PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

Exelon Generation.

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RS-20-010

10 CFR 50.90

February 28, 2020

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

Braidwood Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Byron Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Subject: Application to Revise Technical Specifications 5.6.5, "Core Operating Limits Report (COLR)"

References:

1. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "Response to Request for Information Regarding the Thermal Conductivity Degradation and 10 CFR 50.46 Report," dated March 19, 2012

2. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "ECCS Evaluation Model Error - 10 CFR 50.46 Report," dated May 21, 2012

3. Letter from David M. Gullott (Exelon Generation Company, LLC) to U.S. NRC, "Supplement to RS-12-037 and RS-12-087: Revision to Exelon Generation Company, LLC Commitment Relating to Large Break Loss of Coolant Accident (LBLOCA) Analysis with an NRC Approved Emergency Core Cooling System (ECCS) Evaluation Model that Explicitly Accounts for Thermal Conductivity Degradation (TCD)," dated December 14, 2016

4. Letter from Dwi Murray (Exelon Generation Company, LLC) to U.S. NRC, "Supplement to RS-16-239: Revision to Exelon Generation Company, LLC Commitment Relating to Large Break Loss of Coolant Accident (LBLOCA) Analysis with an NRC Approved Emergency Core Cooling System (ECCS) Evaluation Model that Explicitly Accounts for Thermal Conducitivity Degradation (TCD)," dated September 5, 2019

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Attachment 3 contains Proprietary Information.
When separated from Attachment 3, this document is decontrolled.

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) is submitting a request for an amendment to the Technical Specifications (TS) for Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2 (Braidwood), and Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2 (Byron).

This amendment request proposes to revise TS 5.6.5, "Core Operating Limits Report (COLR)" The proposed change revises TS 5.6.5 to replace the current NRC approved Loss-of-Coolant Accident (LOCA) methodologies with a single, newer NRC approved LOCA methodology, the FULL SPECTRUM™[1] LOCA Evaluation Model (FSLOCA™ EM), that is contained in WCAP-16996-P-A, Rev. 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," and was used for LOCA reanalysis for Braidwood and Byron.

This License Amendment Request completes EGC's action to perform an ECCS LOCA reanalysis in accordance with 10 CFR 50.46(a)(3)(ii) as described in Reference 1 and as amended in References 3 and 4.

Attachment 3 is a proprietary version of the Application of Westinghouse FULL SPECTRUM™ LOCA Evaluation Model to Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2. Attachment 3 contains information proprietary to Westinghouse Electric Company LLC. Westinghouse Electric Company LLC and EGC request that the contents of Attachment 3 be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). Attachment 4 provides the non-proprietary version of the Application of Westinghouse FULL SPECTRUM™ LOCA Evaluation Model to Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2.

The attached request is subdivided as follows:

- Attachment 1 provides a description and evaluation of the proposed changes.
- Attachment 2 provides the Affidavit of Withholding, Proprietary Information Notice, and Copyright Notice for FSLOCA™ LAR Input Document, "APPLICATION OF WESTINGHOUSE FULL SPECTRUM™ LOCA EVALUATION MODEL TO BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2."
- Attachment 3 provides the Proprietary Class 2 Version of FSLOCA™ LAR Input Document, "APPLICATION OF WESTINGHOUSE FULL SPECTRUM™ LOCA EVALUATION MODEL TO BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2."
- Attachment 4 provides the Non-Proprietary Class 3 Version of the FSLOCA™ LAR Input Document, "APPLICATION OF WESTINGHOUSE FULL SPECTRUM™ LOCA EVALUATION MODEL TO BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2."
- Attachment 5a provides the markup of the affected TS pages for Braidwood.

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- Attachment 5b provides the markup of the affected TS pages for Byron.
- Attachment 6a provides the TS Bases pages marked to show the proposed changes for information only for Braidwood.
- Attachment 6b provides the TS Bases pages marked to show the proposed changes for information only for Byron.

The proposed change has been reviewed by Braidwood and Byron Plant Operations Review Committees in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed license amendment request within one year of this submittal date; i.e., by February 28, 2021. Once approved, the amendment shall be implemented according to the following table. It will be implemented during the defueled window in the specified Refueling Outage (Calendar Year (CY) provided for reference):

Unit	Date
Braidwood Unit 1	Refueling Outage A1R22 (April 2021)
Byron Unit 1	Refueling Outage B1R24 (September 2021)
Braidwood Unit 2	Refueling Outage A2R22 (October 2021)
Byron Unit 2	Refueling Outage B2R23 (April 2022)

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

This letter contains no regulatory commitments. Should you have any questions concerning this letter, please contact Ms. Lisa Zurawski at (630) 657-2816.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 28th day of February 2020.

Respectfully,

Dwi Murray
Senior Manager – Licensing
Exelon Generation Company, LLC

Attachments:

1) Evaluation of Proposed Changes
2) Affidavit of Withholding, Proprietary Information Notice, and Copyright Notice for FSLOCA™ LAR Input Document, "APPLICATION OF WESTINGHOUSE FULL SPECTRUM™ LOCA EVALUATION MODEL TO BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2"
3) Proprietary Class 2 Version of FSLOCA™ LAR Input Document, "APPLICATION OF WESTINGHOUSE FULL SPECTRUM™ LOCA EVALUATION MODEL TO BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2"
4) Non-Proprietary Class 3 Version of the FSLOCA™ LAR Input Document, "APPLICATION OF WESTINGHOUSE FULL SPECTRUM™ LOCA EVALUATION MODEL TO BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2"

5a) Proposed Technical Specification Changes (Mark-Up) for Braidwood
5b) Proposed Technical Specification Changes (Mark-Up) for Byron
6a) Proposed Technical Specifications Bases Changes (Mark-Up) for Information Only for Braidwood
6b) Proposed Technical Specifications Bases Changes (Mark-Up) for Information Only for Byron

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, Braidwood Station
NRC Senior Resident Inspector, Byron Station
Illinois Emergency Management Agency – Division of Nuclear Safety

Attachment 1
Evaluation of Proposed Changes

Subject: **License Amendment Request to revise Technical Specifications 5.6.5, "Core Operating Limits Report (COLR)"**

1.0 DESCRIPTION

2.0 DETAILED DESCRIPTION

2.1 Proposed Changes

3.0 TECHNICAL EVALUATION

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements / Criteria

4.2 Precedent

4.3 No Significant Hazards Consideration

4.4 Conclusions

5.0 ENVIRONMENTAL CONSIDERATION

6.0 REFERENCES

1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC, (EGC) requests amendments to Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, (Braidwood) and Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2 (Byron). This amendment request proposes to revise Technical Specifications (TS) 5.6.5, "Core Operating Limits Report (COLR)." The proposed change revises TS 5.6.5 to replace the current NRC approved Loss-of-Coolant Accident (LOCA) methodologies with a single, newer NRC approved LOCA methodology, the FULL SPECTRUM™[2] LOCA Evaluation Model (FSLOCA™ EM), that is contained in WCAP-16996-P-A, Rev. 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," (Reference 1) and was used for LOCA reanalysis for Braidwood and Byron.

The proposed change to the emergency core cooling system (ECCS) LOCA EM for Braidwood and Byron does not involve any changes to fuel type, peaking factors, fuel structural analysis, or boric acid precipitation methodology.

This license amendment request (LAR) completes EGC's action to perform an ECCS LOCA reanalysis in accordance with 10 CFR 50.46(a)(3)(ii) as described in EGC Letter RS-12-037, "Response to Request for Information Regarding Thermal Conductivity Degradation and 10 CFR 50.46 Report," dated March 19, 2012 and as amended in EGC Letter RS-16-239, "Supplement to RS-12-037 and RS-12-087: Revision to Exelon Generation Company, LLC Commitment Relating to Large Break Loss of Coolant Accident (LBLOCA) Analysis with an NRC Approved Emergency Core Cooling System (ECCS) Evaluation Model that Explicitly Accounts for Thermal Conductivity Degradation (TCD)," dated December 14, 2016 and EGC Letter RS-19-091, "Supplement to RS-16-239: Revision to Exelon Generation Company, LLC Commitment Relating to Large Break Loss of Coolant Accident (LBLOCA) Analysis with an NRC Approved Emergency Core Cooling system (ECCS) Evaluation Model that Explicitly Accounts for Thermal Conductivity Degradation (TCD)," dated September 5, 2019.

2.0 DETAILED DESCRIPTION

The proposed changes to TS 5.6.5 reflect the NRC approved LOCA methodology that was used for the LOCA reanalysis for Braidwood and Byron. Attachments 5a and 5b to this amendment request provide the markup pages of the existing TS to show the proposed change.

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2.1 Proposed Changes

The proposed TS changes are described below.

TS 5.6.5.b.6 through 5.6.5.b.8 currently states:

6. WCAP-16009-P-A, Revision 0, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.
7. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.
8. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

Revised TS 5.6.5.b.6 through 5.6.5.b.8 will state:

6. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.
7. Not Used.
8. Not Used.

