) based on vendor alloy-specific data.

The PWROG evaluation utilized several different alloy-specific models. At 50 GWd/MTU fuel burnup, the allowable ECR was 5.7% ECR for ZIRLO (432 wppm), 6.0% ECR for Zry-4 (400 wppm), and 15% ECR for M5 (90 wppm). Citing the significant rod power reduction at high burnup, many of the PWR evaluations were performed at 50 GWd/MTU. For these cases, the allowable ECR corresponding to the corrosion levels at 50 GWd/MTU were compared against the calculated ECR from the analysis-of-record which represents the hottest, low burnup fuel rod in the core. Figure 2-1 illustrates the dramatic reduction in rod power above 50 GWd/MTU relative to the low to intermediate burnup fuel rods. For the purpose of this safety assessment, performing the assessment at 50 GWd/MTU in this fashion is acceptable. For future compliance demonstration, the staff may require explicit evaluations out to the license burnup limit (62 GWd/MTU rod average).

Figure 2-2 illustrates that allowable Cathcart Pawel (CP)-ECR as a function of peak pellet exposure used in the BWROG and PWROG assessments.

Question #8 of the audit plan requested further information on the assumed hydrogen content (and associated allowable CP-ECR) for Group 7. Westinghouse stated that the CE plants comprising Group 7 used conservative Appendix K evaluation models and would have difficulty showing compliance to the bounding ZIRLO analytical limits. As such, new ZIRLO analytical limits were developed using plant-specific fuel duty calculations and the new Westinghouse ZIRLO corrosion model currently under staff review (WCAP-12610-P-A & CENPD-404-P-A, Addendum 2). A hydrogen pickup fraction of 15% was used to convert calculated oxide thickness to cladding hydrogen content (and allowable CP-ECR). This pickup fraction is conservative relative to the 12.5% developed for FRAPCON-3. As a result of these calculations, the Group 7 plants were assessed against a cladding hydrogen content of 267 ppm with an allowable CP-ECR of 10.0%. As stated above, the use of a best-estimate corrosion prediction and hydrogen content is acceptable for this evaluation. However, the staff may require an approved, upper-bound model for future compliance demonstration.

Both BWROG and PWROG evaluations set the minimum time to breakaway oxidation at 5,000 seconds and compared this analytical limit to the calculated time duration above 800°C*.

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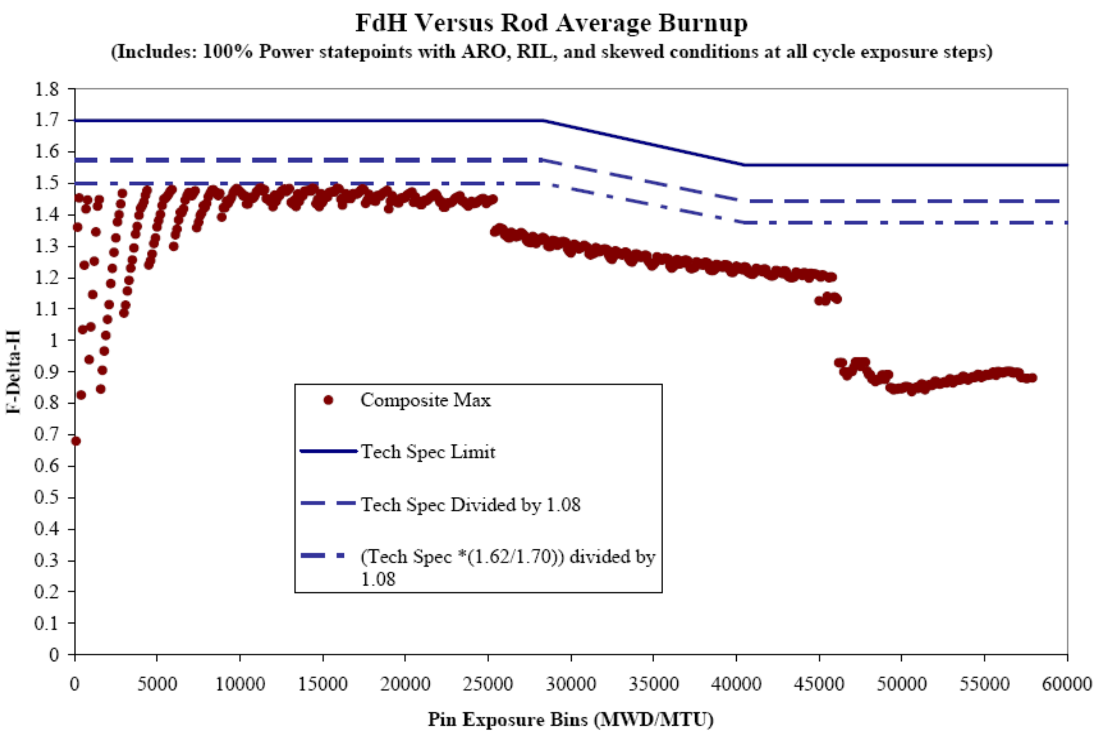
* The draft GL (and earlier "elements" discussed at the public workshop) requested time above 800°C. Whereas, DG-1263 requests time above 650°C. This difference will be further evaluated during the public comment period for the proposed rule and draft RG. However, due to available margin to breakaway, this difference would not change the conclusions of this safety assessment.

Independent breakaway oxidation measurements on ZIRLO cladding segments were conducted by Westinghouse (ADAMS ML091350581). As reported by Westinghouse, sample 128 reached 4,400 seconds with no excessive weight gain or visual evidence of breakaway oxidation. The temperature vs. time profile for this specimen was more similar to SBLOCA profiles than earlier tests at ANL and Westinghouse. No PWR plants reported more than 2,000 seconds above 800°C. As such, any discrepancy between Westinghouse and ANL experimental protocols or test results is insignificant with respect to this margin assessment.

