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JUL 13 2007

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop OP1-17
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED LICENSE AMENDMENT NO. 285
FOR UNIT 1 OPERATING LICENSE NO. NPF-14
AND PROPOSED LICENSE AMENDMENT NO. 253
FOR UNIT 2 OPERATING LICENSE NO. NPF-22
EXTENDED POWER UPRATE APPLICATION
AREVA AUDIT FOLLOWUP RESPONSES TO
REQUEST FOR ADDITIONAL INFORMATION
PLA-6243

Docket Nos. 50-387
and 50-388

References: 1) *PPL Letter PLA-6076, B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 For Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Constant Pressure Power Uprate," dated October 11, 2006.*

2) *PPL Letter PLA-6230, B. T. McKinney (PPL) to USNRC, "Proposed License Amendment Numbers 285 For Unit 1 Operating License No. NPF-14 and 253 for Unit 2 Operating License No. NPF-22 Extended Power Uprate Application Supplement to Request for Additional Information Responses," dated June 27, 2007.*

Pursuant to 10 CFR 50.90, PPL Susquehanna LLC (PPL) requested in Reference 1 approval of amendments to the Susquehanna Steam Electric Station (SSES) Unit 1 and Unit 2 Operating Licenses (OLs) and Technical Specifications (TSs) to increase the maximum power level authorized from 3489 Megawatts Thermal (MWt) to 3952 MWt, an approximate 13% increase in thermal power. The proposed Constant Pressure Power Uprate (CPPU) represents an increase of approximately 20% above the Original Licensed Thermal Power (OLTP).

The purpose of this letter is to supplement the responses to NRC questions identified during the audit at AREVA on June 5-7, 2007 and contained in Reference 2. The Attachments contain the PPL supplemental responses.

Attachment 1 contains AREVA NP, Inc. proprietary information. As such, AREVA NP, Inc. requests that Attachment 1 be withheld from public disclosure in accordance with 10 CFR 2.390 (a)4 and 9.17 (a)4. Affidavits supporting this request are contained in Attachment 3. Attachment 2 contains a non-proprietary version of Attachment 1.

A001

NRR

There are no regulatory commitments associated with this submittal.

PPL has reviewed the "No Significant Hazards Consideration" and the "Environmental Consideration" submitted with Reference 1 relative to the Enclosure. We have determined that there are no changes required to either of these documents.

If you have any questions or require additional information, please contact Mr. Michael H. Crowthers at (610) 774-7766.

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

Executed on: 7-13-07

B. T. McKinney

B. T. McKinney

Attachment 1: Proprietary Version of the Request for Additional Information Responses
Attachment 2: Non-Proprietary Version of the Request for Additional Information Responses
Attachment 3: AREVA NP, Inc. Affidavits

Copy: NRC Region I
Mr. A. J. Blamey, NRC Sr. Resident Inspector
Mr. R. V. Guzman, NRC Sr. Project Manager
Mr. R. R. Janati, DEP/BRP

Attachment 2 to PLA-6243
Non-Proprietary Version of the Request for Additional Information Responses

AREVA Responses to
NRC Audit Action Items

Information requested to supplement response to RAI 3, regarding the Core Operating Limits Report Methods:

1. *Please submit a copy of the presentation "Computer Codes Used for ATRIUM-10 Susquehanna CPPU Analyses," presented to the NRC staff during the audit.*

   A copy of the requested presentation is provided in Attachment 1A.

Information requested to supplement response to RAI 4, regarding the applicability of CASMO4 and MICROBURN-B2 for SSES CPPU operation:

1. *The data provided in support of the staff's review of extended power uprate at Tennessee Valley Authority's Browns Ferry Nuclear Units 2 and 3 provides necessary information regarding the continued applicability of CASMO4 and MICROBURN-B2 for EPU operation at Brown's Ferry. Similar information is requested for the SSES CPPU review. Please submit SSES-specific information comparable to the responses to BFN 2 and 3 EPU RAIs SRXB-A.26, 27, 30, 34, and 35, providing similar documentation of the applicability of CASMO4 and MICROBURN-B2 for uprate operations at SSES.*

   The SSES-specific information comparable to the responses to BFN 2 and 3 EPU RAIs SRXB-A.26, 27, 30, 34, and 35 was included in Reference 2.

2. *Submit the operating conditions presented in tabular format in ANP-2536(P) Tables 3.4 and 3.5.*

   A copy of the requested information is provided in Attachment 1B.

3. *Regarding design criterion section 3.1.7 of ANF-89-98(P)(A), and in the context of the letter dated January 9, 1990, from R. A. Copeland, Advanced Nuclear Fuels Corporation to R. C. Jones, NRC, discuss the continued applicability of the model discussed in the January 9, 1990, letter to the validation of criterion 3.1.7 with respect to the thermal and hydraulic design at SSES Units 1 and 2.*

   [

* ATRIUM is a trademark of AREVA NP.

]

Information requested to supplement response to RAI 8, regarding the statistical rigor of the safety limit minimum critical power ratio.

1. *Regarding information discussed in EMF 2393, Page 2-6, the exit quality correlation instructs the analyst to adjust the feedwater flow if the exit quality is greater than 1. Why would the exit quality be greater than 1?*

[

]

2. *The staff noted that the SLMCPR convergence is based on a certain type of power, requiring a treatment of the power level to obtain proper agreement between the analytical MCPR and the established SLMCPR. Is the SLMCPR adjusted if the treatment discussed above causes the analyzed core power to exceed the rated core power? Provide an example to supplement your discussion.*

[

]

3. *Discuss how departures from the typical uncertainties discussed in ANF-524(P)(A) will affect the safety limit.*

ANF-524(P)(A) (Reference A.5) presents typical uncertainties in Table 5.1. For each unit and cycle, the uncertainties that are appropriate for the cycle are determined and used in the cycle-specific safety limit MCPR analysis. Reactor system uncertainties are provided by the utility. Fuel-specific uncertainties related to power monitoring are based on the neutron physics software used and its ability to represent the measured power. Fuel-specific uncertainties related to critical power correlations are based on the particular approved critical power correlation. Fuel-specific uncertainties due to effects of channel bow are based on measurements of channel bow. Fuel-specific uncertainties associated with flow are based on assembly flow tests.

The general relationship is that the number of fuel rods predicted to be in boiling transition will increase when the uncertainties increase. However, an increase in the number of rods predicted to be in boiling transition will not result in an increase in the safety limit MCPR if the fraction of rods predicted to be in boiling transition remains less than 0.1%. [

]

4. *Tabulate the uncertainties used in the SSES uprate analysis, compare to pre-EPU analysis.*

The uncertainties used in the Susquehanna pre-EPU and SSES uprate safety limit MCPR analyses are compared in the following two tables. As summarized below, there were no changes in uncertainties between the Susquehanna pre-EPU and SSES uprate safety limit MCPR analyses.