3.0 TECHNICAL EVALUATION

3.1 Compliance with the Limitations and Conditions in the Revised NRC Final Safety Evaluation (FSE) for Westinghouse WCAP-16996-P-A, Rev. 1

Attachments 3 and 4 to this amendment request show Braidwood, Units 1 and 2 and Byron, Units 1 and 2 analyses are in compliance with the Limitations and Conditions.

3.2. Changes and Corrections to the FSLOCA™ EM in Westinghouse WCAP-16996-P-A, Revision 1

Westinghouse has issued letter LTR-NRC-18-30 (Reference 2) and LTR-NRC-19-6 (Reference 3) to the NRC which document several changes and corrections that have been made to the FSLOCA™ EM in Westinghouse WCAP-16996-P-A, Revision 1, after the NRC issued the revised Final Safety Evaluation. Those changes and corrections that are applicable to the FSLOCA™ methods amount to minimal impacts and will be handled in accordance with 10 CFR 50.46 requirements. One additional correction involving the treatment of uncertainty in the gamma energy redistribution was identified and corrected in the analysis as outline in Section 2.3 of Attachments 3 and 4. Furthermore, any future changes or corrections to the methods or the specific Braidwood and Byron Analysis of Record being implemented here will also be in accordance with 10 CFR 50.46 requirements.

3.3. Compliance with 10 CFR 50.46

It must be demonstrated that there is a high level of probability that the following criteria in 10 CFR 50.46 are not exceeded:

Peak Cladding Temperature (10 CFR 50.46(b)(1)) - The analysis Peak Cladding Temperature (PCT), corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT is less than 2,200°F, the analyses with the FSLOCA™ EM confirm that 10 CFR 50.46 acceptance criterion (b)(1), i.e., that PCT not exceed 2,200°F, is satisfied.

The results are shown in Tables 8 and 9 of Attachments 3 and 4 for Braidwood and Byron Unit 1 and Unit 2, respectively.

Maximum Cladding Oxidation (10 CFR 50.46(b)(2)) - The analysis Maximum Local Oxidation (MLO) corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting MLO is less than 17 percent when converting the time-at-temperature to an equivalent cladding reacted using the Baker-Just correlation and adding the pre-transient corrosion, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., that the MLO of the cladding not exceed 17 percent of the total cladding thickness before oxidation, is satisfied.

The results are shown in Tables 8 and 9 of Attachments 3 and 4 for Braidwood and Byron Unit 1 and Unit 2, respectively.

Maximum Hydrogen Generation (10 CFR 50.46(b)(3)) - The analysis Core-Wide Oxidation (CWO) corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. Since the resulting CWO is less than 1 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., that CWO not exceed 1 percent of the total hypothetical amount, is satisfied.

The results are shown in Tables 8 and 9 of Attachments 3 and 4 for Braidwood and Byron Unit 1 and Unit 2, respectively.

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

This criterion is met by demonstrating compliance with criteria (b)(1), (b)(2), and (b)(3), and by assuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed. Tables 8 and 9 of Attachments 3 and 4 show Criteria (b)(1), (b)(2), and (b)(3) have been met for Braidwood and Byron Units 1 and 2, respectively.

Section 32.1 of the NRC approved FSLOCA™ EM (WCAP-16996-P-A, Revision 1) documents 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). Inboard grid deformation due to the combined LOCA and seismic loads was calculated to not occur for Braidwood and Byron. The FSLOCA™ EM analyses did not invalidate the existing seismic / LOCA analysis.

Long-Term Cooling (10 CFR 50.46(b)(5)) - This criterion 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 (WCAP-16996-P-A, Revision 1).

In summary, based on the analysis results for the small-break LOCA (SBLOCA, Region I) and large-break LOCA (LBLOCA, Region II) presented in Tables 8 and 9 of Attachments 3 and 4 for Braidwood and Byron Unit 1 and Unit 2, respectively, and the discussions above relative to the criteria in 10 CFR 50.46(b)(4) and (b)(5), it is concluded that Braidwood, Units 1 and 2 and Byron, Units 1 and 2 would continue to comply with the criteria in 10 CFR 50.46 upon approval of this LAR.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Section 182.a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of the TS are contained in 10 CFR 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls.

However, the rule does not specify the particular requirements to be included in a plant's TS. Under 10 CFR 50.36(c)(2)(ii), a limiting condition for operation must be included in TS for any item meeting one or more of the following four criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor, coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The proposed change does not impact the TS safety limits, limiting safety system settings, and limiting control settings; LCOs; surveillance requirements; or design features.

The proposed replacement for NRC approved LOCA methodology will be included in the Administrative Controls section of the Braidwood and Byron TS and would be used to determine a core operating limit. The use of the proposed NRC approved LOCA methodology will continue to ensure that the plant is operated in a safe manner. Therefore, the proposed change is consistent with the Administrative Controls requirement of 10 CFR 50.36(c)(5).

10 CFR 50.46 includes requirements and acceptance criteria pertaining to the evaluation of post accident ECCS performance. This regulation includes the requirement that "... uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria ... there is a high level of probability that the criteria would not be exceeded."

The proposed change requests NRC approval to use the FULL SPECTRUM™ LOCA (FSLOCA™) methodology described in WCAP-16996-P-A, Revision 1 for the performance of full spectrum LOCA analyses, including treatment of uncertainties in the inputs used for the analysis. No change is proposed to the analysis acceptance criteria specified in 10 CFR 50.46. The NRC has reviewed WCAP-16996-P-A, Revision 1 and found it acceptable for referencing in licensing applications for Westinghouse designed four loop Pressurized Water Reactors. WCAP-16996-P-A, Revision 1 is applicable to Braidwood Station and Byron Station, and the plant specific application of the FSLOCA™ methodology to the LOCA analyses have been performed in accordance with the conditions and limitations of the topical report and the associated NRC safety evaluation. The plant specific analyses demonstrate that the requirements of 10 CFR 50.46 will continue to be met, thus ensuring continued safe plant operation.

NRC Generic Letter (GL) 88-16, "Removal of Cycle-Specific, Parameter Limits from Technical Specifications," dated October 4, 1988 (Reference 4), provides that it is acceptable for licensees to control reactor physics parameter limits by specifying an NRC approved calculation methodology. These parameter limits may be removed from the TS and placed in a cycle specific COLR that is required to be submitted to the NRC every operating cycle or each time it is revised.

Consistent with the guidance in NRC GL 88-16, Braidwood and Byron TS 5.6.5, "Core Operating Limits Report (COLR)," requires the following:

- An NRC approved methodology is to be used to determine the core operating limits listed in TS 5.6.5.a;
- The specific NRC approved methodologies used to determine the core operating limits are to be listed in TS 5.6.5.b; and
- The COLR, including any midcycle revisions or supplements, is to be provided upon issuance for each reload cycle to the NRC in accordance with TS 5.6.5.d.

The COLR is defined in Section 1.1 of the TS and the reporting requirements in TS 5.6.5 require that a COLR be submitted to the NRC each operating cycle, or each time the COLR is revised. The GL also required that the TS include a list of references of the NRC approved methodologies that are used to determine the cycle specific core operating limits. TS 5.6.5.b identifies the NRC approved analytical methodologies that are used to determine the core operating limits for Braidwood, Units 1 and 2 and Byron, Units 1 and 2. Upon approval of the proposed change, the guidance in the GL continues to be met since the proposed change will continue to specify the NRC approved methodologies used to determine the core operating limits.

Therefore, the proposed change to replace the previous NRC approved LOCA methodologies with the NRC approved LOCA methodology in WCAP-16996-P-A, Rev. 1, which was used for the Braidwood and Byron LOCA reanalysis, satisfies NRC GL 88-16.

4.2 Precedent

The NRC has approved the following similar license amendment request to revise Core Operating Limits Report for Full Spectrum Loss-of Coolant Accident Methodology:

Letter from B.K. Singal (NRC) to J.M. Welsch (PG&E), "Diablo Canyon Nuclear Power Plant, Units 1 and 2 – Issuance of Amendment Nos. 234 and 236 to Revise Technical Specification 5.6.5b, "Core Operating Limits Report (COLR)," for Full Spectrum Loss-of-Coolant Accident Methodology (EPID L-2018-LLA-0730)," dated January 9, 2020 (ML19316A109). (Reference 5)

The following are three other similar amendment requests currently being reviewed by the NRC:

Letter from J.T. Polickoski (Tennessee Valley Authority) to US NRC, "Application to Implement the FULL SPECTRUM LOCA™ (FSLOCA™) Methodology for Loss-of-Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04)," dated January 17, 2020 (ML20017A338). (Reference 6)

Letter from M.D. Sartain (Dominion Energy Virginia) to US NRC, "Virginia Electric and Power Company North Anna Power Station Units 1 and 2 Proposed License Amendment Request Addition of Analytical Methodology to the Core Operating Limits Report for a Full Spectrum Loss of Coolant Accident (FSLOCA)," dated October 30, 2019 (ML19309D197). (Reference 7)

Letter from M.D. Sartain (Dominion Energy Virginia) to US NRC, "Virginia Electric and Power Company Surry Power Station Units 1 and 2 Proposed License Amendment Request Addition of Analytical Methodology to the Core Operating Limits Report for a Large Break Loss of Coolant Accident (LBLOCA)," dated October 30, 2019 (ML19309D196). (Reference 8)

4.3 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC, (EGC) requests amendments to Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, (Braidwood) and Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2 (Byron). This amendment request proposes to revise Technical Specifications (TS) 5.6.5, "Core Operating Limits Report (COLR)." The proposed change revises TS 5.6.5 to replace the current NRC approved Loss-of-Coolant Accident (LOCA) methodologies with a single, newer NRC approved LOCA methodology, the FULL

SPECTRUM™ LOCA Evaluation Model (FSLOCA™ EM), that is contained in WCAP-16996-P-A, Rev. 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," and was used for LOCA reanalysis for Braidwood and Byron. The NRC approved WCAP-16996-P-A, Revision 1 in a safety evaluation dated October 19, 2016.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises TS 5.6.5.b to replace the current NRC approved Loss-of-Coolant Accident (LOCA) methodologies listed in TS 5.6.5.b with another NRC approved methodology contained in WCAP-16996-P-A, Rev. 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)."

