Shawn K. Gibby
Vice President
Nuclear Engineering

Duke Energy
13225 Hagers Ferry Rd., MG011E
Huntersville, NC 28078

704-519-5138
Shawn.Gibby@duke-energy.com

August 18, 2025
Serial: RA-25-0014

10 CFR 50.12
10 CFR 50.90

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

CATAWBA NUCLEAR STATION, UNITS 1 AND 2
DOCKET NOS. 50-413 AND 50-414 / RENEWED LICENSE NOS. NPF-35 AND NPF-52

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2
DOCKET NOS. 50-369 AND 50-370 / RENEWED LICENSE NOS. NPF-9 AND NPF-17

Subject: License Amendment Request to Adopt the Full Spectrum Loss of Coolant Accident Methodology and AXIOM® Fuel Cladding Topical Reports and Associated 10 CFR 50.46 Exemption Request For Use of AXIOM® Fuel Cladding

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) is submitting a request for an amendment to the Technical Specifications (TS) for Catawba Nuclear Station (CNS), Unit 1 (CNS U1) and McGuire Nuclear Station, Units 1 and 2 (MNS). The proposed amendment requests the addition of the Westinghouse Electric Company LLC (Westinghouse) topical report WCAP-16996-P-A, Revision 1, “Realistic LOCA [Loss of Coolant Accident] Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM™ LOCA Methodology)” (ADAMS Accession No. ML17277A130), to the list of approved analytical methods used to determine the core operating limits provided in TS 5.6.5, “Core Operating Limits Report (COLR).” Additionally, this amendment proposes the annotation of select legacy LOCA methods and the deletion of others listed in TS 5.6.5.b to restrict their future use and allow for a staggered implementation during refueling outages at each unit.

The amendment also requests a modification to the TS to permit the use of the Westinghouse fuel cladding alloy designated as AXIOM®. Specifically, the proposed amendment requests a revision to the TS to update the description of fuel assemblies specified in TS 4.2.1, “Fuel Assemblies,” and add the Westinghouse topical report WCAP-18546-P-A, “Westinghouse AXIOM® Cladding for use in Pressurized Water Reactor Fuel” (ADAMS Accession No. ML23089A063) to the referenced analytical methods in TS 5.6.5.b to allow the use of AXIOM® alloy for fuel rod cladding.

A separate license amendment request to address the adoption of the aforementioned Westinghouse topical reports for the FULL SPECTRUM™ LOCA (FSLOCA™) Evaluation

Methodology (EM) and AXIOM® fuel rod cladding for CNS Unit 2 (CNS U2) will be submitted at a later date. Inclusion of notes within the associated CNS TS are proposed to reflect applicability of the requested license amendment changes to only CNS U1. Deletion of legacy LOCA analysis methods is administrative and is applicable to both CNS U1 and CNS U2.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been concluded that the proposed changes involve no significant hazards consideration.

Enclosure 1 of this license amendment request provides Duke Energy's evaluation of the proposed changes. Attachments 1 and 2 of the enclosure provide a copy of the existing TS pages and TS Bases pages, respectively, marked with the proposed changes. Attachments 3 and 4 of the enclosure provide additional information related to FSLOCA™ plant input parameters. The TS Bases markups are provided for information only and will be incorporated in accordance with each respective site's TS Bases Control Program upon implementation of the approved license amendments. Enclosures 2 and 4 contain the Westinghouse proprietary and non-proprietary versions of a supporting document referenced in the evaluation. An affidavit from Westinghouse attesting to the proprietary nature of the information is provided in Enclosure 3. Duke Energy requests that Enclosure 2 be withheld from public disclosure in accordance with 10 CFR 2.390.

Furthermore, in accordance with the provisions of 10 CFR 50.12, Duke Energy is requesting an exemption from certain requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," as provided in Enclosure 5, to allow the use of the AXIOM fuel rod cladding for CNS U1 and MNS. The exemption request relates solely to the specific types of cladding material specified in the regulation for use in light water reactors. As written, the regulation presumes use of either Zircaloy or ZIRLO High Performance Fuel Cladding.

Approval of the proposed license amendment and corresponding exemption request is requested by March 31, 2026. Once approved, the amendment will be implemented for CNS U1 prior to the Unit 1 Cycle 30 reload campaign in Spring 2026 and implemented for MNS prior to the Unit 1 Cycle 32 reload campaign in Fall 2026. Revisions to the CNS and MNS Updated Final Safety Analysis Reports (UFSAR), necessary to reflect approval of this submittal, will be made in accordance with 10 CFR 50.71(e), with approved exemptions.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated North Carolina and South Carolina Officials.

This letter contains no regulatory commitments.

Please refer any questions regarding this submittal to Ryan Treadway, Director – Nuclear Fleet Licensing, at (980) 373-5873.

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

Executed on August 18, 2025.

Sincerely,

Shawn K. Gibby

Shawn Gibby
Vice President – Nuclear Engineering

Enclosure 1: Evaluation of the Proposed Amendment
  Attachment 1: Proposed Technical Specification Changes (Mark-up)
  Attachment 2: Proposed Technical Specification Bases Changes (Mark-up)
  Attachment 3: Plant Operating Parameters Compared to Technical Specification Limits
  Attachment 4: Comparison of CQD and FSLOCA Input Parameters
Enclosure 2: Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to McGuire Units 1 and 2 and Catawba Unit 1 (Proprietary Version)
Enclosure 3: Westinghouse Affidavit
Enclosure 4: Application of Westinghouse FULL SPECTRUM LOCA Evaluation Model to McGuire Units 1 and 2 and Catawba Unit 1 (Non-proprietary Version)
Enclosure 5: 10 CFR 50.46 Exemption Request for Use of AXIOM Fuel Cladding

cc: USNRC Region II – Regional Administrator
USNRC Resident Inspector – CNS
USNRC Senior Resident Inspector – MNS
USNRC NRR Project Manager – Fleet
USNRC NRR Project Manager – CNS
USNRC NRR Project Manager – MNS
L. Brayboy, Radioactive Materials Branch Manager – NC DHHS
S. Jenkins, Director – Radiological Health Program – SC DES
N. Gauthier, Manager – Nuclear Response Section – SC DES
L. Garner, Manager – Radioactive & Infectious Waste Section – SC DES

ENCLOSURE 1

EVALUATION OF THE PROPOSED CHANGE

15 PAGES PLUS THE COVER

Evaluation of the Proposed Change

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) is submitting a request for an amendment to the Technical Specifications (TS) for Catawba Nuclear Station (CNS), Unit 1 (CNS U1) and McGuire Nuclear Station, Units 1 and 2 (MNS). The proposed amendment requests the addition of the Westinghouse Electric Company LLC (Westinghouse) Topical Report WCAP-16996-P-A, Revision 1, “Realistic LOCA [Loss of Coolant Accident] Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM^{TM} LOCA Methodology)” (ADAMS Accession No. ML17277A130), to the list of approved analytical methods used to determine the core operating limits provided in TS 5.6.5, “Core Operating Limits Report (COLR).” Additionally, this amendment proposes the annotation of select legacy LOCA methods and the deletion of others listed in TS 5.6.5.b to restrict their future use and allow for a staggered implementation during refueling outages at each unit.

The amendment also requests a modification to the TS to permit the use of the Westinghouse fuel cladding alloy designated as AXIOM®. Specifically, the proposed amendment requests a revision to the TS to update the description of fuel assemblies specified in TS 4.2.1, “Fuel Assemblies,” and add the Westinghouse Topical Report WCAP-18546-P-A, “Westinghouse AXIOM® Cladding for use in Pressurized Water Reactor Fuel” (ADAMS Accession No. ML23089A063) to the referenced analytical methods in TS 5.6.5.b to allow the use of AXIOM alloy for fuel rod cladding. A corresponding exemption is being requested from the provisions of 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” in order to support the use of this additional fuel rod cladding material. The exemption request for CNS U1 and MNS is provided in Enclosure 5 of this submittal.

A separate license amendment request to address the adoption of the aforementioned Westinghouse topical reports for the FULL SPECTRUM LOCA (FSLOCA^{TM}) Evaluation Methodology (EM) and AXIOM fuel rod cladding for CNS Unit 2 (CNS U2) will be submitted at a later date. Inclusion of Notes within the associated CNS TS are proposed to reflect applicability of the requested license amendment changes to only CNS U1. Deletion of legacy LOCA analysis methods is administrative and is applicable to both CNS U1 and CNS U2.

2.0 DETAILED DESCRIPTION

2.1 Background

System Design and Operation

The primary function of the Emergency Core Cooling System (ECCS) following a LOCA is to remove the stored and fission product decay heat from the reactor core such that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented.

The principal mechanical components of the ECCS which provide core cooling immediately following a LOCA are the accumulators, the safety injection pumps, the centrifugal charging pumps, the residual heat removal pumps, refueling water storage tank, and the associated valves, and piping.

The ECCS is designed to cool the reactor core as well as to provide additional shutdown capability following initiation of the following accident conditions:

1. A pipe break or spurious valve lifting in the Reactor Coolant System (RCS) which causes a discharge larger than that which can be made up by the normal makeup system, up to and including the instantaneous circumferential rupture of the largest pipe in the RCS.
2. Rupture of a control rod drive mechanism causing a rod cluster control assembly ejection accident.
3. Steam or feedwater system break accident including a pipe break or a spurious valve lifting in the steam system which would result in an uncontrolled steam release or a loss of feedwater.
4. A steam generator tube rupture.

AXIOM Fuel Cladding

Westinghouse developed AXIOM fuel rod cladding material to provide enhanced corrosion resistance when compared to prior zirconium-based fuel cladding materials. The AXIOM alloy is the next generation of robust alloys targeting very high fuel duties. AXIOM cladding is designed to exhibit improved corrosion resistance, lower hydrogen pickup (HPU), and lower creep and growth compared to prior Westinghouse fuel cladding products, ZIRLO and Optimized ZIRLO. AXIOM cladding is a niobium-bearing alloy with reduced tin content to increase corrosion resistance. It also includes vanadium and copper as alloying elements in order to improve specific properties such as HPU. The AXIOM alloy has been processed to be in the partially recrystallized annealed condition, similar to the Optimized ZIRLO cladding, to compensate for the creep strength loss caused by the reduced tin content.

2.2 Current TS Requirements

A description of the current status of the associated TS is provided below.

2.2.1 CNS and MNS TS 4.2.1, *Fuel Assemblies*

The current TS 4.2.1 for CNS and MNS states the following:

> The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of either Zircalloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of ZIRLO®, Optimized ZIRLO™, zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

The CNS TS is modified by a note that states: “A maximum of four lead assemblies containing mixed oxide fuel and M5™ cladding may be inserted into the Unit 1 or Unit 2 reactor core.”

2.2.2 CNS and MNS TS 5.6.5, *Core Operating Limits Report*

The existing TS 5.6.5 for MNS and CNS requires, in part, core operating limits be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and contains references to the approved analytical methods that are used to determine the core operating limits. The current methods listed in TS 5.6.5.b for LOCA analyses are WCAP-10054-P-A, “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,” and WCAP-12945-P-A, Volume 1 and Volumes 2-5, “Code Qualification Document for Best-Estimate Loss of Coolant Analysis.” This TS also references the associated topical report for Optimized ZIRLO, WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, “Optimized ZIRLO.”

2.3 Reason for the Proposed Change

An analysis with the FSLOCA EM has been completed for MNS and CNS U1 to support 18-month cycle operation with AXIOM cladding. This analysis also supports an extended power uprate (EPU), though the EPU will not be implemented at this time. This license amendment request (LAR) for MNS and CNS U1 requests approval to apply the Westinghouse FSLOCA EM, which was developed to address the full spectrum of LOCAs which result from a postulated break in the RCS of a pressurized water reactor (PWR). The break sizes considered in the Westinghouse FSLOCA EM include any break size in which break flow is beyond the capacity of the normal charging pumps, up to and including a double-ended guillotine rupture of an RCS cold leg with a break flow area equal to two times the pipe area, including what traditionally are defined as small-break LOCAs (SBLOCAs) and large-break LOCAs (LBLOCAs).

By letter dated September 22, 2016 (ADAMS Accession No. ML16271A329), Duke Energy committed to submit LBLOCA analyses that apply NRC-approved methods including the effects of fuel pellet thermal conductivity degradation (TCD) to the NRC for review and approval. The Westinghouse FSLOCA EM includes the effects of TCD, and submittal of this LAR fulfills the Commitment for both MNS units and CNS U1.

AXIOM cladding is Westinghouse’s advanced cladding material. The AXIOM alloy is planned to be used as a fuel rod cladding material in all typical Westinghouse PWR production fuel assemblies. Only the cladding material is being changed; and it will be used with existing NRC-approved cladding dimensions, fuel structures, fuel assembly components, and fuel materials. Duke is implementing AXIOM to gain cladding oxidation and hydrogen pickup margin. The margin gain will also assist Duke Energy with core design transitions for proposed future power uprates and 24-month fuel cycles at MNS and CNS U1.

2.4 Description of the Proposed Change

A specific description of each change is provided below. In addition, a mark-up of the CNS and MNS TS for the following proposed changes is provided in Attachment 1 to this enclosure.

2.4.1 CNS and MNS TS 4.2.1, *Fuel Assemblies*

This section is being revised to add the Westinghouse AXIOM alloy to the list of materials that may be used as the fuel rod cladding in CNS U1 and MNS fuel assemblies. The proposed revised specification (with changes in bold and strikethroughs) for MNS TS 4.2.1 reads as follows:

> The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of either Zircalloy, ZIRLO®, ~~or~~ Optimized ZIRLO™**, or AXIOM®** clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of ZIRLO®, Optimized ZIRLO™, **AXIOM®,** zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CNS TS 4.2.1 will receive the same proposed revisions to reflect the addition of the AXIOM alloy to the list of materials, with an additional clarifier that it is only applicable for Unit 1. There is no change proposed to the CNS-specific note addressing mixed oxide fuel and M5 cladding.

2.4.2 CNS and MNS TS 5.6.5, *Core Operating Limits Report*

The proposed change revises MNS TS 5.6.5 to reflect the addition of Westinghouse topical reports WCAP-16996-P-A and WCAP-18546-P-A to the list of approved methods as follows:

> **20. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," (Westinghouse Proprietary).**
> **21. WCAP-18546-P-A, "Westinghouse AXIOM® Cladding for Use in Pressurized Water Reactor Fuel," (Westinghouse Proprietary)**

Since this proposed amendment will also apply to CNS U1, and not CNS U2, the proposed change for CNS TS 5.6.5 similarly reflects the addition of Westinghouse topical reports WCAP-16996-P-A and WCAP-18546-P-A to the list of approved methods, but also includes annotation reflecting that these topical reports are only applicable for Unit 1. This note will remain until a license amendment is issued for CNS U2 to similarly adopt these two topical reports.

> **21. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," (Westinghouse Proprietary). [For use by Unit 1 only.]**
> **22. WCAP-18546-P-A, Revision 0, "Westinghouse AXIOM® Cladding for Use in Pressurized Water Reactor Fuel," (Westinghouse Proprietary) [For use by Unit 1 only.]**

This proposed amendment also addresses administrative changes to clean up the list of analytical methods provided in CNS and MNS TS 5.6.5. For both CNS and MNS, two legacy LOCA methodologies are proposed for deletion, since they have not been used since the transition from Framatome Mark-BW fuel (BAW-10168-P-A) to Westinghouse Robust Fuel Assembly (RFA) fuel in 1998, and implementation of Westinghouse Best Estimate LBLOCA analysis in 2000 which replaced the BASH LBLOCA method (WCAP-10266-P-A).

~~2. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE" (W Proprietary).~~

~~3. BAW-10168-P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants" (B&W Proprietary)~~

For Catawba, it is proposed to also delete the fuel rod design methodology below, since it only supported Framatome Mixed Oxide (MOX) fuel Lead Test Assemblies, which are no longer under consideration for use as batch feed fuel.

~~17. BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," (Framatome ANP Proprietary).~~

In addition to the proposed deletion of these legacy methodologies, it is also proposed to annotate the following two LOCA methodologies to indicate when they will no longer be used to establish core operating limits once all of the Optimized ZIRLO cladding is discharged from the core designs. The proposed schedule for implementation of AXIOM cladding fuel is shown below for each affected unit. Assuming normal reload core design practices, the Optimized ZIRLO cladding fuel would be fully discharged from the core designs two cycles after the implementation of AXIOM cladding.

Unit	Proposed AXIOM Implementation	Full Cores of AXIOM
Catawba Unit 1	Spring 2026, Cycle 30	Spring 2029, Cycle 32
McGuire Unit 1	Fall 2026, Cycle 32	Fall 2029, Cycle 34
McGuire Unit 2	Fall 2027, Cycle 32	Fall 2030, Cycle 34

CNS:

13. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code" (W Proprietary) **[Shall not be used to determine core operating limits after startup of Catawba Unit 1 Cycle 32].**

15. WCAP-12945-P-A, Volume 1 and Volumes 2-5, "Code Qualification Document for Best-Estimate Loss of Coolant Analysis" (W Proprietary) **[Shall not be used to determine core operating limits after startup of Catawba Unit 1 Cycle 32].**

MNS:

13. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code," (W Proprietary) **[Shall not be used to determine core operating limits after startup of McGuire Unit 1 Cycle 34 and McGuire Unit 2 Cycle 34].**

15. WCAP-12945-P-A, Volume 1 and Volumes 2-5, "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," (W Proprietary) **[Shall not be used**

to determine core operating limits after startup of McGuire Unit 1 Cycle 34 and McGuire Unit 2 Cycle 34]

3.0 TECHNICAL EVALUATION

FSLOCA

The technical evaluation for the application of the Westinghouse FSLOCA EM to MNS and CNS U1 is provided in Enclosures 2 (Proprietary) and 4 (Non-proprietary). The application of the FSLOCA EM to MNS and CNS U1 is consistent with the NRC-approved methodology provided in WCAP-16996-P-A, as modified for AXIOM cladding in WCAP-18546-P-A. The application of this topical report to MNS and CNS U1 involves the performance of a composite FSLOCA EM analysis that covers both MNS units and CNS U1.

The proposed amendment updates the listing of approved analytical methods in TS 5.6.5.b. These changes are administrative in nature because the updated list of analytical methods will continue to ensure core operating limits can be established. The updated listing will reflect the adoption of WCAP-16996-P-A, Revision 1, demonstrating the compliance with the ECCS performance criterion of 10 CFR 50.46, subject to the NRC-specified limitations and conditions in the topical report SE. Additionally, the updated listing will reflect the removal of legacy methods that no longer support current fuel designs, and the addition of an annotation to the existing LOCA evaluation methods that continue to support Optimized ZIRLO^{TM} cladding to restrict future use once all Optimized ZIRLO-clad fuel is discharged from the core designs. These changes have no technical impact on the ability to meet COLR limits.

Of note, Duke Energy has an existing Commitment to re-analyze LBLOCA with methods that explicitly consider fuel pellet thermal conductivity degradation (TCD). The Commitment is stated in Duke Energy letter dated September 22, 2016 (ADAMS Accession No. ML16271A329), and is copied below.

> “Duke Energy will submit to the NRC for review and approval a LBLOCA analysis for Catawba and McGuire that apply NRC-approved methods that include the effects of fuel thermal conductivity degradation (TCD). The revised due date for this commitment is 24 months from the latest NRC approval of WCAP-17642-P, WCAP-16996-P, and any supplements that are needed for analysis related to the new 10 CFR 50.46(c) rule.”

The FSLOCA methodology explicitly considers TCD effects. Submission of the subject license amendment request will satisfy the Commitment for MNS and CNS U1. A future CNS U2 license amendment request will similarly request the adoption of the FSLOCA EM and will satisfy the Commitment for the station.

AXIOM

The NRC Staff reviewed Westinghouse topical report for AXIOM fuel rod cladding, WCAP-18546-P-A, Revision 1, and concluded in the issued safety evaluation (SE) dated December 16, 2022 (ADAMS Accession Nos. ML22320A685 (Non-Proprietary) and ML22306A275 (Proprietary)) that the generic topical report was acceptable for licensing applications, subject to the ranges of fuel types, cladding, and reactors identified in the SE being addressed by the licensees. Duke Energy’s proposed implementation of AXIOM fuel rod cladding complies with the NRC limitations and conditions specified in Section 4.0 of the SE for the AXIOM topical report as follows:

Limitation and Condition 1
It is specified that AXIOM cladding must be used with the NRC-approved PWR designs. Both MNS units are PWRs that are currently licensed for operation by the NRC per operating licenses NPF-9 and NPF-17, respectively. Similarly, CNS U1 is a PWR that is currently licensed for operation by the NRC per operating license NPF-35. Duke Energy's use of AXIOM cladding will not challenge this limitation related to PWR design.

Limitation and Condition 2
It is specified that AXIOM cladding must be used with the NRC-approved Westinghouse and CE fuel designs with corresponding pellet and assembly dimensions. Both MNS and CNS U1 utilize the 17 x 17 Westinghouse RFA design, an NRC-approved fuel design. Duke Energy's use of AXIOM cladding will not challenge this limitation related to fuel design.

Limitation and Condition 3
It is specified that AXIOM cladding must be used with the NRC-approved fuel materials and pellet coatings or additives (e.g., ADOPT IFBA, gadolinium). The limitation for use of NRC-approved fuel materials and pellet coatings or additives will be controlled through the fuel procurement process or validated by cycle-specific engineering analyses. Duke Energy's use of AXIOM cladding will not challenge this limitation related to fuel materials and pellet coatings or additives.

Limitation and Condition 4
A fuel limitation is specified for AXIOM use which requires that fuel burnup shall currently be limited to 62 GWd/MTU peak rod average for all cladding types. While there is the potential for an increased limit once additional information is submitted and approved by the NRC, Duke Energy's use of AXIOM cladding for MNS and CNS U1 will apply a peak rod average burnup limit of 62 GWd/MTU. This limitation will be validated by cycle-specific engineering analyses.

Limitation and Condition 5
A fuel limitation is specified for AXIOM use that requires the Best Estimate Oxide Thickness remain below 100 µm. As provided in Section 5.1.1 of WCAP-18546-P-A, the measured maximum oxide thickness of the AXIOM alloys are less than 50 µm for a burnup of close to 75 GWd/MTU. The best estimate oxide thickness will be less than the allowed 100 µm for a peak rod average burnup of 62 GWd/MTU. Furthermore, this limitation will be validated by cycle-specific engineering analyses.

Limitation and Condition 6
A limitation is specified for the best estimate Hydrogen Pickup (HPU) over the operating life of the fuel. As shown in Section 5.2 of WCAP-18546-P-A, the overall maximum hydrogen content for AXIOM is significantly less than the value specified in Limitation and Condition 6, which demonstrates AXIOM's low HPU. This limitation will be validated by cycle-specific engineering analyses to ensure compliance with this condition and limitation.

Furthermore, the content below describes the impact of AXIOM cladding on other safety analyses and methods owned by Duke Energy, similar to the discussion in Sections 3.8.5 and 3.8.6 of the SE for WCAP-18546-P-A.

- Containment Integrity Analyses

The short-term LOCA mass and energy releases (M&E) are used to determine the maximum differential pressure for structural analyses within sub-compartments inside the containment building resulting from postulated pipe ruptures in the primary system piping. This transient lasts for 1 to 3 seconds in duration and the cladding material does not influence the mass and energy releases. The short-term LOCA M&Es are performed with Westinghouse methods, which are evaluated as being unaffected by the use of AXIOM cladding per Section 6.2.3.1 of WCAP-18546-P-A and Section 3.8.5.1 of the corresponding SE.

For long-term LOCA M&E release calculations, Duke Energy has licensed the methodology described in DPC-NE-3004-P-A, “McGuire and Catawba Nuclear Stations Mass and Energy Release and Containment Response Methodology,” used for containment integrity, maximum sump temperature, and equipment qualification at CNS and MNS. It is conservative for the long term LOCA M&E to maximize the rate of transfer of energy from the core into the coolant and out of the break. There is no hot rod or hot assembly modeled when generating long term LOCA M&E, and therefore fuel pellet and cladding interaction and rod burst are not modeled. Fuel thermal performance characteristics are adjusted to maximize fuel temperature and core stored energy. The AXIOM thermal material properties, as provided in Section 3.2 of WCAP-18546-P-A, are similar to other zirconium-based cladding materials, such that the long-term LOCA mass and energy releases are not affected by the use of AXIOM cladding.

The short-term steam line break (SLB) M&Es are performed with Westinghouse methods, which are evaluated as being unaffected by the use of AXIOM cladding per Section 6.2.3.3 of WCAP-18546-P-A, and Section 3.8.5.2 of the corresponding SE.

For long-term SLB M&E release calculations, Duke Energy has licensed the methodology described in DPC-NE-3004-P-A, used for containment integrity and equipment qualification at CNS and MNS. It is conservative for the long term SLB M&E to maximize the rate of transfer of energy from the core into the primary coolant, and maximize primary-to-secondary heat transfer for steam releases out of the secondary-side break. Fuel thermal performance characteristics are adjusted to maximize fuel temperature and core stored energy. The AXIOM thermal material properties, as provided in Section 3.2 of WCAP-18546-P-A, are similar to other zirconium-based cladding materials, such that the long-term SLB mass and energy releases are not affected by the use of AXIOM cladding.

- Radiological Consequences Analyses

Implementation of AXIOM fuel rod cladding will have no impact on models and method used in performing offsite and control room radiological dose consequences analyses for accidents. Radiological consequence analysis does not model cladding. Change of cladding material could impact input to accident radiological consequences. Radiological consequences analyses consider the extent of fuel cladding damage resulting from postulated accidents. The analysis would be incorporated in a plant specific analysis using methods consistent with the analysis of record.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements and Guidance

10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors

10 CFR 50.46 requires, in part, that the calculated ECCS performance for light-water nuclear power reactors with zircaloy or ZIRLO fuel cladding meet certain criteria set forth in 10 CFR 50.46(b). Enclosure 5 of this submittal contains an exemption request being submitted in accordance with 10 CFR 50.12, requesting exemption from certain requirements of 10 CFR 50.46 to allow use of the AXIOM alloy for fuel rod cladding.

As it relates to the proposed change to adopt the FSLOCA EM in WCAP-16996-P-A, the FSLOCA EM satisfies the requirements of 10 CFR 50.46(b) paragraphs (1) through (4), which require, in part, that during a LOCA event, the following criteria are satisfied:

(1) Peak cladding temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.
(2) Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
(3) Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
(4) Coolable geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

The proposed changes either continue to meet the requirements of this regulation or an exemption is justified as described in Enclosure 5.

10 CFR 50.36, Technical specifications

The NRC's regulatory requirements related to the content of the TS are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TS include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant’s TS. 10 CFR 50.36(c)(4) states in part, that design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in other categories of the regulation. The proposed changes continue to meet the requirements of this regulation.

10 CFR 50.36(c)(5) states, “Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the

Commission pursuant to approved technical specifications as specified in § 50.4.” The proposed changes continue to meet the requirements of this regulation.

10 CFR Part 50, Appendix A, General Design Criteria (GDC) 10, 15, and 35

10 CFR 50, Appendix A, GDC 10 (Reactor design) states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

10 CFR 50, Appendix A, GDC 15 (Reactor coolant system design) states that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

10 CFR 50, Appendix A, GDC 35 (Emergency core cooling) states that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. Conformance with GDC 35 is described in more detail in the FSLOCA topical report, WCAP-16996-P-A, Revision 1.

The proposed changes do not affect compliance with these regulations or guidance and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

Conclusion

The proposed change ensures plant compliance with the above regulations and guidance and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

4.2 Precedents

AXIOM

The proposed amendment requests a revision to the TS to update the description of fuel assemblies specified in TS 4.2.1, “Fuel Assemblies,” and add Westinghouse Topical Report WCAP-18546-P-A to the referenced analytical methods in TS 5.6.5.b to allow the use of AXIOM alloy for fuel rod cladding. A corresponding exemption is being requested from the provisions of 10 CFR 50.46 in order to support the use of this additional fuel rod cladding material. The NRC previously issued a license amendment to Turkey Point Nuclear Generating, Unit Nos. 3 and 4 (Turkey Point), by letter dated February 12, 2025 (ADAMS Accession No. ML25043A428) that addressed revisions to the licensing basis to incorporate AXIOM cladding. The Turkey Point submittal also addressed several other changes that are not included within the scope of this request, including incorporation of additional advanced fuel features for pellets and fuel skeleton, application of an updated instrument channel uncertainty evaluation methodology, and changes to facilitate the transition to 24-month fuel cycles.

FSLOCA

The proposed amendment requests the addition of Westinghouse Topical Report WCAP-16996-P-A, Revision 1, to the list of approved analytical methods used to determine the core operating limits provided in TS 5.6.5 for MNS and CNS U1. The NRC has previously issued license amendments for the following sites that have similarly requested approval to include this topical report within their respective TS COLR reference lists:

- Comanche Peak Nuclear Power Plant, Unit Nos. 1 and 2 (Comanche Peak), Amendment Nos. 185 and 185 to Renewed Facility Operating License Nos. NPF-87 and NPF-89, respectively, per letter dated December 20, 2023 (ADAMS Accession No. ML23319A387)
- Beaver Valley Power Station, Unit Nos. 1 and 2 (Beaver Valley), Amendment Nos. 322 and 212 to Renewed Facility Operating License Nos. DPR-66 and NPF-73, respectively, per letter dated October 2, 2023 (ADAMS Accession No. ML23198A359). The Beaver Valley license amendment also added a note to the LOCA methods listed in the TS COLR list of references to restrict their future use.
- Turkey Point Nuclear Generating Unit Nos. 3 and 4, Amendment Nos. 296 and 289 to Renewed Facility Operating License Nos. DPR-31 and DPR-41, respectively, per letter dated May 24, 2022 (ADAMS Accession No. ML22028A066).

The Comanche Peak and Beaver Valley license amendments also addressed the removal of Zircalloy from the list of fuel rod cladding in TS 4.2.1, “Fuel Assemblies,” which is beyond the scope of this request for MNS and CNS U1. Additionally, the Comanche Peak license amendment addressed a change to TS Safety Limit (SL) 2.1.1.2 in “Reactor Core SLs” to reflect the peak fuel centerline melt temperature specified in the Westinghouse performance analysis and design model (PAD5). This particular change is under NRC review for MNS and CNS in a separate submittal (ADAMS Accession No. ML25070A183) and is outside the scope of this license amendment request.

In reviewing the associated submittals for the precedents above, it was determined that additional information was needed as it (1) related to how FSLOCA plant input parameters correspond to TS Limiting Condition for Operation values, and (2) how FSLOCA plant input parameters may differ from previous inputs used in existing LBLOCA analyses. This information for MNS and CNS U1 can be found in Attachments 3 and 4 of this enclosure.

4.3 Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TS) for Catawba Nuclear Station (CNS), Unit 1 (CNS U1) and McGuire Nuclear Station, Units 1 and 2 (MNS). The proposed amendment requests the addition of the Westinghouse Electric Company LLC (Westinghouse) topical report WCAP-

16996-P-A, Revision 1, “Realistic LOCA [Loss of Coolant Accident] Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM^{TM} LOCA Methodology)” (ADAMS Accession No. ML17277A130), to the list of approved analytical methods used to determine the core operating limits provided in TS 5.6.5, “Core Operating Limits Report (COLR).” Additionally, this amendment proposes the annotation of select legacy LOCA methods and the deletion of others listed in TS 5.6.5.b to restrict their future use and allow for a staggered implementation during refueling outages at each unit.

The amendment also requests a modification to the TS to permit the use of the Westinghouse fuel cladding alloy designated as AXIOM®. Specifically, the proposed amendment requests a revision to the TS to update the description of fuel assemblies specified in TS 4.2.1, “Fuel Assemblies,” and add the Westinghouse topical report WCAP-18546-P-A, “Westinghouse AXIOM Cladding for use in Pressurized Water Reactor Fuel” (ADAMS Accession No. ML23089A063) to the referenced analytical methods in TS 5.6.5.b to allow the use of AXIOM alloy for fuel rod cladding. A corresponding exemption is being requested from the provisions of 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” in order to support the use of this additional fuel rod cladding material. The exemption request for CNS U1 and MNS is provided in Enclosure 5 of this submittal.

A separate license amendment request to address the adoption of the aforementioned Westinghouse Topical Reports for the FULL SPECTRUM LOCA (FSLOCA) Evaluation Methodology (EM) and AXIOM fuel rod cladding for CNS Unit 2 (CNS U2) will be submitted at a later date. Inclusion of notes within the associated CNS TS are proposed to reflect applicability of the requested license amendment changes to only CNS U1. Deletion of legacy LOCA analysis methods is administrative and is applicable to both CNS U1 and CNS U2.

Duke Energy has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

(1) *Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?*

Response: No.

The proposed amendment involves a change to utilize the FSLOCA EM, along with the implementation of associated TS changes. The proposed change to TS 5.6.5 permits the use of an NRC-approved methodology for analysis of the LOCAs to determine whether MNS and CNS U1 continue to meet the applicable design and safety analysis acceptance criteria. Restrictions on the future use of legacy LOCA methods are listed in TS 5.6.5.b in order to allow for a staggered implementation during refueling outages at each unit and has no direct impact upon plant operation or configuration. The results of the composite LOCA analyses for MNS and CNS U1 demonstrate the continued fulfilment of the 10 CFR 50.46(b)(1-4) emergency core cooling system (ECCS) performance acceptance criteria using an NRC-approved EM. These changes do not alter plant equipment nor the manner in which equipment is operated and maintained.

The proposed amendment also involves a change to TS 4.2.1 to allow the use of AXIOM-clad nuclear fuel by MNS and CNS U1, and a change to TS 5.6.5 to add the

NRC-approved topical report WCAP-18546-P-A for AXIOM fuel rod cladding to the COLR references. This topical report demonstrates that AXIOM fuel rod cladding has essentially the same properties as Optimized ZIRLO^{TM} fuel rod cladding. Use of AXIOM fuel rod cladding material will not result in adverse changes to the operation or configuration of the facility. The fuel cladding is not an accident initiator and does not affect accident probability. Use of AXIOM cladding meets the fuel design acceptance criteria and therefore does not significantly affect the consequences of an accident.

The proposed changes do not impact either the initiation of an accident or the mitigation of its consequences. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational or public radiation exposure. As a result, the outcomes of accidents previously evaluated are unaffected.

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

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment involves a change to utilize the FSLOCA EM, along with the implementation of associated TS changes and restrictions on the future use of legacy LOCA methods. These changes will not create the possibility of a new or different accident due to credible new failure mechanisms, malfunctions, or accident initiators not previously considered. No physical plant modifications are being made as a result of the change to utilize the FSLOCA EM.

The proposed amendment also involves a change to TS to allow the use of AXIOM-clad nuclear fuel by MNS and CNS U1 and the addition of the NRC-approved topical report for AXIOM fuel rod cladding to the COLR references. This topical report demonstrates that AXIOM fuel rod cladding has essentially the same properties as Optimized ZIRLO fuel rod cladding. In performing similarly to the current fuel rod cladding, this precludes the possibility of the fuel rod cladding becoming an accident initiator and causing a new or different kind of accident.

