Westinghouse

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Our ref: LTR-NRC-05-24

Attn: J. S. Wermiel, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

April 7, 2005

Subject: Overview of Westinghouse BWR Methodology: Vendor Transition and Reload (Proprietary/Non-Proprietary)

Dear Mr. Wermiel:

Enclosed are copies of the Proprietary/Non-Proprietary presentation entitled "Overview of Westinghouse BWR Methodology: Vendor Transition and Reload." This information will be used for the training that is scheduled to be provided by Westinghouse to the NRC on April 14, 2005.

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-05-1978 (Non-Proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to this affidavit or Application for Withholding should reference AW-05-1978 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

R. M. Span

R. M. Span, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: F. M. Akstulewicz/NRR
D. G. Holland/NRR
L. W. Rossbach/NRR
B. J. Benney/NRR
L. M. Feizollahi/NRR

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Our ref: AW-05-1978

April 7, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Overview of Westinghouse BWR Methodology: Vendor Transition and Reload (Proprietary)

Reference: Letter from R. M. Span to J. S. Wermiel, LTR-NRC-05-24, dated April 7, 2005

The Application for Withholding is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of Paragraph (b) (1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-05-1978 accompanies this Application for Withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this Application for Withholding or the accompanying affidavit should reference AW-05-1978 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

R. M. Span

R. M. Span, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

AW-05-1978

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

J. W. Fasnacht, Director
Customer 1st Project

Sworn to and subscribed
before me this 7th day
of April, 2005

Sharon L. Fiori
Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in, "Overview of Westinghouse BWR Methodology: Vendor Transition and Reload," (Proprietary) dated April 14, 2005, for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-05-24) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the training that is scheduled to be provided by Westinghouse to the NRC on April 14, 2005.

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

(a) Provide training to the NRC regarding Westinghouse BWR methodology including aspects that deal with vendor transition and reload activities

Further this information has substantial commercial value as follows:

(a) Westinghouse can sell support and defense of Westinghouse BWR methodology including vendor transition and reload requirements.

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

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

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

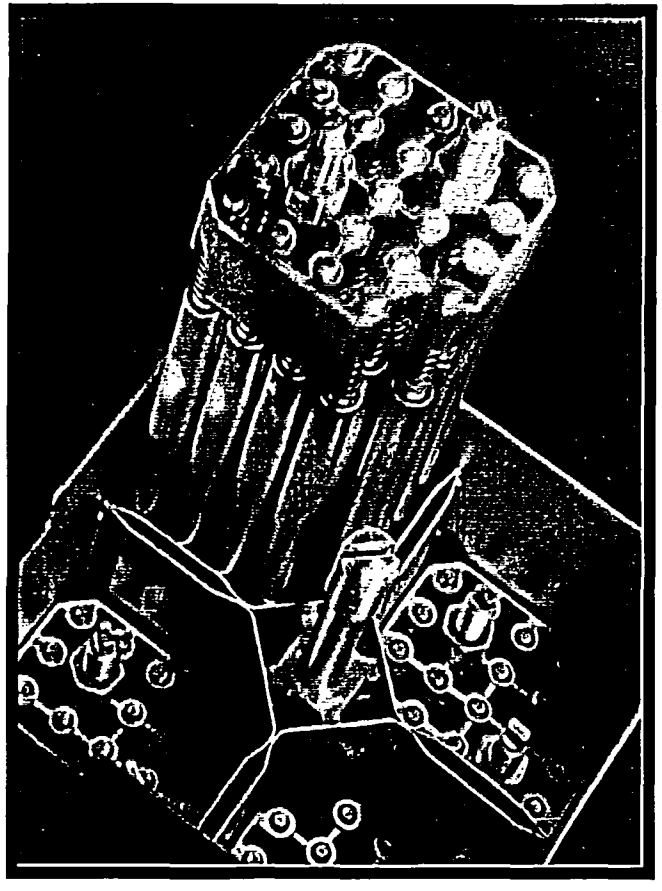
In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.
Further the deponent sayeth not.

Proprietary Information notice

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

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

Copyright Notice



Overview of Westinghouse BWR Methodology: Vendor Transition and Reload

Presented to the USNRC on April 14, 2005

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

Presentation Outline

- Overview
- Design Process
  - –Mechanical Design
  - –Nuclear Design
  - –Thermal Hydraulic Design
- Reload Safety Analysis Process
  - –Events Assessment
  - –Acceptance Criteria
  - –Event Classification by Methods
  - –Anticipated Operational Occurrences
    - –Fast
    - –Slow
- Reload Safety Analysis Process (cont'd)
  - –Accident Analysis
    - –Loss of Coolant Accident
    - –Control Rod Drop Accident
    - –Fuel Handling Accident
    - –Misplaced Assembly Accident
      - –Mislocated
      - –Misoriented
  - –Special Events
    - –Stability
      - –Option I
      - –Option III
    - –Overpressure Protection
    - –Standby Liquid Control System
    - –Anticipated Transient Without Scram

Overview

- Topical Report "Reference Safety Report for Boiling Water Reactor Reload Fuel" (CENPD-300-P-A) describes Westinghouse's Reload Licensing Methodology as well as Fuel Vendor Transition Analyses

- CENPD-300-P-A points to additional Topical Reports for more detailed information on each particular discipline (e.g., Nuclear Analysis Codes, ECCS Methods and Codes, Mechanical Design Methods and Codes)

Overview

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Overview

PHOENIX 2-D Neutronics	Nuclear Design Thermal-Hydraulic Design
POLCA 3-D Core Simulator	
BISON 1-D Kinetics & Thermal Hydraulics	Fast Transient Analysis
GOBLIN 1-D Thermal Hydraulics & Point Kinetics	LOCA Analysis
STAV 1-D Fuel Rod Models	Fuel Rod Mechanical Design
RAMONA 3-D Kinetics & Thermal Hydraulics	Rod Drop Analysis Stability Analysis
ANSYS Finite Element	Assembly Mechanical Design

Overview

- CENPD-300-P-A is divided as follows
  - –Mechanical Design
    - –Fuel Assembly and Rod Methodology
  - –Nuclear Design
    - –Benchmark and Models
    - –Input to other Disciplines
  - –Thermal-Hydraulic Design
    - –Benchmark and Models
    - –SLMCPR
    - –Hydraulic Compatibility

Overview

- CENPD-300-P-A is divided as follows (continued)
  - –Reload Safety Analysis
  - –Anticipated Operational Occurrences
    - –Fast Transient Methodology
    - –Slow Transient Methodology
  - –Accident Analysis
    - –Loss of Coolant Accident
    - –Control Rod Drop Accident
    - –Fuel Handling Accident
    - –Misplaced Assembly Accident

Overview

- CENPD-300-P-A is divided as follows (continued)
  - –Special Events Analysis
    - –Core Thermal-Hydraulic Stability
    - –Overpressurization Protection
    - –SLCS Capability
    - –Anticipated Transients Without Scram

Overview

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Overview

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Overview

- Entire process and methodologies documented in CENPD-300-P-A and discipline specific Topical Reports
- Topical Report SE limitations and conditions addressed in every application (calc notes) as well as in Reload Licensing Report documentation
- Mixed Core treatment discussed in various sections of CENPD-300-P-A; each discipline/analysis handles all legacy fuel constraints as applicable

Mechanical Design
Topical Report Structure

- CENPD-300-P-A
  - Describes Reload Fuel and Safety Analysis Methodology
  - Refers to Supporting Topical Reports
  - Chapter 2 Describes Mechanical Design Methodology
    - Support of other Methodologies (Nuclear, Thermal-hydraulic, Dynamic Analyses) – Primarily: Mechanical design description and fuel rod data
    - Refers to Mechanical Design Methodology LTR (CENPD-287-P-A) and Fuel Rod Design Code Topical Report (CENPD-285-P-A) – STAV, VIK, COLLAPS

Mechanical Design
Topical Report Structure

- Current Licensed Mechanical Design Methodology
  - CENPD-287-P-A: ABB Fuel Assembly Mechanical Design Methodology for BWR Fuel
  - CENPD-285-P-A: Fuel Rod Design Methods for Boiling Water Reactors (STAV, VIK, COLLAPS)
- SER Restrictions on STAV6.2 (CENPD-285-P-A)
  - 50 GWd/MtU Rod Average Burnup
  - Restrictions on Creep Model Uncertainty for No Clad Lift Off
  - Very Conservative FGR model > 40 GWd/MtU
  - Melt Temperature Penalty – Lack of Conductivity Degradation
  - 8 wt/% Maximum Gadolinia
- Restrictions accommodated in application methodology described in CENPD-287-P-A

Mechanical Design
Overview of Goals

- Update STAV 6.2 Code and Methodology to Support 62 MWd/KgU Peak Rod Average Burnup
  - Supplement to CENPD-285-P-A: STAV7.2 Code (WCAP-15836-P)
  - Supplement to CENPD-287-P-A: Methodology (WCAP-15942-P)
- STAV7.2 Improvements – Supplement to CENPD-285-P-A
  - Address NRC concerns, more physically-based models, improved data base, improved numerics
- Methodology - Supplement to CENPD-287-P-A
  - Improved methodology without requirements specific to STAV6.2
  - Justify application methodology to support rod-average burnup of 62 MWd/kgU
- New Information Provided as Supplements to Existing LTRs to Facilitate NRC Review

Mechanical Design

Fuel Performance Methods Supplement - WCAP-15836-P

Report No	Title	Submittal Date	Approval Date	
WCAP-15836	Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1			a,c

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- SER Being Prepared

Mechanical Design

Fuel Performance Methodology - WCAP-15942-P

Report No	Title	Submittal Date	Approval Date	
WCAP-15942	Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors – Supplement 1	[	]	a,c

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Mechanical Design
General Approach - WCAP-15942-P

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Mechanical Design
Chapters – CENPD-287-P-A and WCAP-15942-P

1. Summary and Conclusions
2. General Description
3. Design Criteria
4. Design methodology and Application
5. Technical Data
6. Code Description
7. Operating Experience
8. Prototype Testing
9. Testing, Inspection, and Surveillance Plans
10. References