   The proposed changes to the TS 5.6.5.b core operating limits methodologies, consists of replacing three current LOCA methodologies with a newer, single NRC approved methodology (the FSLOCA™ EM). The NRC review of the FSLOCA™ EM concluded that the analytical methods are acceptable as a replacement for the current LOCA analytical methods listed in TS 5.6.5.b.

   The proposed change does not affect the design or function of any plant structures, systems, and components (SSCs). Thus, the proposed change does not affect plant operation, design features, or the capability of any SSC to perform its safety function. In addition, the proposed change does not affect any previously evaluated accidents in the UFSAR, or any SSCs, operating procedures, and administrative controls that have the function of preventing or mitigating any accident previously evaluated in the UFSAR. Thus, the proposed use of the FSLOCA™ EM will continue to assure that the plant operates in the same safe manner as before and will not involve an increase in the probability of an accident.

The analyses results determined by use of the proposed new methodology will not increase the reactor power level or the core fission product inventory and will not change any transport assumptions or the shutdown margin requirements of the Braidwood and Byron TS. As such, Braidwood and Byron will continue to operate within the power distribution limits and shutdown margins required by the TS and within the assumptions of the safety analyses described in the UFSAR. As such, the proposed changes do not involve a significant increase in the consequences of an accident.

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

2. Does the proposed change create the possibility of a new or different accident from any accident previously evaluated?

   Response: No.

   The proposed change revises TS 5.6.5.b to replace the current NRC approved Loss-of-Coolant Accident (LOCA) methodologies listed in TS 5.6.5.b with a single, newer NRC approved methodology contained in WCAP-16996-P-A, Rev. 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)." The NRC review of the FSLOCA™ EM concluded that the analytical methods are acceptable as a replacement for the current LOCA analytical methods listed in TS 5.6.5.b.

   The proposed change provides revised analytical methods and does not change any system functions or maintenance activities. The change does not involve physical alteration of the plant; that is, no new or different type of equipment will be installed. The change does not impact the ability of any SSC to perform its safety function consistent with the assumptions of the safety analyses and continues to assure the plant is operated within safe limits. As such, the proposed change does not create new failure modes or mechanisms that are not identifiable during testing, and no new accident precursors are generated.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. The setpoints at which protective actions are initiated are not altered by the proposed change. Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The NRC has reviewed and approved the new methodology for the intended use in lieu of the current methodologies; thus, the margin of safety is not reduced due to this change.

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

Based on the above evaluation, EGC concludes that the proposed change does not involve a 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 CONSIDERATION

EGC has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents 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 need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

2. LTR-NRC-18-30, "10 CFR 50.46 Annual Notification and Reporting for 2017," July 2018.

3. LTR-NRC-19-6, "10 CFR 50.46 Annual Notification and Reporting for 2018," February 7, 2019.

4. NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 1988.

5. Letter from B.K. Singal (NRC) to J.M. Welsch (PG&E), "Diablo Canyon Nuclear Power Plant, Units 1 and 2 – Issuance of Amendment Nos. 234 and 236 to Revise Technical Specification 5.6.5b, "Core Operating Limits Report (COLR)," for Full Spectrum Loss-of-Coolant Accident Methodology (EPID L-2018-LLA-0730)," dated January 9, 2020 (ML19316A109).

6. Letter from .T. Polickoski (Tennessee Valley Authority) to US NRC, "Application to Implement the FULL SPECTRUM LOCA™ (FSLOCA™) Methodology for Loss-of-Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04)," dated January 17, 2020 (ML20017A338).

7. Letter from M.D. Sartain (Dominion Energy Virginia) to US NRC, "Virginia Electric and Power Company North Anna Power Station Units 1 and 2 Proposed License Amendment Request Addition of Analytical Methodology to the Core Operating Limits Report for a Full Spectrum Loss of Coolant Accident (FSLOCA)," dated October 30, 2019 (ML19309D197).

8. Letter from M.D. Sartain (Dominion Energy Virginia) to US NRC, "Virginia Electric and Power Company Surry Power Station Units 1 and 2 Proposed License Amendment Request Addition of Analytical Methodology to the Core Operating Limits Report for a Large Break Loss of Coolant Accident (LBLOCA)," dated October 30, 2019 (ML19309D196).

Attachment 2

Affidavit of Withholding, Proprietary Information Notice, and Copyright Notice for FSLOCA™ LAR Input Document, "APPLICATION OF WESTINGHOUSE FULL SPECTRUM™ LOCA EVALUATION MODEL TO BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNITS 1 AND 2"

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Camille T. Zozula, 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 Westinghouse Letter CAE-19-26 / CCE-19-25, Attachment 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) 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.

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(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

AFFIDAVIT

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: 13 Sept 2019

Camille T. Zozula, Manager
Infrastructure & Facilities Licensing

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (5)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

Attachment 5a

Braidwood Station, Units 1 and 2
NRC Docket Nos. 50-456 and 50-457

Proposed Technical Specification Changes (Mark-Up)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

6. ~~WCAP-16009-P-A, Revision 0, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.~~

Not Used.

7. ~~WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.~~

Not Used.

8. ~~WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.~~

9. WCAP-10216-P-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

10. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

11. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

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

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

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; and

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

large break

must be considered, and

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

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 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),

safety injection signal generation,

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 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.

in the modeling

for a large break LOCA

is assumed to inject into the reactor coolant system

intermediate

into small break LOCA, the accumulators, centrifugal charging and SI pumps all

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 and centrifugal charging pumps both~~ 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.

and SI

BASES

APPLICABLE SAFETY ANALYSES (continued)

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 4) will be met following a LOCA:

There is

a. ~~During a small break Loss Of Coolant Accident (LOCA) maximum fuel element cladding temperature is ≤ 2200°F and during a large break LOCA there must be~~ a high level of probability that maximum fuel element cladding temperature is ≤ 2200°F;

There is a high level of probability that the maximum

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

There is a high level of probability that the maximum

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 to the long term cooling requirements of 10 CFR 50.46.

and

per approved methods (Ref.5)

For the small break ~~LOCA analyses, a nominal contained accumulator water volume is used. For the~~ large break LOCA analyses, a contained accumulator water volume range of 920 ft^3 - 980 ft^3 is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. ~~For small breaks, the peak clad temperature is not sensitive to the accumulator water volume. For large breaks, there are two competing effects regarding accumulator water volume: the amount of water available for injection versus the injection rate. A higher water volume results in a larger total injection but at a slower injection rate. Conversely, a lower water volume results in a smaller total injection but at a faster injection rate.~~

BASES

BACKGROUND (continued)

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. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

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 LOCA transient ~~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 (Refs. 2 and 3). 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.

BASES

BACKGROUND (continued)

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR (only when PDMS is inoperable), and control rod insertion.

APPLICABLE SAFETY ANALYSES

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

a. During a ~~small break~~ Loss Of Coolant Accident (LOCA) the ~~peak cladding temperature must not exceed 2200°F and during a large break LOCA there must be a high level of probability that the peak cladding temperature does not exceed 2200°F~~ (Ref. 1);

10 CFR 50.46 acceptance criteria must be met

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 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 $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. ~~Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.~~

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

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

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

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

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

where: F_Q^{RTP} is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For this facility, the actual values of F_Q^{RTP} and K(Z) are given in the COLR; however, F_Q^{RTP} is normally a number on the order of ~~2.50~~ 2.60, and K(Z) is a function that looks like the one provided in Figure B 3.2.1-1.

$F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

When PDMS is inoperable, an $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,

$$F_Q^C(Z) = F_Q^M(Z) * (1.0815)$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

BASES

LCO (continued)

When PDMS is OPERABLE, $F_Q(Z)$ is determined continuously. Then,

$$F_Q^C(Z) = F_Q^M(Z) * U_{FQ}$$

where U_{FQ} is a factor that accounts for measurement uncertainty (Ref. 4) and engineering uncertainty defined in the COLR.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. When PDMS is inoperable, the $F_Q^C(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that ~~precludes peak cladding temperatures above 2200°F during a small break LOCA and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F~~ during a ~~large break~~ LOCA (Ref. 1).

ensure that the 10 CFR 50.46 acceptance criteria are met

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 LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ may produce unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ 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.