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated in the Updated Final Safety Analysis Report. No new single failure mechanisms will be created as a result of the proposed changes, and there are no alterations to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors.

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

(3) *Does the proposed amendment involve a significant reduction in a margin of safety?*

Response: No.

The proposed amendment involves a change to utilize the FSLOCA EM, along with the implementation of associated TS changes and restrictions on the future use of legacy LOCA methods. The analytic technique to be used in the analysis realistically describes the expected behavior of the MNS and CNS U1 reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46, and analysis shows that there is a high level of probability that all criteria contained in 10 CFR 50.46(b) are met. Approved methodologies would continue to be used to ensure that MNS and CNS U1 continue to meet applicable design criteria and safety analysis acceptance criteria.

The proposed amendment also involves a change to TS to allow the use of AXIOM-clad nuclear fuel by MNS and CNS U1 and the addition of the NRC-approved topical report for AXIOM fuel rod cladding to the COLR references. This topical report demonstrates that the material properties of the AXIOM fuel rod cladding are similar to the currently utilized Optimized ZIRLO fuel rod cladding. AXIOM is expected to perform similarly to Optimized ZIRLO for all normal operating and accident scenarios, including both loss of coolant accident (LOCA) and non-LOCA scenarios. The use of AXIOM fuel rod cladding will not result in adverse changes to the operation or configuration of the facility. The proposed changes do not affect the acceptance criteria for any UFSAR safety analysis analyzed accidents or anticipated operational occurrences. All required safety limits will continue to be analyzed using methodologies approved by the NRC. This ensures that applicable design and performance criteria associated with the safety analysis will continue to be met and that the margin of safety is not affected.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based upon the above evaluation, Duke Energy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATIONS

Duke Energy has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined by 10 CFR 20, “Standards for protection against radiation,” or it would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs be prepared in connection with the proposed amendment.

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

6 PAGES PLUS THE COVER

4.0 DESIGN FEATURES

4.1 Site Location

Catawba Nuclear Station is located in the north central portion of South Carolina approximately six miles north of Rock Hill and adjacent to Lake Wylie. The station center is located at latitude 35 degrees, 3 minutes, 5 seconds north and longitude 81 degrees, 4 minutes, 10 seconds west. The corresponding Universal Transverse Mercator Coordinates are E 493, 660 and N 3, 878, 558, zone 17.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of either Zircalloy, ZIRLO®, ~~or~~ Optimized ZIRLO™ , or AXIOM® (Unit 1 only) clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material.* Limited substitutions of ZIRLO®, Optimized ZIRLO™, AXIOM® (Unit 1 only), zirconium alloy, or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

* A maximum of four lead assemblies containing mixed oxide fuel and M5™ cladding may be inserted into the Unit 1 or Unit 2 reactor core.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver indium cadmium and boron carbide as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY" (W Proprietary).
2. ~~WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE" (W Proprietary).~~ Deleted.
3. ~~BAW-10168-P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants" (B&W Proprietary).~~ Deleted.
4. DPC-NE-2011-P-A, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors" (DPC Proprietary).
5. DPC-NE-3001-P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology" (DPC Proprietary).
6. DPC-NF-2010-A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design."
7. DPC-NE-3002-A, "FSAR Chapter 15 System Transient Analysis Methodology."
8. DPC-NE-3000-P-A, "Thermal-Hydraulic Transient Analysis Methodology" (DPC Proprietary).
9. DPC-NE-1004-A, "Design Methodology Using CASMO-3/SIMULATE-3P."
10. DPC-NE-2004-P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01" (DPC Proprietary).
11. DPC-NE-2005-P-A, "Thermal Hydraulic Statistical Core Design Methodology" (DPC Proprietary).

(continued)

[Shall not be used to determine core operating limits after startup of Catawba Unit 1 Cycle 32.]

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

12. DPC-NE-2008-P-A, "Fuel Mechanical Reload Analysis Methodology Using TACO3" (DPC Proprietary).

13. WCAP-10054-P-A, “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code” (W Proprietary).

14. DPC-NE-2009-P-A, “Westinghouse Fuel Transition Report” (DPC Proprietary).

15. WCAP-12945-P-A, Volume 1 and Volumes 2-5, “Code Qualification Document for Best-Estimate Loss of Coolant Analysis” (W Proprietary).

16. DPC-NE-1005P-A, “Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,” (DPC Proprietary).

17. ~~BAW-10231P-A, “COPERNIC Fuel Rod Design Computer Code,” (Framatome ANP Proprietary).~~

Deleted.

18. DPC-NE-1007-PA, “Conditional Exemption of the EOC MTC Measurement Methodology” (Duke and Westinghouse Proprietary).

INSERT 21 and 22 from below.

19. WCAP-12610-P-A, “VANTAGE+ Fuel Assembly Reference Core Report,” April 1995 (Westinghouse Proprietary).

20. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, “Optimized ZIRLO™,” July 2006 (Westinghouse Proprietary).

The COLR will contain the complete identification for each of the Technical Specifications referenced topical reports used to prepare the COLR (i.e., report number, title, revision number, report date or NRC SER date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

21. WCAP-16996-P-A, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)" (Westinghouse Proprietary). [For use by Unit 1 only.]

22. WCAP-18546-P-A, "Westinghouse AXIOM® Cladding for Use in Pressurized Water Reactor Fuel" (Westinghouse Proprietary). [For use by Unit 1 only.]

inued)

4.0 DESIGN FEATURES

4.1 Site Location

The McGuire Nuclear Station site is located at latitude 35 degrees, 25 minutes, 59 seconds north and longitude 80 degrees, 56 minutes, 55 seconds west. The Universal Transverse Mercator Grid Coordinates are E 504, 669, 256, and N 3, 920, 870, 471. The site is in northwestern Mecklenburg County, North Carolina, 17 miles north-northwest of Charlotte, North Carolina.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of either Zircalloy, ZIRLO®, ~~or~~ Optimized ZIRLO™, or AXIOM® clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of ZIRLO®, Optimized ZIRLO™, AXIOM®, zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver indium cadmium (Unit 1) silver indium cadmium and boron carbide (Unit 2) as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.00 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 800 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," (W Proprietary).

2. ~~WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," (W Proprietary).~~

Deleted.

3. ~~BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," (B& W Proprietary).~~

Deleted.

4. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," (DPC Proprietary).

5. DPC-NE-3001PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," (DPC Proprietary).

6. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design ".

7. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology".

8. DPC-NE-3000PA, "Thermal-Hydraulic Transient Analysis Methodology ," (DPC Proprietary).

9. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO - 3/SIMULATE-3 P ".

10. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," (DPC Proprietary).

11. DPC-NE-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," (DPC Proprietary).

12. DPC-NE-2008P-A, "Fuel Mechanical Reload Analysis Methodology Using TACO3," (DPC Proprietary).

13. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code," (W Proprietary)

Add: [Shall not be used to determine core operating limits after startup of McGuire Unit 1 Cycle 34 and McGuire Unit 2 Cycle 34.]

(continued)

Add: [Shall not be used to determine core operating limits after startup of McGuire Unit 1 Cycle 34 and McGuire Unit 2 Cycle 34.]

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

14. DPC-NE-2009-P-A, "Westinghouse Fuel Transition Report," (DPC Proprietary).

15. WCAP-12945-P-A, Volume 1 and Volumes 2-5, " Code Qualification Document for Best-Estimate Loss of Coolant Analysis," (W Proprietary).

16. DPC-NE-1005P-A, "Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," (DPC Proprietary).

17. DPC-NE-1007-PA, "Conditional Exemption of the EOC MTC Measurement Methodology" (Duke and Westinghouse Proprietary).

INSERT 20 and 21 from below.

18. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995. (Westinghouse Proprietary).

19. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006. (Westinghouse Proprietary).

The COLR will contain the complete identification for each of the Technical Specifications referenced topical reports used to prepare the COLR (i.e., report number, title, revision number, report date or NRC SER date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Deleted

5.6.7 PAM Report

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

20. WCAP-16996-P-A, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)" (Westinghouse Proprietary).

21. WCAP-18546-P-A, "Westinghouse AXIOM® Cladding for Use in Pressurized Water Reactor Fuel" (Westinghouse Proprietary).

(continued)

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES (MARK-UP)

80 PAGES PLUS THE COVER

(FOR INFORMATION ONLY)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (~~Ref. 2~~).

Ref. 2 and Ref. 8

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. The unit has four control banks and five shutdown banks.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.3.
4. UFSAR, Section 15.4.
5. UFSAR, Section 4.3.1.5.
6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
7. UFSAR, Section 15.0.

8. WCAP-18546-P-A, March 2023.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (~~Ref. 2~~ Ref. 2 and Ref. 5). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. The plant has four control banks and five shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The shutdown banks must be maintained above designed shutdown bank insertion limits and are typically near the fully withdrawn position during normal full power operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation , the shutdown

ACTIONS (continued)

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.
4. 10 CFR 50.36, Technical Specification, (c)(2)(ii).

5. WCAP-18546-P-A, March 2023.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (~~Ref. 2~~ Ref. 2 and Ref. 7). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. The plant has four control banks and five shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

The control banks are moved in an overlap pattern. When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The fully withdrawn position is defined in the COLR.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.1.
4. UFSAR, Section 15.0.
5. UFSAR, Section 15.4.8.
6. 10 CFR 50.36, Technical Specification, (c)(2)(ii).

7. WCAP-18546-P-A, March 2023.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

10 CFR 50.46 acceptance criteria (Ref. 1 and Ref. 6) must be met

a. During a loss of coolant accident (LOCA), the ~~peak cladding temperature must not exceed 2200°F for small breaks and there is a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1~~);

b. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(X,Y,Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. ~~Other Reference 1 criteria must also be met in LOCAs (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, transient strain, and long term cooling). However, the peak cladding temperature is typically most limiting.~~

$F_Q(X,Y,Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(X,Y,Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(X,Y,Z)$ satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The Heat Flux Hot Channel Factor, $F_Q(X,Y,Z)$, shall be limited by the following relationships:

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

BASES

LCO (continued)

ensure that the 10 CFR 50.46 acceptance criteria (Ref. 1 and Ref. 6) are met

The $F_Q(X,Y,Z)$ limits typically define limiting values for core power peaking that ~~precludes peak cladding temperatures above 2200°F during a small break LOCA and a high level of probability that the peak cladding temperature does not exceed 2200°F~~ for a ~~large break~~ LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the $F_Q(X,Y,Z)$ limits. If $F_Q(X,Y,Z)$ cannot be maintained within the steady state LOCA limits, reduction of the core power is required.

Violating the steady state LOCA limits for $F_Q(X,Y,Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(X,Y,Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(X,Y,Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power. The exception to this is the steam line break event, which is assumed for analysis purposes to occur from very low power levels. At these low power levels, measurements of $F_Q(X,Y,Z)$ are not sufficiently reliable. Operation within analysis limits at these conditions is inferred from startup physics testing verification of design predictions of core parameters in general.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F^M_Q(X,Y,Z)$ exceeds its steady state limit, maintains an acceptable absolute power density. $F^M_Q(X,Y,Z)$ is the measured value of $F_Q(X,Y,Z)$ and the steady state limit includes factors accounting for measurement uncertainty and manufacturing tolerances. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels $\geq$ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".

6. WCAP-18546-P-A, March 2023.

BACKGROUND (continued)

uncontrolled RCCA bank withdrawal (UCBW). For these types of accidents, the event itself causes changes in the power distribution and this LCO alone is not sufficient to preclude DNB. The acceptability of analyses such as the UCBW accident analysis is ensured by LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” LCO 3.1.6, “Control Bank Insertion Limits,” LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits,” in combination with cycle-specific analytical calculations.”

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs.

APPLICABLE SAFETY ANALYSES

Limits on $F_{\Delta H}(X,Y)$ preclude core power distributions that exceed the following fuel design limits:

a. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

b. During a ~~large break~~ loss of coolant accident (LOCA), ~~there must be a high level of probability that the peak cladding temperature (PCT) does not exceed 2200°F~~;

the 10 CFR 50.46 acceptance criteria (Ref. 3 and Ref. 6) must be met

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F_{\Delta H}(X,Y)$ are the core parameters of most importance. The limits on $F_{\Delta H}(X,Y)$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency that do not alter the core power distribution. For transients such as uncontrolled RCCA bank withdrawal, which are characterized by changes in the core power distribution, this LCO alone is not sufficient to preclude DNB. The acceptability of the accident analyses

BASES

APPLICABLE SAFETY ANALYSES (continued)

is ensured by LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” LCO 3.1.6, “Control Bank Insertion Limits,” LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” and LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits,” in combination with cycle-specific analytical calculations. The DNB design basis is met by limiting the minimum DNBR to the design limit value using an NRC approved CHF correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F_{\Delta H}$(X,Y) limit increases with decreasing power level. This functionality in $F_{\Delta H}$(X,Y) is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F_{\Delta H}$(X,Y) in the analyses.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^{N}_{\Delta H}$), the Nuclear Heat Flux Hot Channel Factor (F_Q(Z)), and the axial peaking factors are supported by the LOCA safety analyses that verify compliance with the 10 CFR 50.46 acceptance criteria (Ref. 3 and Ref. 6).

~~The LOCA safety analysis models $F_{\Delta H}$(X,Y) as an input parameter. The Nuclear Heat Flux Hot Channel Factor (F_Q(X,Y,Z)) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).~~

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor (F_Q(X,Y,Z))."

$F_{\Delta H}$(X,Y) and F_Q(X,Y,Z) are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Control Bank Insertion Limits.

$F_{\Delta H}$(X,Y) satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

$F_{\Delta H}$(X,Y) shall be limited by the following relationship:

$$F^{M}_{\Delta H}(X,Y) \le F^{L}_{\Delta H}(X,Y)^{LCO}$$

BASES

LCO (continued)

where: $F^{M}_{\Delta H}(X,Y)$ is defined as the measured radial peak, and

$F^{L}_{\Delta H}(X,Y)^{LCO}$ is defined as the steady state maximum allowable radial peak defined in the COLR.

The $F^{L}_{\Delta H}(X,Y)^{LCO}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB.

$F^{L}_{\Delta H}(X,Y)^{LCO}$ limits are maximum allowable radial peak (MARP) limits which are developed in accordance with the methodology outlined in Reference 5. MARP limits are constant DNBR limits which are a function of both the magnitude and location of the axial peak F(Z), therefore, justifying the X,Y dependence of the $F^{L}_{\Delta H}(X,Y)^{LCO}$ limit.

The limiting value, $F^{L}_{\Delta H}(X,Y)^{LCO}$, is also power dependent and can be described by the following relationship:

$$F^{L}_{\Delta H}(X,Y)^{LCO} = MARP(X,Y) * \left[1.0 + (1/RRH) * (1.0 - P)\right]$$

where: MARP(X,Y) is the maximum allowable radial peaks provided in the COLR,

P is the ratio of THERMAL POWER to RATED THERMAL POWER, and

RRH is the amount by which allowable THERMAL POWER must be reduced for each 1% that $F^{M}_{\Delta H}(X,Y)$ exceeds the limit. The specific value is contained in the COLR.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value, $F^{L}_{\Delta H}(X,Y)^{LCO}$, is allowed to increase approximately 0.3% for every 1% RTP reduction in THERMAL POWER. This increase in the $F^{L}_{\Delta H}(X,Y)^{LCO}$ limit is due to the reduced amount of heat removal required at lower powers.

APPLICABILITY

The $F_{\Delta H}(X,Y)$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and ~~PCT~~. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being

to ensure the 10 CFR 50.46 acceptance criteria (Ref. 3 and Ref. 6) are met

REFERENCES (continued)

3. 10 CFR 50.46.

4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

5. DPC-NE-2004P-A, Rev. 1, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydaulic Methodology Using VIPRE-01," SER Dated February 20, 1997 (DPC Proprietary)

6. WCAP-18546-P-A, March 2023.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

the 10 CFR 50.46 acceptance criteria (Ref. 1 and Ref. 5) must be met

a. During a ~~large break~~ loss of coolant accident (LOCA), ~~there must be a high level of probability that the peak cladding temperature does not exceed 2200°F (Ref. 1)~~;

b. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}(X,Y)$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. WCAP-18546-P-A, March 2023.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for ~~limiting peak clad temperature to < 2200°F~~); and

meeting the 10 CFR 50.46 acceptance criteria (Ref. 16 and Ref. 17)

2. Boration to ensure recovery and maintenance of SDM ($k_{eff} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation;
- Containment Purge and Exhaust Isolation;
- Reactor Trip;
- Turbine Trip;
- Feedwater Isolation;
- Start of motor driven auxiliary feedwater (AFW) pumps;
- Start of control room area ventilation filtration trains;
- Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) suction to containment sump;
- Start of annulus ventilation system filtration trains;
- Start of auxiliary building filtered ventilation exhaust system trains;
- Start of diesel generators

REFERENCES (continued)

15. Attachment 1 to TSTF-569, Rev. 2, “Methodology to Eliminate Pressure Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing.”

16. 10 CFR 50.46.

17. WCAP-18546-P-A, March 2023.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

large break

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

large break

Initial accumulator inventory which is injected into the reactor vessel is lost out the break.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above the permissive circuit P-11 setpoint.

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. The valves will automatically open, however, as a result of an SI signal. The isolation valves between the accumulators and the Reactor Coolant System are required to be open

BACKGROUND (continued)

and power removed during unit operation. In that the subject valves are normally open and do not serve as an active device during a LOCA, the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) is not applicable in this situation. Therefore, the subject valve control circuit is not designed to this standard.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

large break

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The largest break area considered for a large break LOCA is a double ended guillotine break in the RCS cold leg.

~~The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump.~~ During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(for loss of offsite power assumption)

for a large break LOCA

in the modeling

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting, the valves opening, and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed ~~during the blowdown stage of a large break LOCA.~~

APPLICABLE SAFETY ANALYSES (continued)

intermediate

is assumed to inject into the RCS

The ~~worst case~~ small break LOCA analyses also assume a time delay before pumped flow ~~reaches the core~~. For ~~the larger range of small~~ breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators, safety injection pumps, and centrifugal charging pumps all play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

acceptance criteria (Ref. 3 and Ref. 8) are met.

This LCO helps to ensure that the ~~following acceptance criteria established for the ECCS by~~ 10 CFR 50.46 ~~(Ref. 3) will be met following a small break LOCA and there is a high level of probability that the criteria are met following a large break LOCA:~~

~~a.~~ ~~Maximum fuel element cladding temperature is ≤ 2200°F;~~

~~b.~~ ~~Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;~~

~~c.~~ ~~Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and~~

~~d.~~ ~~Core is maintained in a coolable geometry.~~

large break LOCA and the recovery phase of a small break LOCA

Since the accumulators discharge during the blowdown phase of a ~~LOCA~~, they do not contribute directly to the long term cooling requirements of 10 CFR 50.46. However, the boron content of the accumulator water helps to maintain the reactor core subcritical after reflood, thereby eliminating fission heat as an energy source for which cooling must be provided.

For Unit 1, the large and small break LOCA analyses use a range of accumulator water volumes of 6790 gallons to 7422 gallons per the approved method (Ref. 9). For Unit 2, the large break LOCA analysis uses a range of accumulator water volumes of 7550 gallons to 8159 gallons, and the small break LOCA analysis uses a nominal accumulator water volume of 7855 gallons, per approved methods.

Both large and small break LOCA analyses use a nominal accumulator line volume from the accumulator to the check valve.

~~For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used.~~ The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. ~~The large and small break LOCA analyses are performed with accumulator volumes that are consistent with the LOCA evaluation models. To allow for operating margin, values of ± 30 ft^3 are specified.~~

The minimum boron concentration setpoint is used in the post LOCA subcriticality verification during the injection phase. For each reload

APPLICABLE SAFETY ANALYSES (continued)

cycle, the all rods out (ARO) critical boron concentration is verified to be less than the minimum allowed cold leg accumulator boron concentration.

The minimum boron concentration setpoint is also used in the post LOCA sump boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment with all rods in, minus the highest worth rod out (ARI N-1). Of particular interest is the large cold leg break LOCA, since boron accumulation in the core will be maximized during the cold leg recirculation phase due to core boiling. The accumulation of boron in the core prevents the boron from returning to the sump, which leads to a boron diluted sump condition which may cause the core to become re-critical when switching over to hot leg recirculation. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH. In particular, the equilibrium sump pH should be at least 7.5 following the design basis LOCA.

For Unit 1, the large break LOCA analysis uses a range of accumulator nitrogen cover pressures of 555 psig to 708 psig, and the small break LOCA analysis uses a range of accumulator nitrogen cover pressures of 555 psig to 669 psig, per the approved method (Ref. 9). For Unit 2, the large break LOCA analysis uses a range of accumulator nitrogen cover pressures of 555 psig to 708 psig, and the small break LOCA analysis uses a minimum accumulator nitrogen cover pressure of 555 psig, per approved methods.

~~The large and small break LOCA analyses are performed with accumulator pressures that are consistent with the LOCA evaluation models. To allow for operating margin and accumulator design limits, a range from 585 psig to 678 psig is specified.~~ The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Ref. 4).

The accumulators satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (~~Ref. 3~~) could be violated.

large break

Ref. 3 and Ref. 8

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and

LCO (continued)

nitrogen cover pressure must be met. Additionally, the nitrogen and liquid volumes between accumulators must be physically separate.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

the 10 CFR 50.46 acceptance criteria (Ref. 3 and Ref. 8) are met

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure ~~that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F for small break LOCAs and there is a high level of probability that the peak cladding temperature does not exceed 2200°F for large break LOCAs~~.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are allowed to be closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break for the plant. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.5

Verification that power is removed from each accumulator isolation valve operators for NI54A, NI65B, NI76A, and NI88B when the RCS pressure is > 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is ≤ 1000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. IEEE Standard 279-1971.
2. UFSAR, Chapter 6.
3. 10 CFR 50.46.
4. DPC-NE-3004.
5. 10 CFR 50.36, Technical Specification, (c)(2)(ii).
6. WCAP-15049-A, Rev. 1, April 1999.
7. NUREG-1366, February 1990.

8. WCAP-18546-P-A, March 2023.

9. WCAP-16996-P-A, Rev. 1, November 2016.

BASES

BACKGROUND (continued)

The high and intermediate head subsystems of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a safety injection actuation.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the ~~following~~ acceptance criteria for the ECCS, established by 10 CFR 50.46 ~~(Ref. 2), will be met following a small break LOCA and there is a high level of probability that the criteria are met following a large break LOCA:~~

(Ref. 2 and
Ref. 8) are met.

~~a.~~ ~~Maximum fuel element cladding temperature is ≤ 2200°F;~~

~~b.~~ ~~Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;~~

~~c.~~ ~~Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;~~

APPLICABLE SAFETY ANALYSES (continued)

~~d. Core is maintained in a coolable geometry; and~~

~~e. Adequate long term core cooling capability is maintained.~~

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment pressure and temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event has the greatest potential to challenge the limits on runout flow set by the manufacturer of the ECCS pumps. It also sets the maximum response time for their actuation. Direct flow from the centrifugal charging pumps and SI pumps is credited in a small break LOCA event. The RHR pumps are also credited, for larger small break LOCAs, as the means of supplying suction to these higher head ECCS pumps after the switch to sump recirculation. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The MSLB analysis also credits the SI and centrifugal charging pumps. Although some ECCS flow is necessary to mitigate a SGTR event, a single failure disabling one ECCS train is not the limiting single failure for this transient. The SGTR analysis primary to secondary break flow is increased by the availability of both centrifugal charging and SI trains. Therefore, the SGTR analysis is penalized by assuming both ECCS trains are operable as required by the LCO. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

or without

a. A large break LOCA event, with loss of offsite power and a single failure disabling one ECCS train; and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Ref. 3). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA.

SURVEILLANCE REQUIREMENTS (continued)

Inspections will consist of a visual examination of the exterior surfaces of the strainer assembly for any evidence of debris, structural distress or abnormal corrosion. The intent of this surveillance is to ensure the absence of any condition which could adversely affect strainer functionality. Surveillance performance does not require removal of any tophat modules or grating, but the strainer exteriors shall be visually inspected. This surveillance is not a commitment to inspect 100% of the surface area of all tophats, but a sufficiently detailed inspection of exterior strainer surfaces is required to establish a high confidence that no adverse conditions are present. The scope of inspection necessary to provide high confidence includes 100% of the strainer areas that can be accessed and inspected using normal means and tools (i.e., flashlight, extendable mirror, hand held digital camera) without disassembly, and that difficult to access areas will be inspected to the extent possible using these same means.

Any damage detected in the strainer assembly inspection will result in an expansion of the scope of the inspection to include other areas of potential damage. Inspection scope should be expanded, as needed, for degradation of strainer components identified during this inspection that were not considered readily accessible during the inspector's initial evaluation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. UFSAR, Section 6.2.1.
4. UFSAR, Chapter 15.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
7. IE Information Notice No. 87-01.

8. WCAP-18546-P-A, March 2023.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

This LCO is applicable only to those units that utilize the centrifugal charging pumps for safety injection (SI). The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but do not set the limits on their flow requirements. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

and the intermediate head safety injection pumps

This LCO ensures that seal injection flow of ≤ 40 gpm, with centrifugal charging pump operating and charging flow control valve full open, will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical ~~for a large LOCA~~. For smaller LOCAs, the charging pumps ~~alone~~ deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

and the intermediate head safety injection pumps

and the intermediate head safety injection pumps

BASES

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (~~Ref. 3~~).

Ref. 3 and Ref. 4

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a minimum pressure differential and flow must be known. The flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure and that the centrifugal charging pump discharge pressure is greater than or equal to the applicable value specified in the test acceptance criteria. Since the test acceptance criteria head curve ensures the centrifugal charging pumps are capable of delivering the flow assumed in the LOCA analyses, the minimum pressure differential is satisfied by verifying the centrifugal charging pump is operating. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed minimum pressure differential result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the charging flow control valve being full open, is required since the valve is designed to fail open unless motive air is available. With the operating pump and control valve position as specified by the LCO, a flow limit is established. It is this flow limit that is used in the accident analyses.

The limit on seal injection flow, combined with the minimum pressure differential and an open wide condition of the charging flow control valve, must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow might not be as much as assumed in the accident analyses.

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTIONS

A.1

With the seal injection flow exceeding its limit, the amount of high head safety injection flow available to the RCS may be reduced. Under this Condition, action must be taken to restore the flow to below its limit. The

BASES

REFERENCES

1. UFSAR, Chapter 6 and Chapter 15.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. 10 CFR 50.46.

4. WCAP-18546-P-A, March 2023.

BASES

APPLICABLE SAFETY ANALYSES (continued)

There are five conditions which have a potential for resulting in a net external pressure on the containment:

1. Rupture of a hot or high pressure process pipe in the annulus.
2. Inadvertent Containment Spray System initiation during normal operation.
3. Inadvertent containment air return fan initiation during normal operation.
4. Containment purge fan operation with containment purge inlet valves closed.
5. Containment air release fan pressure controller failure resulting in fan not shutting off properly.

The containment design of 1.5 psig negative is not exceeded in the first four conditions due to either equipment limitations or design features, but may be exceeded in the fifth case. The containment air release fan is capable of pulling a negative pressure in containment beyond design limits if allowed to run unchecked. Administrative controls prevent this event from occurring. The Operator Aid Computer (OAC) response to a low pressure alarm ensures a Containment Air Release and Addition System release is terminated, if in progress, and in cases where the OAC is inoperable, the pressure is monitored on a 30-minute frequency during releases. These controls are utilized to ensure the pressure stays within the Technical Specification limit of 0.1 psig negative.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations ~~Therefore, for the reflood phase~~, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with the approved method ~~10 CFR 50, Appendix K~~ (Ref. 2).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.
2. ~~10 CFR 50, Appendix K.~~ WCAP-16996-P-A, Rev. 1, November 2016. (Unit 1)

   WCAP-12945-P-A, March 1998. (Unit 2)
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the ARS being rendered inoperable (Ref. 2).

The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and was calculated to be within the containment environmental qualification temperature during the DBA SLB. The basis of the containment environmental qualification temperature is to ensure the OPERABILITY of safety related equipment inside containment (Ref. 3).

The Containment Spray System actuation modeled in the containment analysis is based on the time associated with reaching the RWST low level setpoint prior to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The Containment Spray System total response time is composed of operator action delay and system startup time.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with ~~10 CFR 50, Appendix K~~ the approved method (Ref. 4).

Inadvertent actuation is precluded by a design feature consisting of an additional set of containment pressure sensors which prevents operation when the containment pressure is below the containment pressure control system permissive.

The Containment Spray System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5).

SURVEILLANCE REQUIREMENTS (continued)

Accumulated gas should be eliminated or brought within the acceptance criteria limits.

Containment Spray System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. UFSAR, Section 6.2.
3. 10 CFR 50.49.
4. ~~10 CFR 50, Appendix K.~~ WCAP-12945-P-A, March 1998. (Unit 2)
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. ASME Code for Operation and Maintenance of Nuclear Power Plants.

APPLICABLE SAFETY ANALYSES (continued)

DBA analyses show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with ~~10 CFR 50, Appendix K~~ (Ref. 2).

the approved method

The analysis for minimum internal containment pressure (i.e., maximum external differential containment pressure) assumes inadvertent simultaneous actuation of both the ARS and the Containment Spray System.

The modeled ARS actuation from the containment analysis is based upon a response time associated with exceeding the containment pressure High-High signal setpoint to achieving full ARS air flow. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The ARS total response time of 600 seconds includes signal delays.

The ARS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

In the event of a DBA, one train of the ARS is required to provide the minimum air recirculation for heat removal assumed in the safety analyses. To ensure this requirement is met, two trains of the ARS must be OPERABLE. This will ensure that at least one train will operate, assuming the worst case single failure occurs, which is in the ESF power supply.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.6 and SR 3.6.11.7

These SRs require verification that each ARS motor operated damper is allowed to open or is prevented from opening and each ARS fan is allowed to start or is de-energized or prevented from starting based on the presence or absence of Containment Pressure Control System start permissive and terminate signals. The CPCS is described in the Bases for LCO 3.3.2, "ESFAS." The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.
2. ~~10 CFR 50, Appendix K.~~ WCAP-16996-P-A, Rev. 1, November 2016. (Unit 1)

   WCAP-12945-P-A, March 1998. (Unit 2)
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors.”

APPLICABLE SAFETY ANALYSES (continued)

conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with ~~10 CFR 50, Appendix K~~ the approved method (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The ice bed satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The ice bed LCO requires the existence of the required quantity of stored ice, appropriate distribution of the ice and the ice bed, open flow paths through the ice bed, and appropriate chemical content and pH of the stored ice. The stored ice functions to absorb heat during the blowdown phase and long term phase of a DBA, thereby limiting containment air temperature and pressure. The chemical content and pH of the stored ice provide core SDM (boron content) and remove radioactive iodine from the containment atmosphere when the melted ice is recirculated through the ECCS and the Containment Spray System, respectively. The limits on boron concentration and pH of the ice are associated with containment sump pH ranging between 7.5 and 9.3 inclusive following the design basis LOCA.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice bed. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ice bed is not required to be OPERABLE in these MODES.

BASES

REFERENCES

1. UFSAR, Section 6.2.

2. ~~10 CFR 50, Appendix K.~~

WCAP-16996-P-A, Rev. 1, November 2016. (Unit 1)

WCAP-12945-P-A, March 1998. (Unit 2)

3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

4. UFSAR, Section 6.3.3.

5. Topical Report ICUG-001, Application of the Active Ice Mass Management Concept to the Ice Condenser Ice Mass Technical Specification, Revision 2.

6. UFSAR, Section 18, Table 18-1.

7. Catawba License Renewal Commitments, CNS-1274.00-00-0016, Section 4.17.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System and ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed with respect to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System and the ARS being rendered inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with ~~10 CFR 50, Appendix K~~ (Ref. 2).

the approved method

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

For very small break events occurring in the lower compartment that do not by themselves produce sufficient breakaway pressure to open the lower inlet doors, slowly released steam will migrate through the Divider Barrier into the upper compartment. In this situation, the Containment ARS will actuate at its defined pressure setpoint (including a defined time delay) and open the lower inlet doors, returning the steam/air mixture to the lower compartment and displacing it into the ice condenser where the steam portion of the flow will be condensed (Ref. 1). The Containment ARS can also be actuated manually.

In addition to calculating the overall peak containment pressures, the DBA analyses include the calculation of the transient differential pressures that would occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand the local transient pressure differentials for the limiting DBAs.

The ice condenser doors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

BASES

REFERENCES

1. UFSAR, Chapter 6.
2. ~~10 CFR 50, Appendix K.~~

WCAP-16996-P-A, Rev. 1, November 2016. (Unit 1)

WCAP-12945-P-A, March 1998. (Unit 2)

3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (~~Ref. 2~~).

Ref. 2 and Ref. 8

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. The unit has four control banks and five shutdown banks.

SURVEILLANCE REQUIREMENTS (continued)

rod drop times, measuring drop times prior to the next criticality following any such removal ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 551°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.3.
4. UFSAR, Section 15.4.
5. UFSAR, Section 4.3.1.5.
6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
7. UFSAR, Section 15.0.

8. WCAP-18546-P-A, March 2023.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (~~Ref. 2~~). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Ref. 2 and Ref. 5

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. The plant has four control banks and five shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The shutdown banks must be maintained above designed shutdown bank insertion limits and are typically near the fully withdrawn position during normal full power operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation , the shutdown

ACTIONS (continued)

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.
4. 10 CFR 50.36, Technical Specification, (c)(2)(ii).

5. WCAP-18546-P-A, March 2023.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (~~Ref. 2~~). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Ref. 2 and Ref. 7

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. The plant has four control banks and five shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

The control banks are moved in an overlap pattern. When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The fully withdrawn position is defined in the COLR.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.1.
4. UFSAR, Section 15.0.
5. UFSAR, Section 15.4.8.
6. 10 CFR 50.36, Technical Specification, (c)(2)(ii).

7. WCAP-18546-P-A, March 2023.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

10 CFR 50.46 acceptance criteria (Ref. 1 and Ref. 6) must be met

a. During a loss of coolant accident (LOCA), the ~~peak cladding temperature must not exceed 2200°F for small breaks and there is a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1~~);

b. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(X,Y,Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. ~~Other Reference 1 criteria must also be met in LOCAs (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, transient strain, and long term cooling). However, the peak cladding temperature is typically most limiting.~~

$F_Q(X,Y,Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(X,Y,Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(X,Y,Z)$ satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The Heat Flux Hot Channel Factor, $F_Q(X,Y,Z)$, shall be limited by the following relationships:

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{P} K(Z) \qquad \text{for P > 0.5}$$

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for P} \le 0.5$$

LCO (continued)

ensure that the 10 CFR 50.46 acceptance criteria (Ref. 1 and Ref. 6) are met

The $F_Q(X,Y,Z)$ limits typically define limiting values for core power peaking that ~~precludes peak cladding temperatures above 2200°F during a small break LOCA and a high level of probability that the peak cladding temperature does not exceed 2200°F~~ for a ~~large break~~ LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the $F_Q(X,Y,Z)$ limits. If $F_Q(X,Y,Z)$ cannot be maintained within the steady state LOCA limits, reduction of the core power is required.