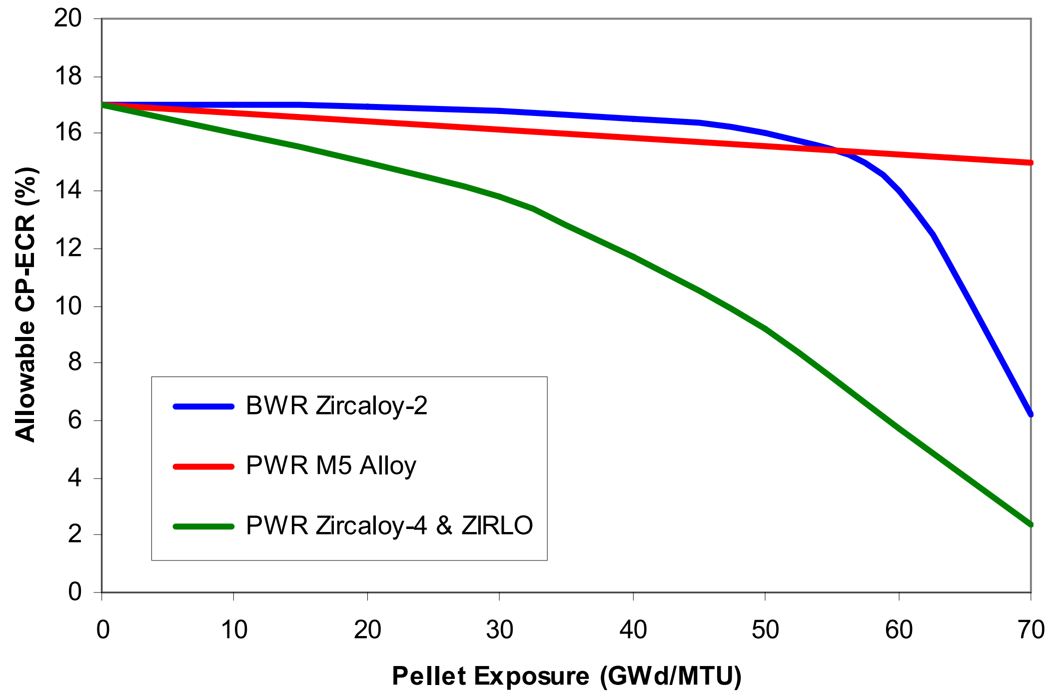
To address the new analytical requirement associated with cladding inside-diameter oxygen ingress, the margin assessment doubled the calculated ECR for fuel rods not predicted to burst. Burst rods would already experience doubled sided oxidation.

Figure 2-1:



(Source: Westinghouse Proprietary Calculation CN-LIS-11-6)

Figure 2-2: Allowable CP-ECR versus Pellet Exposure



2.2 Application of Analytical Credits and New LOCA Analyses

The majority of plants needed no adjustments to show a positive margin to the revised analytical limits. In summary:

Revised PQD Analytical Limit:

- 27 of 35 BWRs needed no adjustments or new calculations.
- 41 of 69 PWR LBLOCA needed no adjustments or new calculations.
- 59 of 69 PWR SBLOCA needed no adjustments or new calculations.

New Breakaway Oxidation Analytical Limit:

- All BWRs and PWRs needed no adjustments or new calculations. Only 1 plant challenges this new requirement.

The audit focused on the quantification, justification, and application of analytical credits and new calculations for the remaining plants which required margin. The staff's assessment of these adjustments is described below.

New LOCA Calculations Using Cathcart-Pawel Correlation

The CP weight gain correlation was used to integrate time-at-temperature in the development of the ductile-to-brittle transition PQD analytical limits. As such, the same CP correlation must be used to integrate time-at-temperature in the LOCA calculations. In some instances, the CP correlation was used solely to integrate time-at-temperature and the Baker-Just (BJ) correlation was retained to estimate exothermic reaction rates for the heat balance calculations. In other instances, the CP correlation was used in for both applications. For the purpose of this safety assessment, the use of CP in one or both applications is acceptable. The CP correlation has been shown to provide an acceptable estimate of reaction rates based on a large empirical database for all modern alloys in the temperature range of interest. For future compliance demonstration, the staff may require the retention of BJ correlation to estimate exothermic reaction rates for the heat balance calculations as required for Appendix K models (CP would be used to integrate time-at-temperature).

New LOCA Calculations Crediting Core Operating Limits Report (COLR)

Using currently approved SAFER/CORCL models, GEH performed new LOCA analyses crediting the existing COLR thermal-mechanical operating limits (TMOL) for several lattice types in both GE11 and GNF2 fuel bundle designs. For the BWR/2 design, COLR maximum average planar linear heat generating rate (MAPLHGR) limits are dictated by calculated PCT early in life and by calculated ECR at higher exposure levels. However, fuel rod power levels may be further restricted because of fuel rod thermal-mechanical design criteria (e.g., rod internal pressure, power-to-melt). Figure 5 of BWROG-TP-11-010 (shown as Figure 2-3 below) illustrates the calculated CP-ECR for both BWR/2 plants and fuel designs. As part of the audit, the staff reviewed GEH calculation DRF 0000-0132-1436, Revision 0. It should be noted that GEH applied a conservative correction factor to the calculated CP-ECR to account for any cladding ID oxygen ingress prior to the time of rod burst. These results demonstrate that the calculated CP-ECR remains below the revised Zry-2 PQD analytical limit at all exposures when the more restrictive TMOL rod power limits are credited.

GEH recently discovered several errors in their LOCA evaluation models. The impact of these errors have been documented in 10 CFR 50.46(a)(3) reports. These errors were corrected in the new BWR/2 LOCA analyses done in support of this margin assessment. Due to the existing margins in BWR/4, /5, and /6 plants, no new LOCA calculations were performed. The impact of these errors is expected to be minimal for these plants and would not change the conclusions of this margin assessment.

Similarly, Westinghouse performed new LOCA analyses crediting the existing COLR thermal-mechanical operating limits for their 4 BWR/3 plants. As with the BWR/2 plants, the allowable fuel rod power level for BWR/3 plants is governed by fuel rod thermal-mechanical design criteria at higher exposure levels. Question #12 of the audit plan requested further information on intermediate power cases. As discussed during the audit and documented in Westinghouse calculation CN-LAM-11-16, Westinghouse provided the results from LOCA calculations at 40, 50, 60, and 70 GWd/MTU (nodal exposure). These results (shown below in Figure 2-4) demonstrate that the calculated CP-ECR remains well below the revised PQD analytical limit at all exposures when the more restrictive TMOL rod power limits are credited.

No new LOCA calculations were included for the remaining 2 BWR/3 plants fueled with GEH fuel designs. The GEH BWR/2 calculations and Westinghouse BWR/3 calculations clearly demonstrate that the TMOL rod power envelope limits transient oxidation to within the new analytical limits at high exposure. This conclusion is equally valid for the remaining 2 BWR/3 plants. During the audit, GEH stated that preliminary BWR/3 calculations were performed and validated this margin; however were never documented.

COLR TMOL rod power envelopes are continuously monitored by the plant computer. In general, TMOL rod power envelopes are developed for each fuel lattice configuration and are not expected to change on a cycle-specific basis (although their inclusion in the COLR would allow frequent changes). Figure 2-5 illustrates the decrease in allowable peak nodal power with exposure for a typical COLR TMOL.

Based on the use of approved models and methods, the staff finds the GEH BWR/2 and Westinghouse BWR/3 LOCA calculations acceptable. Further, crediting reduced fuel rod power at higher exposures that are governed by existing Technical Specifications (i.e., COLR TMOL) is acceptable.

New LOCA Calculations Crediting Diminished Rod Power

As with BWR operations, PWRs are required to limit fuel rod power levels at higher exposure due to fuel rod thermal-mechanical design criteria (e.g., rod internal pressure, power-to-melt). However, unlike the BWR COLR TMOL, most PWR plants do not specify an exposure dependent allowable linear heat generating rate (LHGR) in their COLR. Instead a single value is listed (e.g., 13.1 KW/ft, PVNGS-2 COLR, Revision 17, ADAMS ML11129A0311).