Table A.1 Comparison of Uncertainties Used in Pre-EPU and SSES Uprate SLMCPR

Parameter	Pre-EPU Analysis	SSES Uprate Analysis
Reactor System Uncertainties		
Feedwater flow rate	1.76%	1.76%
Feedwater temperature	0.76%	0.76%
Core pressure	0.5%	0.5%
Total core flow rate	2.5%	2.5%
Fuel-Related Uncertainties		
Radial power	[	]
Assembly flow rate	[	]
Local power	[	]
SPCB additive constant	[	]
Channel bow	See Table 2	See Table 2

[

]

[

]

5. *Explain the reductions in SLMCPR due to channel replacement and assumed uncertainty.*

[

]

6. *Provide a comparison of assembly powers to show where, in the range of available bundle powers, changes in the power distribution occur to support the EPU implementation. This comparison should be provided in the context of a cycle-limiting SLMCPR calculation statepoint.*

Figure A.1 provides the requested comparison. [

]

7. *Discuss the cycle step-through in terms of the calculation. Include a discussion of how many steps are analyzed. Discuss why, sometimes, an analyzed step may fail the safety limit criterion and why, in that regard, the safety limit analysis remains valid.*

[

]

During the audit, the staff also discussed stability analyses with AREVA. The following supplemental information is requested to support the staff's conclusions regarding coupled neutronic and thermal-hydraulic stability:

1. *The NRC staff observed that the planned depletion, from an end-of-cycle stability standpoint, is slightly limiting when compared to other possible operating strategies. Provide qualitative discussion about possible causes for this, including the fuel burning strategies considered.*

   AREVA calculated decay ratios for backup stability protection (BSP) for Susquehanna extended power uprate conditions using cycle depletions based on a MELLLA operating domain and an expanded flow window MELLLA domain. [

   ] BWR core stability is influenced by a combination of factors including core power distribution, void coefficient of reactivity, and boiling boundary. For the MELLLA depletion case, less top-peaked core power distributions and lower boiling boundaries led to higher decay ratios. (The void coefficient of reactivity is very similar between the two cases).

   The expanded flow window MELLLA domain allows lower core flow rates to be used early in the cycle which results in a more bottom-peaked depletion. This leads to a more top-peaked core axial power distribution and higher boiling boundaries near the end-of-cycle when global decay ratio becomes most limiting.

   The MELLLA and the expanded flow domain cases reviewed by the NRC were created to provide a representation of the stability conditions that can be expected under extended power uprate conditions. [

   ]

2. *Provide a comparison of the DIVOM curve when analyzed for EPU and pre-EPU operation and discuss why any changes occur.*

   The purpose of the Option III system is to protect the Technical Specification MCPR Safety Limit if a diverging oscillation event were to be experienced. Cycle-specific DIVOM and setpoint calculations are performed to ensure that cycle-specific variations in parameters that have the potential to impact the protection of the MCPR Safety Limit are explicitly included. Any increase or decrease in

DIVOM value for a specific operating cycle does not mean that the core is more or less stable or more or less likely to experience an oscillation. The change in DIVOM slope only means that the CPR response for a given oscillation magnitude has changed. Since the cycle-specific DIVOM value is explicitly included in the setpoint calculation, any differences due to the EPU power shapes or core loading will be explicitly included in the OPRM setpoint.

Results of the cycle-specific DIVOM calculation for the Susquehanna equilibrium EPU cycle are shown in Figure A.2. Also included in this figure are the results of the previous two licensed Susquehanna cycles (Unit 1 Cycle 15 and Unit 2 Cycle 14). Both cycles are operating at current licensed thermal power (3489 MWt). [

]

[

]

Figure A.2 Comparison of Susquehanna EPU and Pre-EPU DIVOM Data

3. *Qualitatively discuss the stability effects of a fuel assembly misloading.*

A fuel assembly misload can result in a local region of the core operating at a higher power than designed. For the most severe misload configuration, the misloaded assembly could experience operating powers [ ] higher than expected. Due to the localized nature of the event, the power differential decreases rapidly with the neighboring assemblies experiencing elevated powers on the order of [

]

In the case of a very high bundle power characterizing the severe cases, the main concern is unstable single channel oscillations which may occur following a two-pump-trip. The issue of single channel oscillations is well considered in the AREVA DIVOM methodology where calculations preclude channel instabilities [

]

References

A.1 ANP-2536(P) Revision 0, *Susquehanna Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM™-10 Fuel Assemblies for Extended Power Uprate*, AREVA NP, September 2006.

A.2 Letter, R. A. Copeland (AREVA) to R. C. Jones (NRC), "Explicit Modeling of BWR Water Rods in XCOBRA," RAC:002:90, January 9, 1990.

A.3 Letter, R. C. Jones (NRC) to R. Copeland (AREVA), (response to RAC:002:90 listed above as A.2), February 1, 1990.

A.4 EMF-2392(P) Revision 6, *SAFLIM2 Theory, Programmer's and User's Manual*, Framatome ANP, March 2005.

A.5 ANF-524(P)(A) Revision 2 and Supplements 1 and 2, *ANF Critical Power Methodology for Boiling Water Reactors*, Advanced Nuclear Fuels Corporation, November 1990.

Attachment 1A
Information Requested in RAI 3 Item 1

The presentation "Computer Codes Used for ATRIUM™-10 Susquehanna CPPU Analyses," which was presented to the NRC during the audit is attached in response to Item 1 NRC request.

AREVA

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Computer Codes Used for ATRIUM™-10 Susquehanna CPPU Analyses

Michael E. Garrett
Paul D. Wimpy
Daniel R. Tinkler

Richland, Wash.
June 5, 2007

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Reactor Core and Fuel Performance

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Presentation Topics

> Discuss response for RAI 3 for Susquehanna CPPU PUSAR

3. (2.8.1 - Fuel System Design) The staff is unable to determine from TS 5.6.5.b, "Core Operating Limits Report," and PUSAR Table 1-1, which methods specified perform which function. The staff is also unable to determine whether each specified method is being used in a manner consistent with its NRC approval. Supplement both the COLR references list and Table 1-1 with a specific description of the function of each method and explaining why, in some cases, as many as six codes are required to perform a task or group of tasks.