Violating the steady state LOCA limits for $F_Q(X,Y,Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(X,Y,Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(X,Y,Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power. The exception to this is the steam line break event, which is assumed for analysis purposes to occur from very low power levels. At these low power levels, measurements of $F_Q(X,Y,Z)$ are not sufficiently reliable. Operation within analysis limits at these conditions is inferred from startup physics testing verification of design predictions of core parameters in general.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F^M_Q(X,Y,Z)$ exceeds its steady state limit, maintains an acceptable absolute power density. $F^M_Q(X,Y,Z)$ is the measured value of $F_Q(X,Y,Z)$ and the steady state limit includes factors accounting for measurement uncertainty and manufacturing tolerances. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".

6. WCAP-18546-P-A,
March 2023.

BACKGROUND (continued)

(UCBW). For these types of accidents, the event itself causes changes in the power distribution and this LCO alone is not sufficient to preclude DNB. The acceptability of analyses such as the UCBW accident analysis is ensured by LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” LCO 3.1.6, “Control Bank Insertion Limits,” LCO 3.2.4, “QUADRANT POWER TILT RATIO (QPTR),” LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits,” in combination with cycle-specific analytical calculations.”
Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs.

APPLICABLE SAFETY ANALYSES

Limits on $F_{\Delta H}$(X,Y) preclude core power distributions that exceed the following fuel design limits:

a. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

b. During a ~~large break~~ loss of coolant accident (LOCA), ~~there must be a high level of probability that the peak cladding temperature (PCT) does not exceed 2200°F~~;

the 10 CFR 50.46 acceptance criteria (Ref. 3 and Ref. 7) must be met

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F_{\Delta H}$(X,Y) are the core parameters of most importance. The limits on $F_{\Delta H}$(X,Y) ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency that do not alter the core power distribution. For transients such as uncontrolled RCCA bank withdrawal, which are characterized by changes in the core power distribution, this LCO alone is not sufficient to preclude DNB. The acceptability of the accident analyses is ensured by

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.2.3, “AXIAL FLUX DIFFERENCE (AFD),” LCO 3.1.6, “Control Bank Insertion Limits,” LCO 3.2.4, “QUADRANT POWER TILT RATIO

(QPTR),” and LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits,” in combination with cycle-specific analytical calculations. The DNB design basis is met by limiting the minimum DNBR to the design limit value using an NRC approved CHF correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F_{\Delta H}$(X,Y) limit increases with decreasing power level. This functionality in $F_{\Delta H}$(X,Y) is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F_{\Delta H}$(X,Y) in the analyses.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), the Nuclear Heat Flux Hot Channel Factor (F_Q(Z)), and the axial peaking factors are supported by the LOCA safety analyses that verify compliance with the 10 CFR 50.46 acceptance criteria (Ref. 3 and Ref. 7).

~~The LOCA safety analysis models $F_{\Delta H}$(X,Y) as an input parameter. The Nuclear Heat Flux Hot Channel Factor (F_Q(X,Y,Z)) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).~~ The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor (F_Q(X,Y,Z))."

$F_{\Delta H}$(X,Y) and F_Q(X,Y,Z) are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Control Bank Insertion Limits.

$F_{\Delta H}$(X,Y) satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

$F_{\Delta H}$(X,Y) shall be limited by the following relationship:

$$F^M_{\Delta H}(X,Y) \le F^L_{\Delta H}(X,Y)^{LCO}$$

where: $F^M_{\Delta H}$(X,Y) is defined as the measured radial peak, and

$F^L_{\Delta H}(X,Y)^{LCO}$ is defined as the steady state maximum allowable radial peak defined in the COLR.

LCO (continued)

The $F^L_{\Delta H}(X,Y)^{LCO}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB.

$F^L_{\Delta H}(X,Y)^{LCO}$ limits are maximum allowable radial peak (MARP) limits which are developed in accordance with the methodology outlined in Reference 5. MARP limits are constant DNBR limits which are a function of both the magnitude and location of the axial peak F(Z), therefore, justifying the X,Y dependence of the $F^L_{\Delta H}(X,Y)^{LCO}$ limit.

The limiting value, $F^L_{\Delta H}(X,Y)^{LCO}$, is also power dependent and can be described by the following relationship:

$$F^L_{\Delta H}(X,Y)^{LCO} = MARP(X,Y) * \left[1.0 + (1/RRH) * (1.0 - P)\right]$$

where: MARP(X,Y) is the maximum allowable radial peaks provided in the COLR,

P is the ratio of THERMAL POWER to RATED THERMAL POWER, and

RRH is the amount by which allowable THERMAL POWER must be reduced for each 1% that $F^M_{\Delta H}(X,Y)$ exceeds the limit. The specific value is contained in the COLR.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value, $F^L_{\Delta H}(X,Y)^{LCO}$, is allowed to increase approximately 0.3% for every 1% RTP reduction in THERMAL POWER. This increase in the $F^L_{\Delta H}(X,Y)^{LCO}$ limit is due to the reduced amount of heat removal required at lower powers.

to ensure the 10 CFR 50.46 acceptance criteria (Ref. 3 and Ref. 7) are met

APPLICABILITY

The $F_{\Delta H}(X,Y)$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and ~~PCT~~. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that might be expected to be sensitive to $F_{\Delta H}(X,Y)$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F_{\Delta H}(X,Y)$ in these modes. The exceptions to this are the steam line break,

APPLICABILITY (continued)

SURVEILLANCE REQUIREMENTS (continued)

channel factor to the surveillance limit is likely to decrease below the value of that ratio when the measurement was taken.

Each of these extrapolations is applied separately to the enthalpy rise hot channel factor surveillance limit. If both of the extrapolations are unfavorable, i.e., if the extrapolated factor is expected to exceed the extrapolated limit and the extrapolated factor is expected to become a larger fraction of the extrapolated limit than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F^M_{\Delta H}(X,Y)$ limit with the last $F^M_{\Delta H}(X,Y)$ increased by the appropriate factor specified in the COLR, or to evaluate $F^M_{\Delta H}(X,Y)$ prior to the point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F^M_{\Delta H}(X,Y)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_{\Delta H}(X,Y)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending.

$F^M_{\Delta H}(X,Y)$ is verified at power levels 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F^M_{\Delta H}(X,Y)$ is within its limit at high power levels.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR Section 15.4.8
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2005P "Duke Power Company Thermal Hydraulic Statistical Core Design Methodology", September 1992.
6. DPC-NE-2004P-A, Rev. 1, "Duke Power Company McGuire and Catawba Nuclear Statements Core Thermal – Hydraulic Methobology using VIPRE-01, "SER Dated February 20, 1987 (KCP Proprietary.

7. WCAP-18546-P-A, March 2023.

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a ~~large break~~ loss of coolant accident (LOCA), ~~there must be a high level of probability that the peak cladding temperature does not exceed 2200°F (Ref. 1)~~;

the 10 CFR 50.46 acceptance criteria (Ref. 1 and Ref. 5) must be met

b. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}(X,Y)$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

BASES

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

5. WCAP-18546-P-A, March 2023.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

The LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of three or four channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for ~~limiting peak clad temperature to < 2200°F~~); and

meeting the 10 CFR 50.46 acceptance criteria (Ref. 13 and Ref. 14)

2. Boration to ensure recovery and maintenance of SDM (k_{eff} < 1.0).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation;
- Containment Purge and Exhaust Isolation;

SURVEILLANCE REQUIREMENTS (continued)

may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

The response time may be verified for components that replace the components that were previously evaluated in Ref. 8 and Ref. 9, provided that the components have been evaluated in accordance with the NRC approved methodology as discussed in Attachment 1 to TSTF-569, Rev. 2, “Methodology to Eliminate Pressure Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing,” (Ref.12).

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 900 psig in the SGs.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 7.
3. UFSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
7. WCAP-10271-P-A, Supplement 1 and Supplement 2, Rev. 1, May 1986 and June 1990.
8. WCAP 13632-P-A, Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements” Sep., 1995.
9. WCAP-14036-P-A, Revision 1, “Elimination of Periodic Protection Channel Response Time Tests” Oct., 1998.
10. WCAP-14333-P-A, Revision 1, October 1998.
11. WCAP-15376-P-A, Revision 1, March 2003.
12. Attachment 1 to TSTF-569, Rev. 2, “Methodology to Eliminate Pressure Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing.”

13. 10 CFR 50.46.

14. WCAP-18546-P-A, March 2023.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

large break

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

large break

Initial accumulator inventory which is injected into the reactor vessel is lost out the break.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above the permissive circuit P-11 setpoint.

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. The valves will automatically open, however, as a result of an SI signal. The isolation valves between the accumulators and the Reactor Coolant System are required to be open

BACKGROUND (continued)

and power removed during unit operation. In that the subject valves are normally open and do not serve as an active device during a LOCA, the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) is not applicable in this situation. Therefore, the subject valve control circuit is not designed to this standard.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

large break

for large breaks

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. No credit is taken for control rod assembly insertion, except for post-LOCA subcriticality calculation during the sump recirculation phase. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The largest break area considered for a large break LOCA is a double ended guillotine break in the RCS cold leg.

~~The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump.~~ During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(for loss of offsite power assumption)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting, the valves opening, and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency

APPLICABLE SAFETY ANALYSES (continued)

in the modeling

for a large break LOCA

core cooling. No operator action is assumed ~~during the blowdown stage~~ of a large break LOCA.

intermediate

is assumed to inject into the RCS

The ~~worst case~~ small break LOCA analyses also assume a time delay before pumped flow ~~reaches the core~~. For ~~the larger range of small~~ breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators, safety injection pumps, and centrifugal charging pumps all play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

acceptance criteria (Ref. 3 and Ref. 9) are met.

This LCO helps to ensure that the ~~following acceptance criteria established for the ECCS by~~ 10 CFR 50.46 ~~(Ref. 3) will be met following a small break LOCA and there is a high probability that the criteria are met following a large break LOCA:~~

~~a.~~ ~~Maximum fuel element cladding temperature is ≤ 2200°F;~~

~~b.~~ ~~Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;~~

~~c.~~ ~~Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and~~

~~d.~~ ~~Core is maintained in a coolable geometry.~~

large break LOCA and the recovery phase of a small break LOCA

Since the accumulators discharge during the blowdown phase of a ~~LOCA,~~ they do not contribute directly to the long term cooling requirements of 10 CFR 50.46. However, the boron content of the accumulator water helps to maintain the reactor core subcritical after reflood, thereby eliminating fission heat as an energy source for which cooling must be provided.

The large and small break LOCA analyses use a range of accumulator water volumes of 6790 gallons to 7422 gallons per the approved method (Ref. 10).

Both large and small break LOCA analyses use a nominal accumulator line volume from the accumulator to the check valve.

~~For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used.~~ The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. ~~The large and small break LOCA analyses are performed with accumulator volumes that are consistent with the LOCA evaluation models. To allow for operating margin, values of ± 31.5 ft^3 are specified.~~

APPLICABLE SAFETY ANALYSES (continued)

The minimum boron concentration setpoint is used in the post-LOCA subcriticality verification during the injection phase. For each reload cycle, the all rods out (ARO, no credit for control rod assembly insertion) critical boron concentration is verified to be less than the minimum allowed cold leg accumulator boron concentration. No credit is taken for control rod assembly insertion when verifying subcriticality during the injection phase, but credit is taken for control rod assembly insertion in the post-LOCA subcriticality calculation during the sump recirculation phase to offset the boron diluted sump condition described below.

The minimum boron concentration setpoint is also used in the post LOCA sump boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment, with all rods in (crediting control rod assembly insertion), minus the highest worth rod out (ARI N-1). Of particular interest is the large cold leg break LOCA, since boron accumulation in the core will be maximized during the cold leg recirculation phase due to core boiling. The accumulation of boron in the core prevents the boron from returning to the sump, which leads to a boron diluted sump condition. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown, potentially causing the core to become re-critical by injecting boron diluted sump water into the core when switching over to hot leg recirculation. A reduction in the accumulator minimum boron concentration would also increase the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

break LOCA analysis uses a range of accumulator nitrogen cover pressures of 555 psig to 708 psig, and the small break LOCA analysis uses a range of accumulator nitrogen cover pressures of 555 psig to 669 psig, per the approved method (Ref. 10).

The large ~~and small break LOCA analyses are performed with accumulator pressures that are consistent with the LOCA evaluation models. To allow for operating margin and accumulator design limits, a range from 585 psig to 639 psig is specified.~~ The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Ref. 4). The accumulators satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

large break

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (~~Ref. 3~~) could be violated.

Ref. 3 and Ref. 9

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met. Additionally, the nitrogen and liquid volumes between accumulators must be physically separate.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

the 10 CFR 50.46 acceptance criteria (Ref. 3 and Ref. 9) are met

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure ~~that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F for small break LOCAs and there is a high level of probability that the peak cladding temperature does not exceed 2200°F for large break LOCAs~~.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are allowed to be closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS

BASES

SURVEILLANCE REQUIREMENTS (continued)

occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is ≤ 1000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. IEEE Standard 279-1971.
2. UFSAR, Chapter 6.
3. 10 CFR 50.46.
4. DPC-NE-3004.
5. 10 CFR 50.36, Technical Specification, (c)(2)(ii).
6. WCAP - 15049-A, Rev. 1, April 1999
7. NUREG-1366, February 1990.
8. Duke letter to NRC, “Cold Leg Accumulator Isolation Valves”, dated September 8, 1987

9. WCAP-18546-P-A, March 2023.

10. WCAP-16996-P-A, Rev. 1, November 2016.

BACKGROUND (continued)

The high and intermediate head subsystems of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a safety injection actuation.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the ~~following~~ acceptance criteria for the ECCS, established by 10 CFR 50.46 ~~(Ref. 2), will be met following a small break LOCA and there is a high level of probability that the criteria are met following a large break LOCA:~~

(Ref. 2 and Ref. 8) are met.

~~a.~~ ~~Maximum fuel element cladding temperature is ≤ 2200°F;~~

~~b.~~ ~~Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;~~

~~c.~~ ~~Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;~~

APPLICABLE SAFETY ANALYSES (continued)

~~d. Core is maintained in a coolable geometry; and~~

~~e. Adequate long term core cooling capability is maintained.~~

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment pressure and temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event has the greatest potential to challenge the limits on runout flow set by the manufacturer of the ECCS pumps. It also sets the maximum response time for their actuation. Direct flow from the centrifugal charging pumps and SI pumps is credited in a small break LOCA event. The RHR pumps are also credited, for larger small break LOCAs, as the means of supplying suction to these higher head ECCS pumps after the switch to sump recirculation. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The MSLB analysis also credits the SI and centrifugal charging pumps. Although some ECCS flow is necessary to mitigate a SGTR event, a single failure disabling one ECCS train is not the limiting single failure for this transient. The SGTR analysis primary to secondary break flow is increased by the availability of both centrifugal charging and SI trains. Therefore, the SGTR analysis is penalized by assuming both ECCS trains are operable as required by the LCO. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

or without

a. A large break LOCA event, with loss of offsite power and a single failure disabling one ECCS train; and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Ref. 3). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. UFSAR, Section 6.2.1.
4. UFSAR, Chapter 15.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
7. IE Information Notice No. 87-01.
8. WCAP-18546-P-A, March 2023.

APPLICABLE SAFETY ANALYSES (continued)

LOCA sump boron concentration necessary to assure subcriticality, with all rods in (crediting control rod assembly insertion), minus the highest worth rod out (ARI N-1). The large cold leg break LOCA is the limiting case since boron accumulation in the core will be maximized during the cold leg recirculation phase due to core boiling. The accumulation of boron in the core prevents the boron from returning to the sump, which leads to a boron diluted sump condition. A reduction in the RWST minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown, potentially causing the core to become re-critical by injecting boron diluted sump water into the core when switching over to hot leg recirculation.

The RWST minimum boron concentration is also used in the post-LOCA subcriticality verification during the injection phase. For each reload cycle, the all rods out (ARO, no credit for control rod assembly insertion) critical boron concentration is verified to be less than the minimum allowed RWST boron concentration. No credit is taken for control rod assembly insertion when verifying subcriticality during the injection phase, but credit is taken for control rod assembly insertion in the post-LOCA subcriticality calculation during the sump recirculation phase to offset the boron diluted sump condition described above.

The upper limit on boron concentration as listed in the COLR is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

The RWST temperature limits were originally established with containment spray aligned to the RWST and were not revised when the Containment Spray System became a manually actuated system with the initial suction source changed to the Containment Sump. The RWST temperature limits are contained within additional analyses and remain valid, although the basis is historical and no longer relevant. The following paragraph is retained for historical purposes only.

~~In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 70°F. If the lower temperature limit was violated, the containment spray could further reduce containment pressure, which decreases the saturated steam specific volume. This means that each pound of steam generated during core reflood tends to occupy a larger volume, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of 100°F, plus an allowance for temperature measurement uncertainty, is used in the containment~~

APPLICABLE SAFETY ANALYSES (continued)

~~OPERABILITY analysis. Exceeding this temperature will result in higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.~~

The RWST satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

The RWST ensures that an adequate supply of borated water is available to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS OPERABILITY requirements. Since both the ECCS must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

This LCO is applicable only to those units that utilize the centrifugal charging pumps for safety injection (SI). The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but do not set the limits on their flow requirements. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

and the intermediate
head safety injection
pumps

This LCO ensures that seal injection flow of ≤ 40 gpm, with centrifugal charging pump operating and charging flow control valve full open, will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical ~~for a large LOCA~~. For smaller LOCAs, the charging pumps ~~alone~~ deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

and the intermediate
head safety injection
pumps

and the intermediate
head safety injection
pumps

BASES

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (~~Ref. 3~~).

Ref. 3 and Ref. 4

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. In order to establish the proper flow line resistance, a minimum pressure differential and flow must be known. The flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure and that the centrifugal charging pump discharge pressure is greater than or equal to the applicable value specified in the test acceptance criteria. Since the test acceptance criteria head curve ensures the centrifugal charging pumps are capable of delivering the flow assumed in the LOCA analyses, the minimum pressure differential is satisfied by verifying the centrifugal charging pump is operating. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed minimum pressure differential result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the charging flow control valve being full open, is required since the valve is designed to fail open unless motive air is available. With the operating pump and control valve position as specified by the LCO, a flow limit is established. It is this flow limit that is used in the accident analyses.

The limit on seal injection flow, combined with the minimum pressure differential and an open wide condition of the charging flow control valve, must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow might not be as much as assumed in the accident analyses.

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTIONS

A.1

With the seal injection flow exceeding its limit, the amount of high head safety injection flow available to the RCS may be reduced. Under this Condition, action must be taken to restore the flow to below its limit. The

BASES

REFERENCES

1. UFSAR, Chapter 6 and Chapter 15.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. 10 CFR 50.46.

4. WCAP-18546-P-A, March 2023.

APPLICABLE SAFETY ANALYSES (continued)

1. Rupture of a hot or high pressure process pipe in the annulus.
2. Inadvertent Containment Spray System initiation during normal operation.
3. Inadvertent containment air return fan initiation during normal operation.
4. Containment purge fan operation with containment purge inlet valves closed.

The containment design of 1.5 psig negative is not violated in the above conditions due to either equipment limitations or design features.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations ~~Therefore, for the reflood phase~~, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with the approved method ~~10 CFR 50, Appendix K~~ (Ref. 2).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following an event which has the potential to result in a net external pressure on the containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.
2. ~~10 CFR 50, Appendix K.~~ WCAP-16996-P-A, Rev. 1, November 2016.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

BACKGROUND (continued)

The Containment Spray System limits the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the ARS being rendered inoperable (Ref. 2).

The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and was calculated to be within the containment environmental qualification temperature during the DBA SLB. The basis of the containment environmental qualification temperature is to ensure the OPERABILITY of safety related equipment inside containment (Ref. 3).

The Containment Spray System actuation modeled in the containment analysis is based on the time associated with reaching the RWST Low Level Setpoint and operator action prior to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The Containment Spray System total response time is composed of operator action, system startup time, and time for the piping to fill.

~~For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).~~

SURVEILLANCE REQUIREMENTS (continued)

where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. UFSAR, Section 6.2.
3. 10 CFR 50.49.
4. Not used. ~~10 CFR 50, Appendix K.~~
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
6. ASME Code for Operation and Maintenance of Nuclear Power Plants.

APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with ~~10 CFR 50, Appendix K~~ (Ref. 2).

the approved method

The analysis for minimum internal containment pressure (i.e., maximum external differential containment pressure) assumes inadvertent simultaneous actuation of both the ARS and the Containment Spray System.

The modeled ARS actuation from the containment analysis is based upon a response time associated with exceeding the containment pressure High-High signal setpoint to achieving full ARS air flow. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The ARS total response time of 600 seconds includes signal delays.

The ARS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

In the event of a DBA, one train of the ARS is required to provide the minimum air recirculation for heat removal assumed in the safety analyses. To ensure this requirement is met, two trains of the ARS must be OPERABLE. This will ensure that at least one train will operate, assuming the worst case single failure occurs, which is in the ESF power supply.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ARS. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the ARS is not required to be OPERABLE in these MODES.

SURVEILLANCE REQUIREMENTS (continued)

surveillance also tests the circuitry, including time delays, to ensure the system operates properly. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.6.11.4 and SR 3.6.11.5

Verifying the OPERABILITY of the check damper in the air return fan discharge line to the containment lower compartment provides assurance that the proper flow path will exist when the fan is started and that reverse flow can not occur when the fan is not operating. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.6.11.6 and SR 3.6.11.7

These SRs require verification that each ARS motor operated damper opens or is prevented from opening and each ARS fan is allowed to start or is prevented from starting upon receipt of Containment Pressure Control System start permissive and terminate signals. The CPCS is described in the Bases for LCO 3.3.2, "ESFAS." The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.
2. ~~10 CFR 50, Appendix K.~~ WCAP-16996-P-A, Rev. 1, November 2016.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BACKGROUND (continued)

The ice bed limits the temperature and pressure that could be expected following a DBA, thus limiting leakage of fission product radioactivity from containment to the environment.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are not assumed to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the Containment Spray System, RHR Spray System, and the ARS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed in regards to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System, RHR Spray System, and ARS being inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of the transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with ~~10 CFR 50, Appendix K~~ (Ref. 2).

the approved method

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The ice bed satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

REFERENCES

1. UFSAR, Section 6.2.
2. ~~10 CFR 50, Appendix K.~~ WCAP-16996-P-A, Rev. 1, November 2016.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. UFSAR, Section 6.3.3.10.
5. Topical Report ICUG-001, Application of the Active Ice Mass Management Concept to the Ice Condenser Ice Mass Technical Specification, revision 2.
6. UFSAR, Table 18-1 and Section 18.2.14.
7. McGuire License Renewal Commitments MCS-1274.00-00-0016, Section 4.19, Ice Condenser Inspections.

APPLICABLE SAFETY ANALYSES (continued)

is the worst case single active failure and results in one train each of the Containment Spray System and the ARS being rendered inoperable.

The limiting DBA analyses (Ref. 1) show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure. For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the ECCS during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with ~~10 CFR 50, Appendix K~~ the approved method (Ref. 2).

The maximum peak containment atmosphere temperature results from the SLB analysis and is discussed in the Bases for LCO 3.6.5, "Containment Air Temperature."

For very small break events occurring in the lower compartment that do not by themselves produce sufficient breakaway pressure to open the lower inlet doors, slowly released steam will migrate through the Divider Barrier into the upper compartment. In this situation, the Containment ARS will actuate at its defined pressure setpoint (including a defined time delay) and open the lower inlet doors, returning the steam/air mixture to the lower compartment and displacing it into the ice condenser where the steam portion of the flow will be condensed (Ref. 1). The Containment ARS can also be actuated manually.

In addition to calculating the overall peak containment pressures, the DBA analyses include the calculation of the transient differential pressures that would occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand the local transient pressure differentials for the limiting DBAs.

The ice condenser doors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

This LCO establishes the minimum equipment requirements to assure that the ice condenser doors perform their safety function. The ice condenser lower inlet doors, intermediate deck doors, and top deck doors must be closed to minimize air leakage into and out of the ice condenser, with its attendant leakage of heat into the ice condenser and loss of ice

BASES

REFERENCES

1. UFSAR, Chapter 6.
2. ~~10 CFR 50, Appendix K.~~

WCAP-16996-P-A, Rev. 1, November 2016.

3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. MCS-1558.NF-00-0001 "Design Basis Specification for the NF System".

ATTACHMENT 3

PLANT OPERATING PARAMETERS COMPARED TO TECHNICAL SPECIFICATION LIMITS

2 PAGES PLUS THE COVER

Plant Operating Parameters Compared to Technical Specification Limits

Precedent submittals listed in Section 4.2 of Enclosure 1 addressing the adoption of the FSLOCA EM were reviewed to examine requests for additional information (RAIs) that were issued by the NRC staff. The information presented below focuses on an RAI related to how FSLOCA plant input parameters correspond to Technical Specification (TS) Limiting Condition for Operation (LCO) values. Specifically, it was requested that a comparison be provided for the plant operating parameters used in the FSLOCA analysis to the TS limits for the same parameters, where applicable.

Limitation and Condition 11 of the FSLOCA Evaluation Methodology states,

> *In plant-specific reviews, the uncertainty treatment for such plant operating parameters including the sampled distributions and ranges will be considered acceptable if they meet or exceed corresponding design basis and/or Technical Specification limiting conditions for operation limits, with uncertainties included, as appropriate.*

Duke Energy is not proposing any changes to the CNS or MNS TS LCOs as part of this license amendment request. Table 1 below provides a comparison of the plant operating parameters used in the FSLOCA analysis from Tables 1 and 2 of Enclosures 2 (proprietary) and 4 (non-proprietary), versus the corresponding current TS LCO limits.

Table 1: Plant Operating Parameters Compared to the Technical Specification Limits for Catawba Unit 1 and McGuire Units 1 & 2

Parameter	As-Analyzed Value or Range for AXIOM clad fuel	TS Limit	LCO Number
Core power	≤ 3700 MWt + 0.3% calorimetric uncertainty, 3711.1 MWt uncertainty adjusted	3469 MWt Rated Thermal Power	CNS/MNS TS 1.1
Heat Flux Hot Channel Factor $F_Q(Z)$	2.7 (maximum value, with burndown effects included)	Included in the Core Operating Limits Report (COLR)	CNS/MNS TS 3.2.1
Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$	1.72 (maximum value, with burndown effects included)	Included in the COLR	CNS/MNS TS 3.2.2
Axial flux difference band at Full Power	-22.58% / +14.58%	Included in the COLR	CNS/MNS TS 3.2.3
Low pressurizer pressure reactor trip setpoint	1800 psig	≥ 1938 psig (CNS) ≥ 1935 psig (MNS)	CNS/MNS TS 3.3.1
Low pressurizer pressure safety injection actuation setpoint	1700 psig	≥ 1839 psig (CNS) ≥ 1835 psig (MNS)	CNS/MNS TS 3.3.2

Minimum RCS Flow Rate	379,464 gpm	≥ 384,000 gpm (CNS - Unit 1) ≥ 388,000 gpm (MNS)	CNS/MNS TS 3.4.1
Accumulator water volume	$6790 \le V_{ACC} \le 7422$ gal; $907.7 \le V_{ACC} \le 992.2$ ft^3	(CNS) $7630 \le V_{ACC} \le 8079$ gal, $1020 \le V_{ACC} \le 1080$ ft^3; (MNS) $6870 \le V_{ACC} \le 7342$ gal, $918.4 \le V_{ACC} \le 981.5$ ft^3	CNS/MNS TS 3.5.1
Accumulator gas cover pressure	$555 \le P_{ACC} \le 669$ psig (Region I), $555 \le P_{ACC} \le 708$ psig (Region II);	$585 \le P_{ACC} \le 678$ psig (CNS) $585 \le P_{ACC} \le 639$ psig (MNS)	CNS/MNS TS 3.5.1
Minimum accumulator water boron concentration	1950 ppm	Included in the COLR	CNS/MNS TS 3.5.1
Safety Injection (SI) water temperature	60°F ≤ SI Temp ≤ 110°F;	70°F ≤ Temp ≤ 100°F RWST	CNS/MNS TS 3.5.4
Main Steam Safety Valves (MSSV) opening setpoint for second stage	1190 psig + 14.7 = 1204.7 psia. 1.03*1204.7 = 1240.8 psia.	Lift Setting (psig + 3%) CNS MNS 1190 1190	CNS/MNS TS 3.7.1
Minimum initial containment pressure	-0.3 psig (14.4 psia)	-0.1 to +0.3 psig (CNS) -0.3 to +0.3 psig (MNS)	CNS/MNS TS 3.6.4
Maximum initial upper containment temperature	105 °F	75 to 100 °F	CNS/MNS TS 3.6.5
Maximum initial lower containment temperature	125 °F	100 to 120 °F	CNS/MNS TS 3.6.5
Minimum air return fan (deck fan) delay time	8 minutes	8 to 10 minutes	CNS/MNS TS 3.6.11
Maximum initial ice bed temperature	30 °F	27 °F	CNS/MNS TS 3.6.12 and TS 3.6.13

ATTACHMENT 4

COMPARISON OF CQD AND FSLOCA INPUT PARAMETERS

3 PAGES PLUS THE COVER

Comparison of CQD and FSLOCA Input Parameters

Precedent submittals listed in Section 4.2 of Enclosure 1 addressing the adoption of the FSLOCA EM were reviewed to examine requests for additional information (RAIs) that were issued by the NRC staff. The information presented below focuses on an RAI related to how FSLOCA plant input parameters may differ from previous inputs used in existing Large Break LOCA analyses. Specifically, it was requested that a comparison be provided for the input parameters used in the FSLOCA analysis and the UFSAR parameter values, and provide justification for the differences.

The current Code Qualification Document (CQD) Best Estimate Large Break LOCA analyses were implemented at CNS/MNS in 2000, where one composite bounding analysis was applied to all four CNS/MNS units. The FSLOCA Evaluation Methodology (EM) is a new best-estimate method that incorporates new conservatisms requiring a host of new inputs. Several of the FSLOCA analysis inputs were changed from the CQD analysis to improve operating margins, account for instrument uncertainties, add conservatism to safety analysis margins, or maintain compliance with the new FSLOCA methodology. Plant operating ranges considered in the CQD Large Break LOCA analysis are described in CNS UFSAR Table 15-71 and MNS UFSAR Table 15-55. Initial conditions used for the CQD minimum post-LOCA containment pressure are described in CNS UFSAR Table 6-66 and MNS UFSAR Tables 6-64 and 6-65. FSLOCA inputs, including those that were changed relative to the CQD Large Break LOCA analysis, are discussed below and shown in Table 2.

Core thermal power was increased in the FSLOCA analyses to bound an Extended Power Uprate (EPU) to 3700 MWt for MNS Units 1 & 2, and CNS U1. The 0.3% core thermal power uncertainty was included such that the power used in the analysis was 3711.1 MWt. Core peaking factors were adjusted to provide margin to core designs for AXIOM clad fuel. Accumulator parameters were selected to bound an FSLOCA composite model approach for MNS Units 1 & 2 and CNS U1, as described in Section 3 of Enclosures 2 (proprietary) and 4 (non-proprietary). This is the same approach that was used for accumulator parameters in the CQD analysis.

FSLOCA RCS flow rate is based on MNS Unit 1 Thermal Design flow of 379,464 gpm, which is based on use of MNS Unit 1 vessel parameters used in the composite model approach for MNS Units 1 & 2, and CNS Unit 1. The CQD Large Break analysis was originally performed using a total RCS flow rate of 390,000 gpm. A subsequent CQD assessment was performed for a reduction in total RCS flow to 379,464 gpm Thermal Design Flow with a 0°F impact to the CQD analysis. This assessment was described in a letter to the NRC dated April 9, 2003 (ADAMS Accession No. ML031060339) via annual 10 CFR 50.46 reporting.

Minimum injected ECCS flows assumed in CQD LBLOCA analyses are described in CNS UFSAR Table 15-65 and MNS UFSAR Table 15-45. In 2006, the Small Break LOCA was re-analyzed to support decreased intermediate head safety injection (IHSI) and high head safety injection (HHSI) pump flows at all MNS/CNS units. The reduced flows were used in the Small Break LOCA analyses to account for the TS surveillance limits of ± 2% on the frequency of the emergency diesel generators (EDGs), and to obtain additional test acceptance criteria margin for the IHSI and HHSI pumps. The CQD Large Break LOCA analyses were not re-analyzed as sufficient low head safety injection (LHSI) flow margin exists in the current analyses to offset the reduced flow due to the frequency reduction of the EDGs. This assessment was described in a letter to the NRC dated May 22, 2007 (ADAMS Accession No. ML071500297) via 30-Day 10 CFR 50.46 reporting.

Since CNS and MNS have implemented the Water Management Initiative, the transition to cold leg recirculation is delayed since the containment spray pumps do not auto-start in the injection phase. Containment spray pumps are manually started after the RWST reaches the Low Level setpoint. Automatic action of the containment spray system no longer occurs. For the Region II analysis, the analyzed duration is less than or equal to 600 seconds, which is well before the RWST Low Level setpoint would be reached, so containment spray is not actuated for the FSLOCA minimum containment pressure calculation, per Table 2 of Enclosures 2 (proprietary) and 4 (non-proprietary).