Mechanical Design
Chapter 4 - WCAP-15942-P

4	DESIGN METHODOLOGY AND APPLICATION
4.1	Methodology for Evaluation of General Design Criteria
4.2	Methodology and Application - Fuel Assembly Components
4.2.1	Compatibility with Other Fuel Types and Reactor Internals
4.2.2	Geometric Changes in the Assembly During Operation
4.2.3	Transport and Handling Loads
4.2.4	Hydraulic Lifting Loads During Normal Operation and AOOs
4.2.5	Assembly Stress and Strain During Normal Operation and AOOs
4.2.6	Fatigue of Assembly Components
4.2.7	Fretting Wear of Assembly Components
4.2.8	Corrosion of Assembly Components
4.2.9	Hydriding of Zircaloy Assembly Components other than Fuel Rods

Mechanical Design
Chapter 4 - WCAP-15942-P

4.3 Methodology and Application - Fuel Rods
4.3.0 Fuel Rod Power Histories
4.3.1 Rod Internal Pressure
4.3.2 Cladding Stresses
4.3.3 Cladding Strain
4.3.4 Hydriding
4.3.5 Cladding Corrosion
4.3.6 Cladding Collapse (Elastic and Plastic Instability)
4.3.7 Cladding Fatigue
4.3.8 Cladding Temperature
4.3.9 Fuel Temperature
4.3.10 Fuel Rod Bow
4.3.11 Pellet-Cladding Interaction

Mechanical Design
Chapter 4 - WCAP-15942-P

4.4 Steady-state Initialization for Transient and Accident Conditions

4.4.1 LOCA Initialization

4.4.2 Fast transients

4.4.3 Control Rod Drop Accident

4.4.4 Stability Analysis

4.4.5 Dose Calculations

Mechanical Design
Mixed Core Treatment – and EPU

Mechanical compatibility with Legacy Fuel and Plant

- Legacy Fuel
  - Axial Fit-up and Radial Envelopes
- Plant
  - Reactor Internals (Control Blades, detectors, fuel support piece, upper Core Grid)
  - Hydraulic lifting Loads (EPU Conditions)
  - Stress and Strain (EPU Conditions)
  - Fatigue (EPU Conditions)
  - Fretting Wear – Tests bound Plant Conditions
  - Fuel Rod Performance (EPU Conditions)

Mechanical Design
Mixed Core Treatment– Plant Specific

Mechanical compatibility with Legacy Fuel

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Mechanical Design
Mixed Core Treatment– Plant Specific

Mechanical compatibility with Legacy Fuel

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Nuclear Design

- For Nuclear Design, CENPD-300-P-A describes the design bases, reload nuclear design methodology, and nuclear design inputs to other disciplines

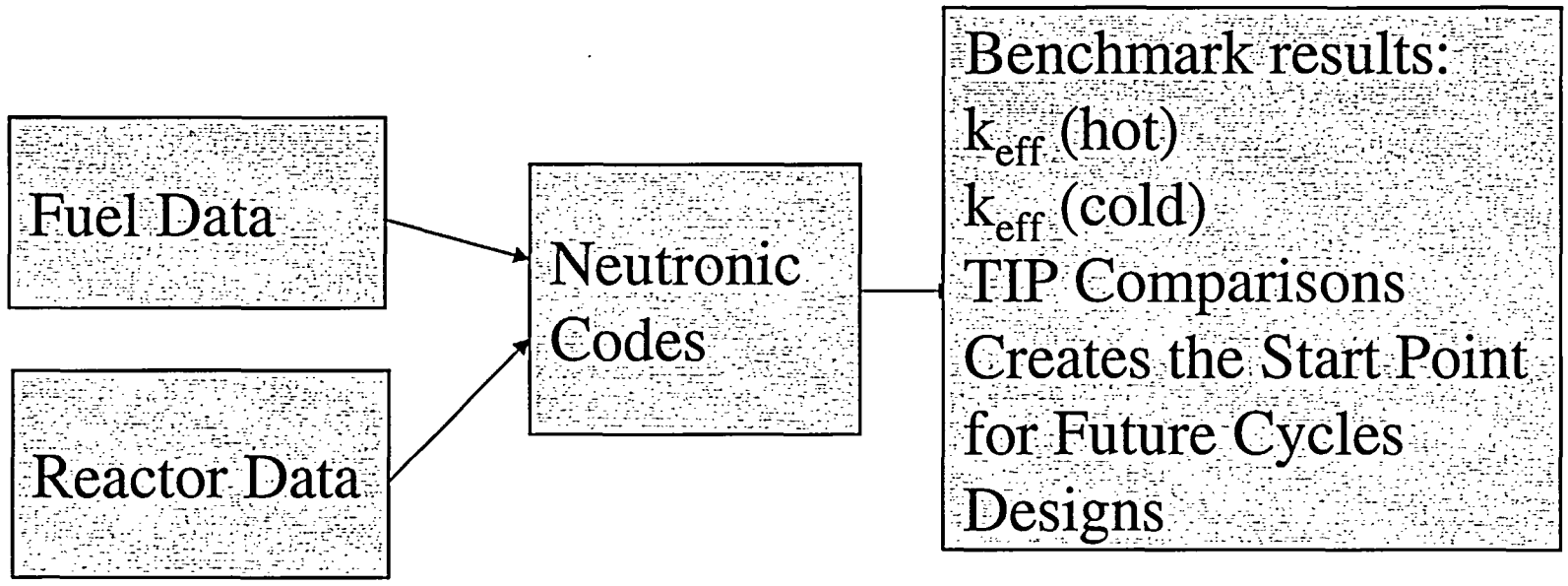
- Prior to Nuclear Design, a detailed nuclear benchmark is performed to "calibrate" the Westinghouse nuclear methods and codes for the plant to be analyzed
  - Detailed data obtained from utility

Nuclear Design

- Licensing Topical Reports
  - –CENPD-300-P-A, Section 4
  - –CENPD-390-P-A
- Computer Codes
  - –POLCA7 (3D Core Simulator)
  - –PHOENIX4 (Lattice Physics Code)

Nuclear Design

- The nuclear benchmark is the modeling of all (or part) of the previous cycles of a reactor



Nuclear Design

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Nuclear Design

- The hot target k_{eff} determined from the nuclear benchmark is used for the design cycle to predict
  - Cycle length
  - Control rod patterns
  - Number of fresh assemblies
  - Enrichment level
  - Hot excess reactivity

- The cold target k_{eff} determined from the nuclear benchmark is used for the design cycle to predict
  - Cold shutdown margin
  - Burnable absorber design

Nuclear Design

- TIP and other nuclear benchmark comparisons are used to verify how well the axial and radial power distributions are computed
  - –Used as a guide in establishing the design thermal margins

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Nuclear Design

- Based on the nuclear benchmark results, the cycle N bundle and core design are established (Reference Core)
  - All Westinghouse methodology requirements and limitations are met for the Reference Core design, as well as those imposed on the legacy fuel or the specific plant
  - The Reference Core is developed in order to support the cycle Reload Safety Analysis. The core operating limits are established based on the Reference Core
  - The Reference Core is developed with the intent that it will represent as closely as possible the as-loaded core for the upcoming cycle (deviations are treated as noted in CENPD-300-P-A Section 4.3.1.3)

Nuclear Design

- Reference Core (3D Model, potentially mixed core) used directly for
  - –Reactivity margins (shutdown margin, hot excess)
  - –Thermal margins (LGHR, MAPLHGR, CPR)
  - –Slow Transients and misplaced assembly accidents
  - –SLCS Evaluation
  - –Cycle startup – reactivity anomaly, period correction, temperature correction, criticality predictions

Nuclear Design

- Reference Core used as the basis for other disciplines
  - –Mechanical analyses
  - –Thermal-hydraulics analyses
  - –Fast Transients analyses
  - –LOCA analyses
  - –CRDA analyses
  - –Stability analyses
  - –Other analyses, as required

Nuclear Design

- Reference Core thermal limits assumptions are corroborated after licensing campaign completed, prior to submitting Reload Licensing Report

- After Cycle N-1 shutdown, licensing analyses are corroborated to be valid for the (to be) As-loaded core

Thermal Hydraulic Design

- Thermal Hydraulic Compatibility
- Safety Limit MCPR (see 12-8-04-slmcpr.ppt)
- Legacy Fuel CPR Correlation

Thermal Hydraulic Design
T/H Compatibility

- Legacy Fuel Hydraulic data from utility
- Detailed mixed core plant hydraulic models established
- Flow splits and DP's evaluated for
  - –Various legacy/Westinghouse fuel fractions
  - –Various points on the power/flow map
- By-pass flow fraction maintained at rated core power/flow
- Internal flow fraction sized to avoid significant boiling
- Same detailed hydraulic modeling in core nuclear models
  - –Thermal performance evaluated during cycle nuclear design

Thermal Hydraulic Design
T/H Compatibility

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Thermal Hydraulic Design
T/H Compatibility

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Thermal Hydraulic Design
T/H Compatibility

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Thermal Hydraulic Design
T/H Compatibility

a,b,c

Thermal Hydraulic Design
Legacy Fuel CPR Correlation

- Legacy Fuel CPR performance data obtained from utility
  - –Parameter Variation Matrix - single parameter variation
  - –Validation Matrix – multi-parameter variation
- Westinghouse correlation adjusted to fit data
- "Conservative adder" applied to the OLMCPR to accommodate uncertainty

Thermal Hydraulic Design
Legacy Fuel CPR Correlation

- Parameter Variation Matrix

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Thermal Hydraulic Design
Legacy Fuel CPR Correlation

- Validation Matrix

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Thermal Hydraulic Design
Legacy Fuel CPR Correlation

- Parameter Variation and Validation Matrix Independent Variable Points in the Active Flow - Assembly Exit Pressure Plane

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Thermal Hydraulic Design
Legacy Fuel CPR Correlation

- Sample Legacy Fuel CPR Correlation

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Thermal Hydraulic Design
Legacy Fuel CPR Correlation

Sample Statistics Summary of Comparison (CPR_Reference/CPR_W)

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Events Assessment

- Reload Licensing Methodology Basis

Documents the process of reviewing the Plant Licensing Bases: FSAR, Plant License and Amendments, Technical Specifications. Identify the events requiring evaluation for the introduction of the new fuel with the main purpose of establishing operating limits.