The “Reload Power History” credit in Section 5.4 of PWROG Report OG-11-143 relies on the bounding rod power histories used in the fuel rod design calculations and verified as part of the reload cycle design process. The 8 CE NSSS design plants using Westinghouse fuel comprise Group 7 and utilize this credit (See Table 7 of PWROG Report OG-11-143). Question #6 of the audit plan requested further information on the derivation of maximum local oxidation (MLO) adjustment and intermediate exposure levels. As discussed during the audit and documented in Westinghouse calculation CN-LAM-11-4, Westinghouse ran explicit STRIKIN hot rod temperature calculations to develop the MLO adjustment based on the thermal-mechanical radial falloff curve. While Table 7 of PWROG Report OG-11-143 only shows the results of an evaluation at 50 GWd/MTU, Westinghouse stated that additional calculations were performed at 32 GWd/MTU (knee of falloff curve).

The “Burnup Study” credit in Section 5.6 of PWROG Report OG-11-143 is based on explicit LOCA calculations performed to evaluate IN 98-29. Only 1 Westinghouse plant comprises Group 5 and utilizes this credit (See Table 5 of PWROG Report OG-11-143). Question #7 of the audit plan requested further information on this burnup study. During the audit, Westinghouse provided results of LOCA calculations performed at several higher exposure points for the most limiting cases from the ASTRUM analysis. For this evaluation, there was no credit taken for a reduction in peaking factors at the increased burnup steps. Figure 2-6 illustrates the results of this evaluation. Examination of this figure reveals that the analysis-of-record (AOR) MLO at low and intermediate exposure (shown as black diamonds) as well as the high exposure cases (shown as green squares) remain below the revised analytical limit. Note that 4.3% ECR which is reported in the PWROG report for Group 5 corresponds to the highest ECR result from the cases run at increased burnup.

It is reasonable to credit lower fuel rod power at higher exposure where fuel rod thermal mechanical design restrictions are more limiting than LOCA restrictions. However, without a specific COLR reference (e.g., BWR TMOL), it is questionable whether the licensee complies with 10 CFR 50.36(c) which requires limiting conditions for operation important to safety be captured in technical specifications. For future compliance demonstration, the staff may require that PWR plants update their COLR and include any exposure dependent rod power limitations important to safety (either thermal-mechanical design or LOCA).

New LOCA Calculations Crediting ANS-1979 Decay Heat Plus 2σ Uncertainty

Appendix K requires the use of ANS-1971 decay heat model plus 20% uncertainty. For realistic ECCS evaluation models, the staff has accepted the use of the ANS-1979 decay heat model plus 2σ uncertainty. However, no realistic models have been approved for SBLOCA. SBLOCA Group 2, consisting of 5 PWR plants, takes credit for this adjustment. Question #10 of the audit plan requested further information on the applicability of this credit to the 5 plants in SBLOCA Group 2. Westinghouse calculation CN-LIS-11-8 documents explicit SBLOCA calculations using the ANS-1979 plus 2σ decay heat model. NOTRUMP results show that by assuming a lower decay heat, less heat is generated in the core while it is uncovered. Less heat generation allows for additional system depressurization. Lower system pressure results in more injected flow both from the ECCS pumps and the accumulators. Therefore, the core exit vapor temperature is significantly reduced. As shown in Figure 2-7, the core uncovers for a shorter period of time and the resulting calculated PCT and local oxidation are significantly lower.

The AOR results, shown in Figure 2-8, illustrate that the calculated MLO exceeds the proposed ECR analytical limits at high exposure. Table 2-1 lists the calculated results for this limiting plant. Examination of this table reveals the significant benefit from switching to the ANS-1979 decay heat plus 2σ uncertainty on calculated results. In addition to calculations performed for the limiting PCT plant, a separate evaluation was completed for the limiting MLO plant in Group 2. This evaluation confirms the applicability of the calculated credit.

For the purpose of this safety assessment, the staff finds the use of the ANS-1979 decay heat plus 2σ uncertainty acceptable.

Transition from Appendix K Model to Realistic Model

Section 5.1 of PWROG Report OG-11-143 defines an Appendix K to best-estimate credit of 60% reduction in MLO. Question #3 of the audit plan requested further information on the derivation of MLO adjustment. As discussed during the audit, Westinghouse compared the calculated MLO for 3 transitions to realistic models. While many more plants have transitioned from Appendix K models to realistic models, they have combined this transition with a major change in plant configuration or power uprate. This makes it more difficult to do a direct comparison. The calculated change in MLO was -73%, -66%, and -59%. Note that a negative change means a reduction in the calculated MLO with the transition to realistic models. Based on the limited survey, Westinghouse chose to apply the smallest observed credit. As part of the audit, the staff reviewed Westinghouse calculation CN-LIS-11-6 which documents the derivation of this credit.

The Westinghouse Appendix K to realistic model transition credit was applied to 5 PWR plants in Group 3. Westinghouse notes that the Appendix K model (BASH EM) may not produce conservative results relative to Best-Estimate methods under certain conditions. These conditions, discussed in LTR-NRC-06-23, relate to downcomer boiling. None of the Group 5 plants surveyed for margin quantification are impacted by the downcomer boiling issue.

The last PWR plant in Group 3 utilizes AREVA fuel. For this assessment, new LOCA calculations were performed using the approved AREVA realistic models and methods. The results are listed in Table 3 of PWROG Report OG-11-143. Hence, this plant does not rely upon an estimated credit, but an explicit re-analysis. During the audit AREVA stated that this plant is transitioning to M5 cladding and a new LOCA AOR is being submitted to the NRC. In the new

AOR, the calculated MLO will be 2.94% CP-ECR. Due to the favorable corrosion properties of M5 cladding, the ECR margin increases dramatically.

Westinghouse does not have an approved realistic evaluation model applicable to CE plants. As a result, the 8 Combustion Engineering (CE) plants in Group 7 credit plant-specific corrosion and exposure dependent rod power history. However, AREVA recently completed a LOCA analysis for SONGS Unit 2 and 3 using their approved realistic methods. The result of this evaluation is a PCT of 1605°F and a maximum local oxidation of less than 0.5% ECR. Compared with the Westinghouse AOR results of 2112°F and 15.6% ECR, the AREVA evaluation shows that significant margin is available for the CE plants in Group 7.

Transition from CQD Model to ASTRUM Model

Section 5.2 of PWROG Report OG-11-143 defines a CQD to ASTRUM credit of 50% reduction in MLO. Question #4 of the audit plan requested further information on the derivation of MLO adjustment. As discussed during the audit, Westinghouse compared the calculated MLO for 9 plants which transitioned from CQD to ATRUM. Table 2-2 lists the calculated changes in MLO for these transitions. Examination of this table reveals a median and average change for this data set of 69% and 59% respectively. In one instance, the calculated MLO increased with the transition to ASTRUM. As part of the audit, the staff reviewed Westinghouse calculation CN-LIS-11-6 which documents the derivation of this credit.