This figure is replaced with the figure on the next page.

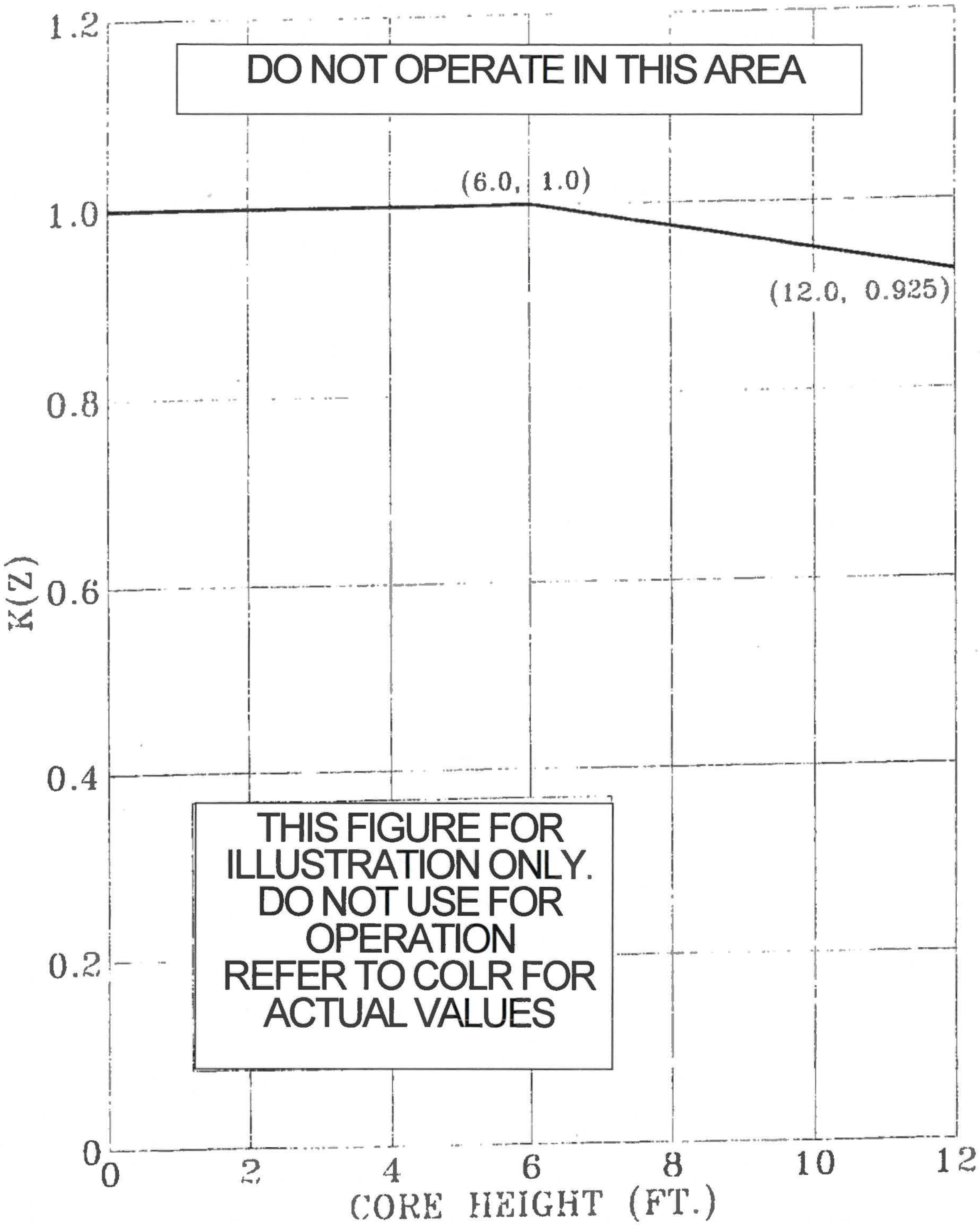

Figure B 3.2.1-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a function of Core Height

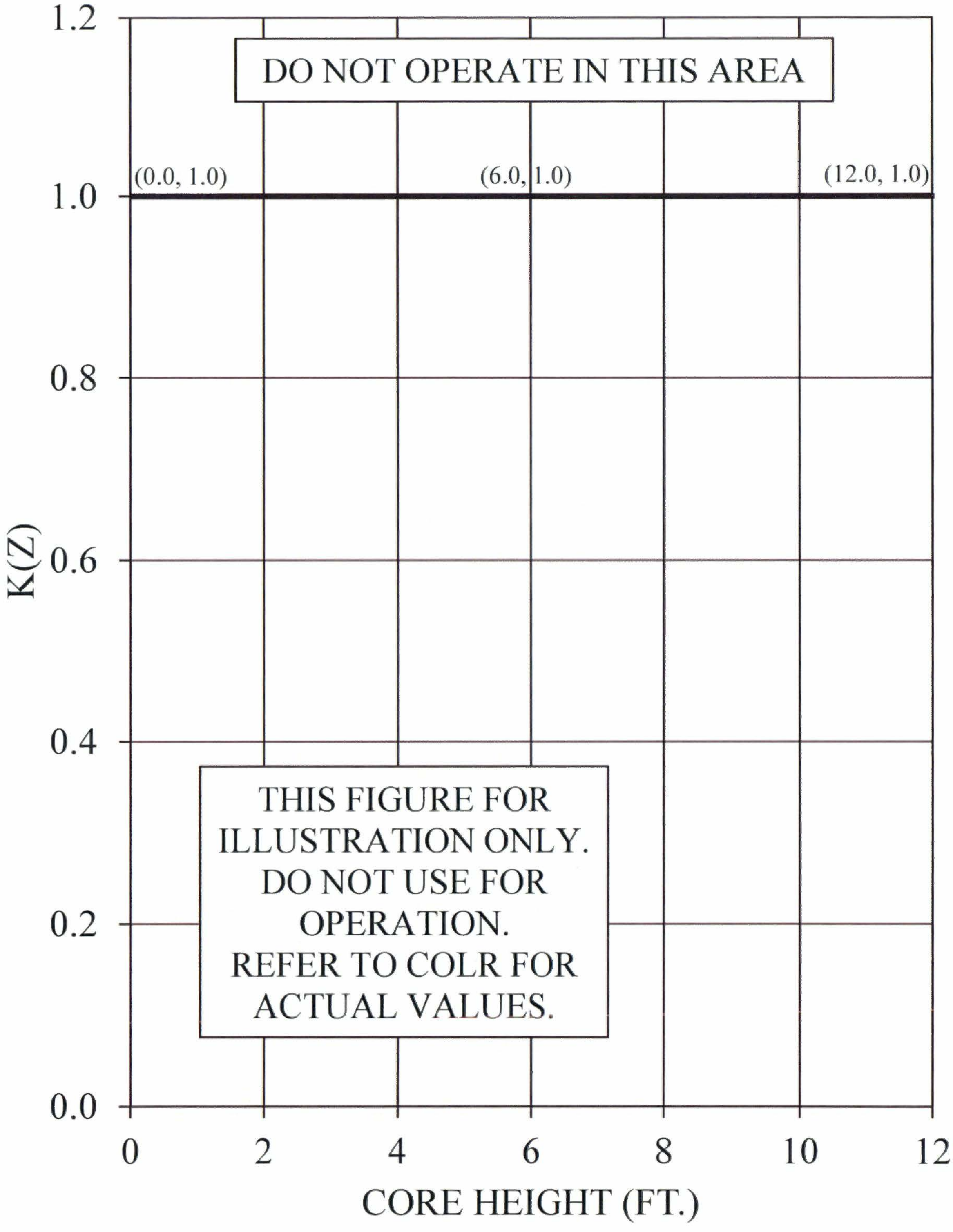

Figure B 3.2.1-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a function of Core Height

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design criterion for the Departure from Nucleate Boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a ~~small break~~ Loss Of Coolant Accident (LOCA) ~~Peak Cladding Temperature (PCT) must not exceed 2200°F and during a large break LOCA there must be a high level of probability that PCT does not exceed 2200°F~~;

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

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 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 Shutdown Margin with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design criterion is met for normal operation, operational transients, and any transients arising from events of moderate frequency. Refer to the Bases for LCO 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits," for a discussion of the applicable Departure from Nucleate Boiling Ratio (DNBR) limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), the Nuclear Heat Flus 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).

~~The LOCA safety analysis indirectly models~~ $\cancel{F^N_{\Delta H}}$ ~~as an input parameter. The Nuclear Heat Flux Hot Channel Factor (~~$\cancel{F_Q(Z)}$~~) 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 3.1.6, "Control Bank Insertion Limits," LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," LCO 3.2.2, LCO 3.2.3, LCO 3.2.4, and LCO 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system when PDMS is inoperable. 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. When PDMS is OPERABLE, $F^N_{\Delta H}$ and $F_Q(Z)$ are determined continuously. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on DNBR and Control Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ 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 a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the plant safety analyses.