Application of the CENPD-300-P-A Topical Report Reload Methodology to the Plant.

Events Assessment

- An engineering evaluation is performed for each licensing basis event to determine whether the event will be reanalyzed to support the introduction of SVEA-96 Optima2 fuel.

- While CENPD-300-P-A identifies certain events that will be evaluated for the introduction of Westinghouse fuel, other events may also be identified as a result of this review.

- Other plant specific licensing commitments are handled by the vendor transition data transfer process and included in the evaluation.

Events Assessment

- The evaluated events are classified as one of following:
  - Affected by introduction of new fuel design and is potentially limiting requiring reanalysis for each reload.
  - Affected by introduction of new fuel design, but not for different cycles requiring a single analysis for the introduction of the new fuel type. The event is not reanalyzed for subsequent reloads.

Events Assessment

- Potentially affected by the introduction of the new fuel type, but is bounded by a more limiting event of the same frequency category. No analyses will be performed for this type of event.

- Not affected by the introduction of a new fuel type. No analyses will be performed for this type of event.

Events Assessment
Categorization of events

- The three frequency categories according to the FSAR:

  A. Incidents of Moderate Frequency: Incidents that may occur with a frequency greater than one per 20 years (Anticipated (expected) Operational Occurrence).

  B. Infrequent Incidents: Incidents that may occur during the life of the particular plant, frequency one in 20 to one in 100 years (Abnormal (unexpected) Operational Occurrences).

  C. Limiting Faults: Incidents not expected to occur, frequency lower than one in 100 years (Design Basis (postulated) Accidents).

Events Assessment
Categorization of events

- Westinghouse methodology:

1. Anticipated Operational Occurrence: Incidents expected to occur one or more times during the life of the Plant. Frequency greater than one per 100 years.
   (Includes Incidents of Moderate Frequency and Infrequent Incidents)
2. Accidents: Postulated events that affect one or more barriers to the release of radioactive materials to the environment.
   (Limiting Faults)
3. Special Events: Additional postulated events that are analyzed to demonstrate different plant capabilities.

Events Assessment
Reload Analysis Potentially Limiting Events

Anticipated Operational Occurrences

- *Generic Analyses*
  - Turbine Trip or Generator Load Rejection without Bypass
  - Pressure Regulator Failure - Closed (BWR/6 Only)
  - Loss of Feedwater Heating
  - Control Rod Withdrawal Error
  - Recirculation Flow Controller Failure - Increasing Flow
  - Feedwater Controller Failure - Maximum Demand
- *Additional Plant Specific Analyses, As Required*

Events Assessment
Reload Analysis Potentially Limiting Events

Design Base Accidents

- *Generic Analyses*
  - Loss of Coolant Accident
  - Control Rod Drop Accident
  - Fuel Handling Accident
  - Fuel Loading Errors
- *Additional Plant Specific Analyses, As Required*

Events Assessment
Reload Analysis Potentially Limiting Events

Special Events

- *Generic Analyses*
  - Core Thermal-Hydraulic Stability
  - Reactor Overpressure Protection (ASME)
  - Shutdown Without Control Rods (Standby Liquid Control System Capability)
  - Spent Fuel Pool Criticality
  - New Fuel Pool Criticality
- *Additional Plant Specific Analyses, As Required*

Acceptance Criteria

Event Classification	Westinghouse Reload Methodology Acceptance Criteria
Moderate Frequency Events	Radioactive Effluents ≤ 10CFR20 Limits Specified Acceptable Fuel Design Limits Satisfied: - MCPR ≥ SLMCPR - LHGR ≤ Overpower Limit (< 1 % plastic strain, no fuel melting) - Average Fuel Pellet Enthalpy ≤ 170 cal/gm Peak Reactor Vessel Pressure ≤ 110 % Design Suppression Pool Temperature ≤ Heat Capacity Temperature Limit
Infrequent Events	AOO Acceptance Criteria are applied

Acceptance Criteria

Event Classification	Westinghouse Methodology Acceptance Criteria
Limiting Faults	Radiological Consequences: Offsite Dose ≤ 10CFR100 Limit Control Room Dose ≤ GDC-19 Limit Barrier Performance Control Rod Drop Accident - Peak Fuel Enthalpy ≤ 280 cal/gm Misplaced Bundle Accident -MCPR ≥ SLMCPR (no fuel cladding damage)

Acceptance Criteria

Event Classification	Westinghouse Methodology Acceptance Criteria
Limiting Faults	Barrier Performance LOCA - 10CFR50.56 Limits Satisfied – Peak clad temp ≤ 2200 F – Max. clad oxidation ≤ 0.17 times clad thickness – Core wide metal-water reaction ≤ 0.01 – Core remains amenable to cooling – Demonstrate long-term cooling capabilities - Containment pressure ≤ Containment design limit

Event Classification by Methods

- The events identified for analysis are grouped by analysis methods into:
  - Fast Transients
  - Slow Transients
  - 3D Transients

Event Classification by Methods

Fast Versus Slow Transient Analyses

- Fast Transients
  - Group of AOOs and accidents for which accurate simulation of the system response is important
  - Generally characterized by rapid changes in neutron flux
  - Typical fast transient AOOs include:
    - Generator Load Rejection
    - Turbine Trip
    - Feedwater Controller Failure
  - Typical fast transients that are not AOOs include:
    - ASME Overpressure Protection
    - ATWS Evaluation

Event Classification by Methods

Fast Versus Slow Transient Analyses

- Slow Transients
  - Group of AOOs and accidents for which dynamic changes are sufficiently slow so that steady state conditions are achieved at each time step during the event
  - Generally characterized by slow changes in neutron flux
  - Typical slow transient AOOs include:
    - Recirculation Flow Controller Failure
    - Control Rod Withdrawal Error
    - Loss of Feedwater Heating
    - Inadvertent HPCI Startup
  - Typical slow transients that are not AOOs include:
    - Mislocated and Misoriented Fuel Assembly

Event Classification by Methods

Fast Versus Slow Transient Analyses

- **Transient Analyses Methodology Matrix**

•Category	•Analysis Codes	•Example AOOs	•Example Accidents and •Special Events
•Fast Transients	•BISON / SLAVE	•Generator Load Rejection	•ASME Overpressure Protection
		•Turbine Trip	•ATWS Evaluation
		•Feedwater Controller Failure	
		•Recirculation Controller Failure	
•Slow Transients	•POLCA	•Control Rod Withdrawal Error	•Mislocated Fuel Assembly
		•Recirculation Flow Controller Failure	•Misoriented Fuel Assembly
		•Inadvertent HPCI Startup	
		•Loss of Feedwater Heating	
•3-D Transients	•RAMONA-3	•- - -	•Stability
			•Control Rod Drop Accident

Anticipated Operational Occurrences Fast Transients - Definition

- Group of AOOs and accidents for which accurate simulation of the system response is important
- Generally characterized by rapid changes in neutron flux
- Events that are of relatively short duration such that the temporal dynamics of nuclear and thermal-hydraulic feedbacks are important, e.g.
  - Generator Load Rejection, Without Bypass
  - Turbine Trip, Without Bypass
  - Feedwater Controller Failure – Maximum Demand
- Typical fast transients that are <u>not</u> AOOs include
  - ASME Overpressure Protection
  - ATWS Evaluation

Anticipated Operational Occurrences Fast Transients - Overview

- Purpose
  - Ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are satisfied during Anticipated Operational Occurrences (AOOs)
- SAFDLs - Acceptance Criteria
  - Clad Temperature: CPR during AOOs > SLMCPR
  - Clad Strain: LHGR such that $\varepsilon < 1\ \%$
  - Pellet Temperature: LHGR such that Peak Fuel $T < T_{\text{fuel Melt}}$
- Methodology
  - Determine plant response to postulated event
  - Determine hot channel response by imposing plant response boundary conditions to hot channel model

Anticipated Operational Occurrences Fast Transients - Acceptance Criteria

- The design bases are met by imposing the following acceptance criteria for all AOOs
  - Radioactive effluents ≤ 10CFR20 limit
  - Peak reactor vessel pressure ≤ 110% vessel design pressure
  - Suppression pool ≤ heat capacity temperature limit
  - Specified Acceptable Fuel Design Limits (SAFDLs) satisfied:
    - MCPR ≥ safety limit MCPR
    - LHGR ≤ overpower limit
    - Average fuel pellet enthalpy ≤ 170 cal/g

Anticipated Operational Occurrences Fast Transients – Mixed Core Treatment

- System analyses currently treated with a one-dimensional, single channel core description based on a radially averaged mixed core (BISON/WPOL/POLBIS)
  - Pressurization AOOs (Turbine Trip, Load Rejection, Feed Water Controller Failure) – Potential CPR limits established for Westinghouse and Legacy Fuel
  - ATWS
  - ASME overpressure evaluation

Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- Main computer codes
  - BISON - for core average system response calculations in 1-D
  - SLAVE - for hot channel CPR response calculations in 1-D
  - WPOL - for 3-D to 1-D collapse of POLCA cross sections for input to BISON
  - POLBIS - for performing volume-weighted averaging of core-dependent inputs, e.g., pellet and cladding diameters, loss coefficients, and flow areas
- Licensing topical reports
  - RPA 90-90-P-A (models and qualification)
  - CENPD-292-P-A (models amendment)
  - CENPD-300-P-A (methodology and sample applications)

Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- BISON
  - Reactor model
    - 1-D 2-group diffusion kinetics model
    - 1-D fuel model with radial heat conduction
    - 1-D TH loop model with single core channel and bypass
    - TH non-equilibrium model

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Anticipated Operational Occurrences
Fast Transients - Analysis Methodology

- Steam line model
  - Free geometry with parallel pipes
  - SV/RV System
  - Turbine Valves
  - Bypass Valves
  - Main Steam Isolation Valves
    etc.