Presentation Topics

> Table 1-1 of the PUSAR identifies tasks and related computer codes used as part of the Susquehanna CPPU project

> Table 1-1 tasks involving ATRIUM-10 fuel and AREVA methods include the following
- Reactor core and fuel performance
- Safety limit MCPR
- Transient analyses
- LOCA-ECCS
- Appendix R – fire protection
- Reactor core stability

Presentation Topics

> For Table 1-1 tasks related to AREVA methods
- Task description
  - Purpose of task
  - Analyses performed
- Computer codes used in analyses
  - Approved topical report
- Calculation description
  - Flow chart showing codes used and major inputs and outputs from each code

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Reactor Core and Fuel Performance
Task Description

> Purpose
  - Model steady-state behavior of core and fuel including calculation of margin to thermal limits for MCPR, MAPLHGR, and LHGR

> Analyses Performed
  - Used for reload core design. Determine core loading pattern, control rod patterns for cycle step-through, evaluation of cold shutdown margin, determination of margins to thermal limits, and quasi-steady-state licensing analyses

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Neutronic Analysis Methodology
Major Computer Codes

Code	Use
CASMO-4	Performs fuel assembly burnup calculations and calculates nuclear data for MICROBURN-B2
MICROBURN-B2	Performs 3-dimensional steady-state reactor core neutronic analyses for assessing impact on thermal limits during localized and quasi-steady-state events

AREVA

Neutronic Analysis Methodology
CASMO-4 Computer Code

Description	Multi-group, 2-dimensional transport theory code
Use	Performs fuel lattice burnup calculations and generates nuclear data for use in MICROBURN-B2
Documentation	EMF-2158(P)(A) Rev 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*, October 1999
Acceptability	The safety evaluation by the NRC for the topical report EMF-2158(P)(A) approves the CASMO-4/ MICROBURN-B2 methodology for licensing applications

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Neutronic Analysis Methodology
MICROBURN-B2 Computer Code

Description	A 3-dimensional, two group, diffusion theory code incorporating microscopic depletion and pin power reconstruction
Use	Performs 3-dimensional steady-state reactor core neutronic analyses for assessing impact on thermal limits during localized and quasi-steady-state events
Documentation	EMF-2158(P)(A) Rev 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*, October 1999
Acceptability	The safety evaluation by the NRC for the topical report EMF-2158(P)(A) approves the CASMO-4/ MICROBURN-B2 methodology for licensing applications

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Neutronic Topical Report SER Restrictions

> EMF-2158(P)(A) **(CASMO-4/MICROBURN-B2)**

1. The CASMO-4/MICROBURN-B2 code system shall be applied in a manner that predicted results are within the range of the validation criteria (Tables 2.1 and 2.2) and measurement uncertainties (Table 2.3) presented in EMF-2158(P)
2. The CASMO-4/MICROBURN-B2 code system shall be validated for analyses of any new fuel design which departs from current orthogonal lattice designs and/or exceed gadolinia and U-235 enrichment limits
3. The CASMO-4/MICROBURN-B2 code system shall only be used for BWR licensing analyses and BWR core monitoring applications
4. The review of the CASMO-4/MICROBURN-B2 code system should not be construed as a generic review of the CASMO-4 or MICROBURN-B2 computer codes
5. The CASMO-4/MICROBURN-B2 code system is approved as a replacement for the CASMO-3/MICROBURN-B code system used in NRC-approved AREVA BWR licensing methodology and in AREVA BWR core monitoring applications. Such replacements shall be evaluated to ensure that each affected methodology continues to comply with its SER restrictions and/or conditions
6. AREVA shall notify any customer who proposes to use the CASMO-4/ MICROBURN-B2 code system independent of any AREVA fuel contract that conditions 1–4 above must be met. AREVA's notification shall provide positive evidence to the NRC that each customer has been informed by AREVA of the applicable conditions for using the code system

Conformance to No. 1 is addressed through benchmarking the core system against previous cycles.

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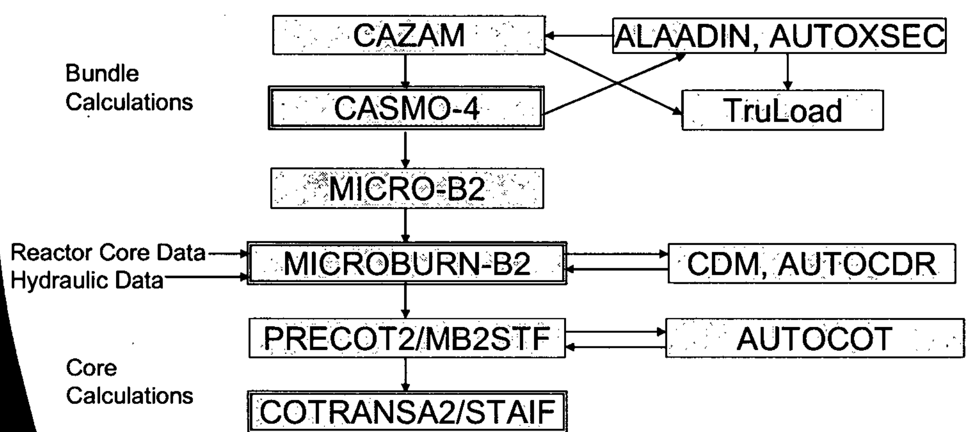
Neutronic Code Input Flow
CASMO-4/MICROBURN-B2



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Neutronic Safety Analysis

> Approved methods generally defined in XN-NF-80-19(P)(A) or subsequently submitted and approved Topical Reports
- XN-NF-80-19(P)(A) Vol 1 Supplements 1 and 2, and Vol 4

(No specific SER restrictions listed)

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Neutronic Safety Analysis Methodology
Cycle-Specific Analyses

> Analyzed with CASMO-4/MICROBURN-B2
- Cold shutdown margin
- Standby boron liquid control
- Control rod withdrawal error
- Loss of feedwater heating
- Control rod drop accident
- Fuel assembly mislocation *
- Fuel assembly misorientation *
- Core flow increase event ($LHGR_f$)
- Neutronic input for SLMCPR, $MCPR_f$, transient analyses, LOCA
- Neutronic input for POWERPLEX®-III CMSS input deck preparation

---

* Cycle-specific confirmation that analysis remains bounding

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Neutronic Safety Analysis Methodology
Cycle-Specific Analyses

> Other Neutronic analyses
- Fuel storage criticality * (CASMO, KENO V.a)
- Fuel handling accident * (ORIGEN-ARP)
- Reactor core stability (STAIF)

* Cycle-specific confirmation that analysis remains bounding

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Neutronic Safety Analyses

> **Cold shutdown margin (CSDM)** – Evaluation of core reactivity at cold conditions with strongest control rod withdrawn, all other rods fully inserted. *(Evaluated on a cycle-specific basis.)*

> **Standby boron liquid control (SLC)** - Reactivity control by injection of boron in moderator (typically 660 ppm B). Must be able to render the core subcritical in event control rods become inoperable. *(This event is assessed on a cycle-specific basis.)*