Table 2 - Comparison of CQD and FSLOCA Input Parameters

Parameter	**Operating Range, CQD Large Break LOCA; CNS UFSAR Tables 15-71 & 6-66, and MNS UFSAR Tables 6-64, 6-65, & 15-55**	**Operating Range, FSLOCA for CNS Unit 1 and MNS Units 1 & 2**
Reactor Power	Core power ≤ 100.3% of 3469 MWt, (≤ 3479.4 MWt)	Core power ≤ 100.3% of 3700 MWt (≤ 3711.1 MWt), bounds EPU conditions
Peak Heat Flux Hot Channel peaking factor	F_Q ≤ 2.7 (≤ 4 ft), 2.5 (> 4 ft); includes TCD burndown effects	F_Q ≤ 2.70; includes burndown effects
Peak Enthalpy Rise Hot Channel peaking factor	$F_{\Delta H}$ ≤ 1.67; includes TCD burndown effects	$F_{\Delta H}$ ≤ 1.72; includes burndown effects
Axial power distribution	Established per EM	Established per EM
RCS T_{avg}	581.1°F ≤ T_{avg} ≤ 593.9°F, 587.5° ± 6.4°F	581.0°F ≤ T_{avg} ≤ 594.0°F, 587.5° ± 6.5°F [Note 1]
Pressurizer pressure	2190 psia ≤ P_{RCS} ≤ 2310 psia, 2250 psia ± 60 psi	2200 psia ≤ P_{RCS} ≤ 2300 psia, 2250 psia ± 50 psi [Note 1]
RCS Loop flow	97,500 gpm/loop; 390,000 gpm total Subsequently evaluated for reduction to 379,464 gpm total flow.	94,866 gpm/loop for all units; 379,464 gpm total
Maximum SG tube plugging level	≤ 10% (CNS 2), ≤ 5% (MNS and CNS 1)	≤ 10% (All Units)
Accumulator fluid temperature	105°F ≤ T_{ACC} ≤ 125°F	105°F ≤ T_{ACC} ≤ 125°F
Accumulator pressure	555 psig ≤ P_{ACC} ≤ 708 psig	555 psig ≤ P_{ACC} ≤ 669 psig (Region I) 555 psig ≤ P_{ACC} ≤ 708 psig (Region II);
Accumulator liquid volume	6790 gal ≤ V_{ACC} ≤ 7422 gal (MNS), 7550 gal ≤ V_{ACC} ≤ 8159 gal (CNS)	6790 gal ≤ V_{ACC} ≤ 7422 gal;
Minimum accumulator boron concentration	≥ 2275 ppm	≥ 1950 ppm
Minimum Injected ECCS flows	CNS UFSAR Table 15-65, MNS UFSAR Table 15-45	Tables 4 and 5 of Enclosures 2 and 4
Safety injection temperature	58°F ≤ SI Temp ≤ 90°F, covers a RWST temperature range of 70-100°F and component cooling water temperature down to 45°F	60°F ≤ SI Temp ≤ 110°F;

Safety injection delay	≤ 17 seconds (with OPA) ≤ 32 seconds (with LOOP)	≤ 17 seconds (with OPA) ≤ 32 seconds (with LOOP)
Initial Containment pressure	14.7 psia	14.4 psia
Minimum containment outside air / ground temperature	N/A	-5°F
Maximum Initial Ice Condenser temperature	27°F	30°F
Minimum Initial Refueling Water Storage Tank temperature	65°F	58°F used for broken loop spilling flow
Maximum containment spray system flow, total	9600 gpm	0 gpm
Maximum number of containment spray pumps operating	2	0
Fastest post-LOCA initiation of spray system (assuming off-site power loss at start of LOCA)	25 seconds	N/A
Fastest post-LOCA initiation of containment air return fans	480 seconds	8 minutes after break

Note 1: New value based on updated measurement uncertainty

ENCLOSURE 3

WESTINGHOUSE AFFIDAVIT

3 PAGES PLUS THE COVER

AFFIDAVIT CAW-25-039

Commonwealth of Pennsylvania:
County of Butler:

(1) I, Jerrod Ewing, Manager, Operating Plants Licensing; Cranberry Township, PA, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of RA-25-0014, Enclosure 2 be withheld from public disclosure under 10 CFR 2.390.

(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 10 CFR 2.390, 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 and is not customarily disclosed to the public.

  (ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

  (iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, 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 technical evaluation justifications 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.

(5) Westinghouse has policies in place to identify proprietary information. 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.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding 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 (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 7/23/2025

Jerrod Ewing

Signed electronically by
Jerrod Ewing

U.S. Nuclear Regulatory Commission
Serial: RA-25-0014
Enclosure 4

ENCLOSURE 4

APPLICATION OF WESTINGHOUSE FULL SPECTRUM LOCA EVALUATION MODEL TO MCGUIRE UNITS 1 AND 2 AND CATAWBA UNIT 1 (NON-PROPRIETARY VERSION)

Note: Proprietary information is identified by bolded brackets and has been removed.

63 PAGES PLUS THE COVER

APPLICATION OF WESTINGHOUSE FULL SPECTRUM LOCA EVALUATION MODEL TO MCGUIRE UNITS 1 AND 2 AND CATAWBA UNIT 1

1.0 INTRODUCTION

An analysis with the **FULL SPECTRUM™** loss-of-coolant accident (**FSLOCA™**) evaluation model (EM) has been completed for McGuire Units 1 and 2 and Catawba Unit 1 to support 18-month cycle operation with **AXIOM**® cladding and the **PRIME™** bottom nozzle. This analysis also supports an extended power uprate (EPU), though the EPU will not be implemented at this time. This license amendment request (LAR) for McGuire Units 1 and 2 and Catawba Unit 1 requests approval to apply the Westinghouse **FSLOCA** EM.

The **FSLOCA** EM (Reference 1) was developed to address the full spectrum of loss-of-coolant accidents (LOCAs) which result from a postulated break in the reactor coolant system (RCS) of a pressurized water reactor (PWR). The break sizes considered in the Westinghouse **FSLOCA** EM include any break size in which break flow is beyond the capacity of the normal charging pumps, up to and including a double-ended guillotine (DEG) rupture of an RCS cold leg with a break flow area equal to two times the pipe area, including what traditionally are defined as small-break LOCAs (SBLOCAs) and large-break LOCAs (LBLOCAs).

The break size spectrum is divided into two regions. Region I includes breaks that are typically defined as SBLOCAs. Region II includes break sizes that are typically defined as LBLOCAs.

The **FSLOCA** EM explicitly considers the effects of fuel pellet thermal conductivity degradation (TCD) and other burnup-related effects by calibrating to fuel rod performance data input generated by the PAD5 code (Reference 2), which explicitly models TCD and is benchmarked to high burnup data in Reference 2. The fuel pellet thermal conductivity model in the WCOBRA/TRAC-TF2 code used in the **FSLOCA** EM explicitly accounts for pellet TCD.

The emergency core cooling system (ECCS) acceptance criteria that apply to this analysis are specified for **AXIOM** cladding in Section 6.2.1.4 of Reference 3. Per Section 6.2.1.4 of Reference 3, the Westinghouse approach used to satisfy the maximum cladding temperature (2,200°F) and maximum hydrogen generation (i.e., core-wide oxidation (CWO)) (1%) acceptance criteria defined in 10 CFR 50.46 (b)(1) and (b)(3) (Reference 4), respectively, is applicable to **AXIOM** cladding. However, the maximum local oxidation (MLO) acceptance criterion (17%) defined in 10 CFR 50.46 (b)(2) is replaced with the Nuclear Regulatory Commission (NRC) approved **AXIOM** cladding performance-based embrittlement acceptance criterion. The Cathcart-Pawel equivalent cladding reacted (ECR) is confirmed to remain below the ductile-to-brittle transition (DBT) limit for **AXIOM** cladding described in Section 3.11 of Reference 3 (i.e., minimum ECR margin (MEM) ≥ 0%). A high probability statement is developed for the peak cladding temperature (PCT), MEM, and CWO that is needed to demonstrate compliance with these acceptance criteria via statistical methods.

Section 6.2.1.4 of Reference 3 indicates that the breakaway time is not a plausibly limiting acceptance criterion for the SBLOCA and LBLOCA analyses with **AXIOM** cladding fuel rods. The coolable geometry acceptance criterion, 10 CFR 50.46 (b)(4), is assured by compliance with the first two acceptance criteria (PCT and local oxidation (MEM)), and demonstrating that fuel assembly grid deformation due to combined seismic and LOCA loads does not extend to the in-board fuel assemblies such that a coolable geometry is maintained. Further, the **FSLOCA** EM does not address the long-term cooling acceptance criterion defined in 10 CFR 50.46 (b)(5). Per Section 6.2.1.4 of Reference 3, the Westinghouse approach used to satisfy the long-term cooling acceptance criterion remains applicable to **AXIOM** cladding.

The **FSLOCA** EM has been generically approved by the NRC for Westinghouse 3-loop and 4-loop plants with cold leg ECCS injection (Reference 1). Since McGuire Units 1 and 2 and Catawba Unit 1 are Westinghouse designed 4-loop plants with cold leg ECCS injection, the approved method is applicable. Information required to address Limitations and Conditions 9 and 10 of the NRC's safety evaluation report (SER) for Reference 1 was docketed in Reference 5 in support of application of the **FSLOCA** EM to Westinghouse 4-loop plants.

This report summarizes the application of the Westinghouse **FSLOCA** EM to McGuire Units 1 and 2 and Catawba Unit 1. The application of the **FSLOCA** EM to McGuire Units 1 and 2 and Catawba Unit 1 is consistent with the NRC-approved methodology (Reference 1), as modified for **AXIOM** cladding in Reference 3, with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The application of the **FSLOCA** EM to McGuire Units 1 and 2 and Catawba Unit 1 is consistent with the limitations and conditions as identified in the NRC's SER for Reference 1, and is also applicable for the McGuire Units 1 and 2 and Catawba Unit 1 plant design and operating conditions.

A composite **FSLOCA** EM analysis was performed to cover McGuire Units 1 and 2 and Catawba Unit 1. The composite model approach is described in Section 3.0.

Both Duke Energy and its analysis vendor (Westinghouse) have interface processes which identify plant configuration changes potentially impacting safety analyses. These interface processes, along with Westinghouse internal processes for assessing EM changes and errors, are used to identify the need for LOCA analysis impact assessments.

The major plant parameter and analysis assumptions used in the McGuire Units 1 and 2 and Catawba Unit 1 analysis with the **FSLOCA** EM are provided in Tables 1 through 5.

2.0 METHOD OF ANALYSIS

2.1 FULL SPECTRUM LOCA Evaluation Model Development

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 (Reference 4 and Reference 6) and Appendix K, "ECCS Evaluation Models," to permit the use of a realistic EM to analyze the performance of the ECCS during a hypothetical LOCA. Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. After the rule change, Westinghouse developed and received approval for a best-estimate LBLOCA EM, which is discussed in Reference 7. The EM is referred to as the Code Qualification Document (CQD), and was developed following Regulatory Guide (RG) 1.157 (Reference 8).

When the **FSLOCA** EM was being developed, the NRC issued RG 1.203 (Reference 9) which expands on the principles of RG 1.157, while providing a more systematic approach to the development and assessment process of a PWR accident and safety analysis EM. Therefore, the development of the **FSLOCA** EM followed the Evaluation Model Development and Assessment Process (EMDAP), which is documented in RG 1.203. While RG 1.203 expands upon RG 1.157, there are certain aspects of RG 1.157 which are more detailed than RG 1.203; therefore, both RGs were used for the development of the **FSLOCA** EM.

2.2 WCOBRA/TRAC-TF2 Computer Code

The **FSLOCA** EM (Reference 1) uses the WCOBRA/TRAC-TF2 code to analyze the system thermal-hydraulic response for the full spectrum of break sizes. WCOBRA/TRAC-TF2 was created by combining a 1D module (TRAC-P) with a 3D module (based on Westinghouse modified COBRA-TF). The 1D and 3D modules include an explicit non-condensable gas transport equation. The use of TRAC-P allows for the extension of a two-fluid, six-equation formulation of the two-phase flow to the 1D loop components. This new code is WCOBRA/TRAC-TF2, where "TF2" is an identifier that reflects the use of a three-field (TF) formulation of the 3D module derived by COBRA-TF and a two-fluid (TF) formulation of the 1D module based on TRAC-P.

This best-estimate computer code contains the following features:

1. Ability to model transient three-dimensional flows in different geometries inside the reactor vessel
2. Ability to model thermal and mechanical non-equilibrium between phases
3. Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes
4. Ability to represent important reactor and plant components such as fuel rods, steam generators (SGs), reactor coolant pumps (RCPs), etc.

A detailed assessment of the computer code WCOBRA/TRAC-TF2 was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the ability of the code to predict key physical phenomena for a LOCA. Modeling of a LOCA introduces additional uncertainties which are identified and quantified in the plant-specific analysis. The reactor vessel and loop noding scheme used in the **FSLOCA** EM is consistent with the noding scheme used for the experiment simulations that form the validation basis for the physical models in the code. Such noding choices have been justified by assessing the model against large and full scale experiments.

2.3 Compliance with FSLOCA EM Limitations and Conditions

The NRC's SER for Reference 1 contains 15 limitations and conditions on the NRC-approved **FSLOCA** EM. A summary of each limitation and condition and how it was met is provided below.

Limitation and Condition Number 1

Summary

The **FSLOCA** EM is not approved to demonstrate compliance with 10 CFR 50.46 acceptance criterion (b)(5) related to the long-term cooling.

Compliance

The analysis for McGuire Units 1 and 2 and Catawba Unit 1 with the **FSLOCA** EM is only being used to demonstrate compliance with the applicable ECCS acceptance criteria discussed in Section 6.0 and is not being used to demonstrate compliance with 10 CFR 50.46 (b)(5).

Limitation and Condition Number 2

Summary

The **FSLOCA** EM is approved for the analysis of Westinghouse-designed 3-loop and 4-loop PWRs with cold-side injection. Analyses should be executed consistent with the approved method, or any deviations from the approved method should be described and justified.

Compliance

McGuire Units 1 and 2 and Catawba Unit 1 are Westinghouse-designed 4-loop PWRs with cold-side injection, so they are within the NRC-approved methodology. The analysis for McGuire Units 1 and 2 and Catawba Unit 1 utilized the NRC-approved **FSLOCA** methodology, except for the changes which were previously transmitted to the NRC pursuant to 10 CFR 50.46 in References 10 through 16 and the changes for applications of **AXIOM** cladding, as described in Reference 3.

The analysis was performed with a code version which incorporated the changes and error corrections described in References 10 through 16, except for the error in the steam/fission gas specific heat calculation described in Reference 16. This error was found to have a negligible impact on analysis results with the **FSLOCA** EM, leading to an estimated PCT impact of 0°F, as described in Reference 16.

Limitation and Condition Number 3

Summary

For Region II, the containment pressure calculation will be executed in a manner consistent with the approved methodology (i.e., the COCO or LOTIC2 model will be based on appropriate plant-specific design parameters and conditions, and engineered safety features which can reduce pressure are modeled). This includes utilizing a plant-specific initial containment temperature, and only taking credit for containment coatings which are qualified and outside of the break zone-of-influence.

Compliance

The containment pressure calculation for the McGuire Units 1 and 2 and Catawba Unit 1 analysis was performed consistent with the NRC-approved methodology. Appropriate design parameters and conditions were modeled, as were the engineered safety features which can reduce the containment pressure. A plant-specific initial temperature associated with normal full-power operating conditions (applicable to all three units) was modeled, and no coatings were credited on any of the containment structures.

Limitation and Condition Number 4

Summary

The decay heat uncertainty multiplier will be **[** **]**^{a,c} The analysis simulations for the **FSLOCA** EM will not be executed for longer than 10,000 seconds following reactor trip unless the decay heat model is appropriately justified. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results will be provided in the analysis submittal in units of sigma and absolute units.

Compliance

Consistent with the NRC-approved methodology, the decay heat uncertainty multiplier was **[** **]**^{a,c} for the McGuire Units 1 and 2 and Catawba Unit 1 analysis. The analysis simulations were all executed for no longer than 10,000 seconds following reactor trip. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results have been provided in units of sigma and approximate absolute units in Table 16.

Limitation and Condition Number 5

Summary

The maximum assembly and rod length-average burnup is limited to **[** **]**^{a,c} respectively.

Compliance

The maximum analyzed assembly and rod length-average burnup were less than or equal to **[** **]**^{a,c} respectively, for McGuire Units 1 and 2 and Catawba Unit 1.

Limitation and Condition Number 6

Summary

The fuel performance data for analyses with the **FSLOCA** EM should be based on the PAD5 code (at present), which includes the effect of thermal conductivity degradation. The nominal fuel pellet average temperatures and rod internal pressures should be the maximum values, and the generation of all the PAD5 fuel performance data should adhere to the NRC-approved PAD5 methodology.

Compliance

PAD5 fuel performance data were utilized in the McGuire Units 1 and 2 and Catawba Unit 1 analysis with the **FSLOCA** EM. The analyzed fuel pellet average temperatures bound the maximum values calculated in accordance with Section 7.5.1 of Reference 2, and the analyzed rod internal pressures were calculated in accordance with Section 7.5.2 of Reference 2.

Limitation and Condition Number 7

Summary

The YDRAG uncertainty parameter should be [

]^{a,c}

Compliance

Consistent with the NRC-approved methodology, the YDRAG uncertainty parameter was [

]^{a,c} for the McGuire Units 1 and 2 and Catawba Unit 1 Region I analysis.

Limitation and Condition Number 8

Summary

The [

]^{a,c}

Compliance

Consistent with the NRC-approved methodology, the [

]^{a,c} for the McGuire Units 1 and 2 and Catawba Unit 1 Region I analysis.

Limitation and Condition Number 9

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to confirm that the **[** **]**^{a,c} for the plant design being analyzed. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Compliance

McGuire Units 1 and 2 and Catawba Unit 1 are Westinghouse-designed 4-loop PWRs. The requested sensitivity study was performed for a 4-loop Westinghouse-designed PWR and is discussed in Reference 5.

Limitation and Condition Number 10

Summary

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to: 1) demonstrate that no unexplained behavior occurs in the predicted safety criteria across the region boundary, and 2) ensure that the **[** **]**^{a,c} must cover the equivalent 2 to 4-inch break range using RCS-volume scaling relative to the demonstration plant. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Additionally, the minimum sampled break area for the analysis of Region II should be 1 ft^2.

Compliance

McGuire Units 1 and 2 and Catawba Unit 1 are Westinghouse-designed 4-loop PWRs. The requested sensitivity study was performed for a 4-loop Westinghouse-designed PWR and is discussed in Reference 5.

The minimum sampled break area for the McGuire Units 1 and 2 and Catawba Unit 1 Region II analysis is 1 ft^2.

Limitation and Condition Number 11

Summary

There are various aspects of this Limitation and Condition, which are summarized below:

1. The [ ]^{a,c} the Region I and Region II analysis seeds, and the analysis inputs will be declared and documented prior to performing the Region I and Region II uncertainty analyses. The [ ]^{a,c} and the Region I and Region II analysis seeds will not be changed throughout the remainder of the analysis once they have been declared and documented.
2. If the analysis inputs are changed after they have been declared and documented, for the intended purpose of demonstrating compliance with the applicable acceptance criteria, then the changes and associated rationale for the changes will be provided in the analysis submittal. Additionally, the preliminary values for PCT, MEM, and CWO which caused the input changes will be provided. These preliminary values are not subject to Appendix B verification, and archival of the supporting information for these preliminary values is not required.
3. Plant operating ranges which are sampled within the uncertainty analysis will be provided in the analysis submittal for both regions.

Compliance

This Limitation and Condition was met for the McGuire Units 1 and 2 and Catawba Unit 1 analysis as follows:

1. The [ ]^{a,c} the Region I and Region II analysis seeds, and the analysis inputs were declared and documented prior to performing the Region I and Region II uncertainty analyses. The [ ]^{a,c} and the Region I and Region II analysis seeds were not changed once they were declared and documented.
2. The analysis inputs were not changed once they were declared and documented.
3. The plant operating ranges which were sampled within the uncertainty analyses are provided for McGuire Units 1 and 2 and Catawba Unit 1 in Table 1.

Limitation and Condition Number 12

Summary

The plant-specific dynamic pressure loss from the SG secondary side to the main steam safety valves (MSSVs) must be adequately accounted for in analysis with the **FSLOCA** EM.

Compliance

A bounding plant-specific dynamic pressure loss from the SG secondary side to the MSSVs was modeled in the McGuire Units 1 and 2 and Catawba Unit 1 analysis.

As discussed in the response to request for additional information (RAI) 132 in Reference 17, [ ]^{a,c} To comply with this requirement, the initial opening pressure of the MSSV was modeled as the plant-specific second stage MSSV set pressure, plus uncertainty (1240.8 psia). For all three units, this value bounds the plant-specific first stage MSSV set pressure, plus uncertainty, plus the plant-specific dynamic pressure loss from the SG secondary side to the MSSVs during a SBLOCA transient.

Limitation and Condition Number 13

Summary

In plant-specific models for analysis with the **FSLOCA** EM: 1) the **[** **]**^{a,c} and 2) the **[** **]**^{a,c}

Compliance

The **[** **]**^{a,c} in the analysis for McGuire Units 1 and 2 and Catawba Unit 1. The **[** **]**^{a,c} in the analysis.

Limitation and Condition Number 14

Summary

For analyses with the **FSLOCA** EM to demonstrate compliance against the current 10 CFR 50.46 oxidation acceptance criterion, the transient time-at-temperature will be converted to an equivalent cladding reacted (ECR) using either the Baker-Just or the Cathcart-Pawel correlation. In either case, the pre-transient corrosion will be summed with the LOCA transient oxidation. If the Cathcart-Pawel correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 13 percent limit. If the Baker-Just correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 17 percent limit.

Compliance

In this analysis, the MLO acceptance criterion of 17% is replaced with the NRC-approved **AXIOM** cladding performance-based embrittlement acceptance criterion. The Cathcart-Pawel ECR is confirmed to remain below the DBT limit for **AXIOM** cladding described in Section 3.11 of Reference 3. Limitation and Condition Number 14 is therefore not applicable to the McGuire Units 1 and 2 and Catawba Unit 1 **FSLOCA** EM analysis.

Limitation and Condition Number 15

Summary

The Region II analysis will be executed twice; once assuming offsite power available (OPA) and once assuming loss-of-offsite power (LOOP). The results from both analysis executions should be shown to be in compliance with the 10 CFR 50.46 acceptance criteria.

The **[** **]**^{a,c}

Compliance

The Region II uncertainty analysis for McGuire Units 1 and 2 and Catawba Unit 1 was performed twice; once assuming OPA and once assuming LOOP. The results from both analyses that were performed are in compliance with the applicable ECCS acceptance criteria (see Section 6.0).

The **[** **]**^{a,c}

3.0 COMPOSITE MODEL APPROACH

The current licensing basis LBLOCA analysis for McGuire Units 1 and 2 and Catawba Units 1 and 2 was performed with the CQD EM. The composite model approach was described in References 18 and 19, and the composite analysis was approved by the NRC in References 20 and 21. The same approach was used in the **FSLOCA** EM analysis for McGuire Units 1 and 2 and Catawba Unit 1, except for the changes noted below.

The composite **FSLOCA** EM analysis does not cover Catawba Unit 2. A separate **FSLOCA** EM analysis will be performed for Catawba Unit 2 due to the different SG design and the importance of the SG design to the progression of Region I transients (see Section 2.3.2.7 of Reference 1).

As described on pages 1 and 2 of the Attachment to Reference 18, two vessel models were built to capture the differences in the upper internals among the units. Other minor differences were bounded in the two vessel models. Sensitivity studies were performed to identify the limiting vessel model. Page 50 of the presentation in Enclosure 2 of Reference 19 indicates that the McGuire Unit 1 vessel model was determined to be limiting, so this model was used in the CQD LBLOCA analysis. It was confirmed that the McGuire Unit 1 vessel model is also appropriate for the Region I and Region II analyses with the **FSLOCA** EM, so this vessel model was also used in these analyses.

As described on page 2 of the Attachment to Reference 18, sensitivity studies were performed to identify the direction of conservatism for the accumulator line resistance and accumulator water volume. Pages 32 and 41 of the presentation in Enclosure 2 of Reference 19 indicate that minimum line resistance and minimum water volume were determined to be limiting. Therefore, the minimum line resistance (based on McGuire Unit 2) and the minimum water volume range (based on McGuire Units 1 and 2) were used in the CQD LBLOCA analysis. The wider accumulator pressure range for Catawba Units 1 and 2, which encompasses the range for McGuire Units 1 and 2, was also used.

Similar sensitivity studies were performed for the Region II analysis with the **FSLOCA** EM, which showed similar trends as the sensitivity studies for the CQD LBLOCA analysis. Therefore, the same accumulator modeling approach used in the CQD LBLOCA analysis was also used in the Region II analysis with the **FSLOCA** EM (minimum line resistance, minimum water volume range, wider pressure range).

Figure 31.3-4 of Reference 1 shows a correlation between accumulator pressure and PCT for the Region I demonstration analysis with the **FSLOCA** EM. Lower accumulator pressure delays the accumulator injection which delays the termination of the boil-off heatup for SBLOCA transients, leading to a higher PCT. The wider accumulator pressure range (555 – 708 psig) was used in the CQD LBLOCA analysis and was also used in the Region II analysis with the **FSLOCA** EM. The narrower accumulator pressure range for McGuire Units 1 and 2 (555 – 669 psig) results in lower sampled pressures and thus higher PCTs for Region I, so the narrower accumulator pressure range was used in the Region I analysis with the **FSLOCA** EM for McGuire Units 1 and 2 and Catawba Unit 1. The accumulator line resistance and accumulator water volume have a negligible impact on the calculated results for Region I, so the same modeling approach used in the Region II analysis with the **FSLOCA** EM was also used in the Region I analysis with the **FSLOCA** EM (minimum line resistance, minimum water volume range).

Reference 22 describes the ongoing process that has been followed to assure that the composite CQD analysis for McGuire Units 1 and 2 and Catawba Units 1 and 2 has remained representative and bounding for these units. The same process will be followed to assure that the composite **FSLOCA** EM analysis for McGuire Units 1 and 2 and Catawba Unit 1 remains representative and bounding for these units.

4.0 REGION I ANALYSIS

4.1 Description of Representative Transient

The small-break LOCA transient can be divided into time periods in which specific phenomena are occurring, as discussed below.

Blowdown

The rapid depressurization of the RCS coincides with subcooled liquid flow through the break. Following the reactor trip on the low pressurizer pressure setpoint, the pressurizer drains, and safety injection (SI) is initiated on the low pressurizer pressure SI setpoint. After reaching this setpoint and applying the SI delays, high head safety injection (HHSI) flow begins. Phase separation begins in the upper head and upper plenum near the end of this period until the entire RCS eventually reaches saturation, ending the rapid depressurization slightly above the SG secondary side pressure near the modeled MSSV setpoint.

Natural Circulation

This quasi-equilibrium phase persists while the RCS pressure remains slightly above the secondary side pressure. The system drains from the top down, and while significant mass is continually lost through the break, the vapor generated in the core is trapped in the upper regions of the primary side (reactor vessel, hot legs, and steam generators) by the liquid remaining in the crossover leg loop seals. Throughout this period, the core remains covered by a two-phase mixture and the fuel cladding temperatures remain at the saturation temperature level.

Loop Seal Clearance

As the system drains, the liquid levels in the downhill side of the pump suction (crossover leg) become depressed all the way to the bottom elevations of the piping, allowing the steam trapped during the natural circulation phase to vent to the break (i.e., a process called loop seal clearance). The break flow and the flow through the RCS loops with a cleared loop seal become primarily vapor. Relief of a static head imbalance allows for a quick but temporary recovery of liquid levels in the inner portion of the reactor vessel.

Boil-Off

With a vapor vent path established after the loop seal clearance, the RCS depressurizes at a rate controlled by the critical flow, which continues to be a primarily high-quality mixture of water and steam. The RCS pressure remains high enough such that SI flow cannot make up for the primary system fluid inventory lost through the break, leading to core uncovery and a fuel rod cladding temperature heatup.

Core Recovery

The RCS pressure continues to decrease, and once it reaches that of the accumulator gas pressure, the introduction of additional ECCS water from the accumulators replenishes the reactor vessel inventory and recovers the core mixture level. The transient is considered over as the break flow is compensated by the SI flow.

4.2 Analysis Results

The McGuire Units 1 and 2 and Catawba Unit 1 Region I analysis was performed in accordance with the NRC-approved methodology in Reference 1, with exceptions identified under Limitation and Condition Number 2 in Section 2.3. [

]^{a,c}

The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Item 5.0a in Table 1. Control rod drop is modeled for breaks less than 1 square foot assuming a 2.0-second signal delay time and a 3.3-second rod drop time. RCP trip is modeled coincident with reactor trip on the low pressurizer pressure setpoint for LOOP transients. When the low pressurizer pressure SI setpoint is reached, there is a delay to account for emergency diesel generator start-up, filling headers, etc., after which SI is initiated into the RCS.

The results of the McGuire Units 1 and 2 and Catawba Unit 1 Region I uncertainty analysis are summarized in Table 6. The sampled decay heat uncertainty multipliers for the Region I analysis cases are provided in Table 16.

Table 7 contains a sequence of events for the transient that produced the Region I analysis PCT result. Figures 1 through 13 illustrate the calculated key transient response parameters for this transient. Table 8 contains a sequence of events for the transient that produced the Region I analysis MEM result. The CWO result in Table 6 is 0.00%. This result applies to all runs in the Region I analysis, so the sequence of events table is not provided for a specific run with this CWO result.

5.0 REGION II ANALYSIS

5.1 Description of Representative Transient

A large-break LOCA transient can be divided into phases in which specific phenomena are occurring. A convenient way to divide the transient is in terms of the various heatup and cooldown phases that the fuel assemblies undergo. For each of these phases, specific phenomena and heat transfer regimes are important, as discussed below.

Blowdown – Critical Heat Flux (CHF) Phase

In this phase, the break flow is subcooled, the discharge rate of coolant from the break is high, the core flow reverses, the fuel rods go through departure from nucleate boiling (DNB), and the cladding rapidly heats up and the reactor is shut down due to the core voiding.

The regions of the RCS with the highest initial temperatures (upper core, upper plenum, and hot legs) begin to flash during this period. This phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture level swells, and a saturated mixture is pushed into the core by the intact loop RCPs, still rotating in single-phase liquid. As the fluid in the cold leg reaches saturation conditions, the discharge flow rate at the break decreases significantly.

Blowdown – Upward Core Flow Phase

Heat transfer is increased as the two-phase mixture is pushed into the core. The break discharge rate is reduced because the fluid becomes saturated at the break. This phase ends as the lower plenum mass is depleted, the fluid in the loops become two-phase, and the RCP head degrades.

Blowdown – Downward Core Flow Phase

The break flow begins to dominate and pulls flow down through the core as the RCP head degrades due to increased voiding, while liquid and entrained liquid flows also provide core cooling. Heat transfer in this period may be enhanced by liquid flow from the upper head. Once the system has depressurized to less than the accumulator cover pressure, the accumulators begin to inject cold water into the cold legs. During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is bypassed around the downcomer and sent out through the break. As the system pressure continues to decrease, the break flow and consequently the downward core flow are reduced. The system pressure approaches the containment pressure at the end of this last period of the blowdown phase.

During this phase, the core begins to heat up as the system approaches containment pressure, and the phase ends when the reactor vessel begins to refill with ECCS water.

Refill Phase

The core continues to heat up as the lower plenum refills with ECCS water. This phase is characterized by a rapid increase in fuel cladding temperature at all elevations due to the lack of liquid and steam flow in the core region. The water completely refills the lower plenum and the refill phase ends. As ECCS water enters the core, the fuel rods in the lower core region begin to quench and liquid entrainment begins, resulting in increased fuel rod heat transfer.

Reflood Phase

During the early reflood phase, the accumulators begin to empty and nitrogen is discharged into the RCS. The nitrogen surge forces water into the core, which is then evaporated, causing system re-pressurization and a temporary reduction of pumped ECCS flow; this re-pressurization is illustrated by the increase in RCS pressure. During this time, core cooling may increase due to vapor generation and liquid entrainment, but conversely the early reflood pressure spike results in loss of mass out through the broken cold leg.

The pumped ECCS water aids in the filling of the downcomer throughout the reflood period. As the quench front progresses further into the core, the PCT elevation moves increasingly higher in the fuel assembly.

As the transient progresses, continued injection of pumped ECCS water refloods the core, effectively removes the reactor vessel metal mass stored energy and core decay heat, and leads to an increase in the reactor vessel fluid mass. Eventually the core inventory increases enough that liquid entrainment is able to quench all the fuel assemblies in the core.

5.2 Analysis Results

The McGuire Units 1 and 2 and Catawba Unit 1 Region II analysis was performed in accordance with the NRC-approved methodology in Reference 1, with exceptions identified under Limitation and Condition Number 2 in Section 2.3. The analysis was performed assuming both OPA and LOOP, and the results of both of the OPA and LOOP analyses are compared to the applicable ECCS acceptance criteria. The most limiting ECCS single failure of one ECCS train is assumed in the analysis as identified in Item 5.0a in Table 1. The results of the McGuire Units 1 and 2 and Catawba Unit 1 Region II OPA and LOOP uncertainty analyses are summarized in Table 6. The sampled decay heat uncertainty multipliers for the Region II analysis cases are provided in Table 16.

Table 9 identifies the break size for the cases that produced the analysis results. Tables 10 through 15 contain a sequence of events for these transients. Figures 14 through 27 illustrate the key response parameters for the transient that produced the analysis PCT result for the uncertainty analysis with OPA.

The containment pressure is calculated using the LOTIC2 code (References 23 and 24) for ice condenser containments. The containment model input is summarized in Tables 2 and 3. The conservatively low containment pressure response used for the Region II analysis is compared to the calculated containment backpressure in Figure 21, consistent with the methodology in Reference 1.

Figures 28 and 29 show the PCT versus effective break area multiplier for the Region II OPA and LOOP uncertainty analyses, respectively. These figures reflect the combined effect of the break size and break flow model uncertainties. Figures 30 and 31 show the transient ECR versus PCT for the Region II OPA and LOOP uncertainty analyses, respectively. Figures 32 and 33 show the CWO versus PCT for the Region II OPA and LOOP uncertainty analyses, respectively. Strong trends of increasing ECR and CWO with increasing PCT occur due to the temperature dependence of the oxidation kinetics.

The uncertainty analysis methodology used in the **FSLOCA** EM is described in Section 30 of Reference 1. A Monte Carlo sampling of all uncertainty contributors leads to the generation of a sample of simulated results from which upper tolerance limits are derived for the analysis figures of merit (PCT, MEM, CWO).

[

]^{a,c}

6.0 COMPLIANCE WITH APPLICABLE ECCS ACCEPTANCE CRITERIA

It must be demonstrated that there is a high level of probability that the following acceptance criteria in 10 CFR 50.46, as modified for **AXIOM** cladding in Section 6.2.1.4 of Reference 3, are met:

(b)(1) The **FSLOCA** EM analysis PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCTs in Table 6 for McGuire Units 1 and 2 and Catawba Unit 1 are less than 2,200°F, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Cladding Temperature does not exceed 2,200°F," is demonstrated.

(b)(2) The maximum MLO acceptance criterion (17%) defined in 10 CFR 50.46 (b)(2) is replaced with the NRC-approved **AXIOM** cladding performance-based embrittlement acceptance criterion.

The **FSLOCA** EM analysis MEM corresponds to a bounding estimate of the 95th percentile MEM at the 95-percent confidence level. Since the resulting Cathcart-Pawel ECRs for McGuire Units 1 and 2 and Catawba Unit 1 remain below the DBT limit for **AXIOM** cladding described in Section 3.11 of Reference 3 (i.e., MEM $\geq$ 0%) per Table 6, the analysis confirms that the **AXIOM** cladding performance-based embrittlement acceptance criterion is demonstrated.

(b)(3) The **FSLOCA** EM analysis CWO corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. Since the resulting CWOs in Table 6 for McGuire Units 1 and 2 and Catawba Unit 1 are less than 1 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation does not exceed 1 percent," is demonstrated.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains in a coolable geometry.

This acceptance criterion is met by demonstrating compliance with the first two acceptance criteria (PCT and local oxidation (MEM)), and by assuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed. The PCT and local oxidation (MEM) acceptance criteria have been met for McGuire Units 1 and 2 and Catawba Unit 1 per Table 6.