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Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- BISON
  - System models
    - Recirculation pumps
    - Models for emergency cooling
    - Safety system and Scram models
  - Other models
    - Measurement systems (level etc)
    - User specified control and safety logic systems
    - Feedwater models

    etc.

Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- SLAVE
  - Single Bundle model
    - Simplified kinetics model based on BISON boundary conditions
    - 1-D fuel model with radial heat conduction
    - TH model with boundary conditions from BISON

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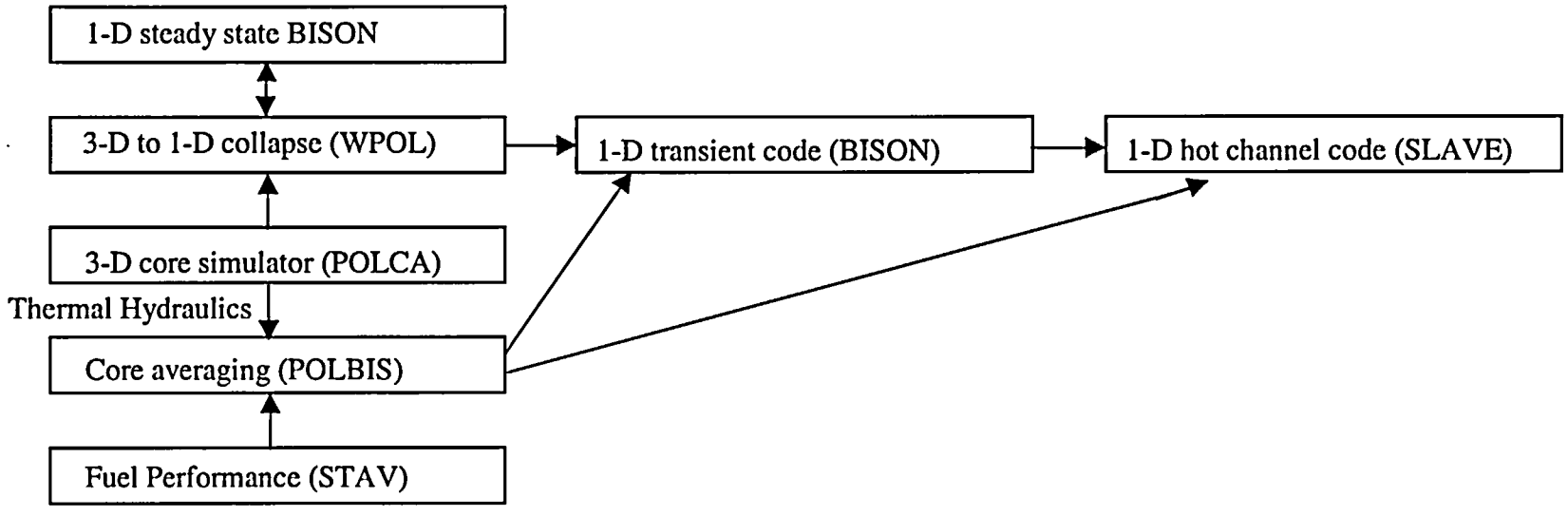
Used for CPR evaluation

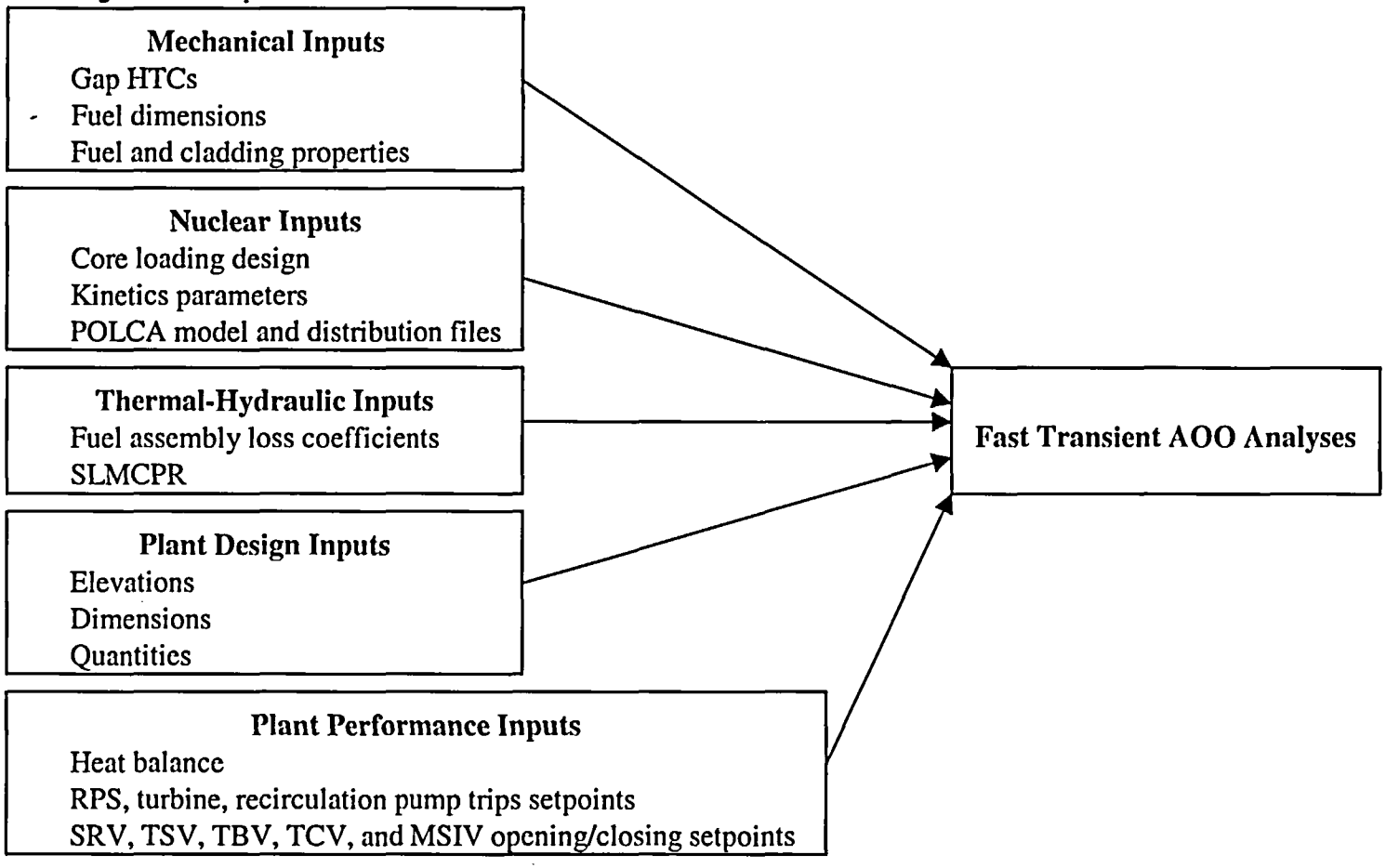
Anticipated Operational Occurrences
Fast Transients - Analysis Methodology

- Fast transient analysis code interfaces



Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- Analysis inputs



Anticipated Operational Occurrences
Fast Transients - Analysis Methodology

- Analysis process - BISON basedeck calculations

Create BISON Basedeck
- Data not expected to change
- Geometric information
- Equipment elevations
- Code options

↓

Create BISON Change Decks
- Core-dependent data
- Burnup-dependent data
- Tuning data

↓

BISON Cross Sections
- Neutronic cross section data
- Operating point data

↓

Benchmark BISON Basedeck
- 100% power / 100% flow calculated values match plant data at time zero
- Steady state operation is maintained with no perturbations introduced
- Calculated transient behavior matches plant data

Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- Analysis process - SLAVE basedeck calculations

Create SLAVE Basedeck
- Data not expected to change
- Geometric information
- Code options

↓

Create SLAVE Change Decks
- Fuel-dependent data
- Burnup-dependent data

↓

BISON Boundary Condition File
- Core inlet pressure
- Core inlet enthalpy
- Core inlet void content
- Core pressure drop
- Core exit void content
- Total core fission power
- Core average axial power shape

Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- Analysis process - BISON cross section calculations

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Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- Analysis process - BISON cross section calculations
  - As part of the cross section generation process, the 3 hot channel candidates for each fuel type are determined from POLCA for use in the subsequent ΔCPR calculations
    - Minimum CPR
    - Maximum LHGR
    - Maximum FRAD
  - R-factor, active flow, nominal bundle power, and axial power shape for each hot channel candidate are determined from POLCA for use in the subsequent ΔCPR calculations

Anticipated Operational Occurrences
Fast Transients - Analysis Methodology

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Anticipated Operational Occurrences
Fast Transients - Analysis Methodology

- Analysis process - BISON system response calculations

BISON Basedeck

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BISON Change Decks
- Core-dependent data
- Burnup-dependent data
- Tuning data

↓

BISON Cross Sections
- Neutronic cross section data
- Operating point data

↓

BISON Perturbation Decks
- Initial water level
- Valve closures / openings
- Reactor, feed water, and pump trips
- Scram rod insertion

↓

BISON System Response
- Boundary conditions recorded for use in hot channel SLAVE case

Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- Analysis process - SLAVE ΔCPR calculations

SLAVE Basedeck

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SLAVE Change Decks
- Fuel-dependent data
- Burnup-dependent data