> **Control rod withdrawal error (CRWE)** - Inadvertent withdrawal of a control rod at power until it is stopped by the Rod Block Monitor (RBM) on BWR/3-5 plants or the Rod Withdrawal Limiter (RWL) on BWR/6 plants (ΔCPR). *(Generic Topical approved for BWR/6 plants.)*

> **Loss of feedwater heating (LFWH)** - A loss of feedwater heating capability due to the closing of a steam extraction line or the bypassing of feedwater flow around a heater, causing insertion of reduced temperature water into the core at power, i.e., reactivity insertion (ΔCPR). *(Generic Topical approved.)*

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Neutronic Safety Analyses
(Continued)

> **Control rod drop accident (CRDA) -** During startup control rod withdrawals a control rod becomes decoupled from its drive, sticks, then falls to the new drive position. High rod worths avoided primarily by implementation of BPWS (banked position withdrawal sequence). This is an accident - current limit is 280 cal/g (max) deposited energy, rods >170 cal/g (offsite dose). *(This event is assessed on a cycle-specific basis.)*

> **Fuel misload -** Inadvertent misplacement of a fuel bundle in either a wrong core location or misrotated in a cell. *Per AREVA methodology, it is an accident (10 CFR 100 offsite dose criteria apply). (Bounding analyses have been performed for both the fuel misload and misorientation events.)*

> **Stability -** Avoidance of core power oscillations by assessment of core loading (decay ratios) with the STAIF code. BWR core stability sensitive to fuel rod thermal time constant, void coefficient, bundle 2- to 1-phase pressure drop, core power distribution, and operating point on the P/F map. *(Methodology capable of supporting interim corrective actions (ICAs), exclusion Z-region, and long-term solutions, e.g., 1D, E1A, III.)*

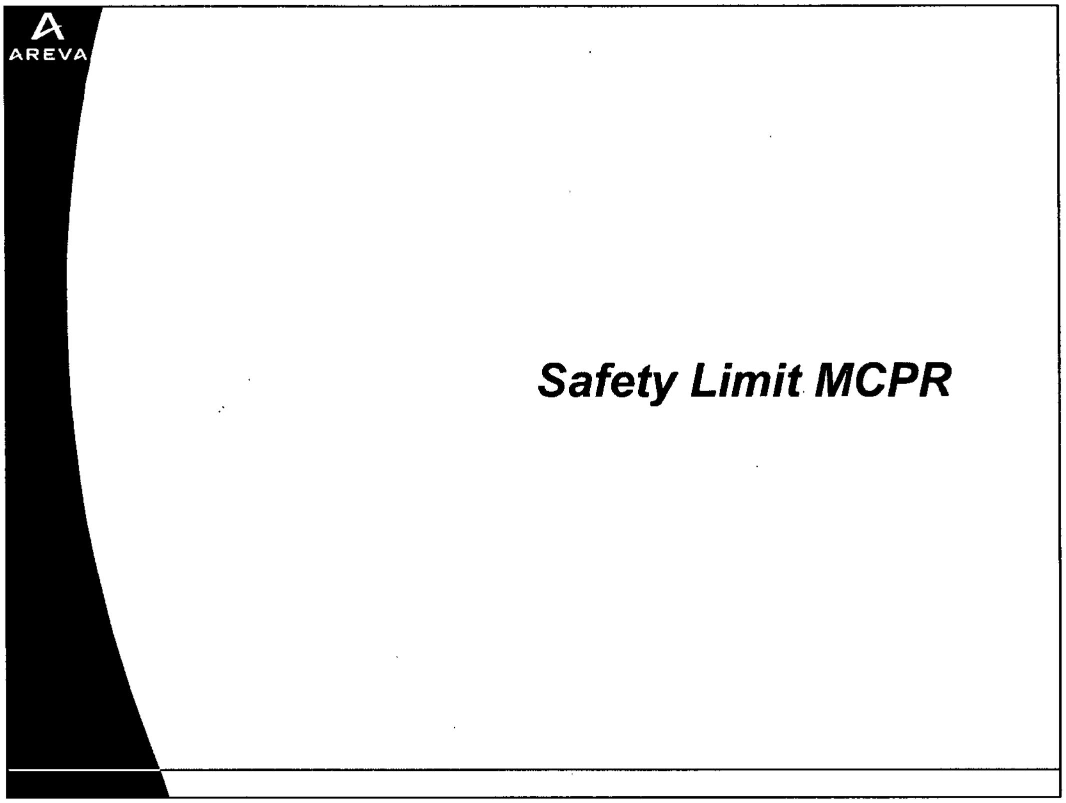
AREVA

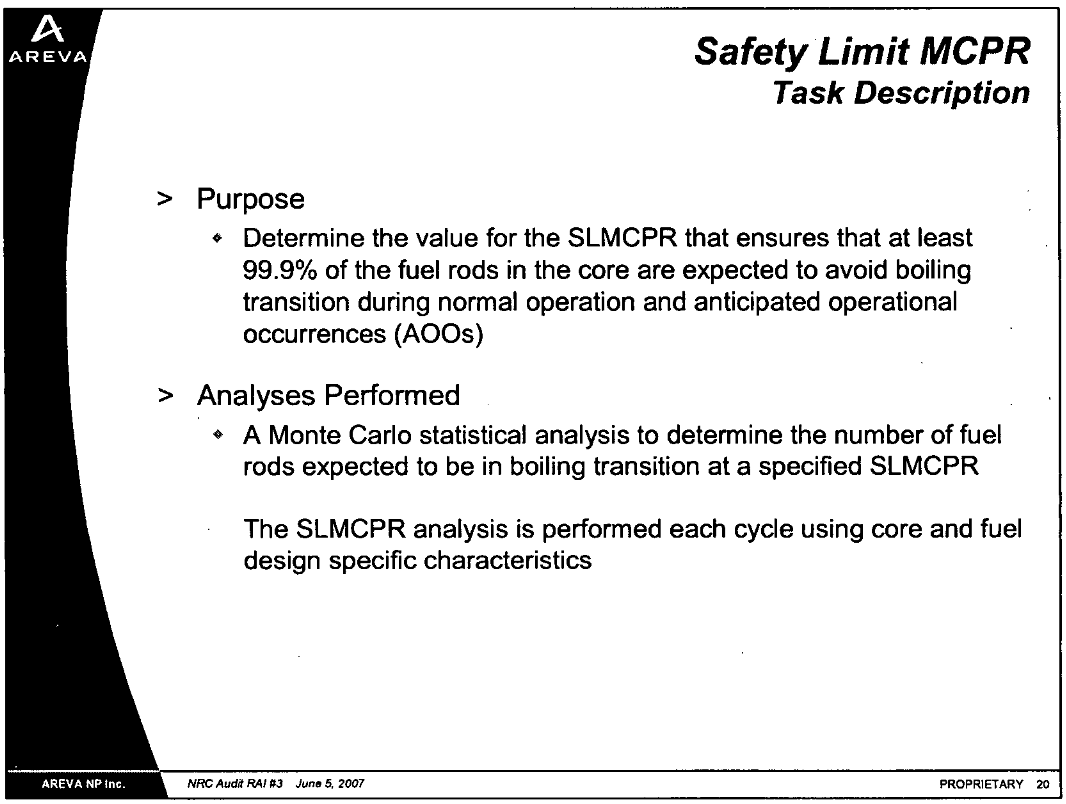
Topical Report SER Restrictions

> ANF-1358(P)(A) Rev 3 (**Loss of Feedwater Heating**)