It is discussed in Section 32.1 of the NRC-approved **FSLOCA** EM (Reference 1) that the effects of LOCA and seismic loads on the core geometry do not need to be considered unless fuel assembly grid deformation extends beyond the core periphery (i.e., deformation in a fuel assembly with no sides adjacent to the core baffle plates). The **FSLOCA** EM analysis does not affect the existing calculations that support the analysis of record related to combined LOCA and seismic loads. The previous calculations on grid deformation due to combined LOCA and seismic loads remain valid. As described in Section 4.2.1.3.2 of the McGuire Nuclear Station Updated Final Safety Analysis Report (UFSAR) and Section 4.2.4.5 of the Catawba Nuclear Station UFSAR, "Grid crush analyses using combined seismic and LOCA loadings show that the fuel assembly will maintain a geometry that is capable of being cooled under the worst-case accident Condition IV event." Therefore, inboard grid deformation due to combined LOCA and seismic loads is not calculated to occur for McGuire Units 1 and 2 and Catawba Unit 1.

(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 the continued delivery of cooling water to the core. The actions that are currently in place to maintain long-term cooling are not impacted by the application of the NRC-approved **FSLOCA** EM (Reference 1).

Based on the above discussion, it is concluded that McGuire Units 1 and 2 and Catawba Unit 1 comply with the acceptance criteria in 10 CFR 50.46, as modified for **AXIOM** cladding in Section 6.2.1.4 of Reference 3.

7.0 ANALYSIS APPLICABILITY

The **FSLOCA** EM analysis for McGuire Units 1 and 2 and Catawba Unit 1 supports 18-month cycle operation with **AXIOM** cladding, the **PRIME** bottom nozzle, and an EPU. This analysis does not support the current cladding material (**Optimized ZIRLO™** cladding).

During fuel transitions, co-resident fuel effects typically require evaluation. For example, differences in fuel assembly hydraulic resistances can lead to a LOCA penalty. In the case of transitioning to **AXIOM** cladding and the **PRIME** bottom nozzle from legacy Westinghouse fuel products, there are no (or negligible) co-resident fuel effects.

The small differences in the bottom nozzle metal mass and pressure loss coefficient have a negligible effect on LOCA analyses. Furthermore, the bottom nozzle loss coefficient is not explicitly modeled in the WCT-TF2 model. The pressure loss information that is used for the WCT-TF2 steady-state calibration, as prescribed by the NRC-approved **FSLOCA** methodology (Section 26.4 of Reference 1), captures the change in the bottom nozzle loss coefficient. Because the bottom nozzle differences are only considered in the WCT-TF2 steady-state calibration and not in the WCT-TF2 model itself, the **PRIME** bottom nozzle is considered to be hydraulically similar to the prior bottom nozzle with respect to mixed core effects within the established Westinghouse mixed core evaluation technique. The cladding material change also does not introduce any hydraulic differences. Therefore, a transition core evaluation for the LOCA analysis is not needed to address a fuel assembly hydraulic resistance mismatch.

The change from **Optimized ZIRLO** cladding to **AXIOM** cladding leads to differences in the fuel pellet temperatures and rod internal pressures, so separate LOCA analyses are required to cover both cladding materials. During the two transition cycles for each unit, the new fuel product (**AXIOM** cladding) will be covered by the **FSLOCA** EM analysis, while the existing fuel product (**Optimized ZIRLO** cladding) will continue to be covered by the current licensing basis LOCA analyses with the NOTRUMP and CQD EMs. For the transition cycles, the portions of the Technical Specifications (TS) that apply to the current licensing basis LOCA analyses will be maintained, while the TS changes to support the new fuel/methodologies will also be incorporated. Following discharge of all fuel with **Optimized ZIRLO** cladding, the **FSLOCA** EM analysis will cover all fuel in the core, and the portions of the TS that apply to the current licensing basis LOCA analyses will no longer be applicable.

The **FSLOCA** EM analysis supports the uprated power level (3700 MWt + 0.3% uncertainty per Item 1.0a in Table 1), while the current licensing basis LOCA analyses support the current power level (maximum power, including uncertainty = ~3479 MWt). The current power level cannot be exceeded during the fuel product transition. The uprated power level cannot be used until all fuel with **Optimized ZIRLO** cladding has been discharged and the **FSLOCA** EM analysis covers all fuel in the core.

It is noted that maintaining the current licensing basis for the outgoing fuel is consistent with the licensing precedent for vendor-to-vendor fuel transitions. For example, Reference 25 documents the license amendment for the transition to Westinghouse fuel for Sequoyah. Westinghouse performed licensing basis analyses (e.g., PAD5, **FSLOCA** EM) that were consistent with current standards of the NRC, and the Framatome licensing basis analyses were maintained for their fuel product. The outgoing fuel in the core was covered under the existing licensing basis analyses, which was deemed to be acceptable by the NRC, and only the Westinghouse fuel was explicitly covered under the new analyses.

8.0 REFERENCES

1. "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (**FULL SPECTRUM** LOCA Methodology)," WCAP-16996-P-A, Revision 1, November 2016.
2. "Westinghouse Performance Analysis and Design Model (PAD5)," WCAP-17642-P-A, Revision 1, November 2017.
3. "Westinghouse **AXIOM**® Cladding for Use in Pressurized Water Reactor Fuel," WCAP-18546-P-A, Revision 0, March 2023.
4. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974.
5. "'Information to Satisfy the **FULL SPECTRUM** LOCA (**FSLOCA**) Evaluation Methodology Plant Type Limitations and Conditions for 4-loop Westinghouse Pressurized Water Reactors (PWRs)' (Proprietary/Non-Proprietary)," LTR-NRC-18-50, Revision 0, July 2018.
6. "Emergency Core Cooling Systems: Revisions to Acceptance Criteria," Federal Register, V53, N180, pp. 35996-36005, September 1988.
7. "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A Volume 1, Revision 2 and Volumes 2-5, Revision 1, March 1998.
8. "Best Estimate Calculations of Emergency Core Cooling System Performance," Regulatory Guide 1.157, USNRC, May 1989.
9. "Transient and Accident Analysis Methods," Regulatory Guide 1.203, USNRC, December 2005.
10. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2017," LTR-NRC-18-30, Revision 0, July 2018.
11. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2018," LTR-NRC-19-6, Revision 0, February 2019.
12. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2020," LTR-NRC-21-5, Revision 0, March 2021.
13. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2021," LTR-NRC-22-8, Revision 0, February 2022.
14. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2022," LTR-NRC-23-5, Revision 0, March 2023.
15. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2023," LTR-NRC-24-6, Revision 0, March 2024.
16. "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2024," LTR-NRC-25-11, Revision 0, March 2025.
17. "Submittal of Westinghouse Responses to 'WCAP-16996-P, 'Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)' Request for Additional Information – Set 8 RAIs 127, 132-135 and 137-139' (Proprietary/Non-Proprietary), Project 700, TAC No. ME5244," LTR-NRC-14-4, Revision 0, January 2014.

18. "Duke Energy Corporation, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, McGuire Nuclear Station, Units 1 and 2, Docket Number 50-369 and 50-370, Implementation of Best-Estimate Large Break LOCA Methodology," April 2000. (ADAMS Accession No. ML003704107)
19. "MEETING SUMMARY - MEETING OF JUNE 12, 2000, REGARDING WESTINGHOUSE BEST-ESTIMATE LARGE BREAK LOCA METHODOLOGY," USNRC, June 2000. (ADAMS Accession No. ML003728321)
20. "MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA8696 AND MA8697)," USNRC, September 2000. (ADAMS Accession No. ML003753895)
21. "CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA8719 AND MA8720)," USNRC, October 2000. (ADAMS Accession No. ML003756631)
22. "Duke Energy Corporation, McGuire Nuclear Station, Units 1 and 2, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, License Amendment Request, Implementation of Best-Estimate Large Break Loss of Coolant Accident (BELBLOCA) Analysis Methodology," August 2000. (ADAMS Accession No. ML003741728)
23. "Westinghouse Emergency Core Cooling System Evaluation Model – Summary," WCAP-8339, Revision 0, June 1974.
24. "Long Term Ice Condenser Containment Code – LOTIC Code," WCAP-8354-P-A, Supplement 1, Revision 0, April 1976.
25. "SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 356 AND 349 REGARDING THE TRANSITION TO WESTINGHOUSE ROBUST FUEL ASSEMBLY-2 (RFA-2) FUEL (EPID L-2020-LLA-0216)," USNRC, October 2021. (ADAMS Accession No. ML21245A267)

Table 1. Plant Operating Range Analyzed and Key Parameters for McGuire Units 1 and 2 and Catawba Unit 1

Parameter		As-Analyzed Value or Range
1.0	**Core Parameters**	
	a) Core power	≤ 3700 MWt + 0.3% uncertainty
	b) Fuel type	17x17 RFA-2 fuel with **AXIOM** cladding, **PRIME** bottom nozzle, and integral fuel burnable absorber (IFBA). Off-spec enriched uranium product (OSEUP) may also be used.
	c) Maximum total core peaking factor (F_Q), including uncertainties	2.70
	d) Maximum hot channel enthalpy rise peaking factor ($F_{\Delta H}$), including uncertainties	1.72
	e) Axial flux difference (AFD) band at 100% power	-22.58% to +14.58%
2.0	**Reactor Coolant System Parameters**	
	a) Thermal design flow (TDF)	94,866 gpm/loop
	b) Vessel average temperature (T_{AVG})	581.0°F ≤ T_{AVG} ≤ 594.0°F Coastdown to 572.5°F
	c) Pressurizer pressure (P_{RCS})	2200 psia ≤ P_{RCS} ≤ 2300 psia
	d) Reactor coolant pump (RCP) model and power	Model 93A, 7000 hp
3.0	**Containment Parameters**	
	a) Containment modeling	Region I: Constant pressure equal to initial containment pressure Region II: Conservatively low containment pressure calculated using the information in Tables 2 and 3
4.0	**Steam Generator (SG) and Secondary Side Parameters**	
	a) Steam generator tube plugging level	≤ 10%
	b) Main steam safety valve (MSSV) set pressures	Pressure at which MSSV begins to open = 1240.8 psia Pressure at which MSSV is fully open = 1343.3 psia These pressures are based on the second stage MSSV, as discussed for Limitation and Condition Number 12 in Section 2.3.
	c) Main feedwater temperature	440°F
	d) Auxiliary feedwater temperature	85°F
	e) Auxiliary feedwater flow rate	220 gpm (OPA) or 760 gpm (LOOP) (total flow to all SGs)

Table 1. Plant Operating Range Analyzed and Key Parameters for McGuire Units 1 and 2 and Catawba Unit 1

Parameter		As-Analyzed Value or Range
5.0	**Safety Injection (SI) Parameters**	
	a) Single failure configuration	Loss of one train of pumped ECCS
	b) Safety injection temperature (T_{SI})	60°F $\leq T_{SI} \leq$ 110°F
	c) Low pressurizer pressure safety injection safety analysis limit	1700 psig
	d) Initiation delay time from low pressurizer pressure SI setpoint to full SI flow	≤ 17 seconds (OPA) or ≤ 32 seconds (LOOP)
	e) Safety injection flow	Minimum flows in Table 4 (Region I) or Table 5 (Region II)
6.0	**Accumulator Parameters**	
	a) Accumulator temperature (T_{ACC})	105°F $\leq T_{ACC} \leq$ 125°F
	b) Accumulator water volume (V_{ACC})	907.69 $ft^3 \leq V_{ACC} \leq$ 992.18 ft^3 (6790 gal $\leq V_{ACC} \leq$ 7422 gal)
	c) Accumulator pressure (P_{ACC})	Region I Analysis Range: 555 psig $\leq P_{ACC} \leq$ 669 psig Region II Analysis Range: 555 psig $\leq P_{ACC} \leq$ 708 psig
	d) Accumulator boron concentration	≥ 1950 ppm
7.0	**Reactor Protection System Parameters**	
	a) Low pressurizer pressure reactor trip signal processing time	≤ 2.0 seconds
	b) Low pressurizer pressure reactor trip setpoint	1800 psig

Table 2. Containment Data Used for Region II Calculation of Containment Pressure for McGuire Units 1 and 2 and Catawba Unit 1

Parameter	Value
Maximum containment upper compartment net free volume	676,255 ft^3
Minimum containment lower compartment net free volume	197,800 ft^3
Minimum containment dead-ended compartment net free volume	146,052 ft^3
Maximum containment upper compartment initial temperature at full power operation	105°F
Maximum containment lower compartment initial temperature at full power operation	125°F
Maximum containment dead-ended compartment initial temperature at full power operation	125°F
Maximum containment ice bed compartment initial temperature at full power operation	30°F
Maximum number of containment air return fans (deck fans) in operation during LOCA transient	2 fans
Minimum air return fan (deck fan) delay time	8.0 minutes
Maximum containment air return fan (deck fan) flow rate per fan	40,000 cfm/fan
Maximum number of containment spray pumps in operation during LOCA transient	0
Minimum refueling water storage tank (RWST) temperature for broken loop spilling SI	58°F
Active sump maximum volume	76,996 ft^3
Containment walls / heat sink properties	Table 3
Minimum containment outside air / ground temperature	-5°F
Minimum initial containment pressure at normal full power operation	14.4 psia
Maximum number of containment venting lines (including purge lines, pressure relief lines or any others) which can be OPEN at onset of transient at full power operation	1 vent line
Maximum effective valve diameter of each containment venting line	6.357 inches
Maximum delay time between SI signal and start of venting valve closure	28 seconds
Maximum venting valve closure time at normal full power operation	5 seconds
SI spilling flows	267.1 lbm/s
Minimum annulus temperature	-5°F

Table 3. Containment Heat Sink Data Used for Region II Calculation of Containment Pressure for McGuire Units 1 and 2 and Catawba Unit 1

Wall	Area (ft^2)	Thickness (ft)	Material
1	21,142	1.34	Concrete
2	5,017	0.0156 1.5	Stainless Steel Concrete
3	24,391	0.058	Carbon Steel
4	31,035	0.0290	Carbon Steel
5	801	0.0625	Stainless Steel
6	57,387	1.97	Concrete
7	9,019	2.04	Concrete
8	3,541	2.5	Concrete
9	2,361	0.0156 1.5	Stainless Steel Concrete
10	768	0.04207 1.5	Carbon Steel Concrete
11	21,278	0.0535	Carbon Steel
12	35,273	0.0535	Carbon Steel
13	14,445	0.0625	Carbon Steel
14	9,040	0.0625	Carbon Steel
15	32,640	0.0026	Stainless Steel
16	51,000	0.00042	Copper
17	9,600	0.00833	Stainless Steel

Table 4. Safety Injection Flow Used for Region I Calculation for McGuire Units 1 and 2 and Catawba Unit 1

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm/loop)	Intermediate Head Safety Injection (IHSI) Flow (gpm/loop)	Low Head Safety Injection (LHSI) Flow (gpm/loop)
14.7	89.8	129.6	844.9
34.7	89.0	129.2	801.2
54.7	88.3	128.6	747.3
74.7	87.7	128.1	683.4
94.7	87.1	127.5	609.3
114.7	86.6	126.9	525.0
134.7	86.3	126.2	430.7
154.7	85.3	125.5	326.2
174.7	84.8	124.8	211.5
194.7	84.3	124.0	86.8
194.71	84.3	124.0	0.0
214.7	83.8	123.2	0.0
314.7	81.0	118.6	0.0
414.7	77.9	113.2	0.0
514.7	74.5	106.8	0.0
614.7	70.9	99.6	0.0
714.7	66.9	91.5	0.0
814.7	62.6	82.5	0.0
914.7	58.0	72.6	0.0
1014.7	53.1	61.8	0.0
1114.7	47.9	50.1	0.0
1214.7	42.4	37.6	0.0
1314.7	36.7	24.1	0.0
1414.7	33.7	9.8	0.0
1414.71	33.7	0.0	0.0
1514.7	30.6	0.0	0.0
1614.7	24.1	0.0	0.0
1614.71	0.0	0.0	0.0
2500.0	0.0	0.0	0.0

Table 5. Safety Injection Flow Used for Region II Calculation for McGuire Units 1 and 2 and Catawba Unit 1

Pressure (psia)	High Head Safety Injection (HHSI) Flow (gpm/loop)	Intermediate Head Safety Injection (IHSI) Flow (gpm/loop)	Low Head Safety Injection (LHSI) Flow (gpm/loop)
14.7	89.8	132.0	853.5
34.7	89.0	130.8	700.5
54.7	88.3	129.6	545.8
74.7	87.7	128.5	389.5
94.7	87.1	127.4	231.6
114.7	86.6	126.5	72.1
114.71	86.6	126.5	0.0
134.7	86.3	125.6	0.0
154.7	85.3	121.8	0.0
174.7	84.8	121.0	0.0
194.7	84.3	120.3	0.0
214.7	83.8	119.5	0.0
314.7	81.0	114.7	0.0
414.7	77.9	108.7	0.0
514.7	74.5	101.5	0.0
614.7	70.9	93.1	0.0
714.7	66.9	83.5	0.0
814.7	62.6	72.8	0.0
914.7	58.0	60.8	0.0
1014.7	53.1	47.7	0.0
1114.7	47.9	33.3	0.0
1214.7	42.4	17.8	0.0
1214.71	42.4	0.0	0.0
1314.7	36.7	0.0	0.0
1514.7	30.6	0.0	0.0
1614.7	24.1	0.0	0.0
1614.71	0.0	0.0	0.0
2500.0	0.0	0.0	0.0

Table 6. McGuire Units 1 and 2 and Catawba Unit 1 Analysis Results with the FSLOCA EM

Outcome	Region I Value	Region II Value (OPA)	Region II Value (LOOP)
95/95 PCT	1199°F	1662°F	1647°F
95/95 MEM	5.33%	5.07%	4.82%
95/95 CWO	0.00%	0.09%	0.08%

Table 7. Sequence of Events for the McGuire Units 1 and 2 and Catawba Unit 1 Region I Analysis PCT Case

Event	Time after Break (s)
Start of Transient	0.0
Reactor Trip Signal	16.4
Safety Injection Signal	27.9
Safety Injection Begins	59.9
Loop Seal Clearing Occurs	768
Top of Core Uncovered	1028
Accumulator Injection Begins	1610
PCT Occurs	1611
Top of Core Recovered	1998

Table 8. Sequence of Events for the McGuire Units 1 and 2 and Catawba Unit 1 Region I Analysis MEM Case

Event	Time after Break (s)
Start of Transient	0.0
Reactor Trip Signal	22.6
Safety Injection Signal	35.5
Safety Injection Begins	67.5
Loop Seal Clearing Occurs	922
Top of Core Uncovered	1136
Accumulator Injection Begins	1436
PCT Occurs	1438
Top of Core Recovered	1602

Table 9. Break Sizes for McGuire Units 1 and 2 and Catawba Unit 1 Region II Analysis Cases

Analysis Case	Region II Analysis with OPA		Region II Analysis with LOOP	
	Break Type	Effective Break Area Multiplier	Break Type	Effective Break Area Multiplier
PCT	DEG	2.6613	DEG	2.4245
MEM	DEG	2.4586	DEG	2.4586
CWO	DEG	2.7980	Split	2.6349

Table 10. Sequence of Events for the McGuire Units 1 and 2 and Catawba Unit 1 Region II Analysis PCT Case with OPA

Event	Time after Break (s)
Start of Transient	0.0
Fuel Rod Burst Occurs	2.0
Safety Injection Signal	5.5
Accumulator Injection Begins	12.0
End of Blowdown	22.0
Safety Injection Begins	22.5
Accumulator Empty	55.5
PCT Occurs	112
All Rods Quenched	273

Table 11. Sequence of Events for the McGuire Units 1 and 2 and Catawba Unit 1 Region II Analysis PCT Case with LOOP

Event	Time after Break (s)
Start of Transient	0.0
Safety Injection Signal	5.4
Fuel Rod Burst Occurs	5.4
PCT Occurs	6.3
Accumulator Injection Begins	11.5
End of Blowdown	15.0
Safety Injection Begins	37.4
Accumulator Empty	53.5
All Rods Quenched	325

Table 12. Sequence of Events for the McGuire Units 1 and 2 and Catawba Unit 1 Region II Analysis MEM Case with OPA

Event	Time after Break (s)
Start of Transient	0.0
Fuel Rod Burst Occurs	2.1
Safety Injection Signal	5.4
Accumulator Injection Begins	11.5
End of Blowdown	21.5
Safety Injection Begins	22.4
Accumulator Empty	55.5
PCT Occurs	112
All Rods Quenched	379

Table 13. Sequence of Events for the McGuire Units 1 and 2 and Catawba Unit 1 Region II Analysis MEM Case with LOOP

Event	Time after Break (s)
Start of Transient	0.0
Fuel Rod Burst Occurs	2.1
Safety Injection Signal	5.4
Accumulator Injection Begins	11.0
End of Blowdown	21.5
Safety Injection Begins	37.4
Accumulator Empty	55.0
PCT Occurs	119
All Rods Quenched	291

Table 14. Sequence of Events for the McGuire Units 1 and 2 and Catawba Unit 1 Region II Analysis CWO Case with OPA

Event	Time after Break (s)
Start of Transient	0.0
Fuel Rod Burst Occurs	2.9
Safety Injection Signal	5.1
Accumulator Injection Begins	12.0
End of Blowdown	15.5
Safety Injection Begins	22.1
Accumulator Empty	61.6
PCT Occurs	88.5
All Rods Quenched	358

Table 15. Sequence of Events for the McGuire Units 1 and 2 and Catawba Unit 1 Region II Analysis CWO Case with LOOP

Event	Time after Break (s)
Start of Transient	0.0
Safety Injection Signal	5.3
Accumulator Injection Begins	12.5
End of Blowdown	22.5
Safety Injection Begins	37.3
Fuel Rod Burst Occurs	38.5
Accumulator Empty	54.0
PCT Occurs	87.3
All Rods Quenched	411

Table 16. Sampled Value of Decay Heat Uncertainty Multiplier, DECAY_HT, for McGuire Units 1 and 2 and Catawba Unit 1 Region I and Region II Analysis Cases

Region	Case	DECAY_HT (units of σ)	DECAY_HT (absolute units)*
Region I	PCT	+0.3642σ	2.03%
	MEM	+1.0004σ	5.58%
	CWO	N/A**	N/A**
Region II (OPA)	PCT	+0.1338σ	0.71%
	MEM	+1.2212σ	6.78%
	CWO	+0.6222σ	3.27%
Region II (LOOP)	PCT	+0.5801σ	3.02%
	MEM	+1.2212σ	6.78%
	CWO	+0.4004σ	2.14%

*Approximate uncertainty in total decay heat power at 1 second after shutdown as defined by the ANSI/ANS-5.1-1979 decay heat standard for ^{235}U, ^{239}Pu, and ^{238}U assuming infinite operation.

**No decay heat uncertainty value is provided for the Region I CWO case since the analysis result for all runs is 0.00%.

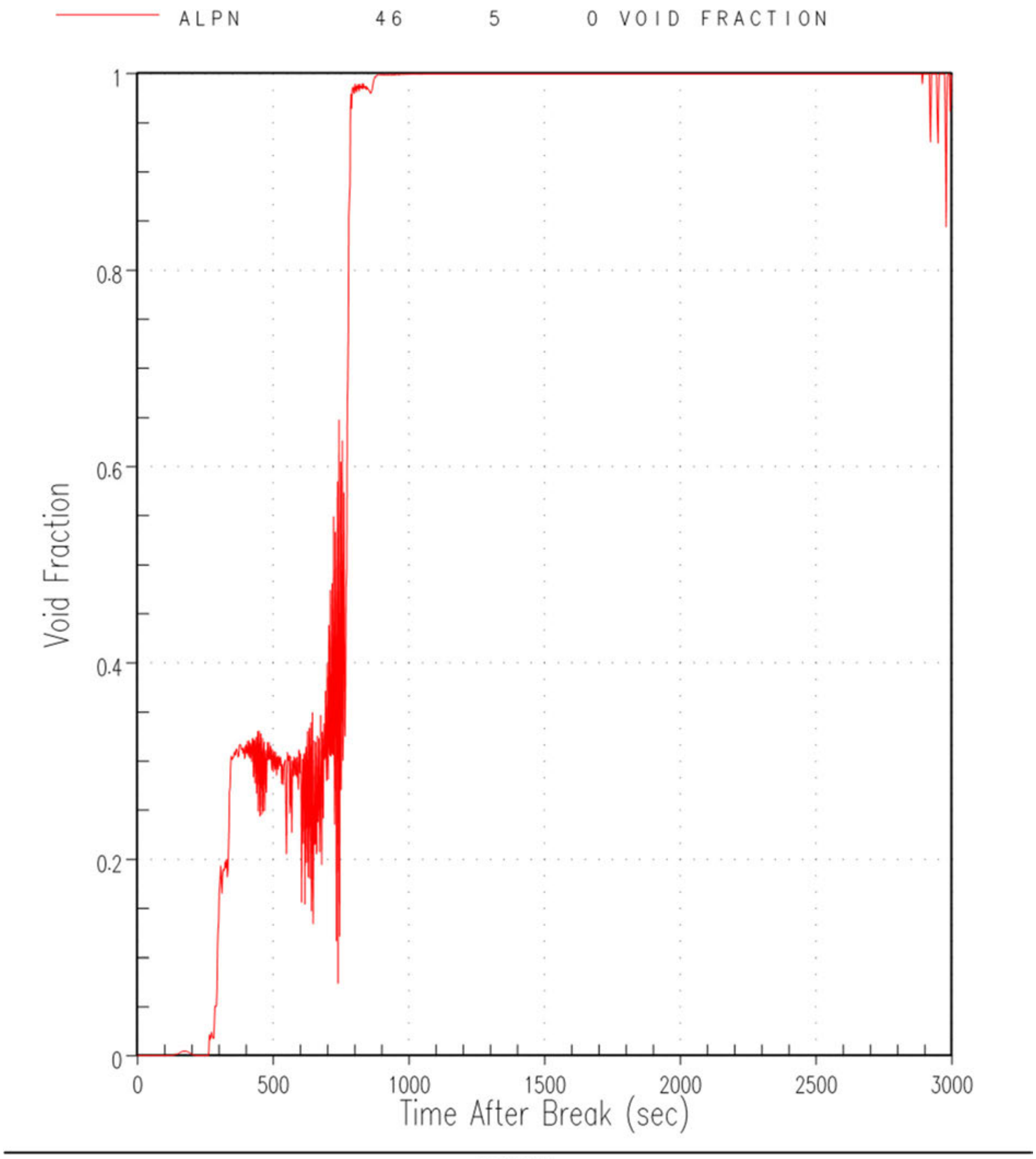


Figure 1: McGuire Units 1 and 2 and Catawba Unit 1 Break Flow Void Fraction for the Region I Analysis PCT Case

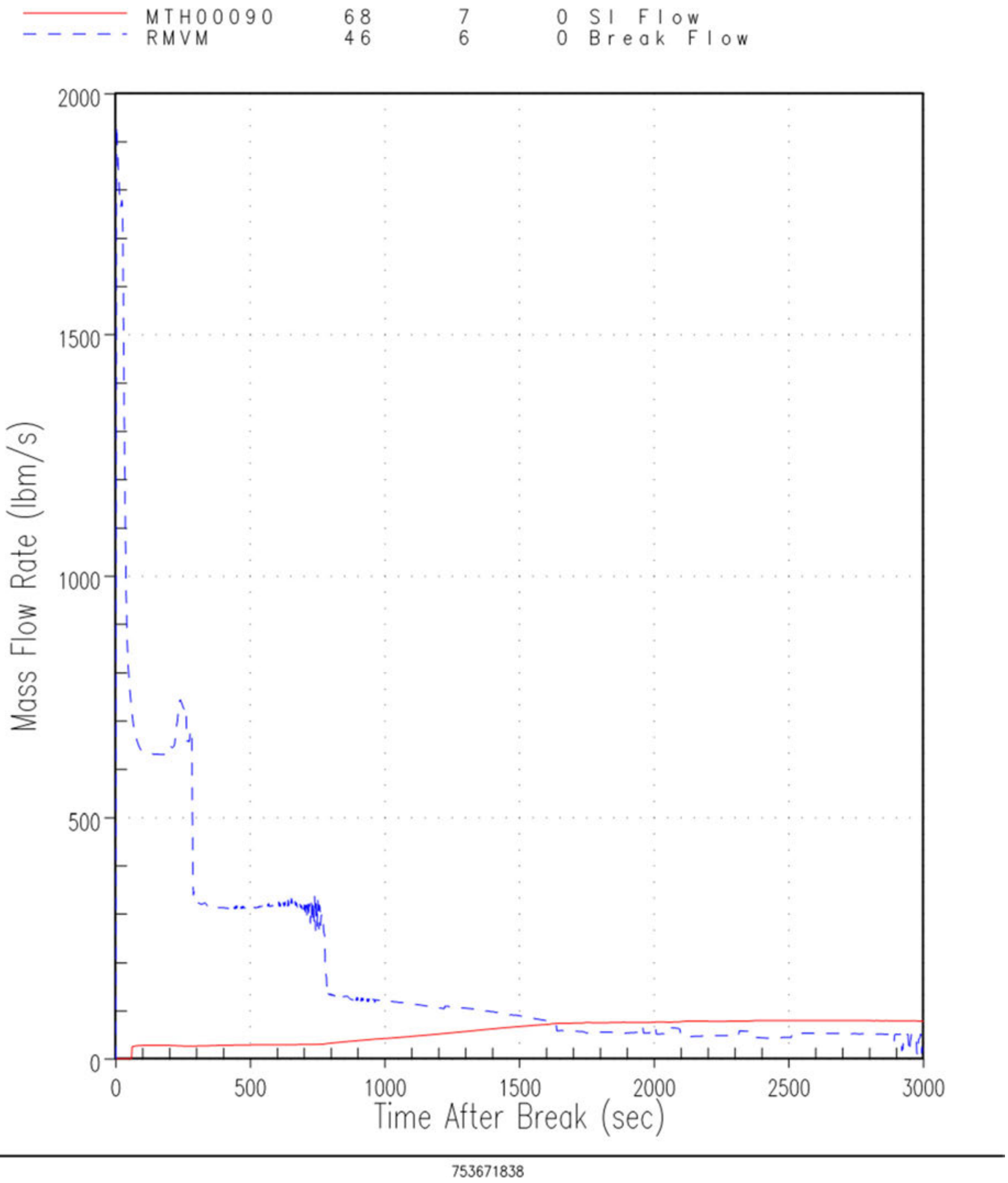


Figure 2: McGuire Units 1 and 2 and Catawba Unit 1 Total Safety Injection Flow (not including Accumulator Injection Flow) and Total Break Flow for the Region I Analysis PCT Case

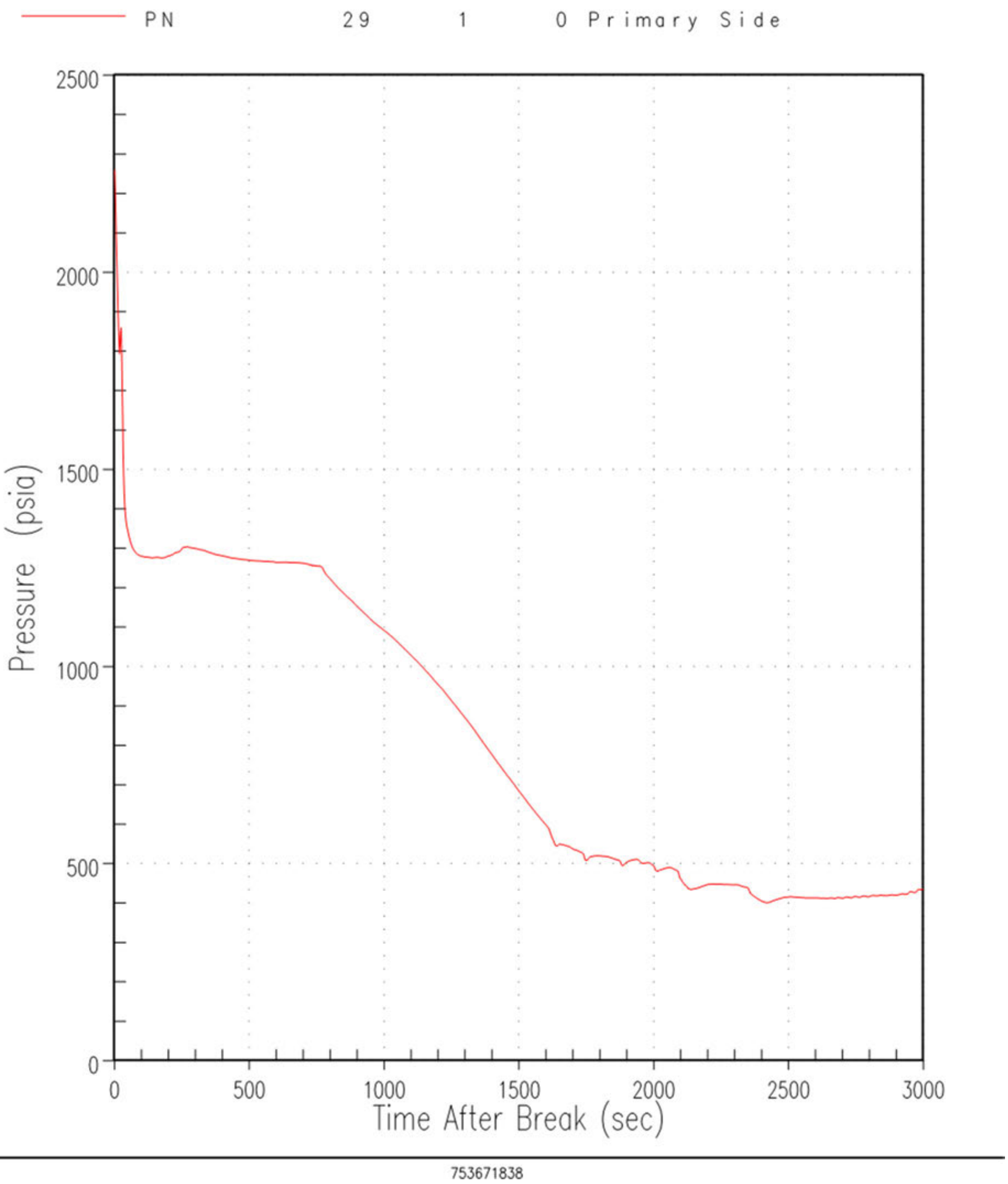


Figure 3: McGuire Units 1 and 2 and Catawba Unit 1 RCS Pressure for the Region I Analysis PCT Case