↓

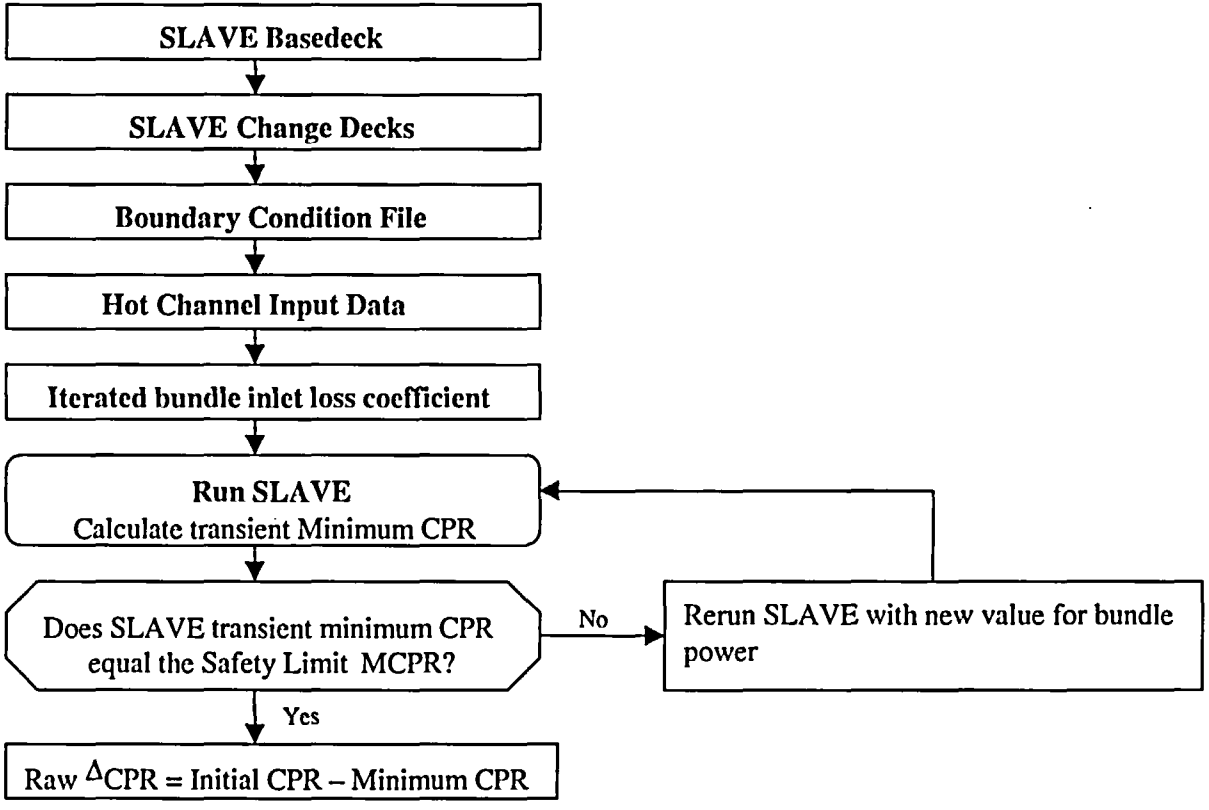
Boundary Condition File
- Core inlet pressure
- Core inlet enthalpy
- Core inlet void content
- Core pressure drop
- Core exit void content
- Total core fission power
- Core average axial power shape

↓

Hot Channel Input Data
- Bundle R-Factor
- Bundle active flow
- Bundle nominal power
- Hot channel axial power shape

Anticipated Operational Occurrences
Fast Transients - Analysis Methodology

- Analysis process - SLAVE ΔCPR calculations
- Bundle power iteration



Anticipated Operational Occurrences
Fast Transients - Analysis Methodology

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Anticipated Operational Occurrences
Fast Transients - Analysis Methodology

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Anticipated Operational Occurrences Fast Transients - Analysis Methodology

- Analysis outputs
  - Operating Limit Minimum CPRs (OLMCPRs)
    - Full power fast transient OLMCPRs - to be auctioneered against those OLMCPRs determined by the slow transients
    - Low power fast transient OLMCPRs - to be used to determine the power-dependent OLMCPRs
  - Maximum increase in Linear Heat Generation Rate (LHGR) during the transients

Anticipated Operational Occurrences Slow Transients

- In CENPD-300-P-A, "Slow Transients" refers to AOOs which can be described in terms of a sequential series of steady-state conditions
  - Recirculation Flow Controller Failure (RFCF)
  - Control Rod Withdrawal Error (RWE)
  - Loss of Feedwater Heating (LOFH)
- Slow transient analyses are performed directly on the 3D core model
- Approach A (CENPD-300-P-A Section 7.3.3) used

Anticipated Operational Occurrences Slow Transients

- ΔCPRs based on these AOOs are used to establish limiting OLMCPRs which protect SLMCPRs based on all licensing basis events

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- Additionally, it must be verified that the TTMOL (transient thermal mechanical operating limit) is not exceeded during the transient

Anticipated Operational Occurrences
Slow Transients - RFCF

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Anticipated Operational Occurrences
Slow Transients - RFCF

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Anticipated Operational Occurrences Slow Transients - RFCF

- Licensing Topical Report
  - –CENPD-300-P-A, Section 7.5.3
- Computer Codes
  - –POLCA7 (3D Core Simulator)

Anticipated Operational Occurrences
Slow Transients - RFCF

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Anticipated Operational Occurrences
Slow Transients - RFCF

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Anticipated Operational Occurrences
Slow Transients - RWE

- Rod Withdrawal Error
  - –Local power increase due to a control rod being withdrawn with no operator action
  - –Causes increase in core and local power
  - –Event may be terminated by the Rod Block Monitor System
  - –Increase in power leads to decrease in CPR

Anticipated Operational Occurrences Slow Transients - RWE

- Licensing Topical Report
  - –CENPD-300-P-A, Section 7.5.4
- Computer Codes
  - –POLCA7 (3D Core Simulator)

Anticipated Operational Occurrences
Slow Transients - RWE

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Anticipated Operational Occurrences
Slow Transients - RWE

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Anticipated Operational Occurrences
Slow Transients - RWE

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Anticipated Operational Occurrences
Slow Transients - LOFH

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Anticipated Operational Occurrences Slow Transients - LOFH

- Licensing Topical Report
  - –CENPD-300-P-A, Section 7.5.5
- Computer Codes
  - –POLCA7 (3D Core Simulator)

Anticipated Operational Occurrences
Slow Transients - LOFH

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Accident Analysis

- Definition
  - Postulated events that affect one or more radioactive material barriers and are not expected to occur during the plant lifetime
  - Used to establish design bases for certain systems
- Potentially Limiting Accidents Re-evaluated
  - Loss of Coolant Accident
  - Control Rod Drop Accident
  - Fuel Handling Accident
  - Misplaced Assembly Accident
    - Mislocated
    - Rotated
  - Plant-specific

Accident Analysis
LOCA Methodology

- GOBLIN Code
  - Determines core-wide response
    - Reactor Trip
    - MSIV Closure
    - ECCS actuation
    - Boundary conditions for hot assembly and heat-up analyses
  - Determines hot assembly response ('DRAGON' mode)
    - Boiling transition
    - Loss of convective cooling
    - Restoration of two-phase cooling
    - Boundary conditions for heat-up analysis
- CHACHA Code
  - Determines heat-up response of hot plane

Accident Analysis
LOCA Methodology (Flow of Information)

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Accident Analysis
LOCA Licensing Topical Reports

- RPB 90-93-P-A, "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," October 1991
- RPB 90-94-P-A, "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity," October 1991
- CENPD-293-P-A, "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," July 1996
- CENPD-283-P-A, "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel," July 1996
- WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," April 2003
- WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," November 2004

Accident Analysis
LOCA Process

- LOCA Inputs
  - Receive plant-specific inputs (geometry, system performance, uncertainties, etc)
    - Document LOCA input parameters in LOCA Inputs Report and obtain customer concurrence
  - Receive design inputs (nuclear, thermal-hydraulics, fuel performance)
    - Document in Calculation Note
- Develop Plant Models
  - GOBLIN
  - DRAGON
  - CHACHA

Accident Analysis
LOCA Process (Verify Plant Models)

- Jet Pump Model Verification
  - M and N ratio
- Recirculation Pump Modeling Verification
  - Plant-specific pump coastdown data
- Core Model Verification
  - Verify model predicts pressure distribution, flow distribution and enthalpy distribution similar to thermal-hydraulic analysis using 3D core simulator
- System Model Verification
  - Verify model predicts ECCS performance as described in the LOCA Inputs Report
  - Verify reactor trip occurs as described in the LOCA Inputs Report
  - Verify SRVs function as described in the LOCA Inputs Report

Accident Analysis
LOCA Process (Reports)

- LOCA Report (at fuel transition)
  - Break Spectrum Analysis
    - Results of analysis using bounding plant model ('Plant 5') to determine limiting break size, break location, single failure combination
      - Plant 5 model accounts for plant-to-plant variations in delay times, ECCS performance, valve stroke times, etc. by assuming bounding values.
    - Results cover operating domain (e.g., MELLLA/ICF upper boundary)

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    - Sensitivity studies
      - Limiting ECCS temperature
      - Axial power distribution

Accident Analysis
LOCA Process (Reports)

- LOCA Report (at fuel transition)

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Accident Analysis
LOCA Process (Reports)

- MAPLHGR Report (cycle-specific)

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Accident Analysis
Control Rod Drop Accident Analysis Description

- Design basis event of Reactivity Insertion Accident
- Drop of the highest worth control rod out of the reactor core under most severe conditions for reactivity insertion.

Accident Analysis

Control Rod Drop Accident Design Bases and Acceptance Criteria

- Design Bases
  - –General Design Criteria (GDC) 28
  - –Effects of postulated reactivity accidents neither:
    - –Result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor
    - –Cause sufficient damage to impair significantly the capacity to cool the core.