1. The methodology applies to BWR/3–6 plants and the fuel types which were part of the database (GNF-8x8, 9/9B and 11; ANF-8x8 and 9/9; and ATRIUM-9B and 10) , provided that the exposure, the ratio of rated power and rated steam generation rate, rated feedwater temperature, and change in feedwater temperature are within the range covered by the data points presented in ANF-1358(P)(A) Rev 3.
2. To confirm applicability of the correlation to fuel types outside the database, AREVA will perform additional calculations using the methodology, as described in Section 3.0 of the SER. In addition, AREVA calculations will be consistent with the methodology described in EMF-2158(P)(A) Rev 0 and comply with the guidelines and conditions identified in the associated NRC SER.
3. The methodology applies only to the MCPR operating limit and the LHGR for the LFWH event.

Conformance addressed through engineering guideline requirements for analysis.


AREVA
Safety Limit MCPR


AREVA
Safety Limit MCPR
Task Description
> Purpose
Determine the value for the SLMCPR that ensures that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition during normal operation and anticipated operational occurrences (AOOs)
> Analyses Performed
A Monte Carlo statistical analysis to determine the number of fuel rods expected to be in boiling transition at a specified SLMCPR
The SLMCPR analysis is performed each cycle using core and fuel design specific characteristics
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Safety Limit MCPR
Computer Codes

Code	Use
MICROBURN-B2	Provides radial peaking factor and exposure for each bundle in the core and the core average axial power shape
CASMO-4	Provides local peaking factor distribution for each fuel type
XCOBRA	Provides hydraulic demand curves for each fuel type
SAFLIM2	Calculates the fraction of rods in boiling transition (BT) for a specified core MCPR and exposure

AREVA

Safety Limit MCPR
XCOBRA Computer Code

Description	XCOBRA predicts the steady-state thermal-hydraulic performance of BWR cores at various operating conditions and power distributions
Use	Evaluate core thermal-hydraulic performance (core pressure drop, core flow distribution, bypass flow, MCPR, etc.) and fuel assembly hydraulic demand curves (HDC)
Acceptability	XN-NF-80-19(P)(A) Vol 3 Rev 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description,* January 1987
	NRC accepts the use of XCOBRA based on the similarity of the computational models to those used in the approved code XCOBRA-T
	SER restrictions are implemented in engineering analysis guidelines

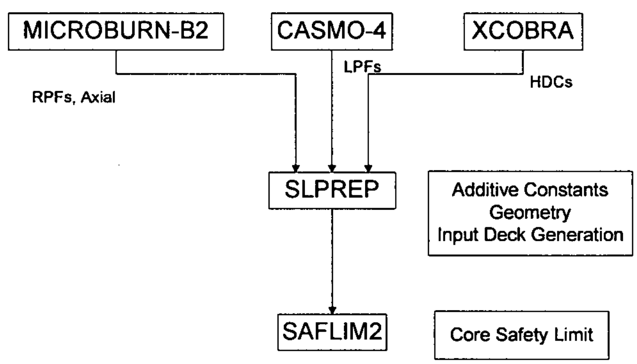
AREVA

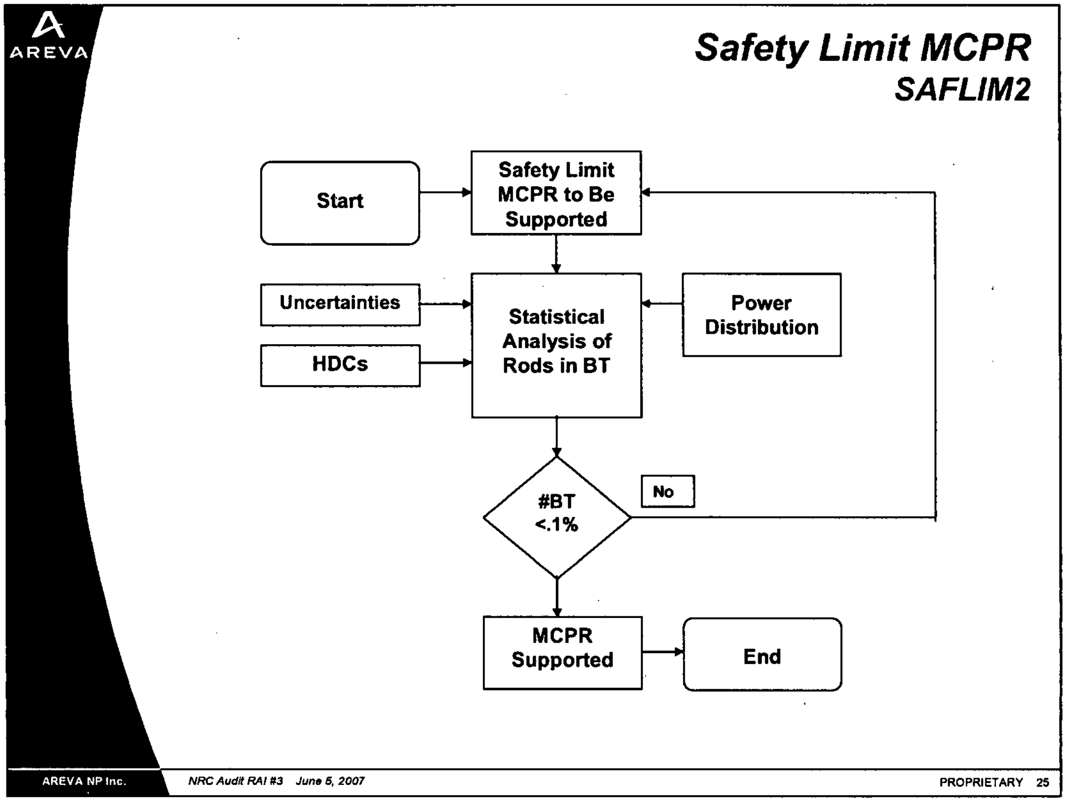
Safety Limit MCPR
SAFLIM2 Computer Code

Description	SAFLIM2 is a computer code used to determine the number of fuel rods in the core expected to experience boiling transition for a specified core MCPR
Use	Evaluate the safety limit MCPR (SLMCPR) which ensures that at least 99.9% of the fuel rods in the core are expected to have a MCPR value greater than 1.0
Acceptability	ANF-524(P)(A) Rev 2 and Supplements, *ANF Critical Power Methodology for Boiling Water Reactors,* November 1990 The safety evaluation by the NRC for the topical report approves the SAFLIM2 methodology for licensing applications SER restrictions are implemented in engineering analysis guidelines and automation tools