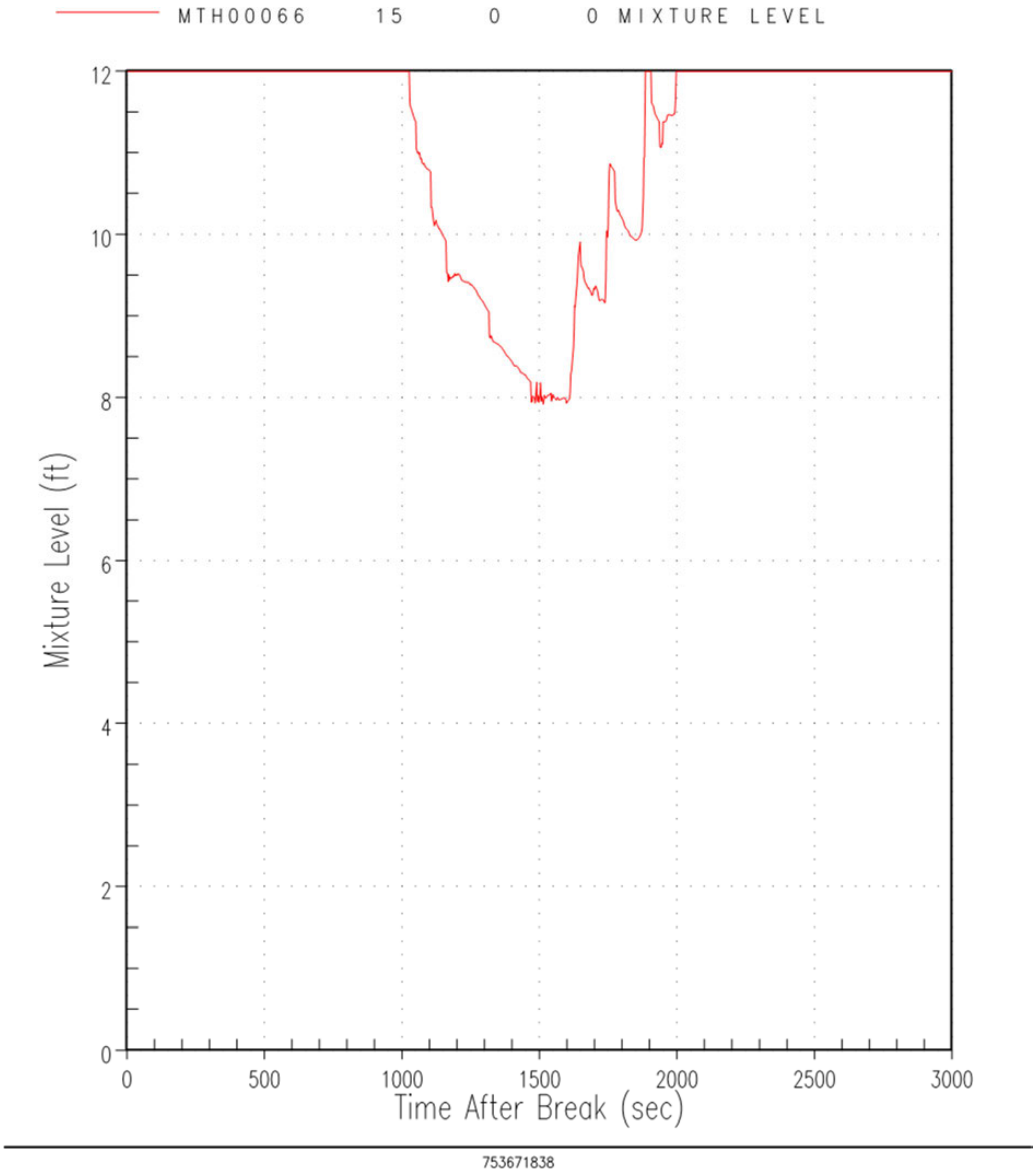


Figure 4: McGuire Units 1 and 2 and Catawba Unit 1 Hot Assembly Two-Phase Mixture Level (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case

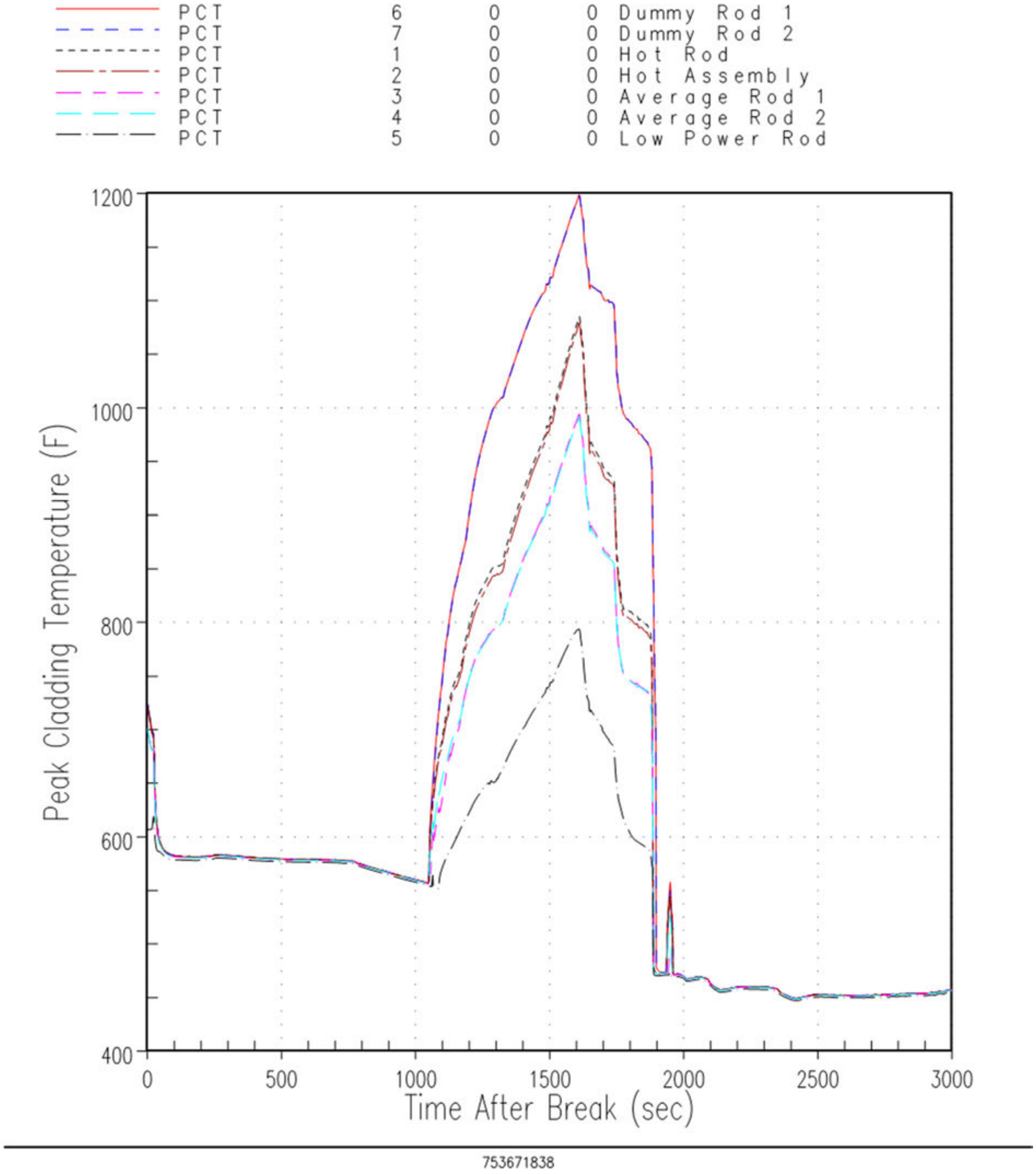
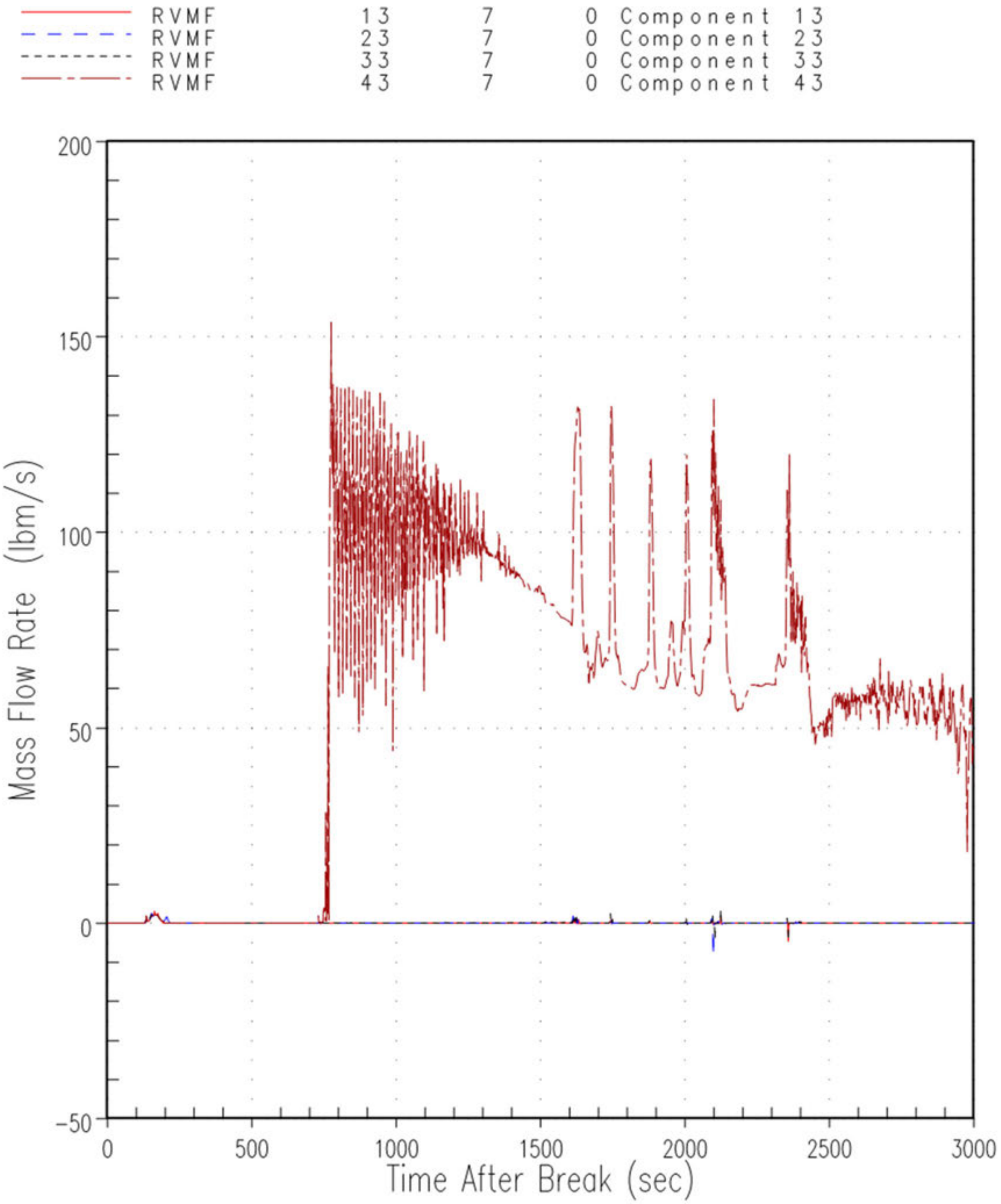
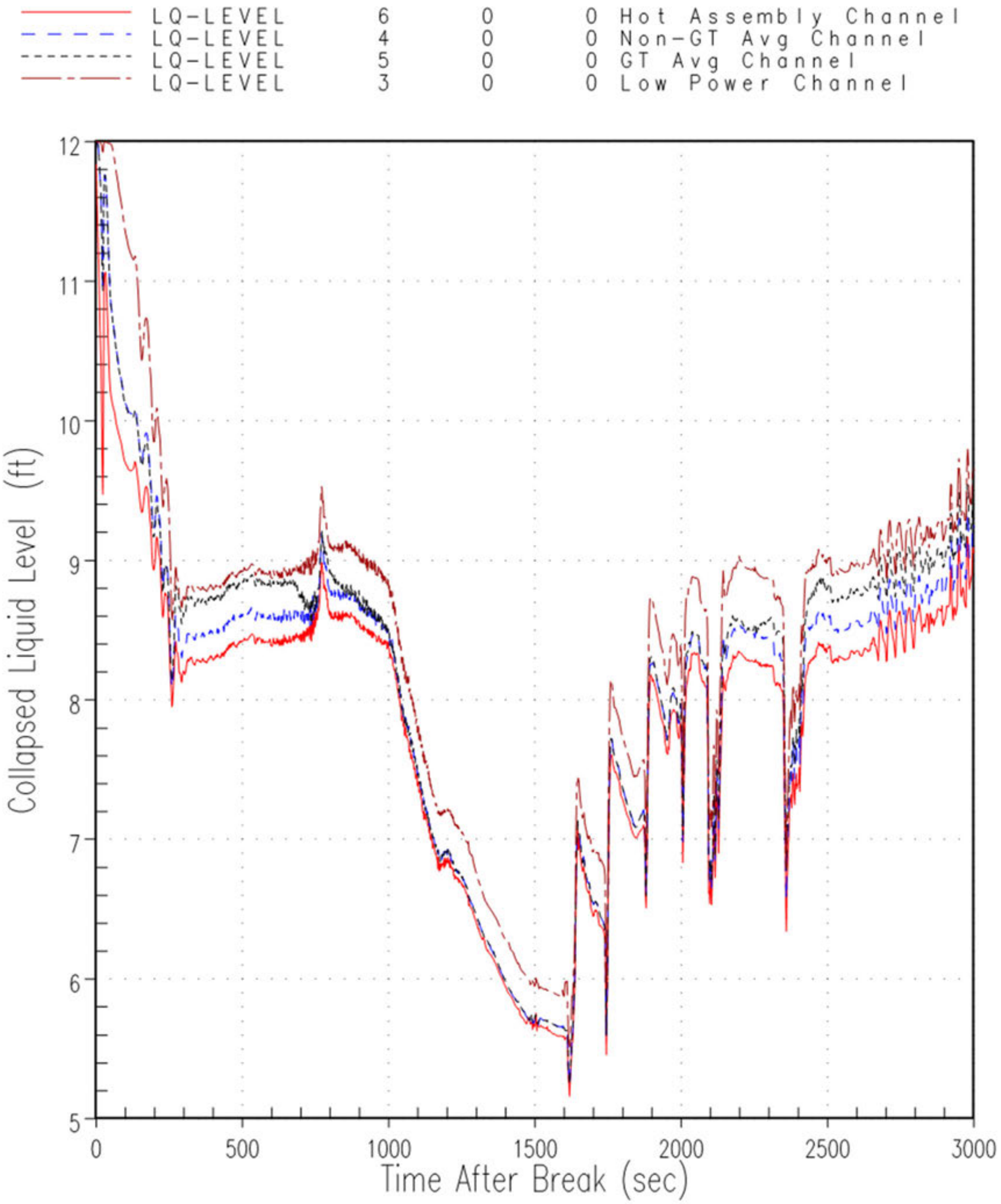


Figure 5: McGuire Units 1 and 2 and Catawba Unit 1 Peak Cladding Temperature for All Rods for the Region I Analysis PCT Case



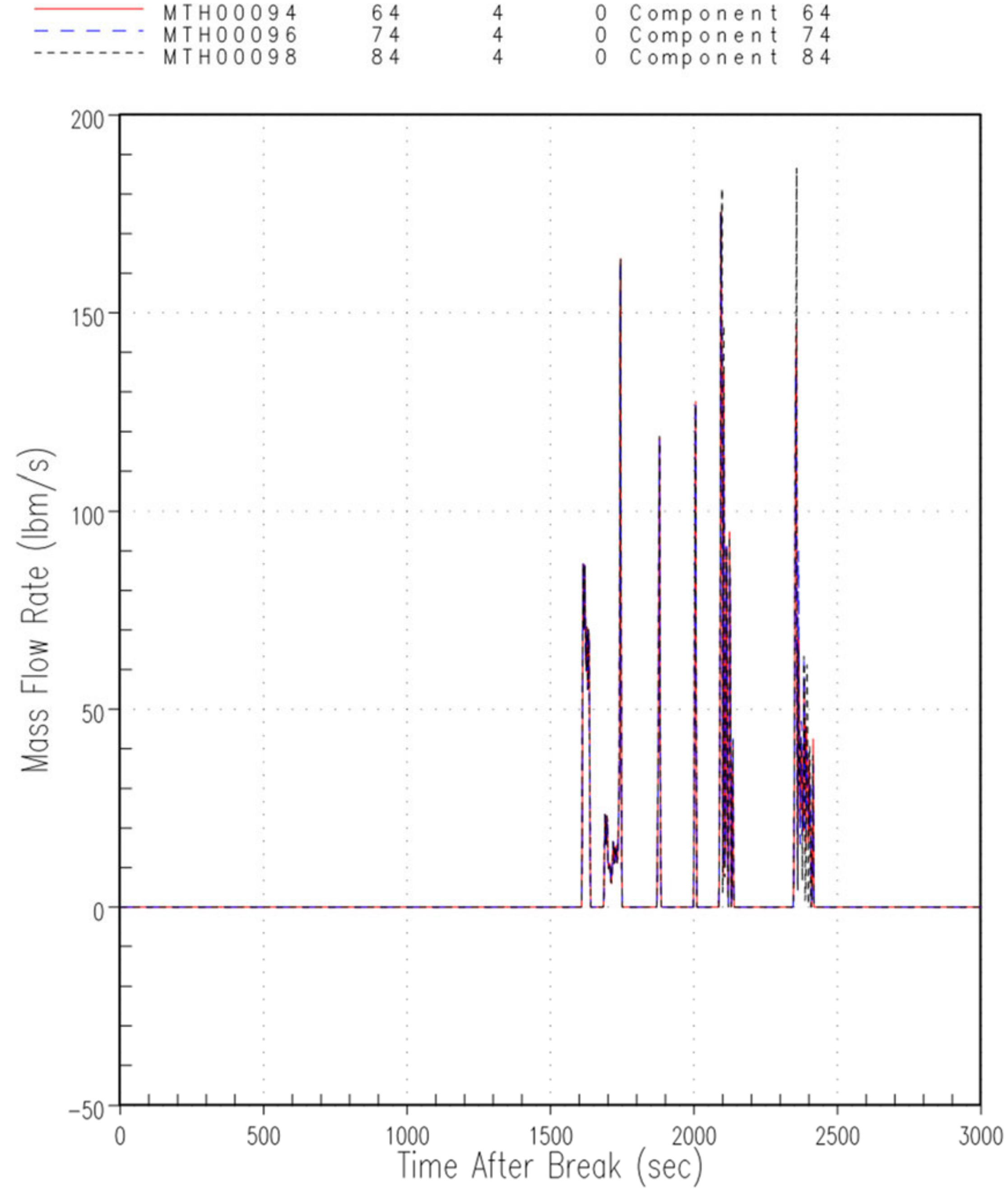
753671838

Figure 6: McGuire Units 1 and 2 and Catawba Unit 1 Vapor Mass Flow Rate through the Crossover Legs for the Region I Analysis PCT Case



753671838

Figure 7: McGuire Units 1 and 2 and Catawba Unit 1 Core Collapsed Liquid Levels (Relative to Bottom of Active Fuel) for the Region I Analysis PCT Case



753671838

Figure 8: McGuire Units 1 and 2 and Catawba Unit 1 Accumulator Injection Flow per Loop for the Region I Analysis PCT Case

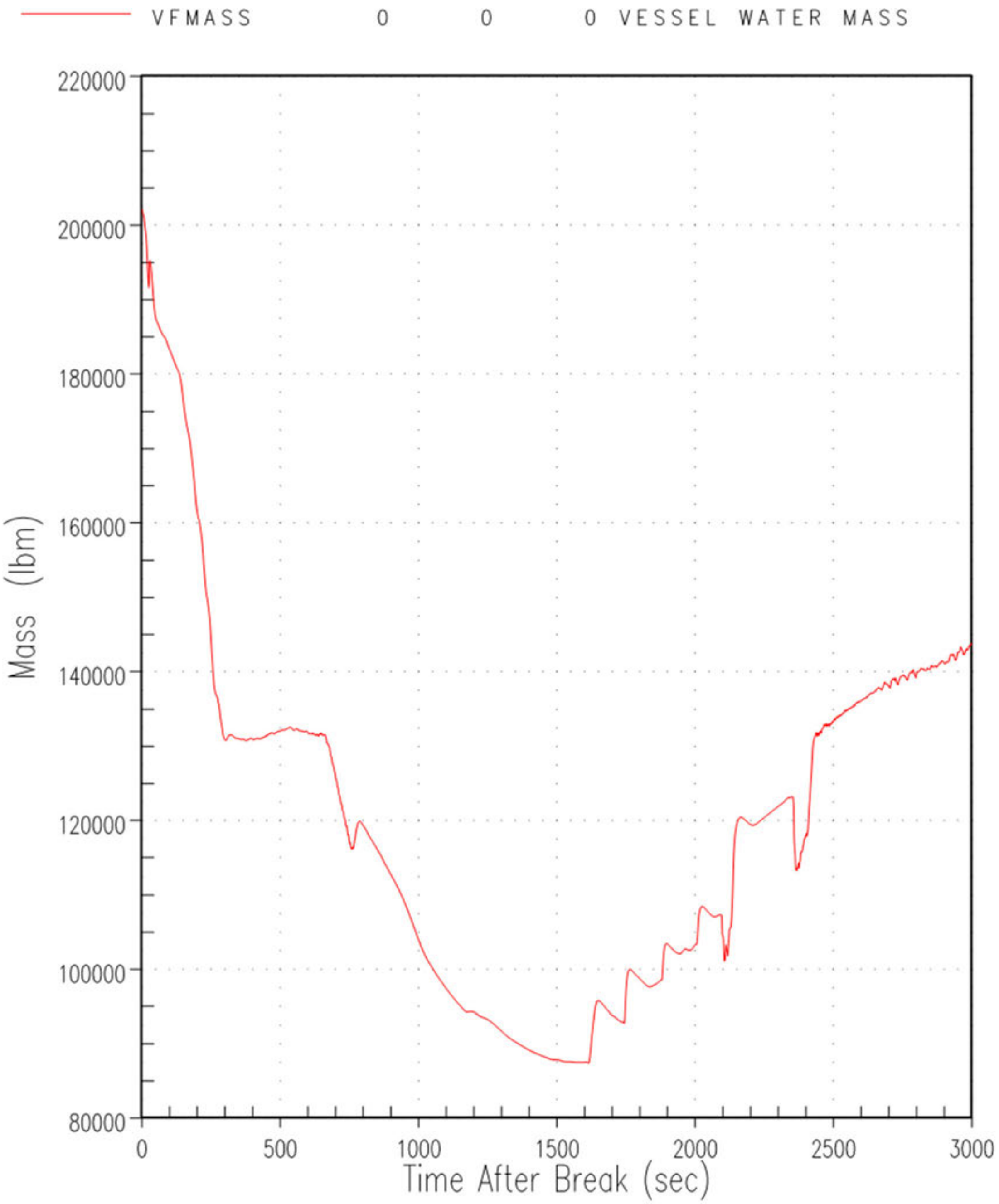


Figure 9: McGuire Units 1 and 2 and Catawba Unit 1 Vessel Fluid Mass for the Region I Analysis PCT Case

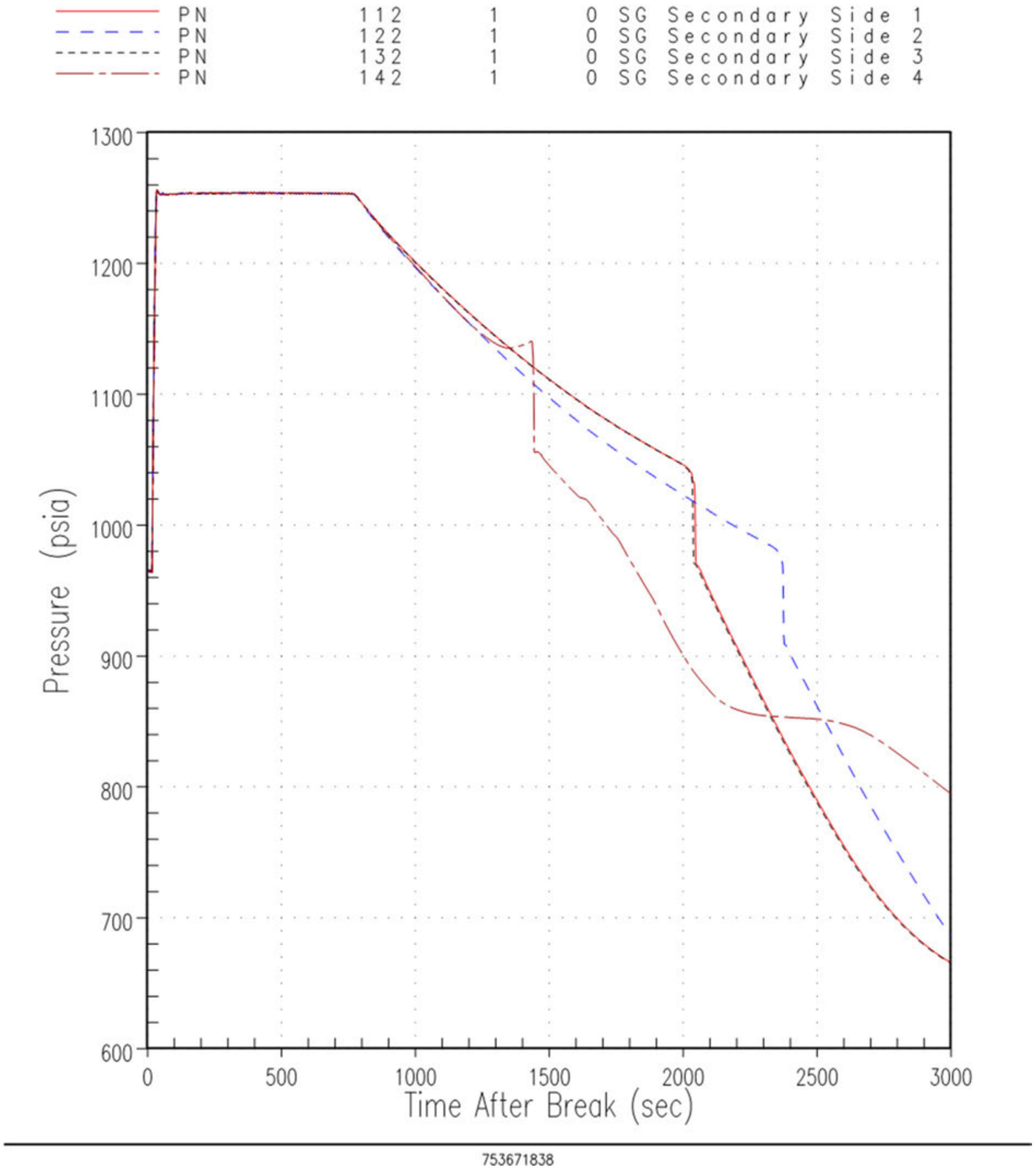


Figure 10: McGuire Units 1 and 2 and Catawba Unit 1 Steam Generator Secondary Side Pressure for the Region I Analysis PCT Case

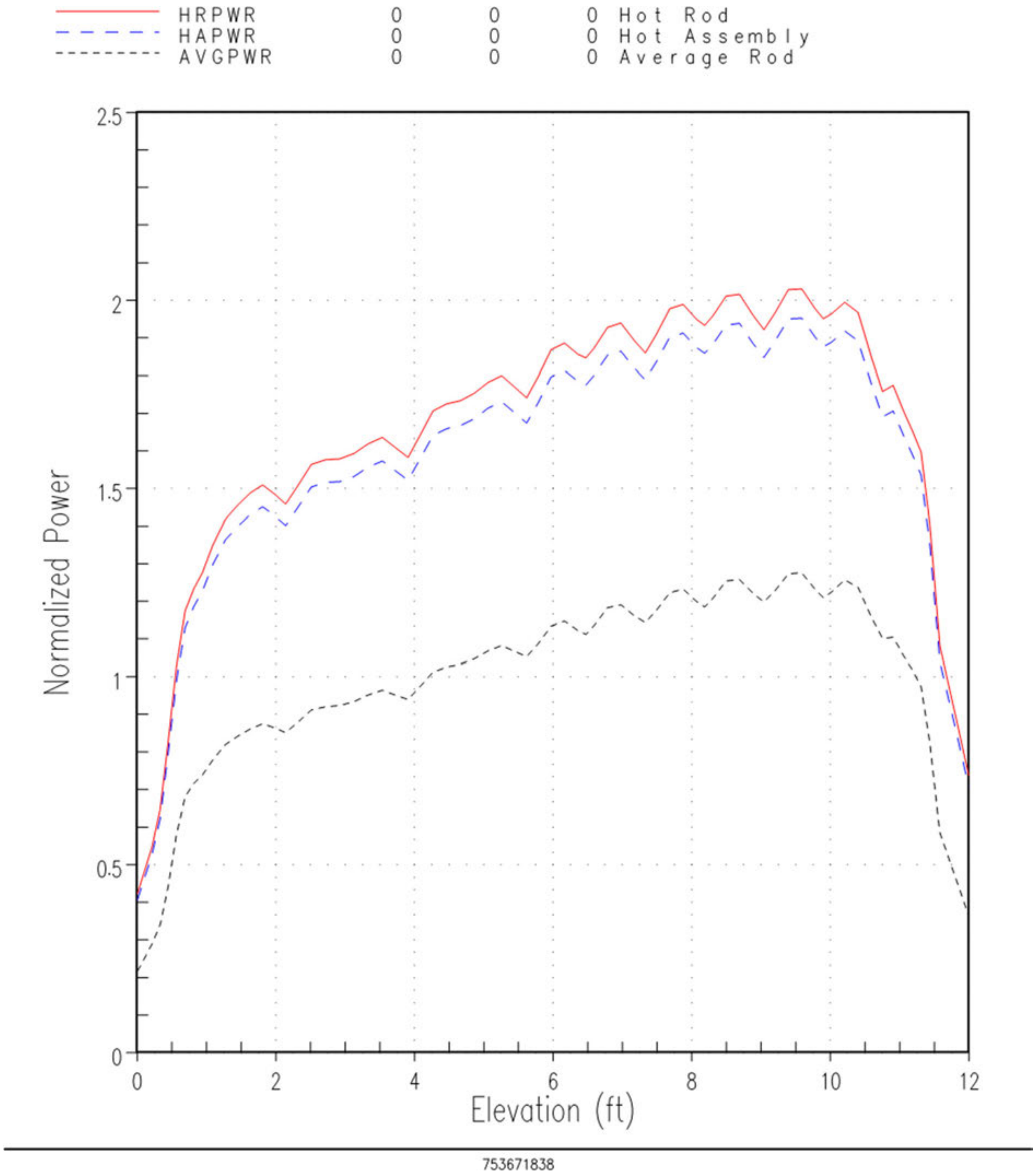


Figure 11: McGuire Units 1 and 2 and Catawba Unit 1 Normalized Core Power Shapes for the Region I Analysis PCT Case

Note: The localized power decreases occur at grid elevations.