Accident Analysis

Control Rod Drop Accident Design Bases and Acceptance Criteria

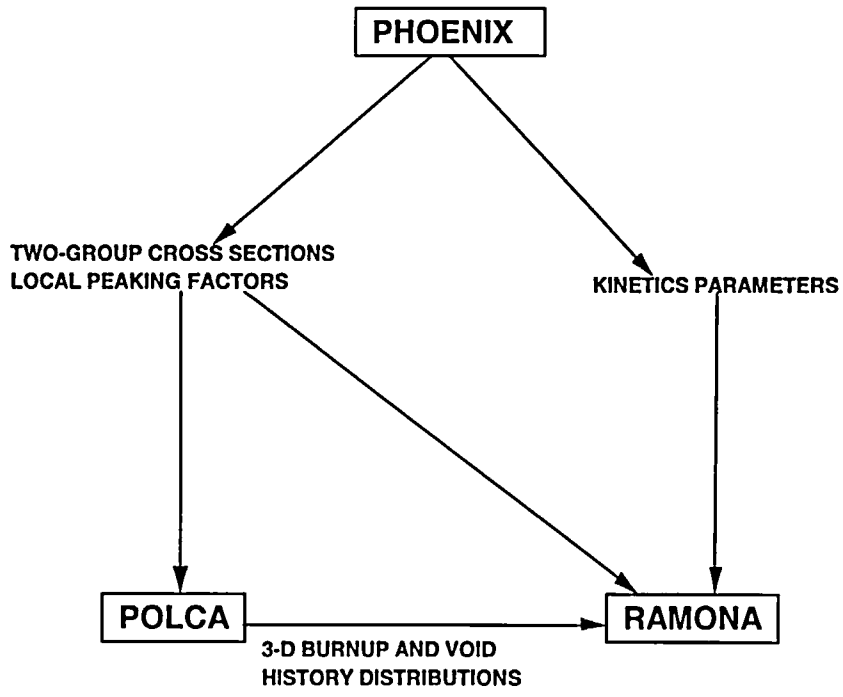
- Design Acceptance Criteria
  - –In accordance with in NUREG-0800, Standard Review Plan
    - –Reactivity excursions should not result in a radially averaged fuel rod enthalpy greater than 280 calories/gm at any axial location in any fuel rod.
    - –The maximum reactor pressure <ASME Code Section III "Service Limit C" stress limits.
    - –The number of fuel rods predicted to reach radially average enthalpy of 170 cal/gm are assumed failed and input to a radiological evaluation.

Accident Analysis
Control Rod Drop Accident Analysis Methodology

- Computer Codes
  - –POLCA - Selecting potentially limiting Rod Configurations and Worths
  - –RAMONA - Dynamic calculation of Peak Fuel Enthalpy
  - –PHOENIX - Cross Sections input to Dynamic calculation
  - –POLCA - Initial Burnup distribution input to Dynamic calculations
- Licensing Topical Report
  - –CENPD-284-P-A (Code Methods, Design Bases, Methodology, Sample Application)

Accident Analysis
Control Rod Drop Accident Analysis Methodology

- POLCA Analysis Inputs
  - –Core Configuration
  - –Rod Withdrawal Sequencing
- RAMONA Analysis
  - –Core Configuration and Burnup Distribution
  - –Lattice Cross Sections
  - –Kinetics Parameters
  - –Gap Heat Transfer Coef.
  - –Limiting Control Rod Configuration
  - –Scram Speed
  - –Flux Scram Setpoint



Accident Analysis
Control Rod Drop Accident Analysis Methodology

- 2-Step Analysis Process
  - –Rod Withdrawal Plant Sequence Evaluated with POLCA to Identify Potentially Limiting Dropped Rods
  - –Potentially Limiting Cases Evaluated Dynamically with RAMONA
  - –Systematic Approach to Minimize Cycle-Specific Evaluation
  - –Emphasis on Modeling Phenomena as Accurately as Possible to Avoid Unnecessary Conservatisms
  - –Thorough Benchmarking
  - –Uncertainty Analysis Included in Methodology

Accident Analysis
Control Rod Drop Accident Analysis Methodology

- Analysis Outputs
  - –Peak Fuel Rod Enthalpy in cal/gm
  - –No vessel overpressurization confirmation
  - –Fuel Rod Failures, if applicable

Accident Analysis

Control Rod Drop Accident Sample Application - Initial Conditions

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Accident Analysis

Control Rod Drop Accident Sample Application – Drop Rod Worth

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Accident Analysis

Control Rod Drop Accident Sample – Potential Limiting Configurations

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Accident Analysis

Control Rod Drop Accident Sample Application – Power Transient

a,c

Accident Analysis

Control Rod Drop Accident Sample Application – Reactivity Components

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Accident Analysis

Control Rod Drop Accident Sample Application – Peak Enthalpy

a,c

Accident Analysis

Control Rod Drop Accident Sample Application – Uncertainty Evaluation

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Accident Analysis

Control Rod Drop Accident Sensitivities/Experience/Observations

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Accident Analysis

Control Rod Drop Accident Sensitivities/Experience/Observations

a,c

Accident Analysis

Control Rod Drop Accident Sensitivities/Experience/Observations

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Accident Analysis

Control Rod Drop Accident Sensitivities/Experience/Observations

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Accident Analysis
Fuel Handling Accident

- Described in CENPD-300-P-A
- Comparative Analysis with Plant Analysis of Record
- Potential Energy gained by dropped assembly absorbed by dropped and impacted assemblies
- All fuel rods in dropped assembly assumed to fail
- Depending on design, energy transferred to impacted assemblies shared between channel and rods
- Additional rods in impacted assemblies fail at 1 % strain determined form simple energy balance

Accident Analysis
Misplaced Assembly Accident

- Mislocated Fuel Assembly Accident
  - –Fuel assembly placed in incorrect location
- Misoriented (Rotated) Fuel Assembly Accident
  - –Fuel assembly placed in proper location in wrong orientation

Accident Analysis
Misplaced Assembly Accident

- Classified as accidents for which the SLMCPR is the figure of merit to preclude the possibility of long-term operation in transition boiling

- ΔCPRs based on these accidents considered with AOO results to establish limiting OLMCPRs which protect SLMCPRs based on all licensing basis events

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Accident Analysis
Misplaced Assembly Accident - Mislocated

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Accident Analysis
Misplaced Assembly Accident - Mislocated

- Licensing Topical Reports
  - –CENPD-300-P-A, Section 8.5.1
- Computer Codes
  - –POLCA7 (3D Core Simulator)
  - –PHOENIX4 (Lattice Physics Code)

Accident Analysis
Misplaced Assembly Accident - Mislocated

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Accident Analysis
Misplaced Assembly Accident - Misoriented

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Accident Analysis
Misplaced Assembly Accident - Misoriented

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Accident Analysis
Misplaced Assembly Accident - Misoriented

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Accident Analysis
Misplaced Assembly Accident - Misoriented

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Accident Analysis
Misplaced Assembly Accident - Misoriented

- Licensing Topical Reports
  - –CENPD-300-P-A, Section 8.5.2
- Computer Codes
  - –POLCA7 (3D Core Simulator)
  - –PHOENIX4 (Lattice Physics Code)

Accident Analysis
Misplaced Assembly Accident - Misoriented

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Special Events

- Definition
  - Events evaluated to demonstrate compliance with certain regulatory requirements, industry codes, licensing commitments
- Potentially Limiting Special Events Re-evaluated:
  - Core Thermal Hydraulic Stability
  - Overpressure Protection
  - Standby Liquid Control System Capability
  - Anticipated Transients Without Scram

Special Events

Core Thermal Hydraulic Stability - Overview

- Two types of stability analyses are performed to protect the fuel from coupled neutronic – thermal hydraulic instabilities
  - Option I-A – Exclusion Region
    - Identify regions on the power-flow map that restrict plant operation
      - Warning region – plant takes action to exit region
      - Scram region – plant is scrammed to avoid unstable situations
    - Used as a backup in the event OPRM is out of service
  - Option III – LPRM-base Detect and Suppress
    - Automatic scram based on detected oscillation magnitude
    - Protects the safety limit

Special Events
Core Thermal Hydraulic Stability Option I-A

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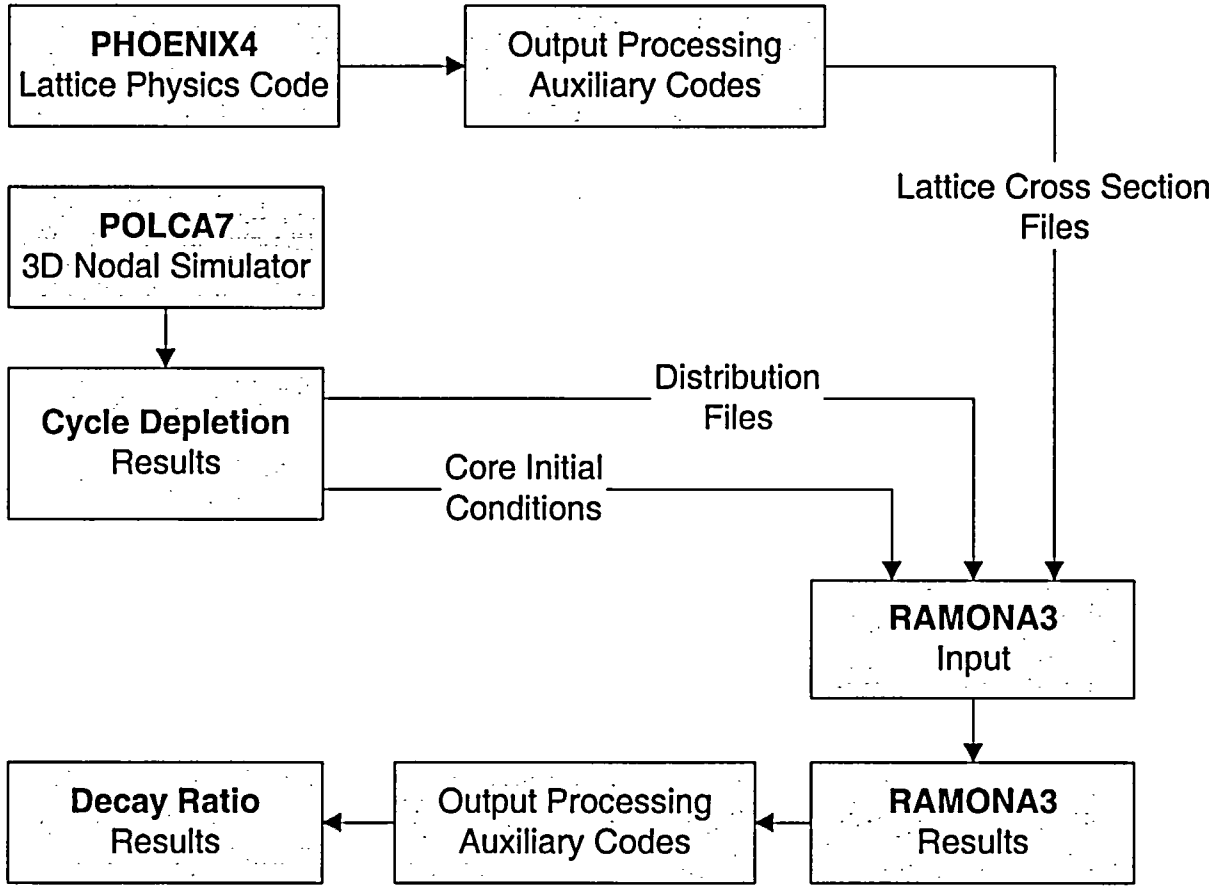
- Codes Used
  - RAMONA3 – Full mixed core model used to simulate the response to perturbation
  - POLCA7 – Full mixed core model used to derive the distribution file and the initial conditions at each statepoint analyzed
  - PHOENIX4 – Used to derive cross sections for each lattice type