A large portion of the benefit in transitioning to ASTRUM is achieved by removing an artificial time stretch which was imposed by the staff in the approved CQD model (not part of approved ASTRUM). Extending the transient has a much larger impact on calculated MLO than on PCT.

Westinghouse stated that some of the difference was likely attributable to model updates and input differences between the analyses, while some of it is also due to the random sampling in ASTRUM. The nature of the sampling in ASTRUM as well as the potential input differences makes it difficult to assign a plant-specific benefit, but a generic benefit for the methodology transition from CQD to ASTRUM can be estimated. In terms of the percentage benefit, most plants show a benefit between 61 and 93 percent with several plants showing somewhat less. Eliminating the two cases which showed the largest percentage reduction in MLO (arbitrary value to add some element of conservatism in margin quantified), the average of the remaining cases is a 50% benefit. Crediting a 50% benefit would be less than the average (due to the exclusion of the two most beneficial cases), and it would also be less than or equal to the benefit observed for all cases except for three.

Improved Statistics for Best-Estimate Analyses

Section 5.5 of the PWROG Report OG-11-143, discusses credit taken for the manner in which the uncertainty analysis data set is used to determine the maximum local oxidation. The ASTRUM methodology uses highest values from a calculated data set of 124 cases to determine the 95th quartile values at 95% confidence for the peak cladding temperature, maximum local oxidation, and core-wide oxidation. Question #5 of the audit plan requested further information on the derivation of MLO adjustment. Referencing the approved ASTRUM methodology in WCAP-16009-P-A, Westinghouse stated that the ASTRUM Topical demonstrates that 124 random samples allows the estimation of three independent output variables (PCT, MLO, and core wide oxidation (CWO)) to the 95th percentile with 95% confidence (95/95). However, it has been suggested by several authors that this approach is overly conservative. Westinghouse stated that the 3rd most limiting result provides sufficient confidence.

As discussed during the audit, Westinghouse compared the calculated 1st and 3rd worse MLO cases for 26 ASTRUM analyses. Table 2-3 lists the calculated changes in MLO for these 26 ASTRUM analyses. Examination of this table reveals a median and average change for this data set of 40% and 41% respectively. One method would be to consider the explicit third case values for each plant in a bin rather than the limiting case. This method would be applicable if all the plants in a given bin had ASTRUM analyses. This approach was applied to the 4 PWRs in Group 4. For plants which credited the benefit from a prior method to ASTRUM, explicit values would not be available. For such cases, the estimated margin to be credited for moving from the first most limiting MLO to the third is a 23% benefit. This is well below the average percentage change of 41%, and captures all the cases except for two (at lower limiting MLO values of about 2% ECR). This approach was applied to the 7 PWRs in Group 6 (along with other credits).

The PWROG report takes credit for using the third highest value of that same data set to represent the 95th quartile value at a 95% confidence level for the single parameter, maximum local oxidation. Proper statistical approaches would not permit using the same data set for two purposes, rather a new data set would have to be generated. In this case, the staff will permit use of the same data set in this manner since this is not a licensing basis analysis but an assessment of margins in the analysis of maximum local oxidation. The staff also notes that regenerating the complete data set would not likely produce a significantly different result based on experience in reviewing large numbers of analysis data sets. This decision on the part of the NRC does not establish a precedent for using this approach in licensing basis analyses.

Increased Allowable ECR – Expanded PQD Empirical Database

The empirical database used to develop the PQD analytical limits in DG-1263 is comprised mostly of ring-compression tests performed on cladding segment exposed to steam oxidation at 2200 °F. Results from EPRI-sponsored research conclude that testing conducted at a maximum temperature below 2200 °F yield a higher allowable CP-ECR at the ductile to brittle transition. This finding is consistent with results from earlier research. The 7 PWR plants in Group 6 credit an increase in allowable CP-ECR of 2.5% (5.7% to 8.2%) based on the results of the EPRI-sponsored research. Figure 2-9 plots the results of this research. Examination of the data suggests that the 2.5% increase in allowable CP-ECR is supported by the testing. This conclusion hinges on the PCT for high burnup fuel remaining below 1922 °F. The maximum PCT reported for the Group 6 plants is 2028 °F. This calculated PCT represent the highest power, low burnup fuel rod. Results from the reload power history calculations above illustrate the likely reduction in PCT for higher burnup fuel rods. Hence, it is reasonable to conclude that fuel rods at 50 GWd/MTU would have a calculated PCT less than 1922 °F for these plants.

Figure 2-3: BWR/2 LOCA Calculations at COLR TMOL
(Source: Figure 5 of BWROG-TP-11-010)

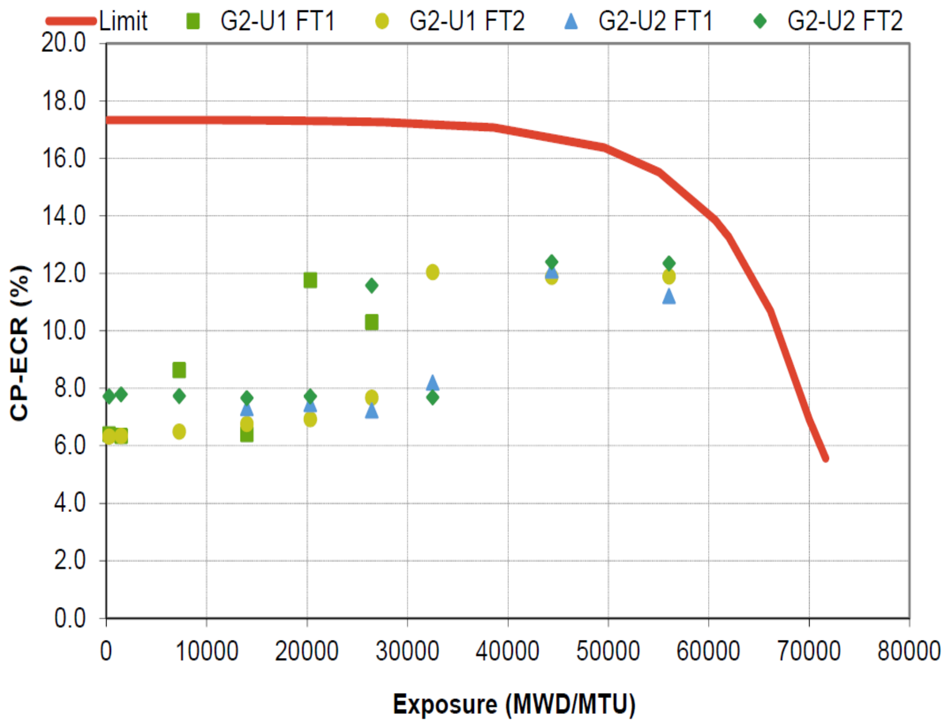


Figure 2-4: BWR/3 LOCA Calculations at COLR TMOL
(Source: Figure 1 of Westinghouse Proprietary Calculation CN-LAM-11-23)