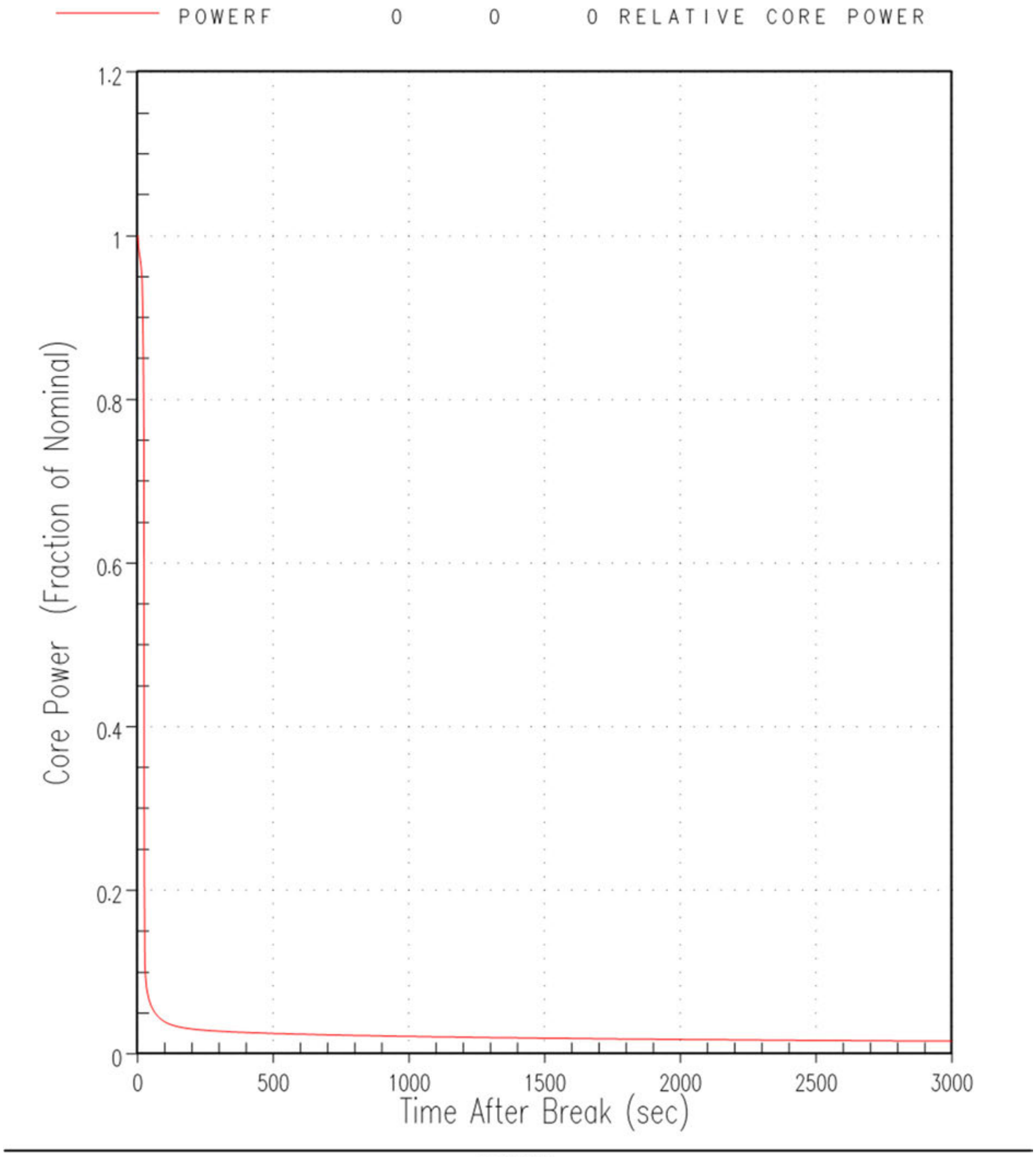
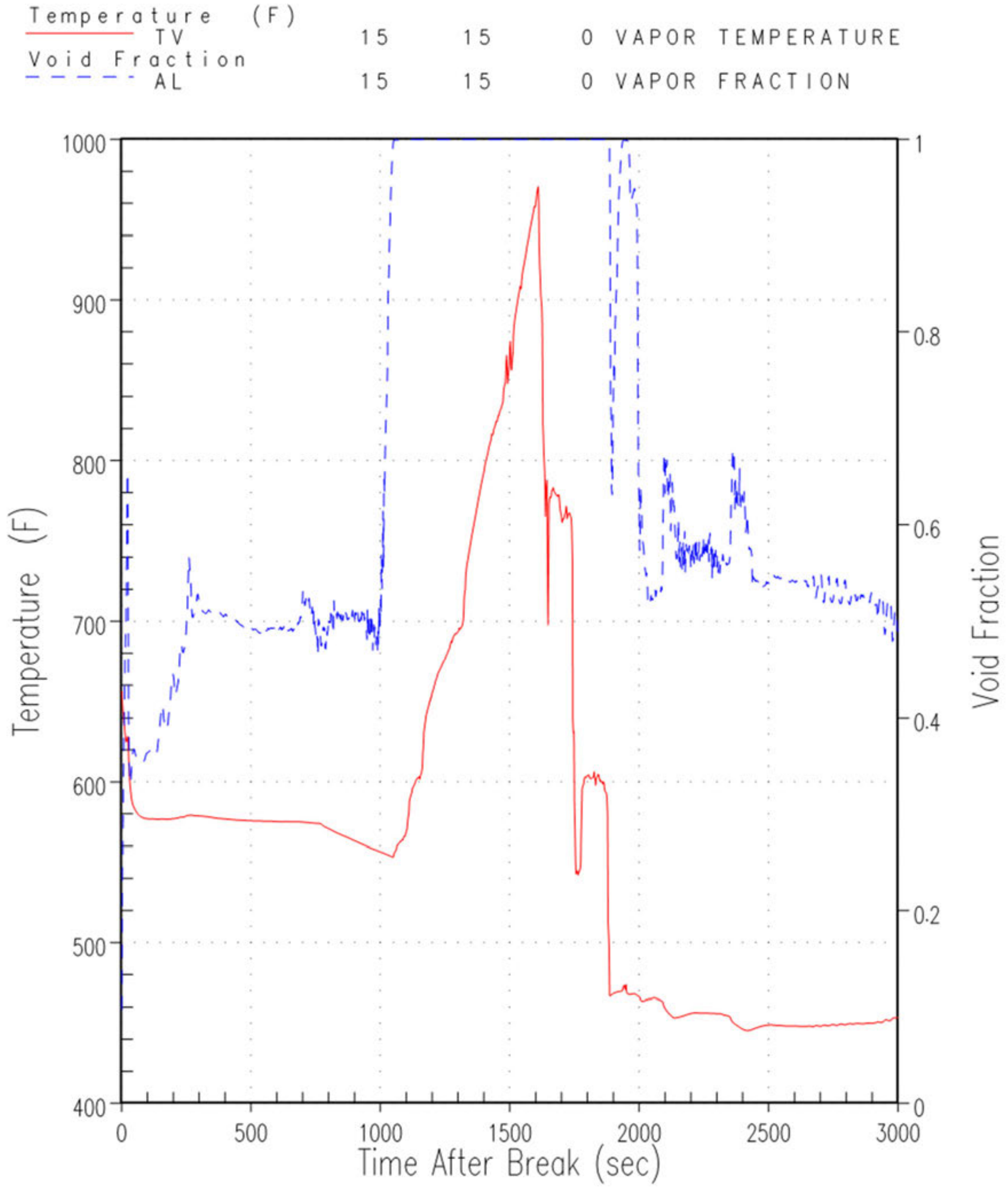


Figure 12: McGuire Units 1 and 2 and Catawba Unit 1 Relative Core Power for the Region I Analysis PCT Case



753671838

Figure 13: McGuire Units 1 and 2 and Catawba Unit 1 Vapor Temperature and Void Fraction at Core Outlet (Hot Assembly Channel) for the Region I Analysis PCT Case

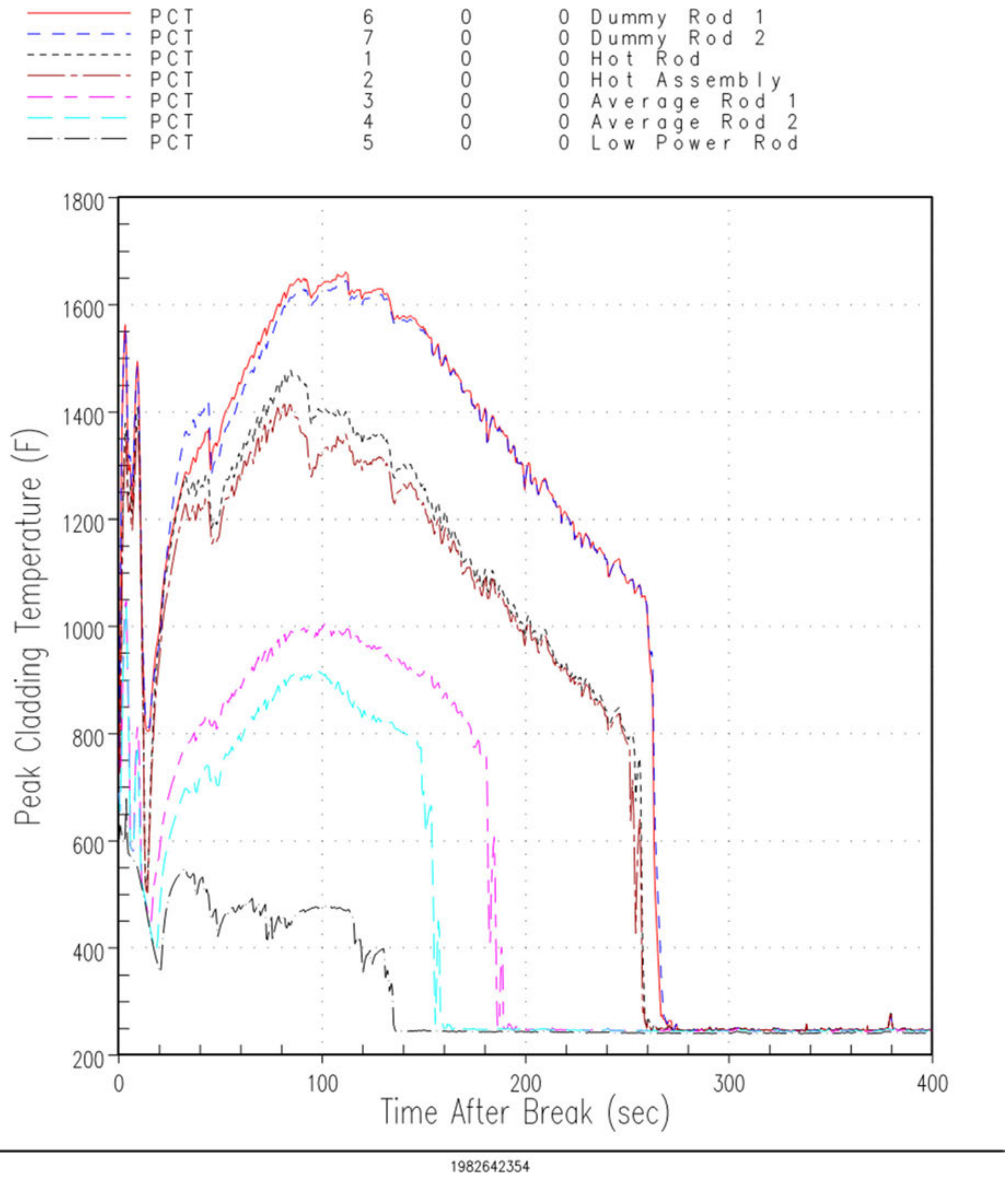


Figure 14: McGuire Units 1 and 2 and Catawba Unit 1 Peak Cladding Temperature for All Rods for the Region II Analysis PCT Case

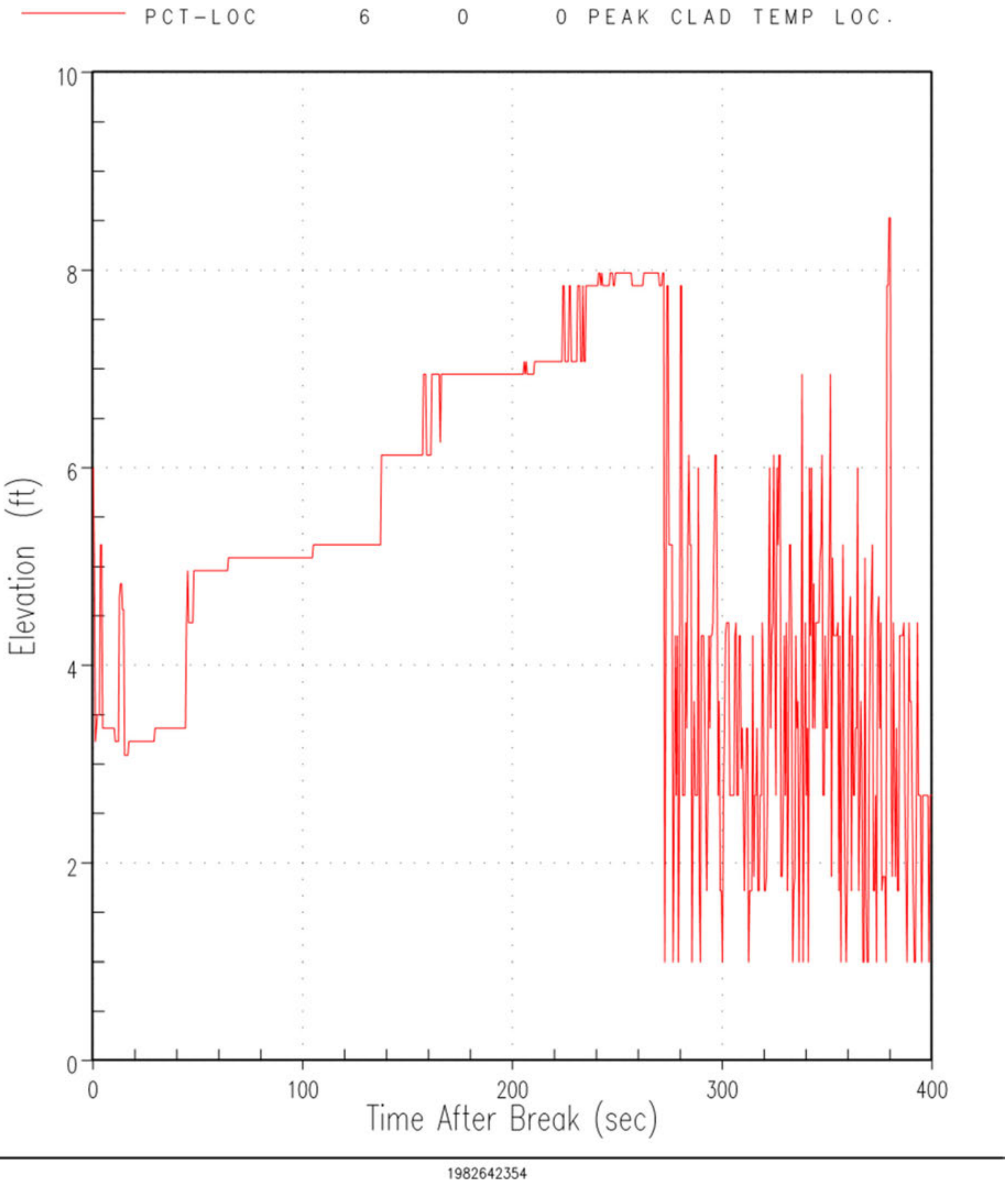


Figure 15: McGuire Units 1 and 2 and Catawba Unit 1 Peak Cladding Temperature Elevation (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

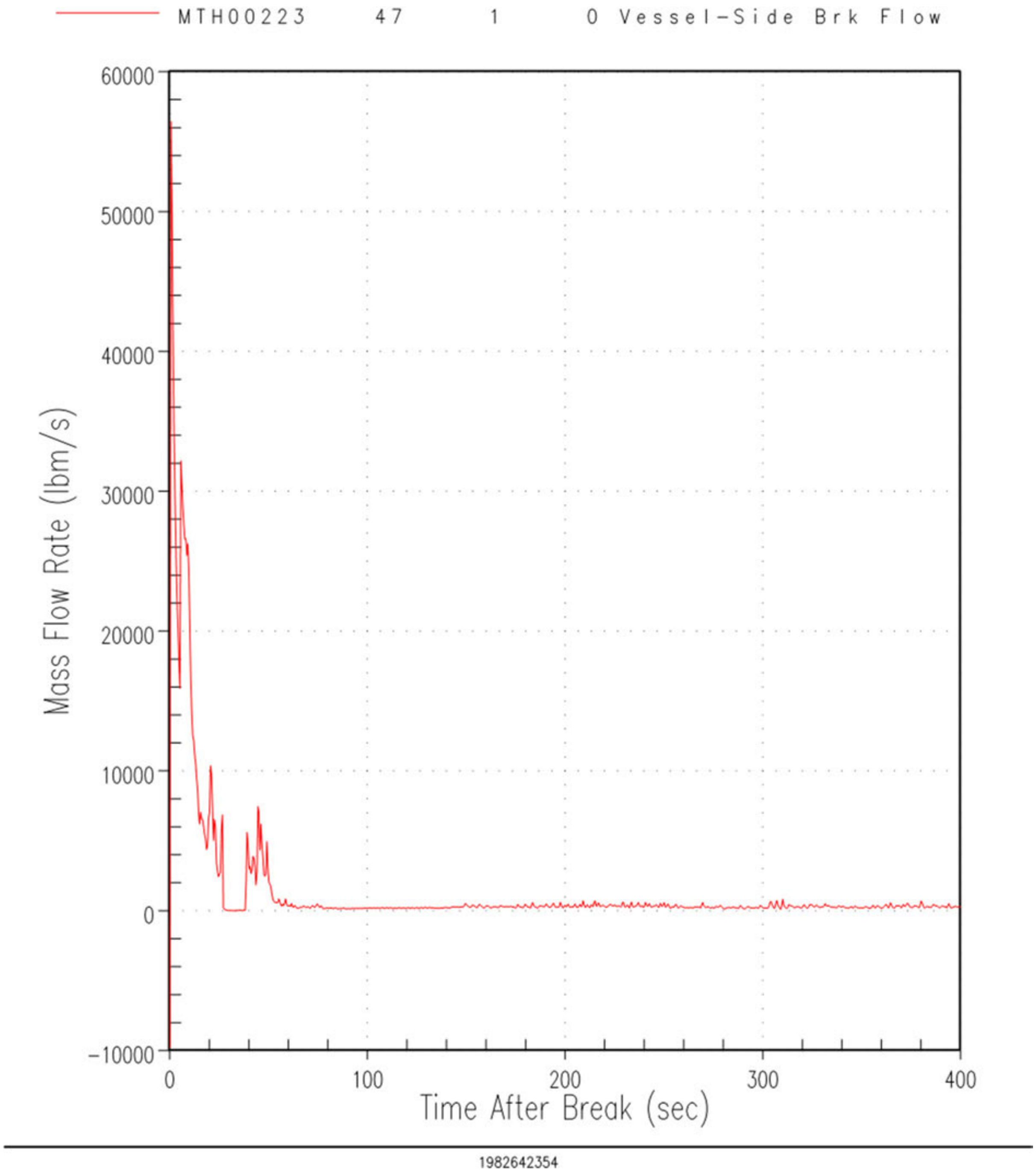


Figure 16a: McGuire Units 1 and 2 and Catawba Unit 1 Vessel-Side Break Mass Flow Rate for the Region II Analysis PCT Case

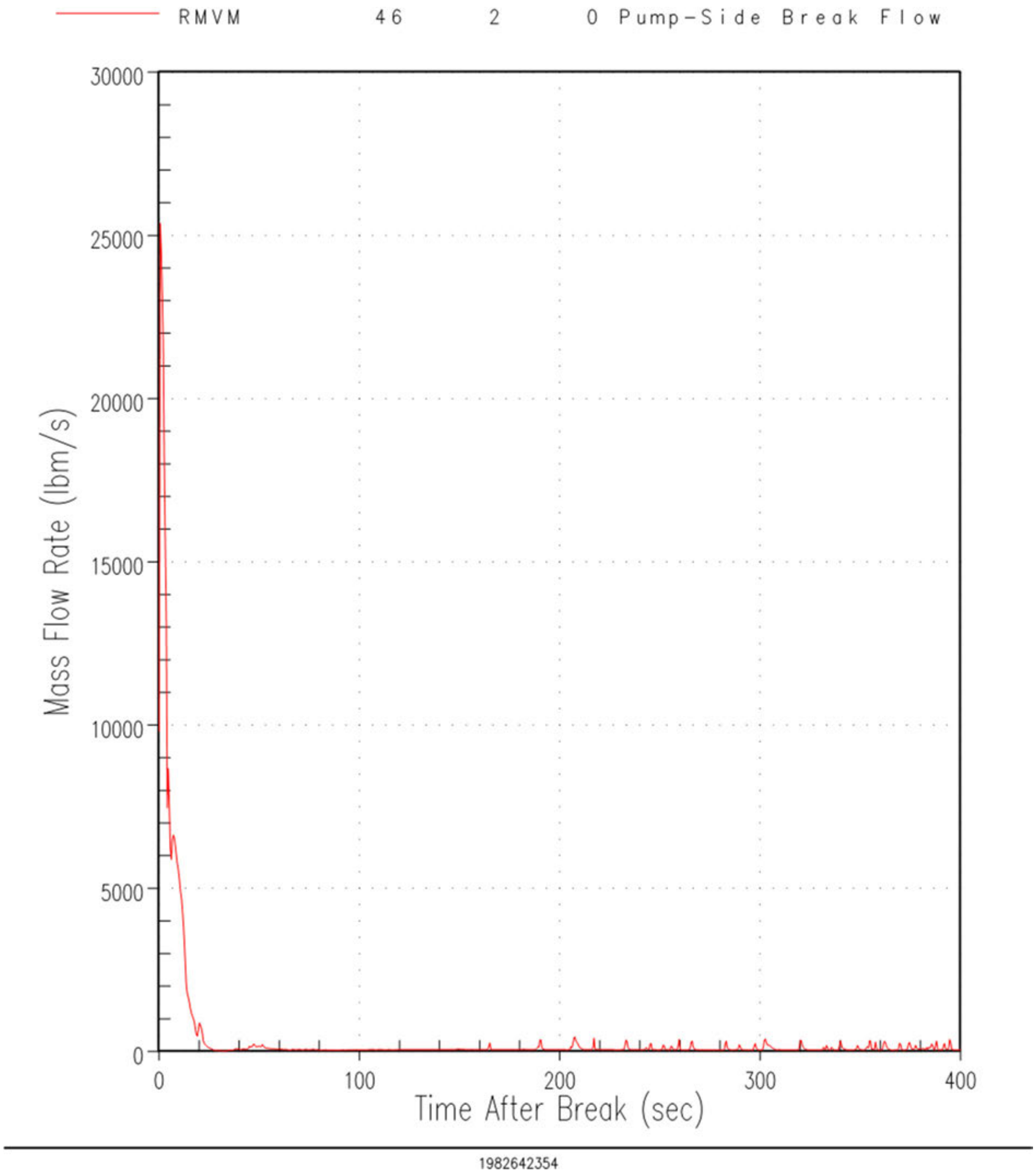


Figure 16b: McGuire Units 1 and 2 and Catawba Unit 1 Pump-Side Break Mass Flow Rate for the Region II Analysis PCT Case

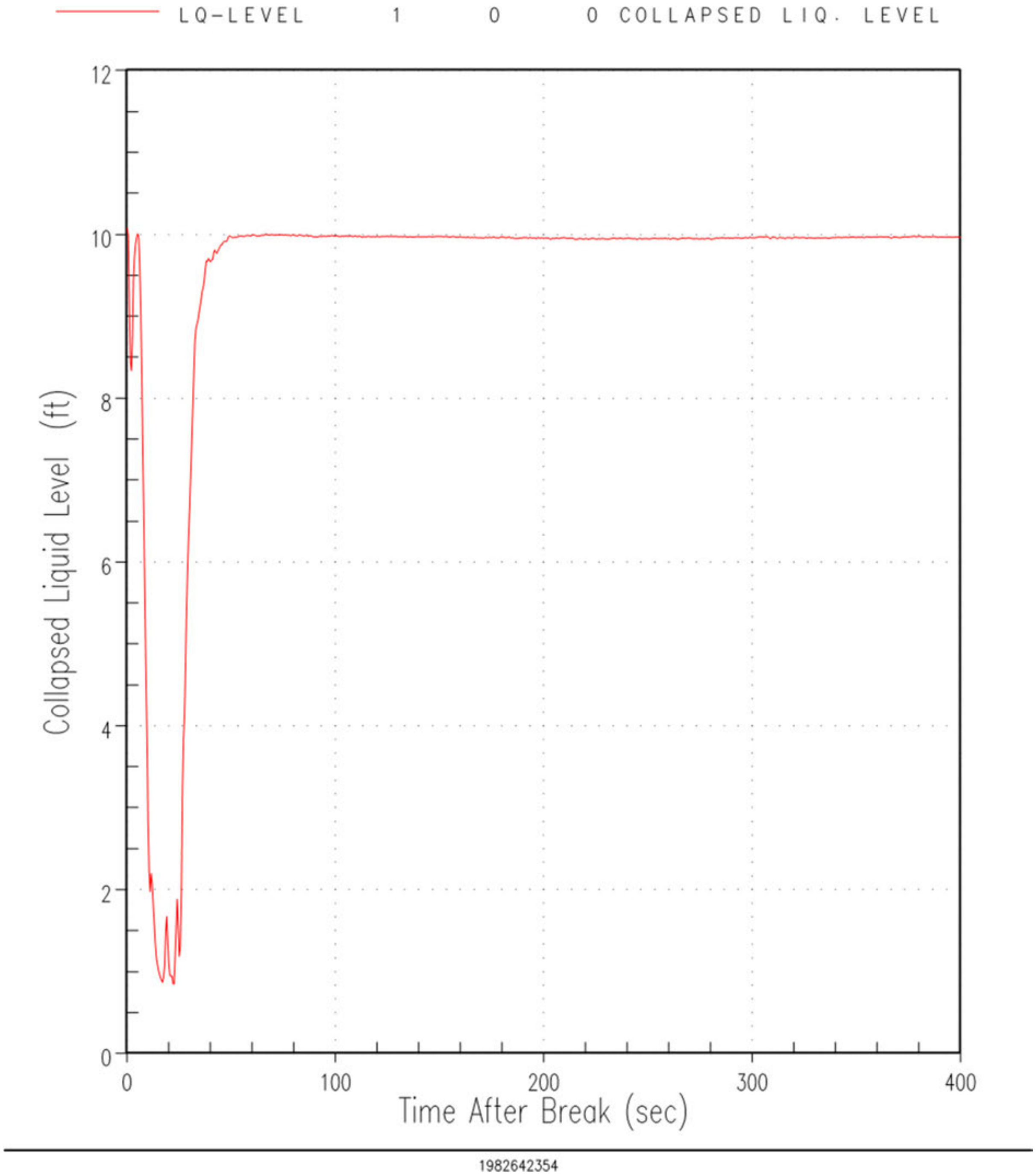


Figure 17: McGuire Units 1 and 2 and Catawba Unit 1 Lower Plenum Collapsed Liquid Level (Relative to Inside Bottom of Vessel) for the Region II Analysis PCT Case

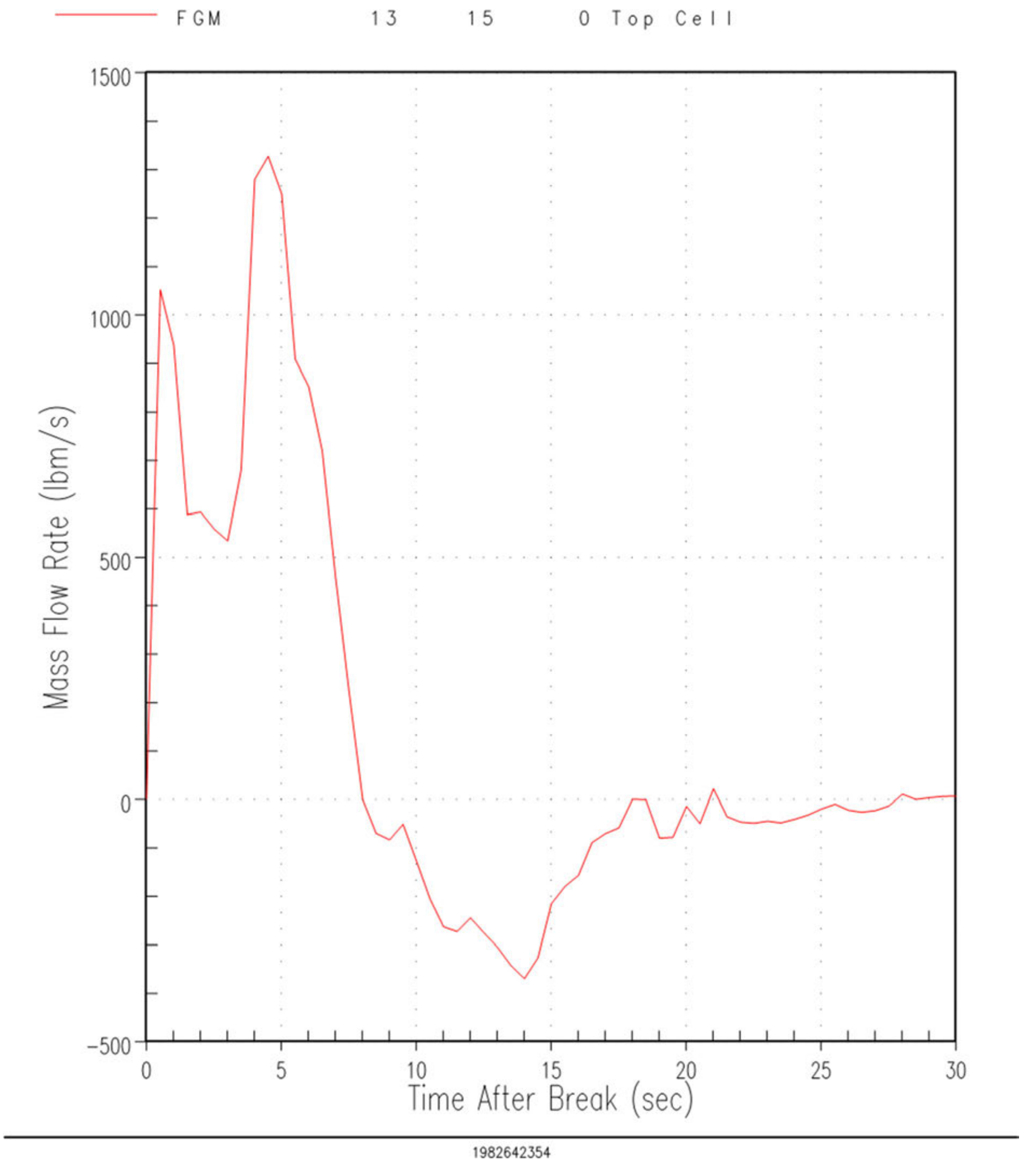


Figure 18: McGuire Units 1 and 2 and Catawba Unit 1 Vapor Mass Flow Rate at the Top Cell Face of the Core Average Channel not Under Guide Tubes for the Region II Analysis PCT Case

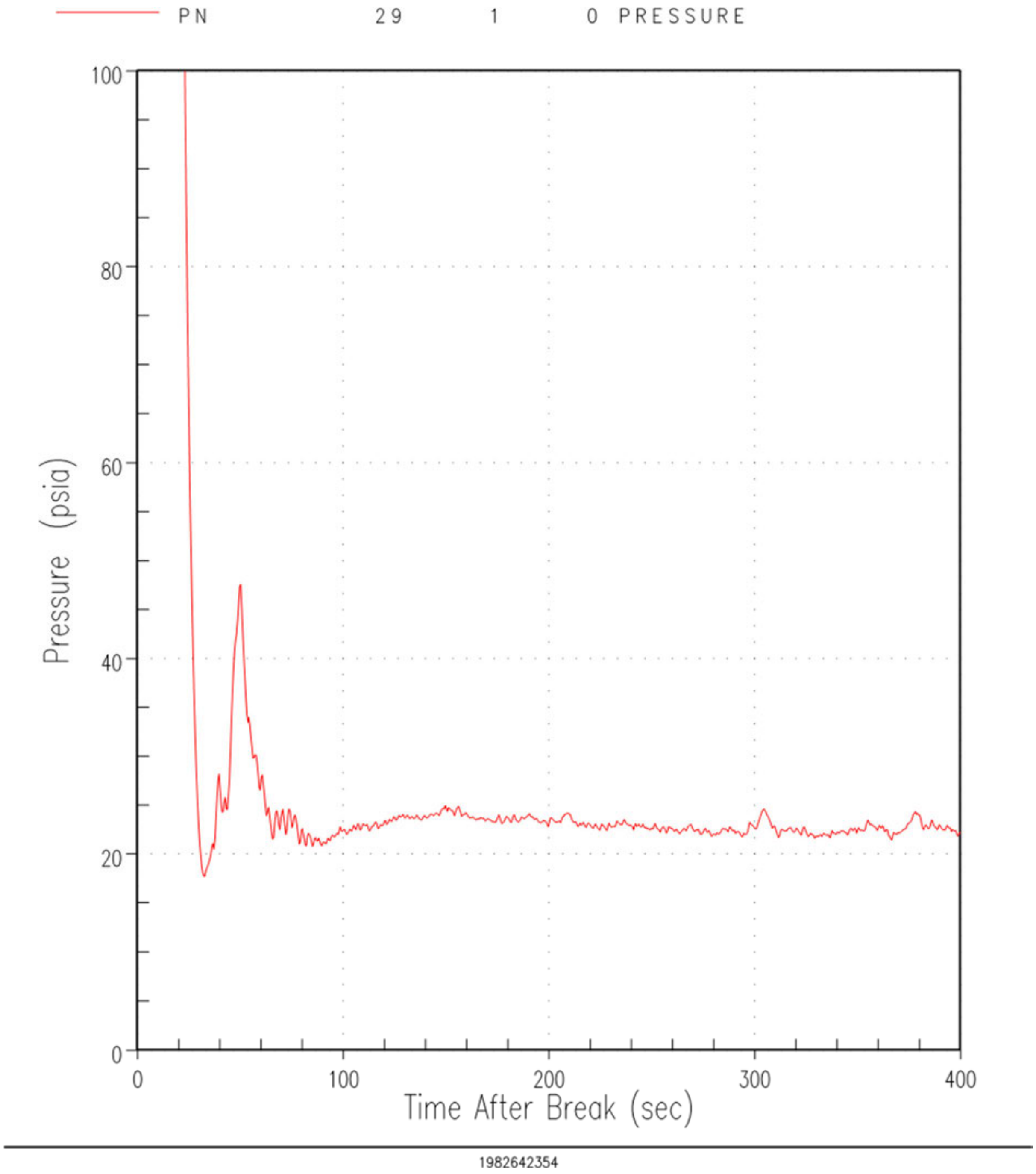


Figure 19: McGuire Units 1 and 2 and Catawba Unit 1 RCS Pressure for the Region II Analysis PCT Case

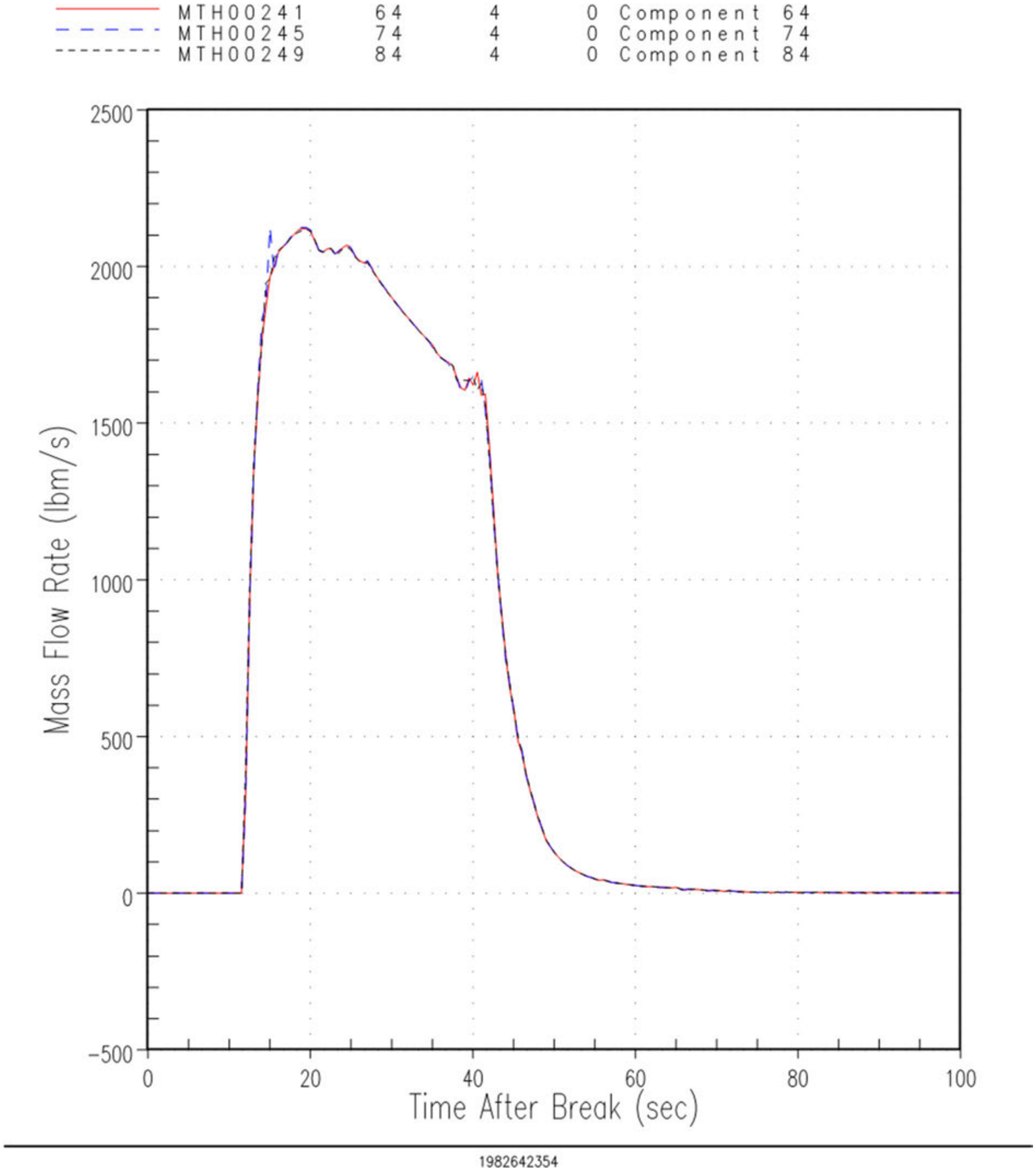


Figure 20: McGuire Units 1 and 2 and Catawba Unit 1 Accumulator Injection Flow per Loop for the Region II Analysis PCT Case