Special Events
Core Thermal Hydraulic Stability Option I-A

Information Flow



Special Events
Core Thermal Hydraulic Stability Option I-A

Licensed Topical Reports

- CENPD-294-P-A, "Thermal-Hydraulic Stability Analysis Methods for Boiling Water Reactors," July 1996
- CENPD-295-P-A, "Thermal-Hydraulic Stability Analysis Methodology for Boiling Water Reactors," July 1996

Special Events
Core Thermal Hydraulic Stability Option I-A

Important Parameters

- Single-phase and two-phase pressure drops
- Radial and axial power distributions
- Void reactivity
- Core power and flow rate
- Control rod pattern
- Core inlet subcooling

Special Events

Core Thermal Hydraulic Stability Option I-A

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Special Events

Core Thermal Hydraulic Stability Option III

- Methodology
  - Based on BWROG 'Plant-Specific Regional Mode DIVOM Procedure Guideline,' GE-NE-0000-0028-9714-R0

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- Codes Used
  - POLCA7 for steady-state calculations
  - POLCA7/RAMONA3 to establish limiting power/flow conditions (also used for DR calculations)
  - BISON (SLAVE) to calculate fractional change in CPR in hot assembly (also used for fast transient hot channel calculations)

Special Events

Core Thermal Hydraulic Stability Option III (Example)

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Special Events
Core Thermal Hydraulic Stability Option III

Analysis Procedure

- Generate plant-specific models
  - POLCA7, PHOENIX4, RAMONA3, BISON (SLAVE)

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Special Events
Core Thermal Hydraulic Stability Option III

Analysis Assumptions

- Assumptions (GE-NE-0000-0028-9714-R0)
  - Xenon Condition
    - Rated core power equilibrium xenon used at off-rated condition
    - Consistent with runback into region susceptible to instability
  - Feedwater Temperature
    - Performed at off-rated equilibrium feedwater temperature
    - Nominal feedwater heating

Special Events
Core Thermal Hydraulic Stability Option III

DIVOM Calculation

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Special Events
Core Thermal Hydraulic Stability Option III

DIVOM Calculation

- DIVOM Calculation (continued)
  - Hot Channel Calculation
    - Boundary conditions from RAMONA3 used to drive BISON hot channel analysis

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Special Events
Core Thermal Hydraulic Stability Option III

DIVOM Calculation

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Special Events
Overpressure Protection - Acceptance Criteria

- The design bases are met by imposing the following acceptance criterion
  - Peak reactor vessel pressure ≤ 110% vessel design pressure

Special Events

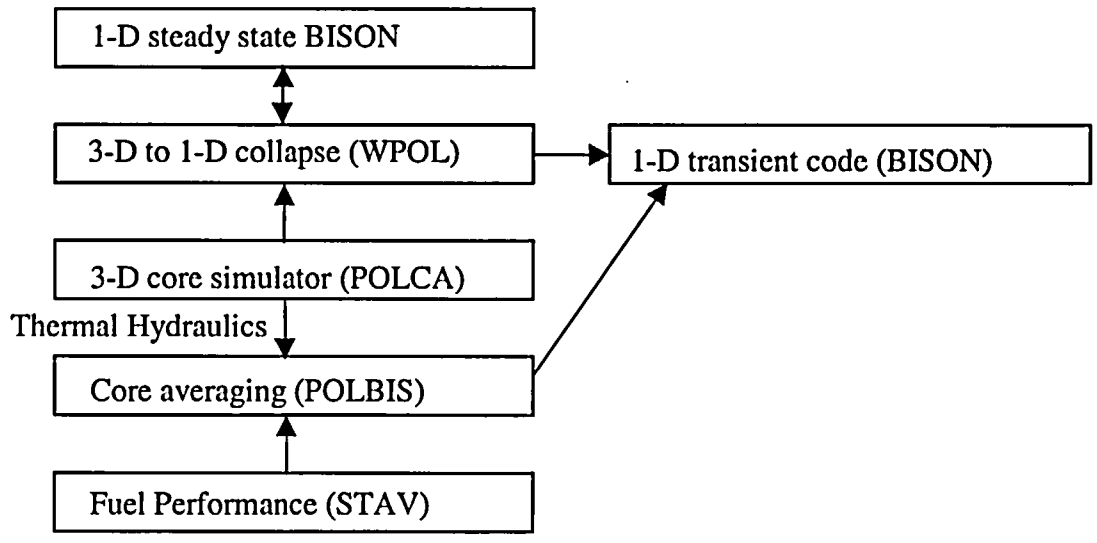
Overpressure Protection - Analysis Methodology

- Main computer codes
  - BISON - for core average system response calculations in 1-D
  - WPOL - for 3-D to 1-D collapse of POLCA cross sections for input to BISON
  - POLBIS - for performing volume-weighted averaging of core-dependent inputs, e.g., pellet and cladding diameters, loss coefficients, and flow areas
- Licensing topical reports
  - RPA 90-90-P-A (models and qualification)
  - CENPD-292-P-A (models amendment)
  - CENPD-300-P-A (methodology and sample applications)

Special Events

Overpressure Protection - Analysis Methodology

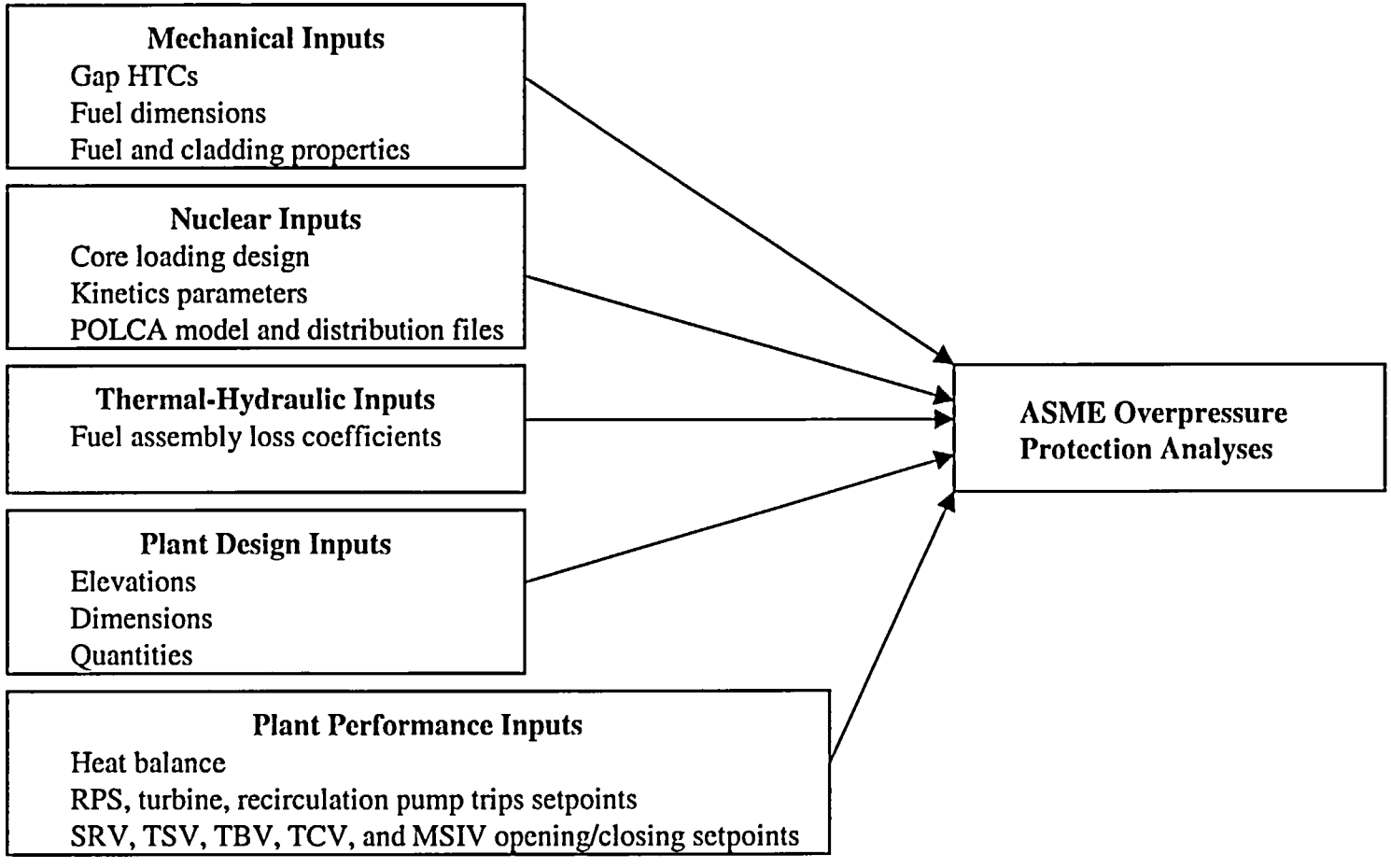
- ASME overpressure protection analysis code interfaces