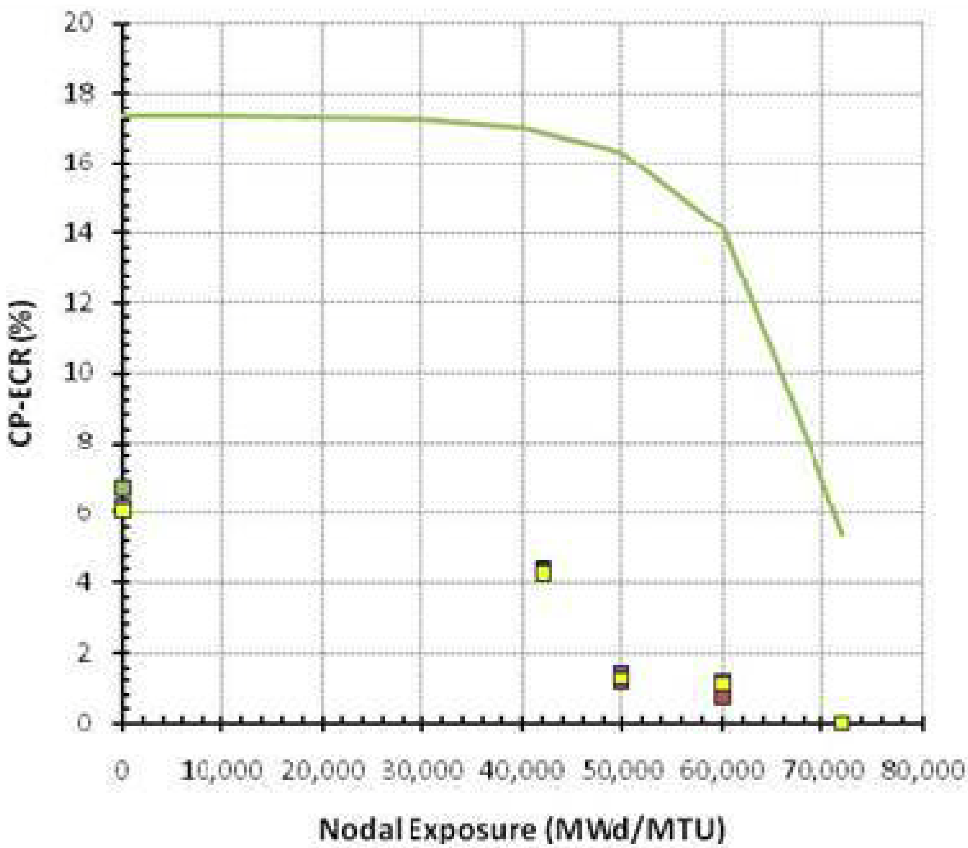


Figure 2-5: Typical BWR COLR Thermal-Mechanical Operating Limits
(Based on ADAMS ML110940268)

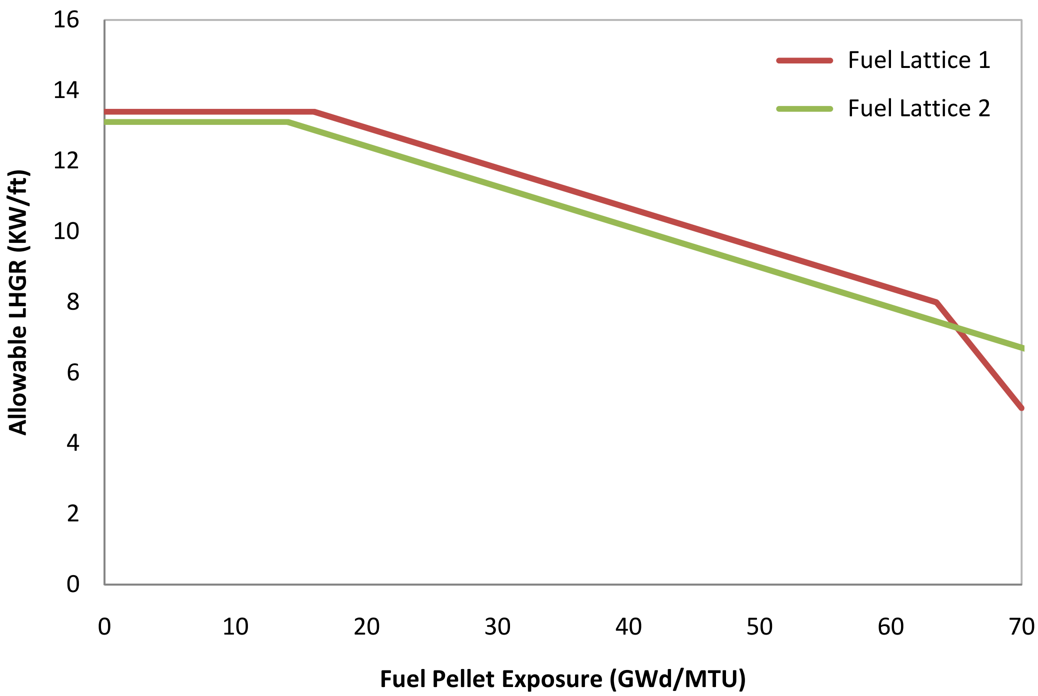


Figure 2-6: Group 5 Burnup Study
(Source: Westinghouse Audit)