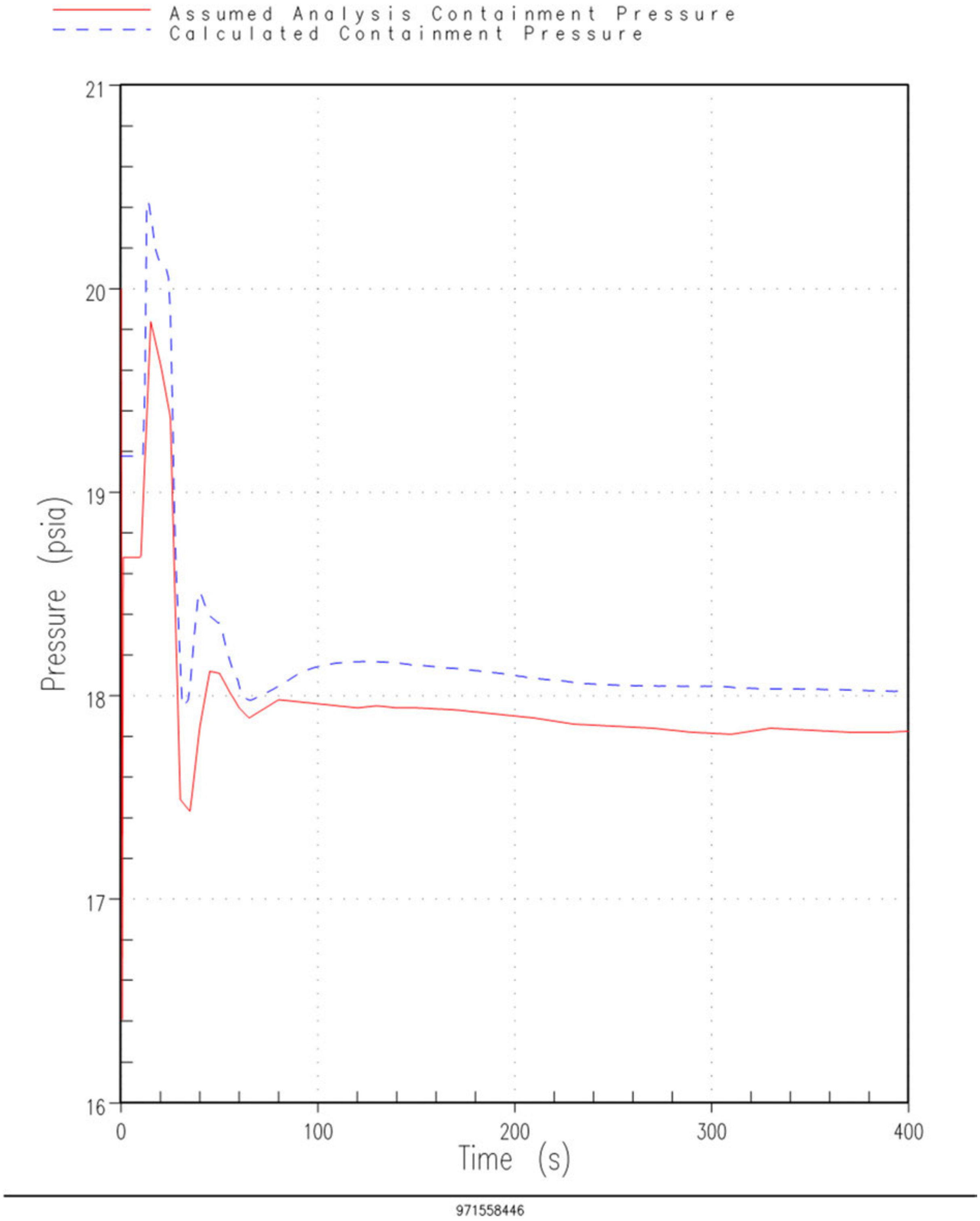


Figure 21: McGuire Units 1 and 2 and Catawba Unit 1 Containment Pressure Comparison for the Region II Analysis

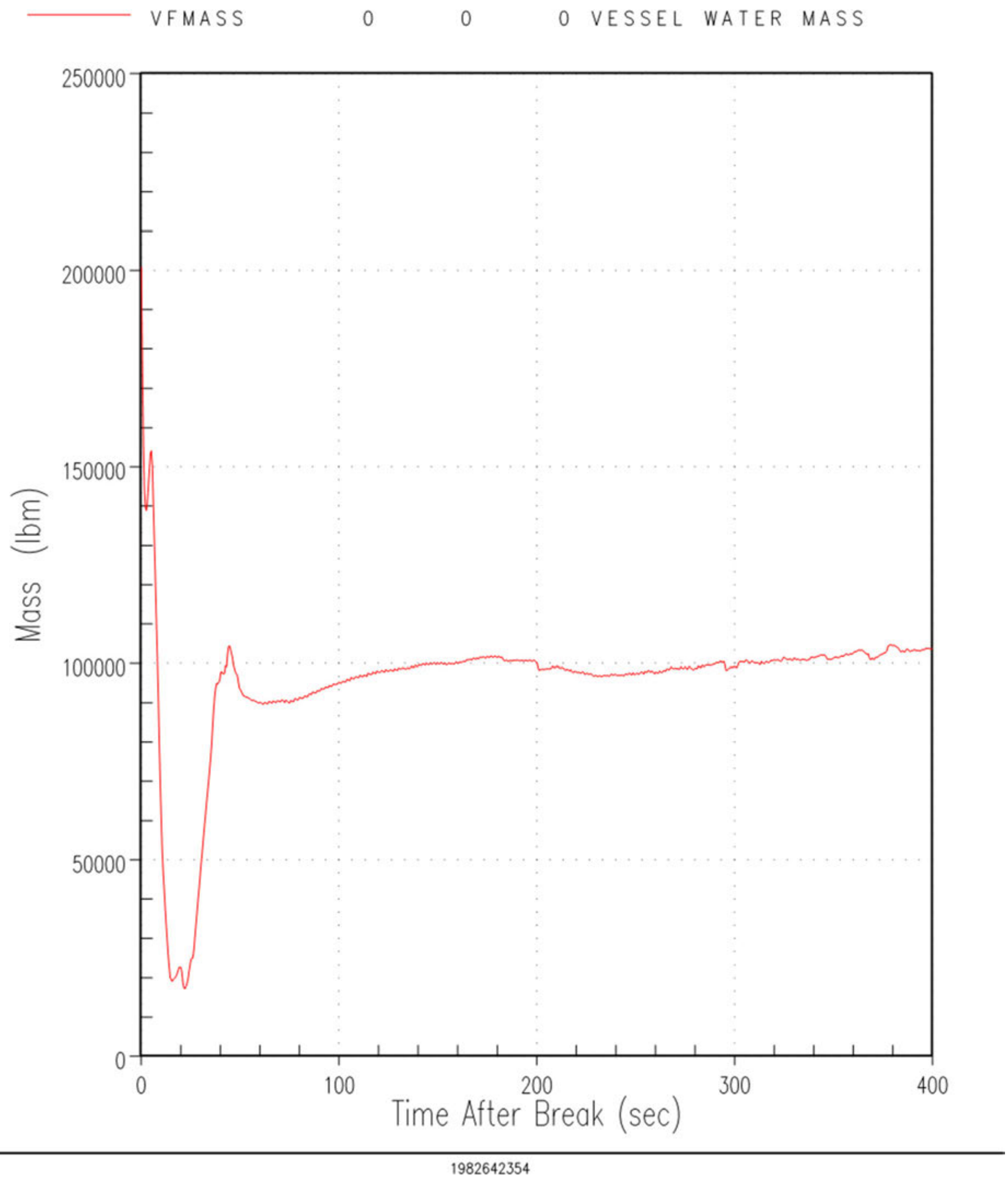


Figure 22: McGuire Units 1 and 2 and Catawba Unit 1 Vessel Fluid Mass for the Region II Analysis PCT Case

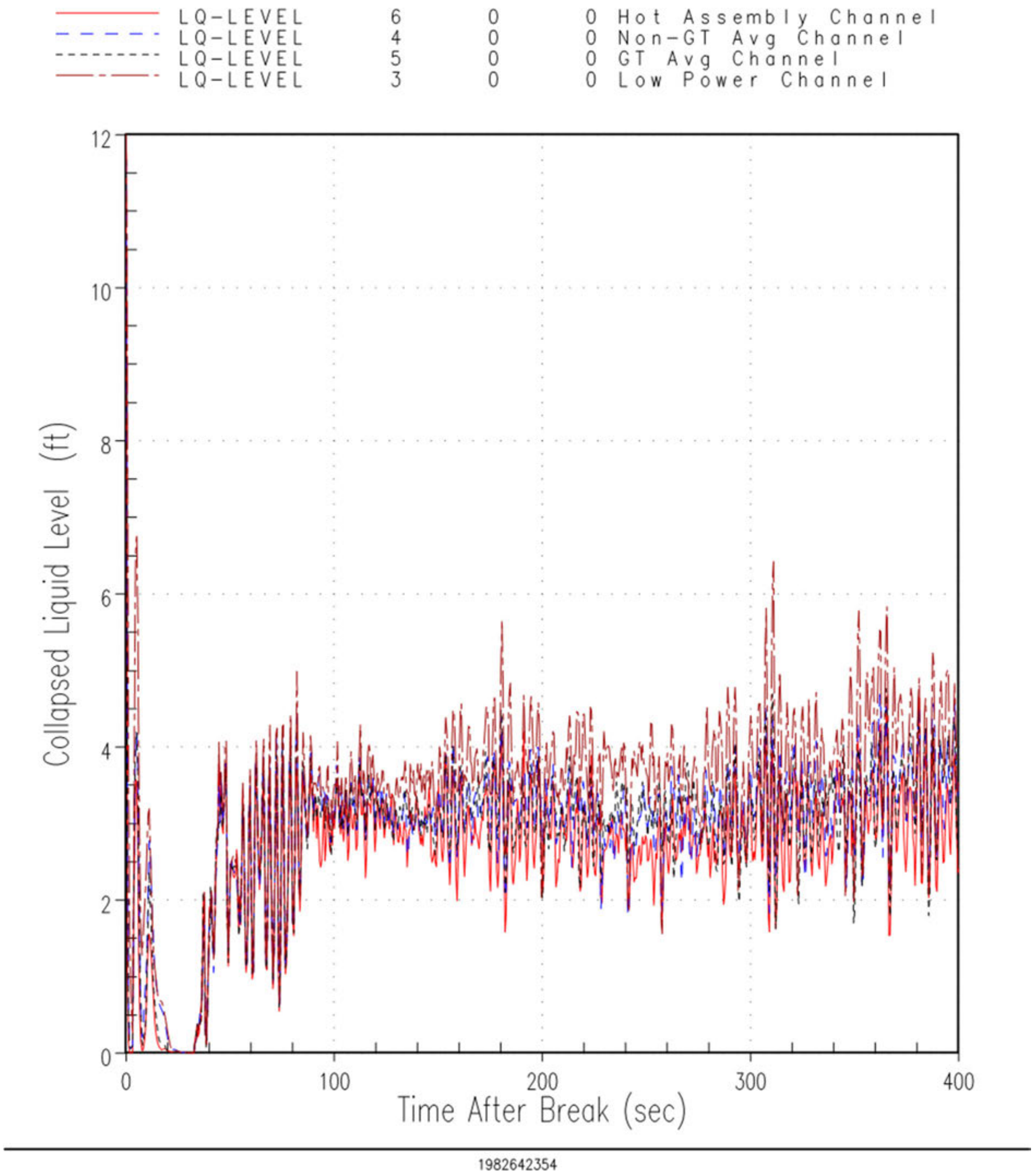


Figure 23: McGuire Units 1 and 2 and Catawba Unit 1 Collapsed Liquid Level for Each Core Channel (Relative to Bottom of Active Fuel) for the Region II Analysis PCT Case

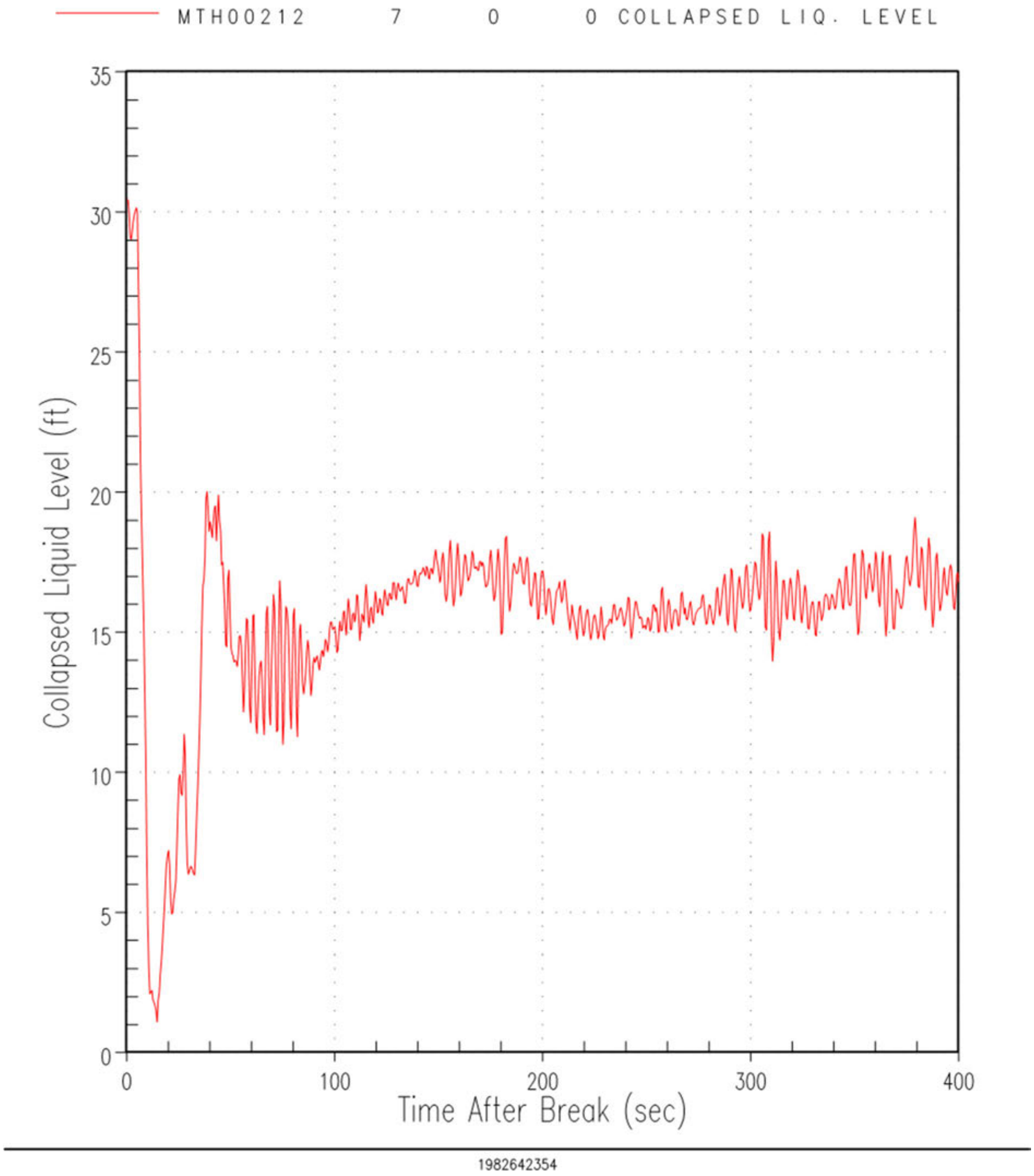


Figure 24: McGuire Units 1 and 2 and Catawba Unit 1 Average Downcomer Collapsed Liquid Level (Relative to Bottom of Upper Tie Plate) for the Region II Analysis PCT Case

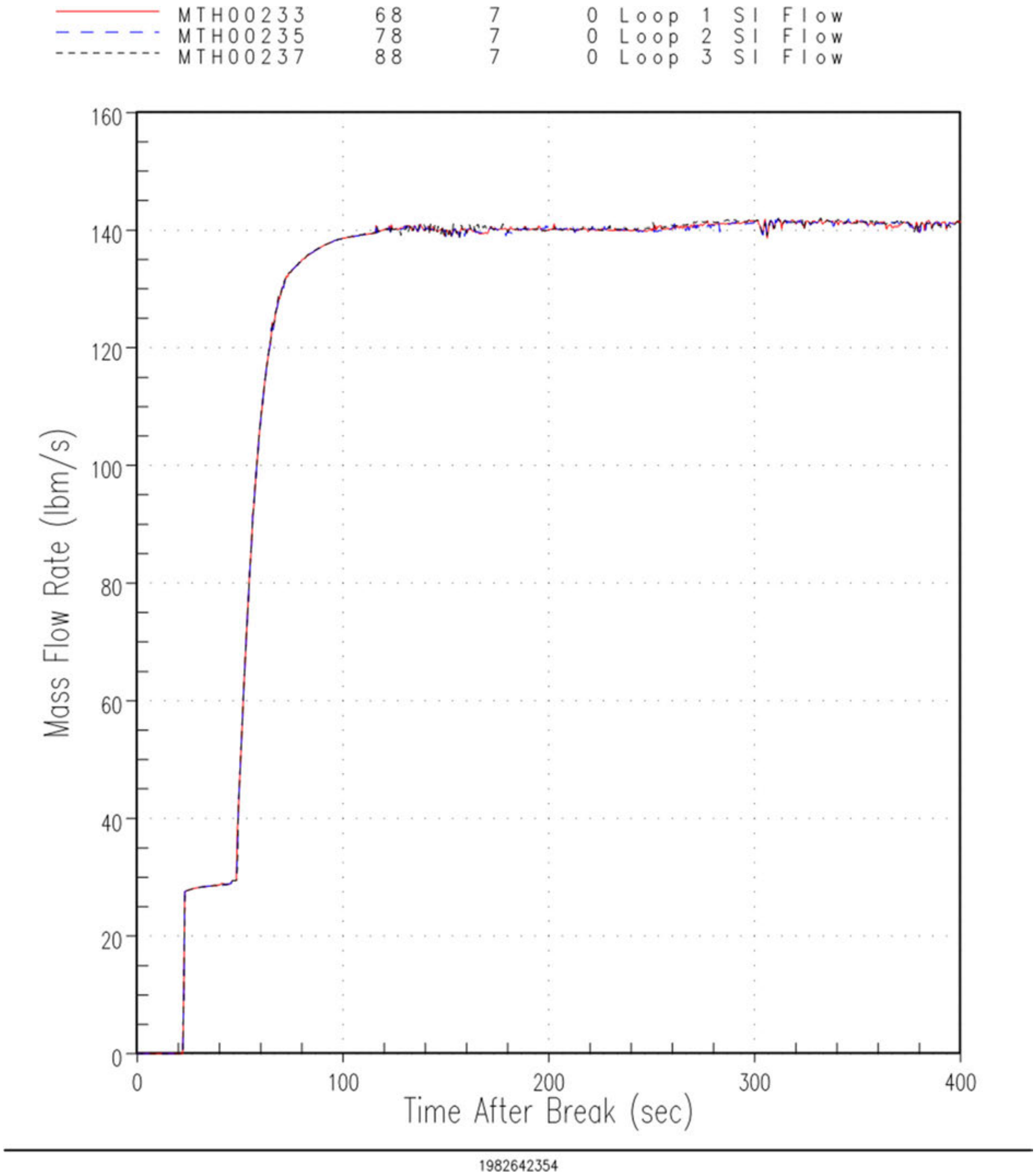


Figure 25: McGuire Units 1 and 2 and Catawba Unit 1 Safety Injection Flow per Loop (not including Accumulator Injection Flow) for the Region II Analysis PCT Case

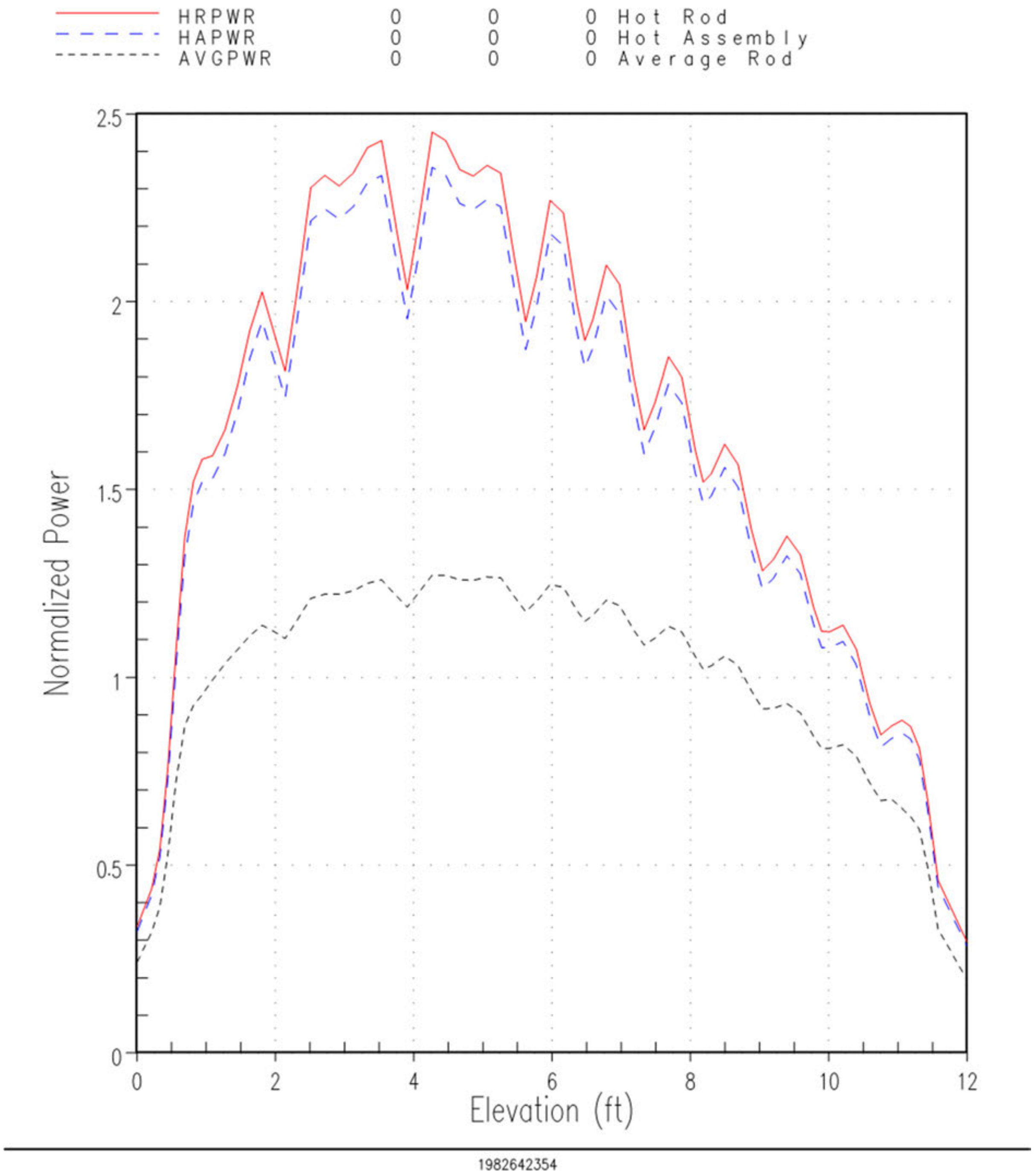


Figure 26: McGuire Units 1 and 2 and Catawba Unit 1 Normalized Core Power Shapes for the Region II Analysis PCT Case

Note: The localized power decreases occur at grid elevations.

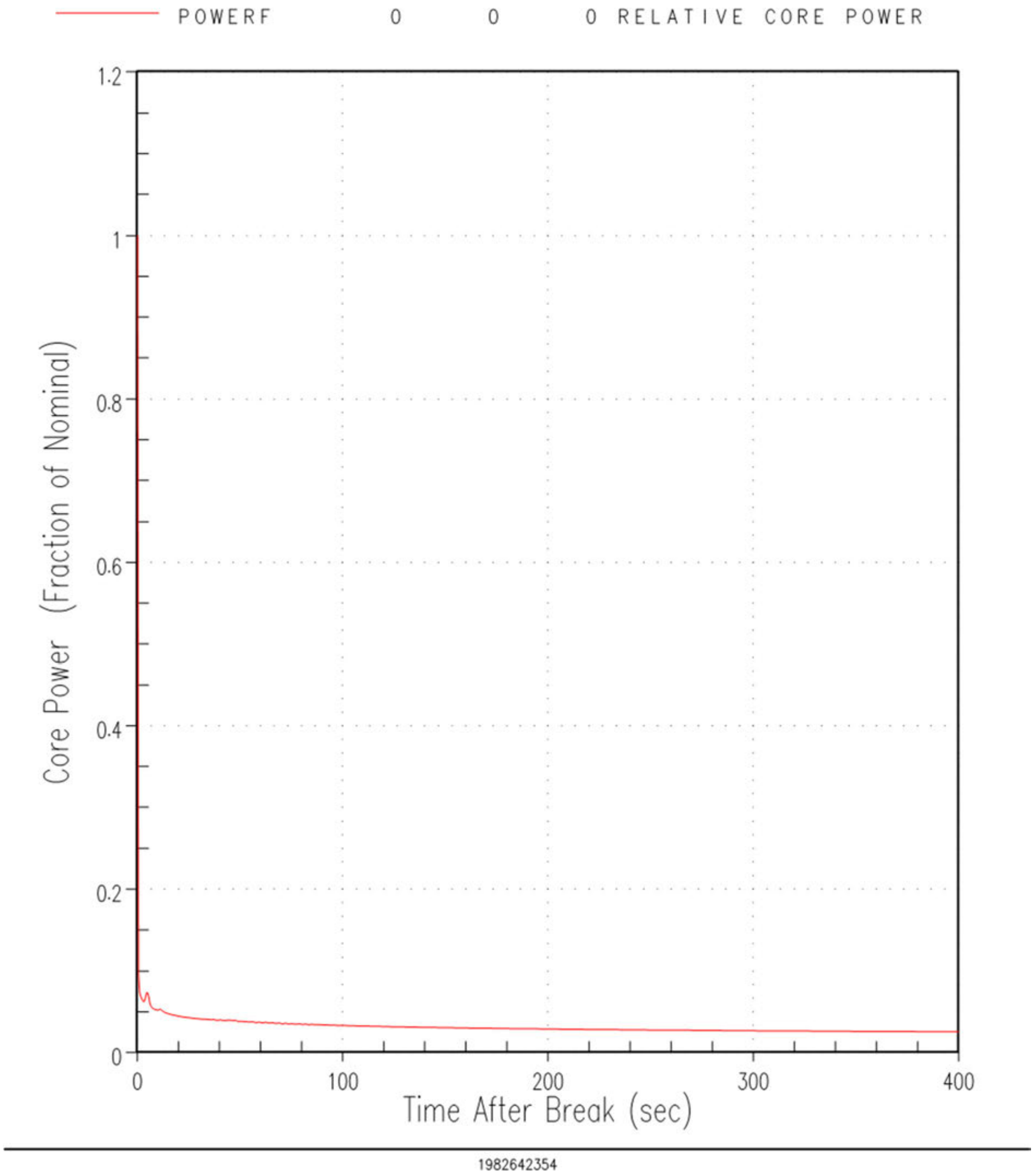


Figure 27: McGuire Units 1 and 2 and Catawba Unit 1 Relative Core Power for the Region II Analysis PCT Case

a,c

Figure 28: [**]**^{a,c}

[ ]a,c

Figure 29: [ ]a,c

a,c

Figure 30: [ ]^{a,c}

a,c

Figure 31: [

]^{a,c}

a,c

Figure 32: [ ]^{a,c}

a,c

Figure 33: [

]^{a,c}

ENCLOSURE 5

10 CFR 50.46 EXEMPTION REQUEST FOR USE OF AXIOM FUEL CLADDING

3 PAGES PLUS THE COVER

10 CFR 50.46 Exemption Request for Use of AXIOM Cladding

1.0 PURPOSE

Pursuant to 10 CFR 50.12, “Specific exemptions,” Duke Energy Carolinas, LLC (Duke Energy), requests an exemption from the requirements of 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” for Catawba Nuclear Station, Unit 1 (CNS U1) and McGuire Nuclear Station, Units 1 and 2 (MNS). This exemption request is related to the proposed use of the AXIOM® fuel rod cladding material by CNS U1 and MNS since it does not conform to the specifications for either zircaloy or ZIRLO®, both of which are explicitly identified as required fuel rod cladding material in 10 CFR 50.46, Section (a)(1)(i).

Consequently, an exemption is required from specific portions of 10 CFR 50.46 in order to support the application of AXIOM fuel rod cladding to the fuel for CNS U1 and MNS. The CNS U1 and MNS reloads scheduled to contain fuel rods with AXIOM cladding are proposed beginning Spring 2026 with CNS U1. This exemption request relates solely to the specific cladding material identified in these regulations (fuel rods with zircaloy or ZIRLO cladding) and will provide for the application of 10 CFR 50.46 to the use of AXIOM fuel rod cladding at CNS U1 and MNS.

2.0 BACKGROUND

As the nuclear industry pursues longer operating cycles, with increased fuel discharge burnup and fuel duty, the corrosion performance requirements for nuclear fuel cladding become more demanding. AXIOM cladding is designed to exhibit improved corrosion resistance, lower hydrogen pickup (HPU), and lower creep compared to other Westinghouse Electric Company, LLC (Westinghouse) cladding products (e.g., ZIRLO and Optimized ZIRLO™). AXIOM cladding is a niobium-bearing alloy with reduced tin content to increase corrosion resistance like Optimized ZIRLO alloy. AXIOM cladding material has alloying elements including vanadium and copper to improve specific properties such as HPU.

In addition, fuel rod internal pressures (resulting from the increased fuel duty, use of integral fuel burnable absorbers, and corrosion/temperature feedback effects) have become more limiting with respect to fuel rod design criteria. Reducing the associated corrosion buildup, and thus, minimizing the temperature feedback effects, provides additional margin to the fuel rod internal pressure design limit.

As documented in the NRC’s safety evaluation (SE) for Westinghouse topical report WCAP-18546-P-A, “Westinghouse AXIOM Cladding for Use in Pressurized Water Reactor Fuel,” (ADAMS Accession No. ML22306A248), AXIOM cladding has been approved for use in PWR fuel. Additionally, an exemption request has previously been approved for use of AXIOM cladding by Turkey Point Nuclear Generating, Unit Nos. 3 and 4 (Turkey Point), by letter dated September 13, 2024 (ADAMS Accession No. ML24207A034).

Technical Specification (TS) changes for CNS U1 and MNS are required to allow the use of AXIOM fuel rod cladding for core reload applications. The request for these changes is provided in Enclosure 1 of this submittal.

3.0 TECHNICAL JUSTIFICATION OF ACCEPTABILITY

Westinghouse topical report WCAP-18546-P-A provides the details and results of tests for AXIOM cladding along with the material properties proposed for use in various models and methodologies when analyzing AXIOM fuel cladding, including the use of the FULL SPECTRUM™ Loss-of-Coolant Accident (LOCA)(FSLOCA™) evaluation model (EM) (ADAMS Accession No. ML17277A130). As described in the SE for WCAP-18546-P-A, the NRC staff reviewed the licensing criteria assessment, which included various fuel rod design criteria, safety analyses for both LOCA and non-LOCA transients, and radiological consequence analyses, and found that the topical report is acceptable for referencing in licensing applications to the extent specified under the limitations and conditions associated with the topical report. The CNS U1 and MNS LOCA analysis for the fuel assemblies with AXIOM cladding was performed using the FSLOCA EM and adheres to the limitations of the associated topical reports. Enclosures 2 (proprietary) and 4 (non-proprietary) of the license amendment request attendant to this exemption request describes the CNS U1 and MNS LOCA evaluation performed for the fuel assemblies with AXIOM cladding.

4.0 JUSTIFICATION OF EXEMPTION

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that: 1) the exemption is authorized by law, 2) the exemption will not present an undue risk to public health and safety, 3) the exemption is consistent with the common defense and security; and (4) special circumstances, as defined in 10 CFR 50.12(a)(2) are present. The requested exemption to allow the use of an advanced zirconium alloy other than zircaloy or ZIRLO for fuel cladding material, in this case AXIOM, at MNS and CNS U1 satisfies these criteria as described below.

1. <u>This exemption is authorized by law</u>

As required by 10 CFR 50.12(a)(1), this requested exemption is “authorized by law.” The NRC has the authority under 10 CFR 50.12 to grant exemptions from the requirements of Part 50 upon showing proper justification. Further, it should be noted that CNS U1 and MNS are not seeking an exemption from the acceptance and analytical criteria of 10 CFR 50.46. The intent of this request is solely to allow the use of the criteria set forth in 10 CFR 50.46 for application to the AXIOM fuel rod cladding material, as it is currently not explicitly covered by the regulation.

2. <u>This exemption will not present an undue risk to public health and safety</u>

The reload evaluations will ensure that acceptance criteria are met for future reload cores after the transition to fuel rods clad with AXIOM material. Fuel assemblies using AXIOM fuel rod cladding will be evaluated using NRC-approved analytical methods and plant-specific models to address the changes in the cladding material properties. The safety analyses for CNS U1 and MNS are supported by the applicable site-specific TSs. Reload cores are required to be operated in accordance with the operating limits specified in the TSs. Thus, the granting of this exemption request will not pose an undue risk to public health and safety.

3. This exemption is consistent with common defense and security

As noted above, this exemption request is only to allow the application of the aforementioned regulations to an improved fuel rod cladding material. All the requirements and acceptance criteria will be maintained. The special nuclear material in these assemblies is required to be handled and controlled in accordance with approved procedures. Use of AXIOM fuel rod cladding will not affect plant operations and is consistent with common defense and security.

4. Special circumstances are present which necessitate the request of an exemption to the regulations of 10 CFR 50.46

10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption to the regulations unless special circumstances are present. The requested exemption meets the special circumstance criteria of 10 CFR 50.12(a)(2)(ii), which states: “Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.” For MNS and CNS U1, application of the subject regulations is not necessary to achieve the underlying purpose of the rule.

10 CFR 50.46 identifies acceptance criteria for ECCS performance at nuclear power plants. Westinghouse has performed an evaluation using LOCA methods as described in Enclosures 2 (proprietary) and 4 (non-proprietary) of this submittal to ensure that assemblies with AXIOM fuel rod cladding material meet all LOCA safety criteria.

5.0 CONCLUSION

The acceptance criteria and requirements of 10 CFR 50.46 are currently limited in applicability to the use of fuel rods with zircaloy or ZIRLO cladding. 10 CFR 50.46 does not apply to the proposed use of AXIOM fuel rod cladding material because AXIOM has a slightly different composition than zircaloy or ZIRLO. With the approval of this exemption request, these regulations will be applied to the use of AXIOM fuel rod cladding at CNS U1 and MNS.

In order to support the use of AXIOM fuel rod cladding material at CNS U1 and MNS, an exemption from the requirements of 10 CFR 50.46 is requested. The acceptance criteria applicable to AXIOM cladding have been established in WCAP-18546-P-A and the plant specific response is shown to comply with the acceptance criteria. Pursuant to 10 CFR 50.12, the requested exemption is authorized by law, does not present undue risk to public health and safety, and is consistent with the common defense and security. Approval of this exemption request does not violate the underlying purpose of the rule. In addition, special circumstances exist to justify the approval of an exemption from the subject requirements.