The power multiplication factor in this equation provides margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the fuel design limits for DNBR and ~~PCT~~ the 10 CFR 50.46 acceptance criteria (Ref. 3). 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. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1, A.2, A.3, and A.4

With $F^N_{\Delta H}$ exceeding its limit, Condition A is entered. $F^N_{\Delta H}$ may be restored to within its limits within 4 hours, through, for example, realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. If the value of $F^N_{\Delta H}$ is not restored to within its specified limit, THERMAL POWER must be reduced to < 50% RTP in accordance with Required Action A.1. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses.

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 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, 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. When Power Distribution Monitoring System (PDMS) is OPERABLE, Peak Linear Heat Rate and the linear power along the fuel rod with the highest integrated power are measured continuously.

APPLICABLE SAFETY ANALYSES

Limits on QPTR preclude core power distributions that violate the following fuel design criteria:

a. During a ~~small break~~ Loss Of Coolant Accident (LOCA) the ~~Peak Cladding Temperature (PCT) must not exceed 2200°F and during a large break LOCA there must be a high level of probability that the PCT does not exceed 2200°F~~ (Ref. 1);

10 CFR 50.46 acceptance criteria must be met

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 2); and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure-Low is a primary actuation signal for ~~small~~ Loss Of Coolant Accidents (LOCAs) and a backup actuation signal for Steam Line Breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, 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 when the unit status is within the Applicability. A channel is OPERABLE with a trip setpoint outside its calibration tolerance band provided the trip setpoint "as found" value does not exceed its associated Allowable Value and provided the trip setpoint "as left" value is adjusted to a value within the calibration tolerance band of the Nominal Trip Setpoint. A trip setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions. 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.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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)~~ compliance with the 10 CFR 50.46 acceptance criteria (Ref. 17); and

2. Boration to ensure recovery and maintenance of SDM.

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 Isolation;
- Reactor Trip;
- Turbine Trip;
- Feedwater Isolation;
- Start of Auxiliary Feedwater (AF) pumps;
- Control room ventilation isolation; and
- Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) pump suction to containment sump.

BASES

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 7.
3. UFSAR, Chapter 15.
4. IEEE-279-1971.
5. Technical Requirements Manual.
6. WCAP-12523, "Bases Document for Westinghouse Setpoint Methodology for Protection Systems, Zion/Byron/Braidwood Units" October 1990.
7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
8. WCAP-13632 Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," August 1995.
9. UFSAR, Section 7.3.
10. WCAP-12583, "Westinghouse Setpoint Methodology For Protection Systems, Byron/Braidwood Stations," May 1990.
11. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.
12. Not used.
13. Not used.
14. Not used.
15. WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.
16. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2000.
17. 10 CFR 50.46.

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 large break 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.

In the refill phase of a large break 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. Initial accumulator inventory which is injected into the reactor vessel is lost out the break. 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 initiate ~~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.

BASES

BACKGROUND (continued)

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. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

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 LOCA transient ~~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 (Refs. 2 and 3). 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

large break

must be considered, and

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),

safety injection signal generation,

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 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.

in the modeling

for a large break LOCA

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 and centrifugal charging pumps both~~ 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.

is assumed to inject into the reactor coolant system

intermediate

into small break LOCA, the accumulators, centrifugal charging and SI pumps all

and SI

BASES

APPLICABLE SAFETY ANALYSES (continued)

There is

There is a high level of probability that the maximum

There is a high level of probability that the maximum

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

and

per approved methods (Ref. 5)

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 4) will be met following a LOCA:

a. ~~During a small break Loss Of Coolant Accident (LOCA) maximum fuel element cladding temperature is ≤ 2200°F and during a large break LOCA there must be~~ a high level of probability that 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.

Since the accumulators discharge during the blowdown phase of ~~a~~ ~~LOCA~~, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For the small break ~~LOCA analyses, a nominal contained accumulator water volume is used.~~ ~~For the~~ large break LOCA analyses, a contained accumulator water volume range of 920 ft^3 - 980 ft^3 is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. ~~For small breaks, the peak clad temperature is not sensitive to the accumulator water volume.~~ ~~For large breaks, there are two competing effects regarding accumulator water volume: the amount of water available for injection versus the injection rate.~~ ~~A higher water volume results in a larger total injection but at a slower injection rate.~~ ~~Conversely, a lower water volume results in a smaller total injection but at a faster injection rate.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both the large and the small break LOCA analyses model the pipe water volume from the accumulator to the SI accumulator discharge header downstream cold leg injection check valve (SI8948). However, an evaluation was performed neglecting the pipe water volume between the SI accumulator discharge header upstream cold leg injection check valve (SI8956) to the SI accumulator discharge header downstream cold leg injection check valve (SI8948) to address gas accumulation. This evaluation determined that the impact on peak clad temperature is minimal for the large break LOCA analyses and is a 20°F penalty for ~~was minimal for both the large break and~~ the small break LOCA analyses. ~~Since the range of the allowed accumulator volumes is relatively small and has a minimal effect on peak clad temperature, a nominal water volume is used in the small break LOCA analysis.~~ The ~~The small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the~~ Technical Specification (TS) minimum and maximum limits are ~~of~~ 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). The small break and large break LOCA analyses assume ~~The large break LOCA analysis assumes~~ a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. 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.

The small break and ~~LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.~~ ~~The~~ large break LOCA analyses are performed at a nitrogen cover pressure range of ~~587 psia to 692 psia~~ 586.3 psig to 662.3 psig per approved methods (Ref.5). The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 3).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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. 4) could be violated.

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

For an accumulator to be considered OPERABLE, the isolation valve must be fully open with power removed, a contained volume ≥ 31% and ≤ 63% (6995 gallons to 7217 gallons) with a boron concentration ≥ 2200 ppm and ≤ 2400 ppm, and a nitrogen cover pressure ≥ 602 and ≤ 647 psig, must be met.

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.

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. 4) limit of 2200°F~~.

the 10 CFR 50.46 (Ref. 4) acceptance criteria are met

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator since the static design of the accumulators limits the ways in which the concentration can be changed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.5

Sampling the affected accumulator within 6 hours after a 1% volume increase (nominally 70 gallons or 10% of indicated level) will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration of the accumulator after a 1% volume increase (10% indicated level increase) if the added water inventory is from the Refueling Water Storage Tank (RWST) and the boron concentration of the RWST is ≥ 2200 ppm and ≤ 2400 ppm. With the water contained in the RWST within the boron concentration requirements of the accumulators, any added inventory would not cause the accumulator's boron concentration to exceed the limits of this LCO.

With the only indication available to the operators in the control room being level indication in percent, a required accumulator volume increase of 1% or an increase of 10% of indicated level would require the accumulator to be sampled to verify the accumulator boron concentration is within the limits. The ~~small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the~~ TS minimum and maximum limit~~s of~~ are 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). These volumes are also indicated in the specific tank curves for the SI accumulators. The ~~large break LOCA analysis assumes~~ small break and large break LOCA analyses assume a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits. The 10% indicated level increase is considered a conservative indication for a 70 gallon increase in the accumulator volume requiring an increase in the sampling requirement to verify accumulator boron concentration remains within the specified limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.6

Verification that power is removed from each accumulator isolation valve operator 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 power to the accumulator motor operated isolation valves is removed by opening the motor control center breaker and tagging it out administratively. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. IEEE Standard 279-1971.

2. UFSAR, Chapter 15.

3. UFSAR, Chapter 6.

4. 10 CFR 50.46.

5. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

BASES

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 LOCA:

There is

a. ~~During a small break LOCA maximum fuel element cladding temperature is ≤ 2200°F and during a large break LOCA there must be~~ a high level of probability that maximum fuel element cladding temperature is ≤ 2200°F;

There is a high level of probability that the maximum

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

There is a high level of probability that the maximum

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;

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 temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Ref. 3). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

FSLOCA™

a. For the large break LOCA event, the ~~ASTRUM~~ methodology examines LOOP and no-LOOP cases with a single failure disabling one train of SI pumps. No failure of containment heat removal system is modeled; and

BASES

APPLICABLE SAFETY ANALYSES (continued)

a range of RWST temperatures of 32°F to 120°F is used for the containment spray temperature per approved methods (Ref.4).

In the ECCS analysis, ~~the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 35°F~~. If the lower temperature limit is violated, the containment spray further reduces containment pressure. The reduced containment pressure lowers the quality of steam exiting the break thus decreasing the rate which the steam is vented to the containment atmosphere. The decreased rate of steam vented to the containment atmosphere results in a corresponding decrease in the rate the Reactor Coolant System pressure drops and the rate ECCS fluid is injected in the core thereby causing a rise in peak clad temperature. ~~The~~ upper temperature limit of 100°F is used in the ~~small break LOCA analysis and~~ containment OPERABILITY analysis. Exceeding this temperature will result in ~~a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA and~~ 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 limits on RWST level and boron concentration also ensure that the post-LOCA sump pH will be between 8.0 and 11.0. The minimum and maximum pH values are verified for each fuel cycle using conservative maximum and minimum RWST volumes and the maximum and minimum allowed RWST boron concentrations. The LOCA offsite dose analysis assumes a conservatively low sump pH for the re-evolution of iodine from the sump. Ensuring that the minimum sump pH is at least 8.0 protects mechanical components and equipment inside containment from the effects of chloride induced stress corrosion cracking. Ensuring that the maximum sump pH is no greater than 11.0 limits the production of hydrogen due to the corrosion of aluminum and zinc inside containment. Finally, the limits on RWST boron concentration also ensure that the containment spray pH is acceptable. The calculation of the iodine removal effectiveness of the containment spray assumes a conservatively low containment spray pH.