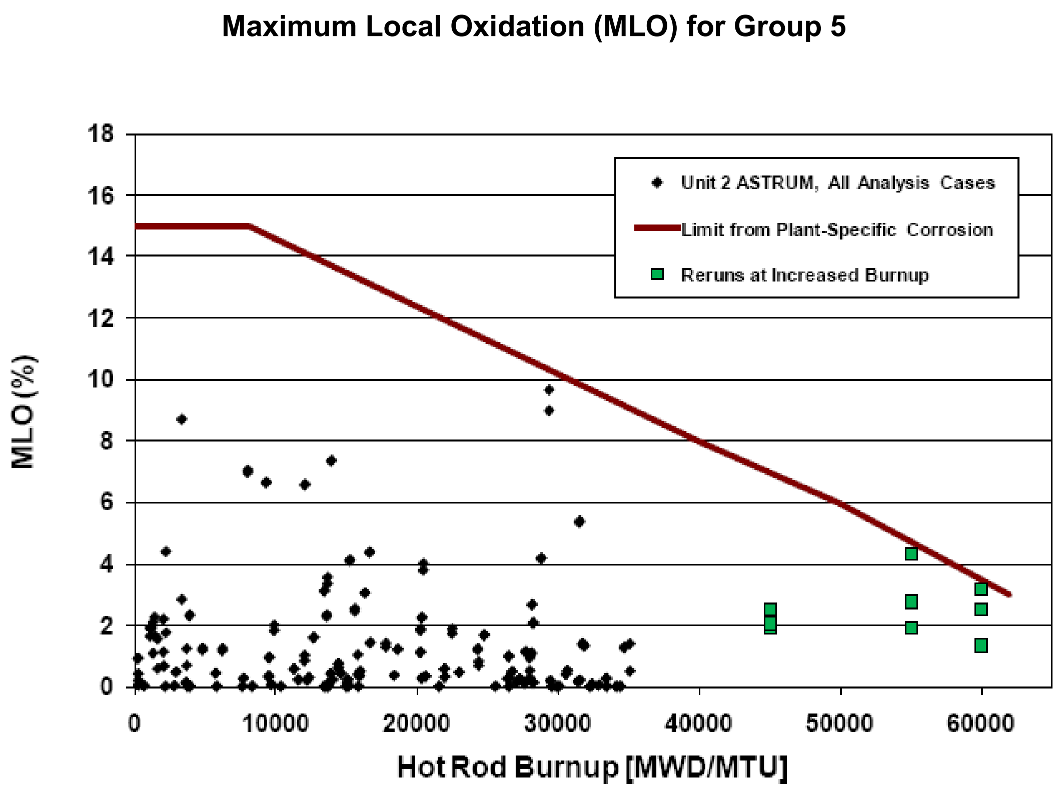


Figure 2-7: SBLOCA Calculations with ANS-1979 +2σ Decay Heat

(Source: Westinghouse Proprietary Calculation CN-LIS-11-8)

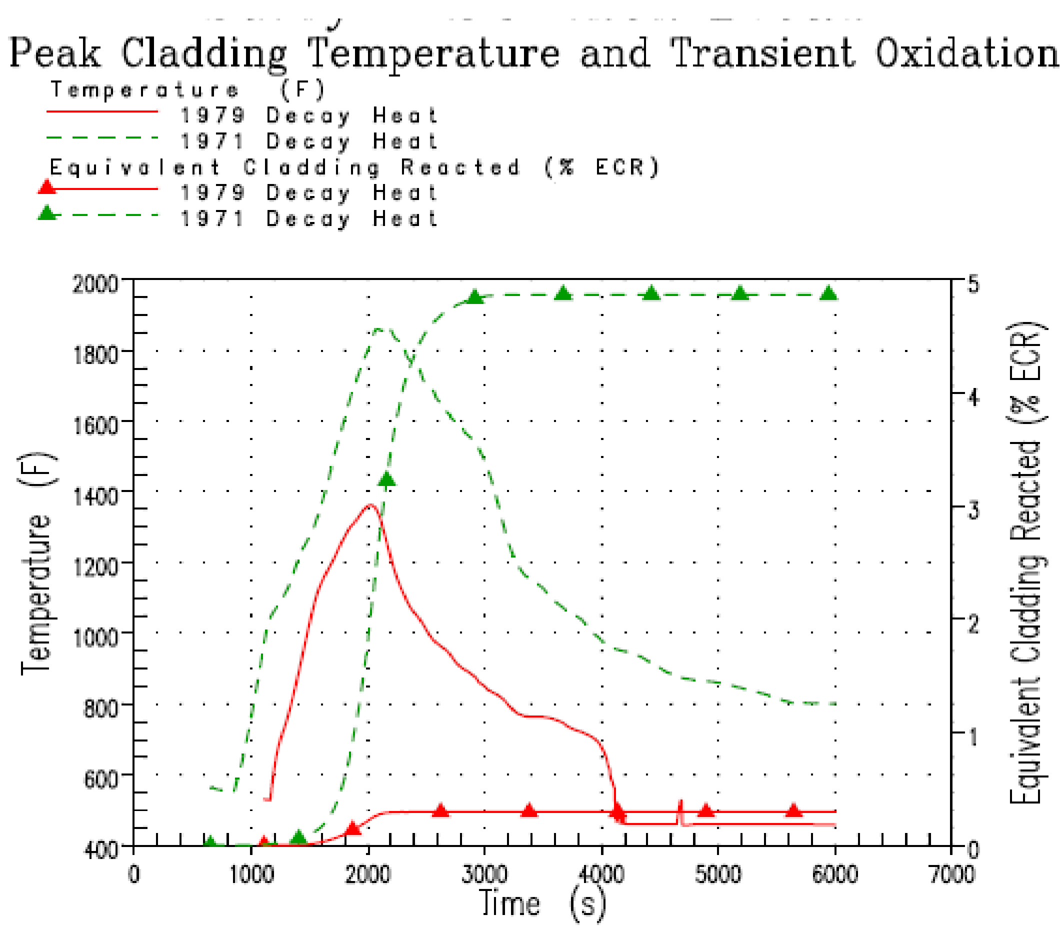


Figure 2-8: SBLOCA Group 2 Limiting Plant AOR Results
(Source: Westinghouse Proprietary Calculation CN-LIS-11-8)

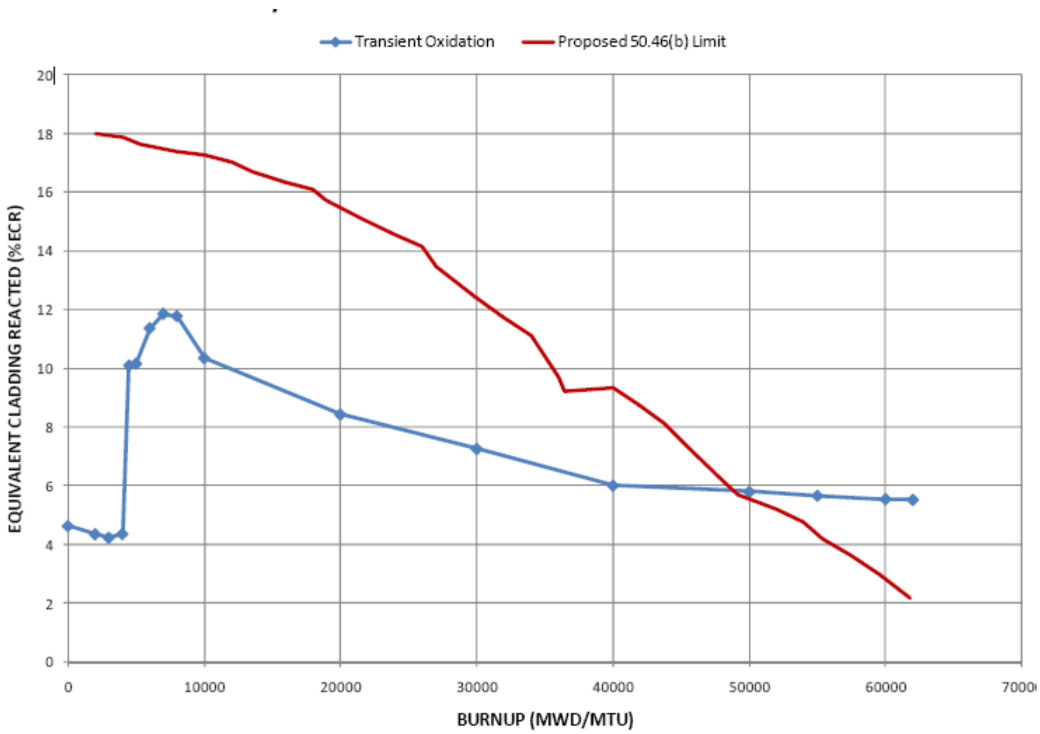


Figure 2-9: PQD Measurements from EPRI-Sponsored Research
(Zircaloy-4 Oxidized at 1922 °F)
(Source: Westinghouse Proprietary Calculation CN-LIS-11-6)