AREVA

Safety Limit MCPR
Calculation Process




AREVA
Safety Limit MCPR
SAFLIM2
Start
Safety Limit MCPR to Be Supported
Uncertainties
HDCs
Statistical Analysis of Rods in BT
Power Distribution
#BT <.1%
No
MCPR Supported
End
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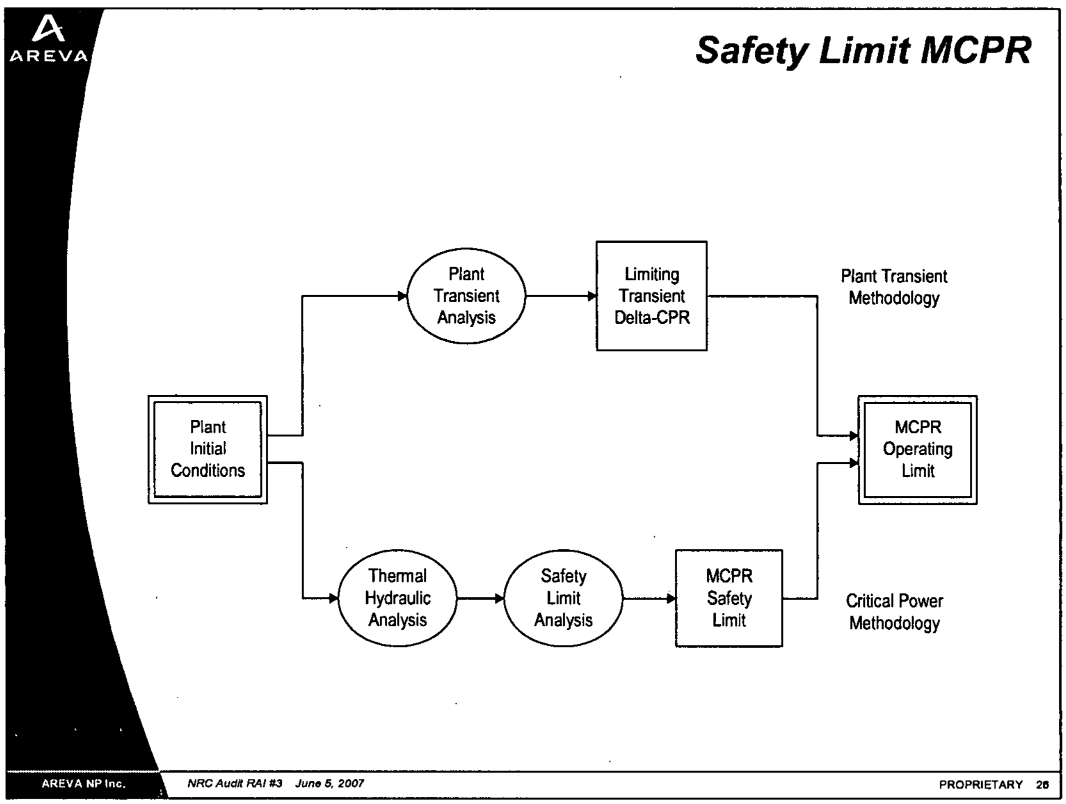
AREVA
Safety Limit MCPR
Plant Initial Conditions
Plant Transient Analysis
Limiting Transient Delta-CPR
Plant Transient Methodology
MCPR Operating Limit
Thermal Hydraulic Analysis
Safety Limit Analysis
MCPR Safety Limit
Critical Power Methodology
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Transient Analysis

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Transient Analysis
Task Description

> Purpose
- Calculate the ΔCPR for limiting anticipated operation occurrences (AOOs) for use in determining the power dependent operating limit MCPR
- Transient analysis methodology is used for "core-wide" events that do not involve local power/reactivity excursions

> Analyses Performed
- Generator load rejection without bypass
- Turbine trip without bypass
- Feedwater controller failure
- Pressure regulator failure – down scale
- MSIV closure
- Inadvertent HPCI startup
- Loss of feedwater flow
- Recirculation flow controller failure

AREVA

Transient Analyses
Major Computer Codes

Code	Use
RODEX2	Gap conductance for core and hot channel
XCOBRA	Hot channel active flow
COTRANSA2	System and core average transient response
XCOBRA-T	ΔCPR calculation
MICROBURN-B2	Cross-sections at state point of interest

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Transient Analysis
COTRANSA2 Computer Code

Description	COTRANSA2 is a BWR system transient analysis code with models representing the reactor core, reactor vessel, steam lines, recirculation loops, and control systems
Use	Evaluate key reactor system parameters such as power, flow, pressure, and quality during core-wide BWR transient events Provide boundary conditions for hot channel analyses performed to calculate ΔCPR
Acceptability	The safety evaluation by the NRC for the topical report ANF-913(P)(A) approves COTRANSA2 for licensing applications SER restrictions are implemented in engineering analysis guidelines and automation tools

AREVA

Transient Analysis
XCOBRA-T Computer Code

Description	XCOBRA-T predicts the transient-thermal hydraulic performance of BWR cores during postulated system transients
Use	Evaluate the transient thermal-hydraulic response of individual fuel assemblies in the core during transient events Evaluate the ΔCPR for the limiting fuel assemblies in the core during system transients
Acceptability	The safety evaluation by the NRC for the topical report XN-NF-84-105(P)(A) approves XCOBRA-T for licensing applications SER restrictions are implemented in engineering analysis guidelines or through computer code controls (defaults, override warning messages)

AREVA

Transient Analysis
RODEX2 Computer Code

Description	Predicts the thermal and mechanical performance of BWR fuel rods as a function of power history
Use	Used to provide initial conditions for transient and accident analyses (hot channel and core average fuel rod gap conductance)
Acceptability	The safety evaluation by the NRC for XN-NF-81-58(P)(A) Rev 2 and Supplements approves RODEX2 for licensing applications SER restrictions are implemented in engineering analysis guidelines, automation tools, or through computer code controls (defaults, override warning messages)

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Transient Analysis
Calculation Process