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

The small break and large break LOCA analyses use a range of RWST temperatures of 32°F to 120°F per approved methods (Ref. 4). Exceeding the maximum temperature of this range could result in a higher peak cladding temperature, because there is less heat transfer from the core to the injected water. The containment

BASES

REFERENCES

1. WCAP-13964, Revision 2, "Commonwealth Edison Company, Byron/Braidwood Units 1 & 2, Increased Steam Generator Tube Plugging/Reduced Thermal Design Flow/Positive Moderator Temperature Coefficient Analysis Program, Engineering/Licensing Report," September 1994.

2. UFSAR, Chapter 15.

3. UFSAR, Section 6.2.1.

4. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The initial pressure condition used in the containment analysis was 1.0 psig. This resulted in a maximum peak pressure from a LOCA of 42.8 psig for Unit 1 and 38.4 psig for Unit 2. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA does not exceed the containment design pressure, 50 psig.

The containment was also evaluated for an external pressure load equivalent to -3.5 psig (Ref. 2). The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0.0 psig. This resulted in a minimum pressure inside containment of -3.48 psig, which is less than the design load.

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. ~~Therefore, for the reflood phase~~ For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with ~~10 CFR 50, Appendix K~~ WCAP-16996-P-A, Revision 1 (Ref. 3).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

ACTIONS (continued)

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit 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 controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.

2. Safety Evaluation Report Related to the Operation of Byron Station Units 1 and 2, Supplement 2.

3. ~~10 CFR 50, Appendix K.~~

WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.8 psig for Unit 1 and 38.4 psig for Unit 2 (experienced during a LOCA). The analysis shows that the peak containment temperature is 333.6°F for Unit 1 and 330.8°F for Unit 2 (experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level of 3672.6 MWt, one containment spray train and one containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the 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 WCAP-16996-P-A, Revision 1 ~~10 CFR 50, Appendix K~~ (Ref. 4).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -3.48 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the LOCA containment analysis is based on a response time associated with exceeding the containment High-3 pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time of 110.2 seconds (for the limiting case) includes Diesel Generator (DG) startup (for loss of offsite power), sequencing of equipment, containment spray pump startup, and spray line filling (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency 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 9.4.8.
3. UFSAR, Section 6.5.2.
4. ~~10 CFR 50, Appendix K.~~ WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.
5. UFSAR, Section 6.2.1.1.3.
6. UFSAR, Section 6.2.2.
7. UFSAR, Section 6.2.
8. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In addition, the minimum available AF flow and system characteristics are serious considerations in the analysis of a ~~small break Loss Of Coolant Accident (LOCA) and~~ loss of offsite power (Ref. 3).

The AF System design is such that it can perform its function following an FWLB between the main feedwater isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of one AF pump. The AF lines to the SGs are orificed such that sufficient flow is delivered to the non faulted SGs. Reactor trip is assumed to occur when the faulted SG reaches the low-low level setpoint. Sufficient flow would be delivered to the intact steam generators by the other AF pump.

During the loss of all AC power events, the Engineered Safety Feature Actuation System (ESFAS) automatically actuates the AF diesel driven pump and associated controls to ensure an adequate supply to the steam generators during loss of power. Valves which can be manually controlled are provided for each AF line to control the AF flow to each steam generator during loss of all AC power events.

AF flow is also an important consideration in the Steam Generator Tube Rupture (SGTR) event (Ref. 4). For the SGTR event, isolation of AF is an important recovery action. The flow path is normally isolated with the motor operated AF013 valves. Prior to isolation with AF013 valves, flow to the ruptured SG is limited by the control function of the AF005 valves.

The AF System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Since the AF005 valve is a backup and diverse method to isolate flow to the faulted SG in a SGTR event, this function does not satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). Administrative requirements for this function are addressed in the Technical Requirements Manual.

Attachment 5b

Byron Station, Units 1 and 2
NRC Docket Nos. 50-454 and 50-455

Proposed Technical Specification Changes (Mark-Up)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The initial pressure condition used in the containment analysis was 1.0 psig. This resulted in a maximum peak pressure from a LOCA of 42.8 psig for Unit 1 and 38.4 psig for Unit 2. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA does not exceed the containment design pressure, 50 psig.

The containment was also evaluated for an external pressure load equivalent to -3.5 psig (Ref. 2). The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0.0 psig. This resulted in a minimum pressure inside containment of -3.48 psig, which is less than the design load.

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. ~~Therefore, for the reflood phase~~ For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with ~~10 CFR 50, Appendix K~~ WCAP-16996-P-A, Revision 1 (Ref. 3).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

ACTIONS (continued)

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit 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 controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 6.2.
2. Safety Evaluation Report Related to the Operation of Byron Station Units 1 and 2, Supplement 2.
3. ~~10 CFR 50, Appendix K~~.

WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both the large and the small break LOCA analyses model the pipe water volume from the accumulator to the SI accumulator discharge header downstream cold leg injection check valve (SI8948). However, an evaluation was performed neglecting the pipe water volume between the SI accumulator discharge header upstream cold leg injection check valve (SI8956) to the SI accumulator discharge header downstream cold leg injection check valve (SI8948) to address gas accumulation. This evaluation determined that the impact on peak clad temperature ~~was minimal for both the large break and~~ is minimal for the large break LOCA analyses and is 20°F penalty for the small break LOCA analyses. ~~Since the range of the allowed accumulator volumes is relatively small and has a minimal effect on peak clad temperature, a nominal water volume is used in the small break LOCA analysis.~~ ~~The small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the~~ The Technical Specification (TS) minimum and maximum limits are ~~of~~ 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). The ~~large break LOCA analysis assumes~~ small break and large break LOCA analyses assume a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. 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.

The small break and ~~LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit.~~ ~~The~~ large break LOCA analyses are performed at a nitrogen cover pressure range of ~~587 psia to 692 psia~~ 586.3 psig to 662.3 psig per approved methods (Ref. 5). The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

6. ~~WCAP-16009-P-A, Revision 0, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.~~

Not Used.

7. ~~WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.~~

Not Used.

8. ~~WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.~~

9. WCAP-10216-P-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

10. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

11. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

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

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

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; and

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.8 psig for Unit 1 and 38.4 psig for Unit 2 (experienced during a LOCA). The analysis shows that the peak containment temperature is 333.6°F for Unit 1 and 330.8°F for Unit 2 (experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level of 3672.6 MWt, one containment spray train and one containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the 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~~ WCAP-16996-P-A, Revision 1 (Ref. 4).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -3.48 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the LOCA containment analysis is based on a response time associated with exceeding the containment High-3 pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time of 110.2 seconds (for the limiting case) includes Diesel Generator (DG) startup (for loss of offsite power), sequencing of equipment, containment spray pump startup, and spray line filling (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency 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 9.4.8.
3. UFSAR, Section 6.5.2.
4. ~~10 CFR 50, Appendix K~~. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016
5. UFSAR, Section 6.2.1.1.3.
6. UFSAR, Section 6.2.2.
7. UFSAR, Section 6.2.

APPLICABLE SAFETY ANALYSES (continued)

In addition, the minimum available AF flow and system characteristics are serious considerations in the analysis of a ~~small break Loss Of Coolant Accident (LOCA) and~~ loss of offsite power (Ref. 3).

The AF System design is such that it can perform its function following an FWLB between the main feedwater isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of one AF pump. The AF lines to the SGs are orificed such that sufficient flow is delivered to the non faulted SGs. Reactor trip is assumed to occur when the faulted SG reaches the low-low level setpoint. Sufficient flow would be delivered to the intact steam generators by the other AF pump.

During the loss of all AC power events, the Engineered Safety Feature Actuation System (ESFAS) automatically actuates the AF diesel driven pump and associated controls to ensure an adequate supply to the steam generators during loss of power. Valves which can be manually controlled are provided for each AF line to control the AF flow to each steam generator during loss of all AC power events.

AF flow is also an important consideration in the Steam Generator Tube Rupture (SGTR) event (Ref. 4). For the SGTR event, isolation of AF flow is an important recovery action. This flow path is normally isolated with the motor operated AF013 valves. Prior to isolation with AF013 valves, flow to the ruptured SG is limited by the control function of the AF005 valves.

The AF System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Since the AF005 valve is a backup and diverse method to isolate flow to the faulted SG in a SGTR event, this function does not satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). Administrative requirements for this function are addressed in the Technical Requirements Manual.

Attachment 6a

Braidwood Station, Units 1 and 2
NRC Docket Nos. 50-456 and 50-457

Proposed Technical Specifications Bases Changes (Mark-Up) for Information Only

BASES

APPLICABLE SAFETY ANALYSES (continued)

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 3).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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~~ large break LOCA and the recovery phase of a small break LOCA ~~LOCA~~, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 4) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open with power removed, a contained volume ≥ 31% and ≤ 63% (6995 gallons to 7217 gallons) with a boron concentration ≥ 2200 ppm and ≤ 2400 ppm, and a nitrogen cover pressure ≥ 602 and ≤ 647 psig, must be met.