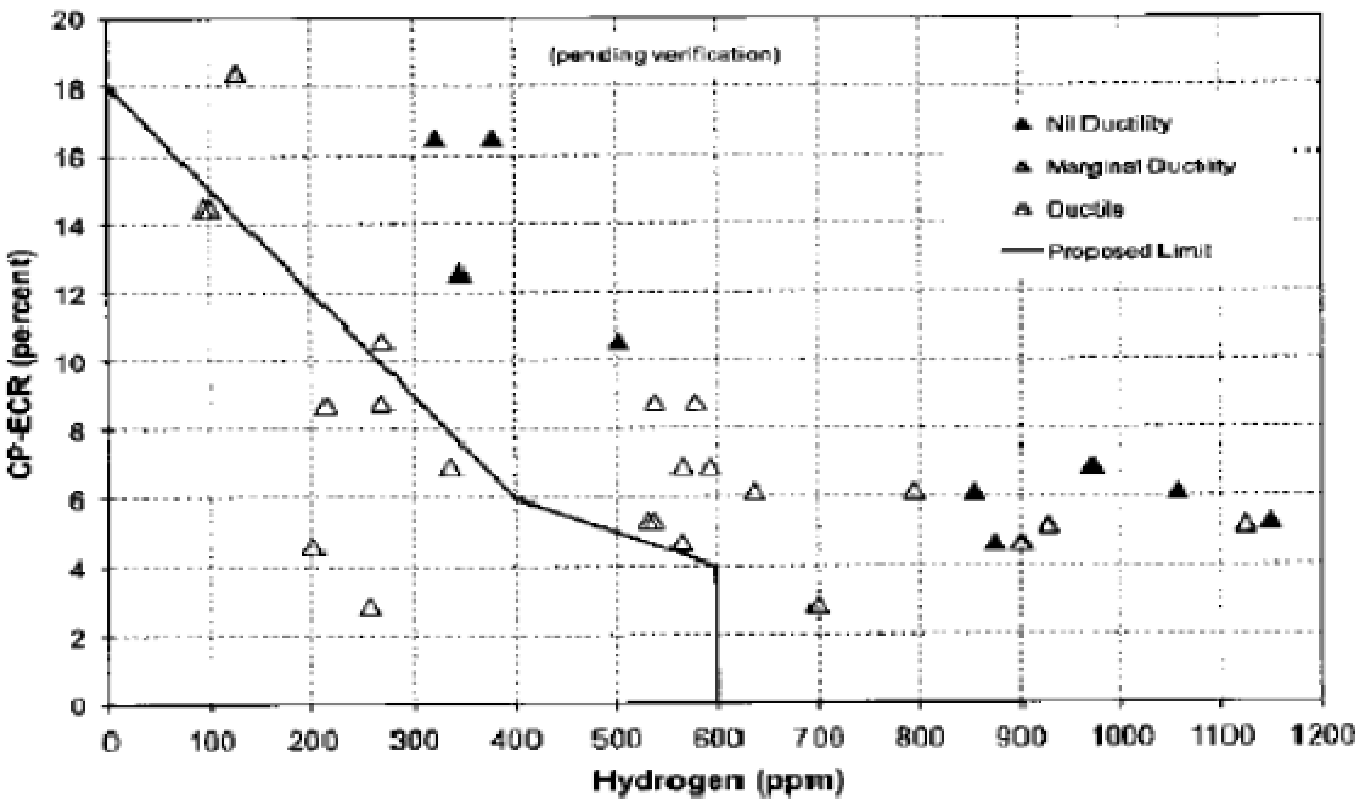


Table 2-1: SBLOCA Calculations with ANS-1979 +2σ Decay Heat
(Source: Westinghouse Proprietary Calculation CN-LIS-11-8)

Results at Different Break Sizes at BOL

	2.5-Inch		**2.75-Inch**		**3-Inch**	
	AOR	**1979 Decay Heat**	**AOR**	**1979 Decay Heat**	**AOR**	**1979 Decay Heat**
PCT, °F	1869.8	1362.7	1906.7	1361.5	1765.6	1364.3
PCT Time, sec	2174.3	2034.2	1687.6	1651.9	1451.6	1373.0
PCT Elevation, ft	12.00	11.50	12.00	11.5	11.75	11.50
Maximum Local ZrO_2, %	**4.87**	**0.29**	**4.64**	**0.29**	**2.47**	**0.23**
Max. Local ZrO_2 Elevation, ft	12.00	11.50	11.75	11.50	11.75	11.25
HR Average ZrO_2, %	0.65	0.04	0.66	0.04	0.38	0.04

Results for 2.75 inch break at Higher Exposure

	40,000 MWD/MTU		60,000 MWD/MTU	
	AOR	**1979 Decay Heat**	**AOR**	**1979 Decay Heat**
HR Burst?	Yes	No	Yes	No
PCT, °F	1892.6	1322.9	1888.0	1319.4
PCT Time, sec	1687.5	1610	1687.4	1610.0
PCT Elevation, ft	11.75	11.25	11.75	11.25
Maximum Local ZrO_2, %	10.55	4.55	15.42	9.87
Max. Local ZrO_2 Elevation, ft	11.25	11.25	11.25	11.25
HR Average ZrO_2, %	4.83	4.54	10.07	9.87
SS ZrO_2, %	4.54		9.87	
Max. HR Transient ZrO_2, %	**6.01**	**0.01**	**5.55**	**0.00**
HR Average Trans ZrO_2, %	0.29	6.71	0.20	0.00

Table 2-2: CQD to ASTRUM Transition
(Source: Westinghouse Proprietary Calculation CN-LIS-11-6)

Plant	**CQD MLO (%)**	**ASTRUM MLO (%)**	**Absolute Delta MLO (%)**	**Fractional MLO Benefit (-)**
Beaver Valley Unit 1	8.8	9.22	-0.42	-0.05
Byron/Braidwood Unit 1	10.5	5.51	4.99	0.48
Byron/Braidwood Unit 2	11.5	8.27	3.23	0.28
Diablo Canyon Unit 2	11.3	1.64	9.66	0.85
Indian Point Unit 2	13	2.39	10.61	0.82
Point Beach Unit 1	8.5	2.61	5.89	0.69
Point Beach Unit 2	8.5	2.57	5.93	0.70
Turkey Point	11	4.24	6.76	0.61
Watts Bar	15	1.04	13.96	0.93