- Use MICROBURN-B2 in conjunction with PRECOT2 to build the appropriate core cross-section deck (COTRAN deck)
- Use COTRANSA2 to read the unnormalized COTRAN input deck and produce a normalized cross-section set
- Use COTRANSA2 to analyze the transient and produce a system boundary conditions output file
- Use XCOBRA to generate hot channel flow rates as a function of hot channel radial peaking factors
- Read the COTRANSA2 boundary conditions output file using XCOBRA-T and perform an iterative MDNBR analysis to calculate the ΔCPR

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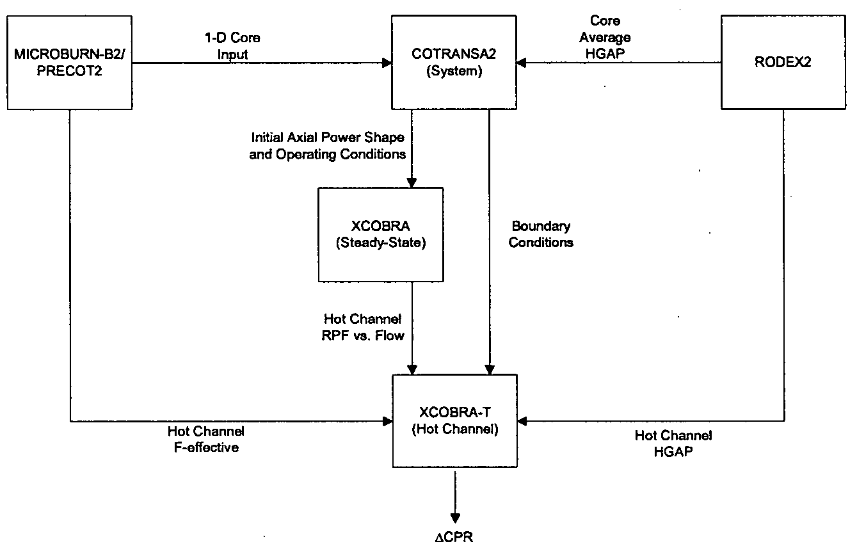
Transient Analysis



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Transient Analysis
Calculation Process

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Transient Analysis
Calculation Process *(Continued)*

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Transient Analysis

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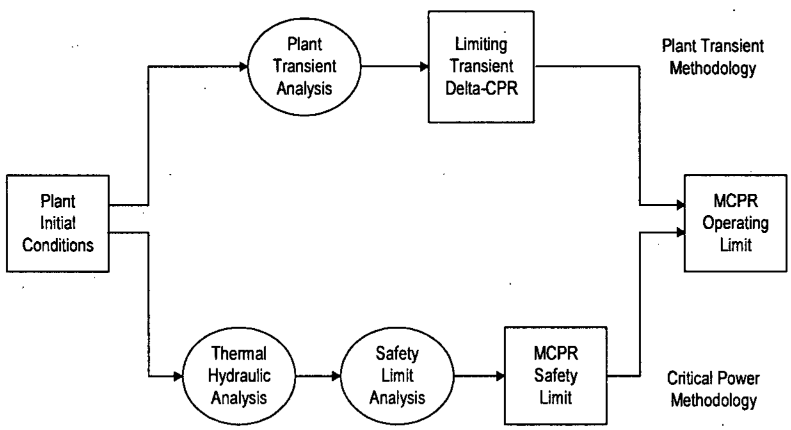
Thermal Analysis



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LOCA-ECCS

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LOCA-ECCS
Task Description

- Purpose
  - Demonstrate that fuel MAPLHGR limits ensure that all 10 CFR 50.46 acceptance criteria are met during a postulated LOCA
- Analyses Performed
  - LOCA break spectrum analyses are performed to determine characteristics of the limiting LOCA event – single failure, break location, break size, axial power shape
  - LOCA MAPLHGR analyses are performed to determine the PCT and metal-water reaction (cladding oxidation) for each nuclear fuel design in the core

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LOCA-ECCS
Major Computer Codes

Code	Purpose
RODEX2	Fuel rod performance code used to predict the thermal-mechanical behavior of BWR fuel rods as a function of exposure
RELAX	BWR system analysis code used to calculate the reactor system and hot channel response during the blowdown, refill, and reflood phases of a LOCA
HUXY	Heat transfer code used to calculate the heatup of a BWR fuel assembly during all phases of a LOCA

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LOCA-ECCS
RODEX2 Computer Code

Description	Fuel rod performance code used to predict the thermal-mechanical behavior of BWR fuel rods as a function of exposure and power history
Use	Fuel rod stored energy Initial fuel rod thermal and mechanical properties
Acceptability	The safety evaluation by the NRC for XN-NF-81-58(P)(A) Rev 2 and Supplements approves RODEX2 for licensing applications SER restrictions are implemented in engineering analysis guidelines, automation tools, or through computer code controls (defaults, override warning messages)

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LOCA-ECCS
RELAX Computer Code

Description	RELAX is a BWR systems analysis code used to calculate the reactor system and core hot channel response during a LOCA
Use	Evaluate the time required to reach the end of the blowdown phase and to reach core reflood during the refill/reflood phase of the LOCA analysis Evaluate hot channel fluid conditions during the blowdown phase of LOCA and time to reach hot channel reflood during the refill/reflood phase of the LOCA analysis
Acceptability	The safety evaluation by the NRC for the topical report EMF-2361(P)(A) approves RELAX for licensing applications SER restrictions are implemented in engineering analysis guidelines and automation tools

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LOCA-ECCS
HUXY Computer Code

Description	Heat transfer code used to calculate the heatup of the peak power plane in a BWR fuel assembly during the blowdown, refill, and reflood phases of a LOCA
Use	Evaluate the peak clad temperature and metal-water reaction in the fuel assembly resulting from a LOCA
Acceptability	The safety evaluation by the NRC for the topical report XN-CC-33(A) Rev 1 approves HUXY for licensing applications SER restrictions are implemented in engineering analysis guidelines or directly in engineering computer codes

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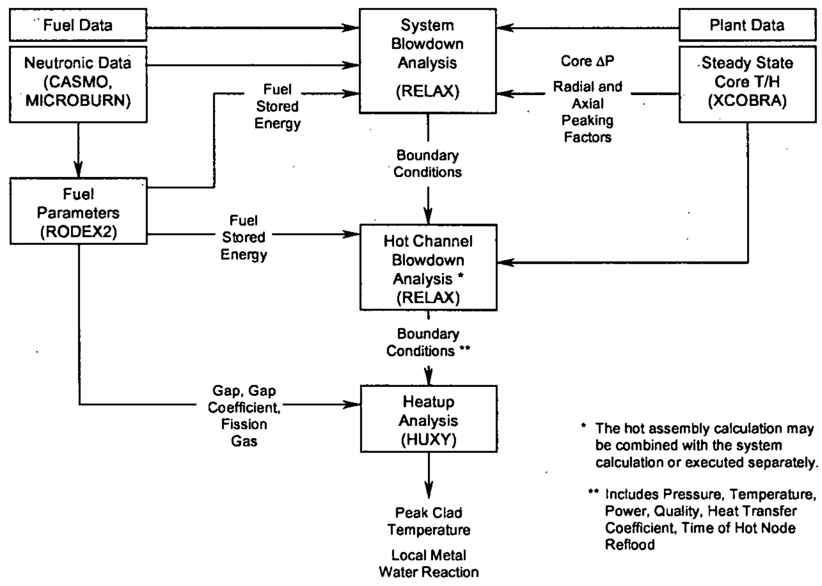
LOCA-ECCS



* The hot assembly calculation may be combined with the system calculation or executed separately.