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.

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 the 10 CFR 50.46 (Ref. 4) acceptance criteria are met ~~peak clad temperature remains below the 10 CFR 50.46 (Ref. 4) limit of 2200°F~~.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator since the static design of the accumulators limits the ways in which the concentration can be changed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.5

Sampling the affected accumulator within 6 hours after a 1% volume increase (nominally 70 gallons or 10% of indicated level) will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration of the accumulator after a 1% volume increase (10% indicated level increase) if the added water inventory is from the Refueling Water Storage Tank (RWST) and the boron concentration of the RWST is ≥ 2200 ppm and ≤ 2400 ppm. With the water contained in the RWST within the boron concentration requirements of the accumulators, any added inventory would not cause the accumulator's boron concentration to exceed the limits of this LCO.

With the only indication available to the operators in the control room being level indication in percent, a required accumulator volume increase of 1% or an increase of 10% of indicated level would require the accumulator to be sampled to verify the accumulator boron concentration is within the limits. The ~~small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the~~ TS minimum and maximum limits are ~~of~~ 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). These volumes are also indicated in the specific tank curves for the SI accumulators. The ~~large break LOCA analysis assumes~~ small break and large break LOCA analyses assume a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits. The 10% indicated level increase is considered a conservative indication for a 70 gallon increase in the accumulator volume requiring an increase in the sampling requirement to verify accumulator boron concentration remains within the specified limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.6

Verification that power is removed from each accumulator isolation valve operator 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 power to the accumulator motor operated isolation valves is removed by opening the motor control center breaker and tagging it out administratively. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. IEEE Standard 279-1971.

2. UFSAR, Chapter 15.

3. UFSAR, Chapter 6.

4. 10 CFR 50.46.

5. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

BASES

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 LOCA:

There is

a. ~~During a small break LOCA maximum fuel element cladding temperature is ≤ 2200°F and during a large break LOCA there must be~~ a high level of probability that maximum fuel element cladding temperature is ≤ 2200°F;

There is a high level of probability that the maximum

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

There is a high level of probability that the maximum

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;

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 temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Ref. 3). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

FSLOCA™

a. For the large break LOCA event, the ~~ASTRUM~~ methodology examines LOOP and no-LOOP cases with a single failure disabling one train of SI pumps. No failure of containment heat removal system is modeled; and

BASES

APPLICABLE SAFETY ANALYSES (continued)

a range of RWST temperatures of 32°F to 120°F is used for the containment spray temperature per approved methods (Ref. 4).

In the ECCS analysis, ~~the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 35°F.~~ If the lower temperature limit is violated, the containment spray further reduces containment pressure. The reduced containment pressure lowers the quality of steam exiting the break thus decreasing the rate which the steam is vented to the containment atmosphere. The decreased rate of steam vented to the containment atmosphere results in a corresponding decrease in the rate the Reactor Coolant System pressure drops and the rate ECCS fluid is injected in the core thereby causing a rise in peak clad temperature. ~~The~~ upper temperature limit of 100°F is used in the ~~small break LOCA analysis and~~ containment OPERABILITY analysis. Exceeding this temperature will result in ~~a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA and~~ 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 limits on RWST level and boron concentration also ensure that the post-LOCA sump pH will be between 8.0 and 11.0. The minimum and maximum pH values are verified for each fuel cycle using conservative maximum and minimum RWST volumes and the maximum and minimum allowed RWST boron concentrations. The LOCA offsite dose analysis assumes a conservatively low sump pH for the re-evolution of iodine from the sump. Ensuring that the minimum sump pH is at least 8.0 protects mechanical components and equipment inside containment from the effects of chloride induced stress corrosion cracking. Ensuring that the maximum sump pH is no greater than 11.0 limits the production of hydrogen due to the corrosion of aluminum and zinc inside containment. Finally, the limits on RWST boron concentration also ensure that the containment spray pH is acceptable. The calculation of the iodine removal effectiveness of the containment spray assumes a conservatively low containment spray pH.

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

The small break and large break LOCA analyses use a range of RWST temperatures of 32°F to 120°F per approved methods (Ref. 4). Exceeding the maximum temperature of this range could result in a higher peak cladding temperature, because there is less heat transfer from the core to the injected water. The containment

BASES

REFERENCES

1. WCAP-13964, Revision 2, "Commonwealth Edison Company, Byron/Braidwood Units 1 & 2, Increased Steam Generator Tube Plugging/Reduced Thermal Design Flow/Positive Moderator Temperature Coefficient Analysis Program, Engineering/Licensing Report," September 1994.

2. UFSAR, Chapter 15.

3. UFSAR, Section 6.2.1.

4. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

Attachment 6b

Byron Station, Units 1 and 2
NRC Docket Nos. 50-454 and 50-455

Proposed Technical Specifications Bases Changes (Mark-Up) for Information Only

BASES

BACKGROUND (continued)

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR (only when PDMS is inoperable), and control rod insertion.

APPLICABLE SAFETY ANALYSES

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

a. During a ~~small break~~ Loss Of Coolant Accident (LOCA) the ~~peak cladding temperature must not exceed 2200°F and during a large break LOCA there must be a high level of probability that the peak cladding temperature does not exceed 2200°F~~ 10 CFR 50.46 acceptance criteria must be met (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 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 $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. ~~Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.~~

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

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

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

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

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

where: F_Q^{RTP} is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For this facility, the actual values of F_Q^{RTP} and K(Z) are given in the COLR; however, F_Q^{RTP} is normally a number on the order of ~~2.50~~ 2.60, and K(Z) is a function that looks like the one provided in Figure B 3.2.1-1.

$F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

When PDMS is inoperable, an $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,

$$F_Q^C(Z) = F_Q^M(Z) * (1.0815)$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

BASES

LCO (continued)

When PDMS is OPERABLE, $F_Q(Z)$ is determined continuously. Then,

$$F_Q^C(Z) = F_Q^M(Z) * U_{FQ}$$

where U_{FQ} is a factor that accounts for measurement uncertainty (Ref. 4) and engineering uncertainty defined in the COLR.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. When PDMS is inoperable, the $F_Q^C(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that ~~precludes peak cladding temperatures above 2200°F during a small break LOCA and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F~~ during a ~~large break~~ LOCA (Ref. 1).

ensure that the 10 CFR 50.46 acceptance criteria are met

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 LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ may produce unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ 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.

This figure is replaced with the figure on the next page.