Table 2-3: ASTRUM Calculated MLO, Comparison of 1st and 3rd Worst Case (Source: Westinghouse Proprietary Calculation CN-LIS-11-6)

Plant	First Case MLO (%)	Third Case MLO (%)	Absolute Delta MLO (%)	Fractional Delta MLO (-)
Almaraz Units 1 and 2	3.30	2.46	0.84	0.25
Angra Unit 1	2.16	0.61	1.55	0.72
Beaver Valley Unit 1	9.22	5.48	3.74	0.41
Byron/Braidwood Unit 1	5.51	2.66	2.85	0.52
Byron/Braidwood Unit 2	8.27	4.97	3.30	0.40
Comanche Peak Unit 1	0.23	0.10	0.13	0.57
Comanche Peak Unit 2	0.71	0.43	0.28	0.39
D. C. Cook Unit 1	9.98	4.27	5.71	0.57
D. C. Cook Unit 2	9.69	7.39	2.30	0.24
Diablo Canyon Unit 2	1.64	1.50	0.14	0.09
J. M. Farley Units 1 and 2	2.86	2.04	0.82	0.29
R. E. Ginna	3.43	2.13	1.30	0.38
Indian Point Unit 2	2.39	2.09	0.30	0.13
Maanshan Units 1 and 2	1.58	0.83	0.75	0.47
Millstone Unit 3	3.50	1.23	2.27	0.65
North Anna Unit 1	4.67	2.60	2.07	0.44
North Anna Unit 2	3.53	2.65	0.88	0.25
Point Beach Unit 1	2.61	1.71	0.90	0.34
Point Beach Unit 2	2.57	1.19	1.38	0.54
Ringhals Unit 3	6.37	3.85	2.52	0.40
Surry Units 1 and 2 (2008)	3.74	2.89	0.85	0.23
Surry Units 1 and 2 (2010)	2.45	1.56	0.89	0.36
Tihange Unit 3	4.85	2.10	2.75	0.57
Turkey Point Units 3 and 4	4.24	3.15	1.09	0.26
Watts Bar Unit 2	1.04	0.48	0.56	0.54
Wolf Creek	0.97	0.32	0.65	0.67

2.3 Plant-Specific Data

The BWROG ECCS margin assessment groups plants based on plant design (e.g., BWR/2). The PWROG ECCS margin assessment groups plants based on plant design, fuel cladding material, and/or ECCS evaluation model. The Owners Group reports document the margin assessment for only the limiting plant within each grouping. As part of the audit, the staff collected and evaluated each individual plant margin assessment. The results of this investigation, including plant design, fuel vendor, cladding material, AOR results (PCT, MLO, and time spent above 800 °C), evaluation model, plant grouping, available margin to new analytical limits, and type of credit, were tabulated for each plant and documented in ADAMS ML11262A010.

Question #9 of the audit plan requested further information on the SBLOCA margin assessment for plants fueled with AREVA fuel designs. Note that the limiting plant in each SBLOCA grouping was evaluated with Westinghouse information (Table 8, 9, and 10 of PWROG report). During the audit, AREVA staff identified that all of their plants fall into Group 1 and do not rely on any credits to demonstrate sufficient margin to the revised analytical limits. This information is captured in ML11262A010.

Question #11 of the audit plan requested further information on the minimal margin to breakaway oxidation for the BWR/2 plants. During the audit, GEH staff described the unique LOCA progression for this plant design and why the DEG break was limiting with respect to time duration above 800 °C. In summary, the BWR/2 break location (recirculation piping at bottom of vessel) results in an unrecoverable core liquid level. As such, the maximum break size will uncover the core quicker and result in a longer time at elevated temperature. GEH identified that the margin to breakaway (300 seconds) increases dramatically (approx. 3600 seconds) when the Appendix K required ANS-1971 decay heat model plus 20% uncertainty is replaced with a best-estimate decay heat model. GEH also identified that the second BWR/2 plant has substantially more margin (3400 seconds) due to higher core spray delivery.

As part of the audit, the NRC staff collected plant-specific information for all operating reactors. However, the OG margin assessment reports did not include new reactors which are expected to begin operation in the next few years. Watts Bar Unit 2 is a Westinghouse designed 4 Loop PWR with a rated thermal power of 3,411 MWt. Its operating license is currently under review by the NRC staff. The Watts Bar Unit 2 LBLOCA analysis was performed with the Westinghouse ASTRUM methodology and resulted in a 1552 °F PCT and 1.04% ECR. The SBLOCA analysis was performed with an approved Appendix K model and resulted in a 1184 °F PCT and less than 0.1% ECR. The time duration above 800 °C is minimal. Hence, Watts Bar Unit 2 has sufficient margin to the revised analytical limits.

Bellefonte Units 1 and 2 are B&W designed PWRs with a rated power of 3,763 MWt. The Bellefonte Cycle 1 safety analyses have not been completed. During the audit, AREVA staff stated that the Bellefonte units will use fuel designs with M5 alloy cladding. As shown in Figure 2-1, the allowable CP-ECR for M5 alloy remains above 15% at end of life. AREVA staff also stated that the Bellefonte units will have similar power densities to the existing fleet. Combined with an improved ECCS, including a high pressure safety injection, the Bellefonte units are not expected to be susceptible to either PQD or breakaway oxidation. The NRC staff will confirm that the Bellefonte plants will begin operation with sufficient margin to the revised analytical limits once the Cycle 1 LOCA analysis is complete.

3. Need for Additional Regulatory Action - Generic Letter

To obtain the necessary plant-specific information to complete a more details ECCS performance safety assessment, the staff developed a draft GL entitled, “Potential Embrittlement of Fuel Rods During Postulated Loss-of-Coolant Accidents” (ADAMS ML102650015). Based upon the information provided in the PWROG and BWROG margin assessment reports and the information collected during the staff audits of the Westinghouse, AREVA, and GEH engineering calculations, the staff concludes that sufficient plant-specific information has been documented to complete the safety assessment. Therefore, no further regulatory action to request information is required and the draft GL need not be issued.

4. Conclusions

As part of the audit, NRC staff met with representatives of Westinghouse, AREVA, and GEH, discussed the PWROG and BWROG margin assessment reports and the audit plan focus topics, reviewed the underlying vendor engineering calculations, and collected plant-specific data. All of the LOCA evaluations and calculations were performed and documented in accordance with the fuel vendor’s 10 CFR 50 Appendix B quality assurance program. Based upon the information provided in the Owner’s Group report and collected during this audit, the staff concludes that no further regulatory action to request information is required and the draft GL need not be issued.