** Includes Pressure, Temperature, Power, Quality, Heat Transfer Coefficient, Time of Hot Node Reflood

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Appendix R – Fire Protection

Appendix R – Fire Protection

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AREVA

Reactor Core Stability

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Reactor Core Stability
Task Description

> Purpose
- Determine OPRM setpoints required to ensure that power oscillations are suppressed prior to exceeding acceptable fuel design limits

> Analyses Performed
- DIVOM (Delta over Initial Versus Oscillation Magnitude)
- Initial MCPR

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Reactor Core Stability
Computer Codes

Code	Use
MICROBURN-B2	Provides initial state point information to STAIF and RAMONA5-FA. Also calculates the initial MCPRs for state points of interest
STAIF	Used to determine the limiting channel decay ratio exposure
RAMONA5-FA	Performs the DIVOM calculation

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Reactor Core Stability

STAIF Computer Code

Description
STAIF is the computer code used to calculate the core-wide, out-of-phase, and single-channel decay ratios

Use
Determine the limiting single-channel decay ratio exposure

Acceptability
EMF-CC-074(P)(A) Vol 4 Rev 0, *BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2*, Siemens Power Corporation, August 2000

The safety evaluation by the NRC for the topical report approves the STAIF methodology for licensing applications

SER restrictions are implemented in the engineering analysis guidelines

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Reactor Core Stability

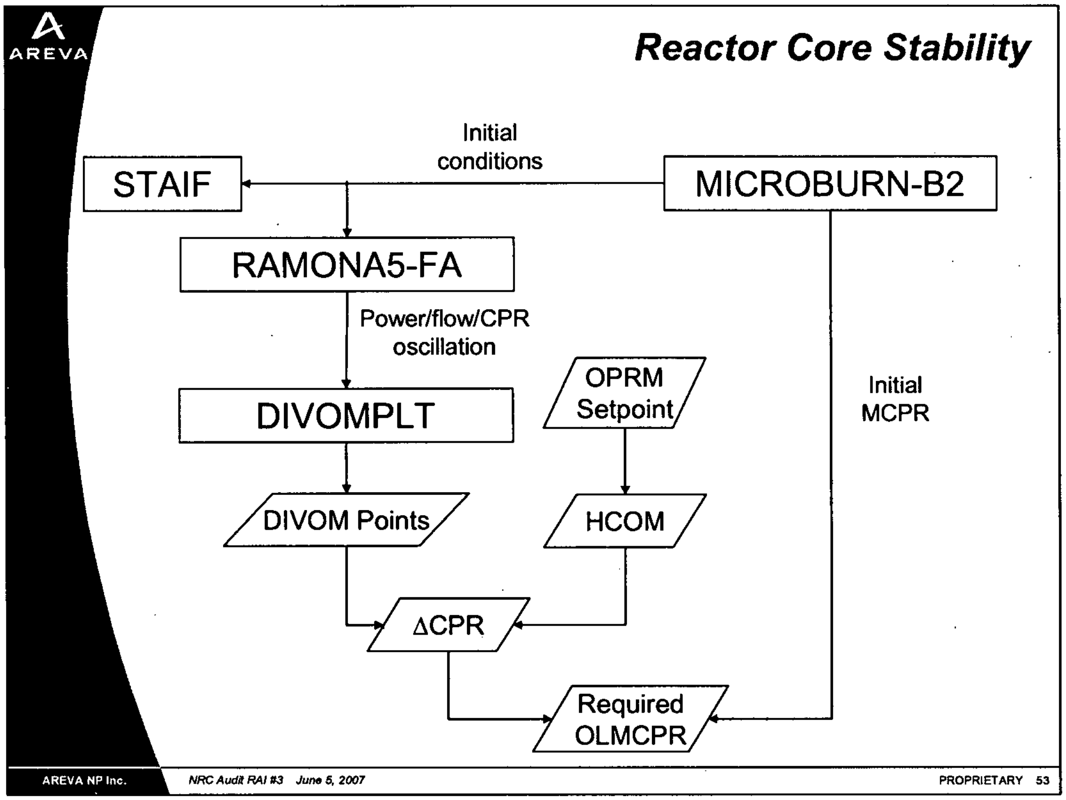
RAMONA5-FA Computer Code

Description
RAMONA5-FA is the computer code used to calculate the DIVOM relationship

Use
Simulate out-of-phase core power and flow oscillations to determine the relationship between power oscillations and CPR response

Acceptability
BAW-10255(P) Rev 2, *Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code*, January 2006

RAMONA5-FA implements the BWROG DIVOM methodology guidelines. This code has been audited by the NRC.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Reactor Core Stability
Initial conditions
STAIF
MICROBURN-B2
RAMONA5-FA
Power/flow/CPR oscillation
OPRM Setpoint
Initial MCPR
DIVOMPLT
DIVOM Points
HCOM
ΔCPR
Required OLMCPR
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Attachment 1B
Information Requested in RAI 4 Item 2

Operating conditions are presented in tabular format provided from ANP-2536(P) as requested.

AREVA PROPRIETARY

Susquehanna Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM™-10 Fuel Assemblies for Extended Power Uprate

ANP-2536(P)
Revision 0
Page 3-14

Table 3.4 Susquehanna Thermal-Hydraulic Design Conditions

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AREVA NP Inc.

Table 3.5 Susquehanna Thermal-Hydraulic Results

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Attachment 3 to PLA-6243
AREVA NP, Inc. Affidavits

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Jerald S. Holm. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the PPL letter PLA-6243, *Susquehanna Steam Electric Station Proposed License Amendment No. 285 for Unit 1 Operating License No. NPF-14 and Proposed License Amendment No. 253 for Unit 2 Operating License No. NPF-22 Extended Power Uprate Application AREVA Audit Followup Responses to Request for Additional Information*, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be

withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information".

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available,

on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Jerald S. Holm

SUBSCRIBED before me this 6th

day of July, 2007.

Leslie M. Koep

Leslie M. Koep
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 6/18/2011

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COMMISSION EXPIRES
NOTARY
PUBLIC
06-18-11
STATE OF WASHINGTON