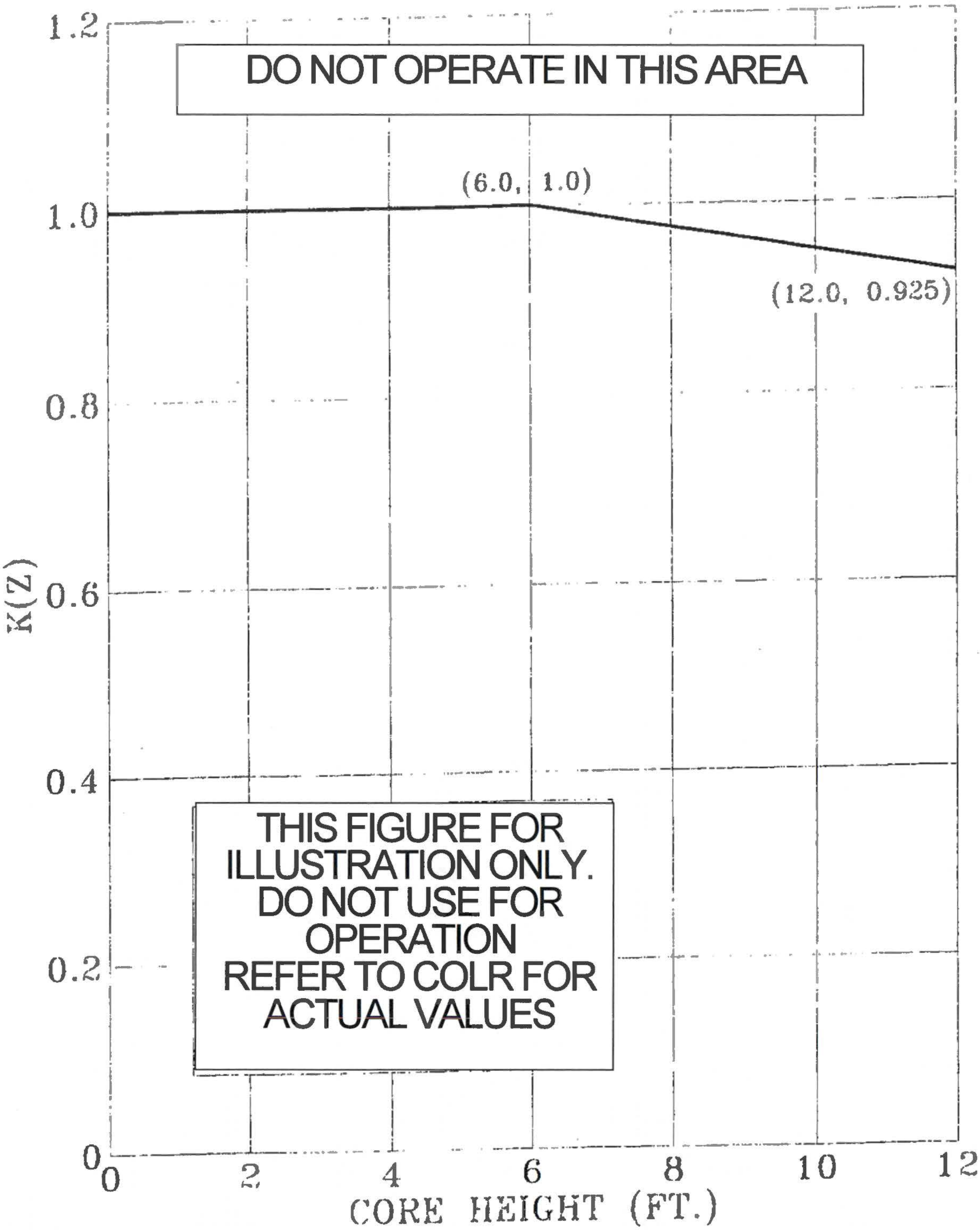

Figure B 3.2.1-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a function of Core Height

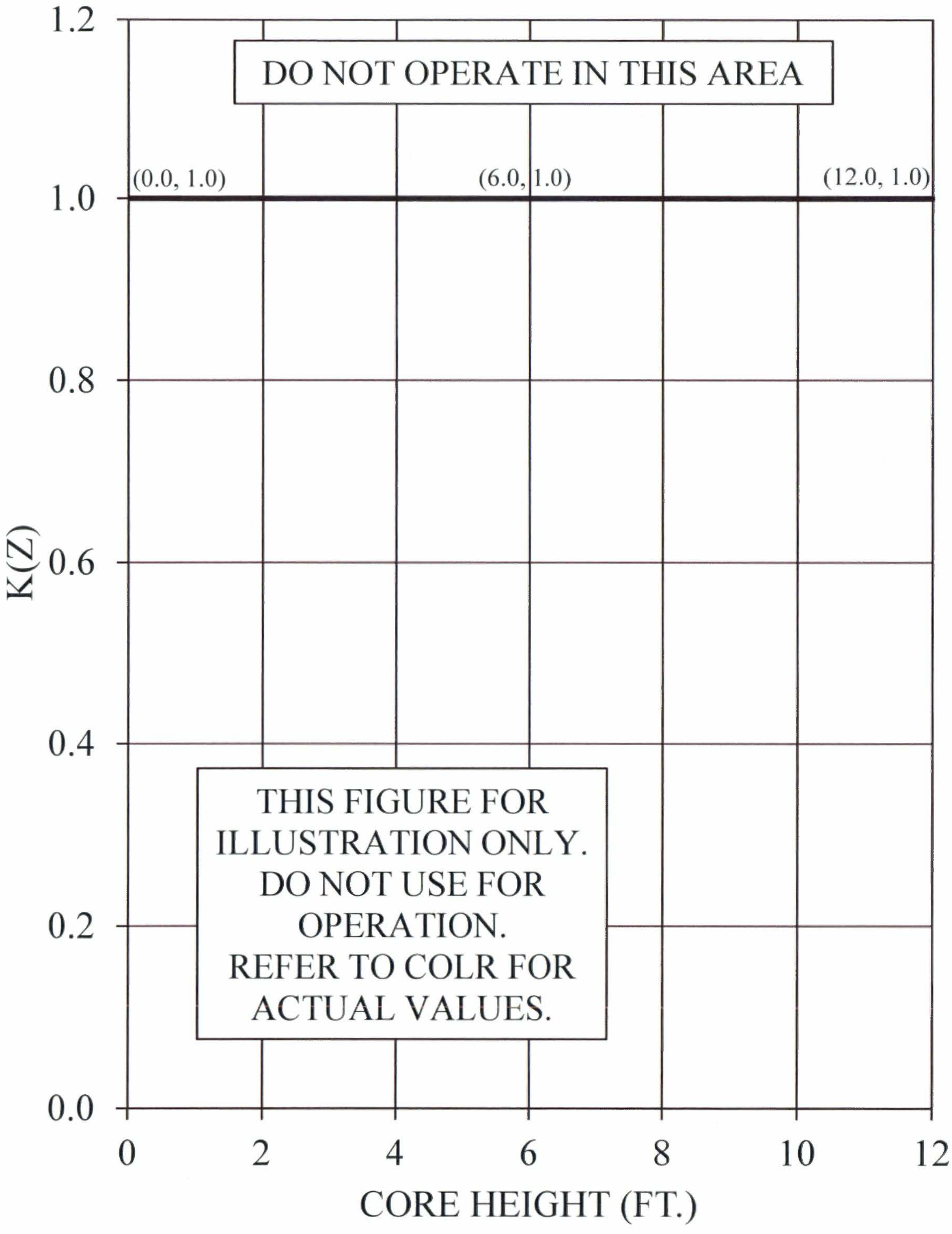

Figure B 3.2.1-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a function of Core Height

BASES

BACKGROUND (continued)

The COLR provides peaking factor limits that ensure that the design criterion for the Departure from Nucleate Boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a ~~small break~~ Loss Of Coolant Accident (LOCA) ~~Peak Cladding Temperature (PCT) must not exceed 2200°F and during a large break LOCA there must be a high level of probability that PCT does not exceed 2200°F~~;

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

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 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 Shutdown Margin with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design criterion is met for normal operation, operational transients, and any transients arising from events of moderate frequency. Refer to the Bases for LCO 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits," for a discussion of the applicable Departure from Nucleate Boiling Ratio (DNBR) limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

~~The LOCA safety analysis indirectly models~~ $\cancel{F^N_{\Delta H}}$ ~~as an input parameter. The Nuclear Heat Flux Hot Channel Factor~~ $\cancel{(F_Q(Z))}$ ~~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 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)

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 3.1.6, "Control Bank Insertion Limits," LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," LCO 3.2.2, LCO 3.2.3, LCO 3.2.4, and LCO 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system when PDMS is inoperable. 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. When PDMS is OPERABLE, $F^N_{\Delta H}$ and $F_Q(Z)$ are determined continuously. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on DNBR and Control Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ 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 a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the plant safety analyses.

The power multiplication factor in this equation provides margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

the 10 CFR 50.46 acceptance criteria (Ref. 3)

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to prevent 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 reactor coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1, A.2, A.3, and A.4

With $F^N_{\Delta H}$ exceeding its limit, Condition A is entered. $F^N_{\Delta H}$ may be restored to within its limits within 4 hours, through, for example, realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. If the value of $F^N_{\Delta H}$ is not restored to within its specified limit, THERMAL POWER must be reduced to < 50% RTP in accordance with Required Action A.1. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses.

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 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, 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. When Power Distribution Monitoring System (PDMS) is OPERABLE, Peak Linear Heat Rate and the linear power along the fuel rod with the highest integrated power are measured continuously.

APPLICABLE SAFETY ANALYSES

Limits on QPTR preclude core power distributions that violate the following fuel design criteria:

a. During a ~~small break~~ Loss Of Coolant Accident (LOCA) the ~~Peak Cladding Temperature (PCT) must not exceed 2200°F and during a large break LOCA there must be a high level of probability that the PCT does not exceed 2200°F~~ (Ref. 1);

10 CFR 50.46 acceptance criteria must be met

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 2); and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure-Low is a primary actuation signal for ~~small~~ Loss Of Coolant Accidents (LOCAs) and a backup actuation signal for Steam Line Breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, 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 when the unit status is within the Applicability. 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.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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:

compliance with the 10 CFR 50.46 acceptance criteria (Ref. 17)

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

2. Boration to ensure recovery and maintenance of SDM.

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 Isolation;
- Reactor Trip;
- Turbine Trip;
- Feedwater Isolation;
- Start of Auxiliary Feedwater (AF) pumps;
- Control room ventilation isolation; and
- Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) pump suction to containment sump.

BASES

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 7.
3. UFSAR, Chapter 15.
4. IEEE-279-1971.
5. Technical Requirements Manual.
6. WCAP-12523, "RTS/ESFAS Setpoint Methodology Study," October 1990.
7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
8. WCAP-13632 Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," August 1995.
9. UFSAR, Section 7.3.
10. WCAP-12583, "Westinghouse Setpoint Methodology For Protection Systems, Byron/Braidwood Stations," May 1990.
11. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.
12. WCAP-13877, Revision 2-P, "Reliability Assessment of Westinghouse Type AR Relays Used as SSPS Slave Relays," October 1999.
13. WCAP-13878-P, Revision 2, "Reliability Assessment of Potter & Brumfield MDR Series Relays," October 1999.
14. WCAP-13900, Revision 0, "Extension of Slave Relay Surveillance Test Intervals," April 1994.
15. WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.
16. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2000.
17. 10 CFR 50.46

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

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.

initiate

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.

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