Special Events

Overpressure Protection - Analysis Methodology

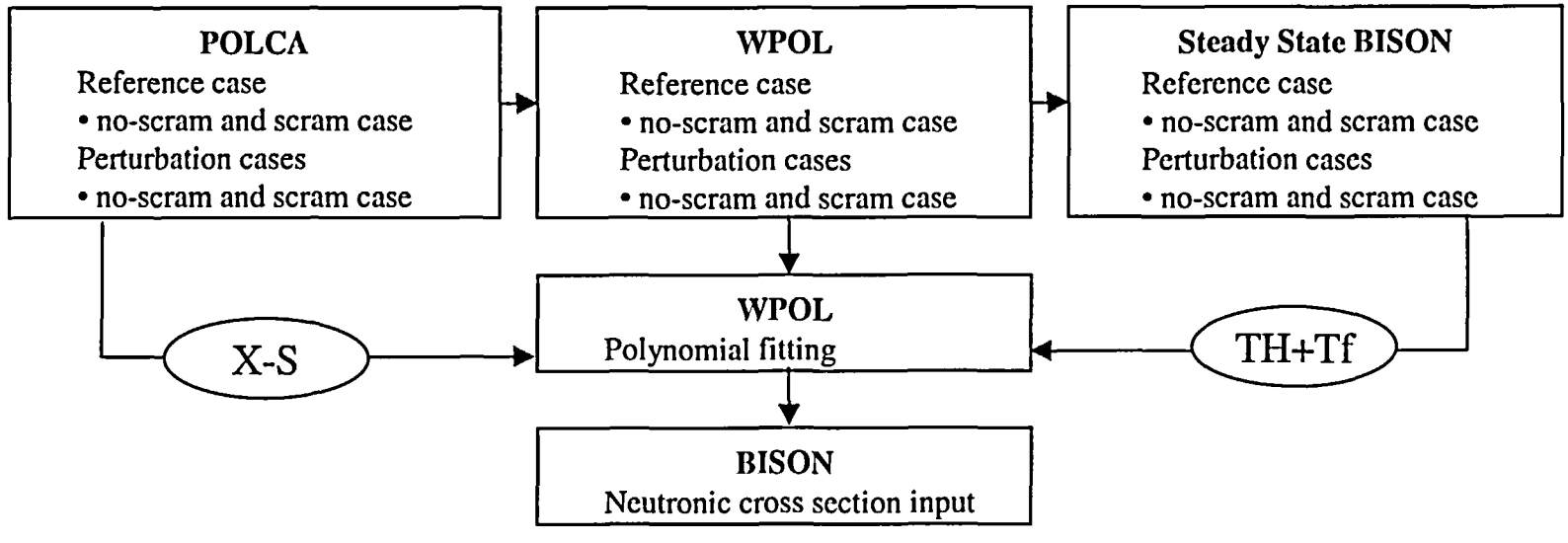
- Analysis inputs



Special Events

Overpressure Protection - Analysis Methodology

- Analysis process - BISON cross section calculations



Special Events

Overpressure Protection - Analysis Methodology

- Analysis process - BISON system response calculations

BISON Basedeck

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BISON Change Decks
- Core-dependent data
- Burnup-dependent data
- Tuning data

↓

BISON Cross Sections
- Neutronic cross section data
- Operating point data

↓

BISON Perturbation Decks
- Initial water level
- Valve closures / openings
- Reactor, feedwater, and pump trips
- Scram rod insertion

↓

BISON System Response
- Peak system pressure determined for comparison to plant design pressure

Special Events

Overpressure Protection - Analysis Methodology

- Analysis outputs
  - Peak system pressure
    - Steam dome pressure
    - Lower plenum pressure
    - Steam line pressure

Special Events

Overpressure Protection - Analysis Methodology

- ASME overpressure protection analyses
  - Typically set by the Main Steam Isolation Valve Closure (MSIVC) event

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Special Events
Standby Liquid Control System Capability

- Two independent reactivity control systems are required in BWRs
  - –Control blades (normal operation/emergency mechanical system)
  - –SLCS (emergency chemical backup)
- As part of the reload licensing methodology, Westinghouse verifies the viability of the Standby Liquid Control System

Special Events
Standby Liquid Control System Capability

- Design Bases
  - The SLCS shall be capable of shutting the reactor down from the most reactive operating state at any time in cycle life

- Acceptance Criteria
  - The acceptance limit is a calculated reactivity demonstrating that the reactor is shutdown (with adequate margin) for the most reactive moderator temperature at any time during the cycle for the minimum boron concentration in the plant SLCS

Special Events
Standby Liquid Control System Capability

- Licensing Topical Report and Supporting Documents
  - –CENPD-300-P-A, Section 9.4
  - –10 CFR 50, Appendix A, General Design Criteria 26 and 27
  - –BWROG-98001
- Computer Codes
  - –POLCA7 (3D Core Simulator)
  - –PHOENIX4 (Lattice Physics Code)

Special Events
Standby Liquid Control System Capability

- Analysis Goal
  - –The core must be shutdown at any temperature between hot operating and cold shutdown conditions (subcriticality demonstrated at various temperatures)

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Special Events
Standby Liquid Control System Capability

- Analysis Process
  - –CENPD-300-P-A specifies that SLCS calculations can be

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Special Events
Standby Liquid Control System Capability

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Special Events
ATWS

The licensed Reload Methodology is to confirm that Westinghouse fuel is bounded by the existing ATWS analysis of record.

- Evaluation of the core response to postulated events having rapid changes in neutron flux; the limiting pressurization transients are analyzed.

- Evaluation aimed to confirm that introduction of new fuel design does not invalidate the plant's existing ATWS analysis.

Special Events
ATWS

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Special Events
ATWS

Core response:

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Special Events
ATWS

Core response (continued):

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Special Events
ATWS

Containment response:

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Special Events
ATWS

Containment response (continued):

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Special Events
ATWS

Conclusions:

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Special Events
ATWS

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Special Events
ATWS

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Special Events
ATWS

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Attachment of Additional Support Information from Previous Westinghouse-NRC Meetings

Mixed Core Treatment Summary

NRC/Westinghouse Meeting
Rockville, MD
December 8, 2004

Mixed Core Treatment Summary - Topics

- Plant and Legacy Fuel-type Specific Analyses
  - Mechanical compatibility with legacy Fuel
  - T-H compatibility with legacy Fuel
  - Legacy Fuel CPR correlation
  - Nuclear Model Calibration
  - LOCA Systems Calculation
  - Seismic/LOCA Evaluation
- Cycle-Specific Analyses
  - Confirmation of plant specific conclusions
  - Nuclear Design and Licensing Evaluation
  - Dynamic Analyses

Mixed Core Treatment – Plant Specific

Mechanical compatibility with Legacy Fuel and Plant

- Legacy Fuel
  - Axial Fit-up and Radial Envelopes
- Plant
  - Reactor Internals (Control Blades, detectors, fuel support piece, upper Core Grid)
  - Bound Hydraulic lifting Loads
  - Bound Stress and Strain
  - Bound Fatigue
  - Bound Fretting Wear
  - Bound Fuel Rod Performance

Mixed Core Treatment– Plant Specific
Mechanical compatibility with Legacy Fuel

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Mixed Core Treatment– Plant Specific
Mechanical compatibility with Legacy Fuel

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Mixed Core Treatment– Plant Specific
T-H compatibility with legacy Fuel

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Mixed Core Treatment– Plant Specific

Legacy Fuel CPR correlation

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Mixed Core Treatment– Plant Specific

Nuclear Model Calibration

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Mixed Core Treatment– Plant Specific

LOCA Systems Calculation

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Mixed Core Treatment– Plant Specific Seismic-Loca Evaluation

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Mixed Core Treatment– Cycle-specific

Confirmation of Plant-specific conclusions

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Mixed Core Treatment– Cycle-specific

Nuclear Design and Licensing Evaluation

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Mixed Core Treatment– Cycle-specific

Nuclear Design and Licensing Evaluation

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Mixed Core Treatment– Cycle-specific

Dynamic analyses

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Mixed Core Treatment– Cycle-specific

Dynamic analyses

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Westinghouse BWR SLMCPR Analysis

NRC/Westinghouse Meeting
Rockville, MD
December 8, 2004

SLMCPR Analysis - Topics

- SLMCPR Definition
- Licensing Basis and Westinghouse Correspondence
- Westinghouse Methodology
  - Westinghouse Fuel
  - Legacy Fuel
- Sample Analyses
  - Typical Results
  - Understanding the Results

Specified Acceptable Fuel Design Limits

- CENPD-300-P-A identifies SAFDLs in Westinghouse Methodology:
  - –< 1% Clad Strain (Fuel Rod Failure) - LHGR limit
  - –No Fuel centerline melt (Pellet Integrity)- LHGR Limit
  - –170 cal/g (Rapid Energy Deposition) - Pellet Enthalpy Limit
  - –Avoid Boiling Transition (Clad Overheating) - CPR Limit

$$CPR = \frac{Power\ (Dryout)}{Power\ (Operating)}$$

SLMCPR Regulatory Basis

•NRC Standard Review Plan (Sections 4.2 and 4.4) Provides Guidance on demonstrating dryout margin

- "Uncertainties in the values of process parameters, core design parameters, and calculational methods used in the assessment of thermal margin should be treated with at least 95% probability at a 95% confidence level."
- "..assurance that there be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) or transition condition during normal operation or anticipated operational occurrence."

SLMCPR Regulatory Basis

•NRC Standard Review Plan (Sections 4.2 and 4.4) provides two approaches for demonstrating compliance with 95%/95% criterion on avoiding boiling transition

- ... "a limiting (minimum) DNBR, CHFR, or CPR is to be established such that at least 99.9% of the fuel rods in the core would not be expected to experience departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences" = Approach followed in U.S. BWRs
- direct demonstration that hot rod at the DNB, CHF, or CPR limit will avoid dryout during normal operation and AOOs 95% of the time with 95% confidence

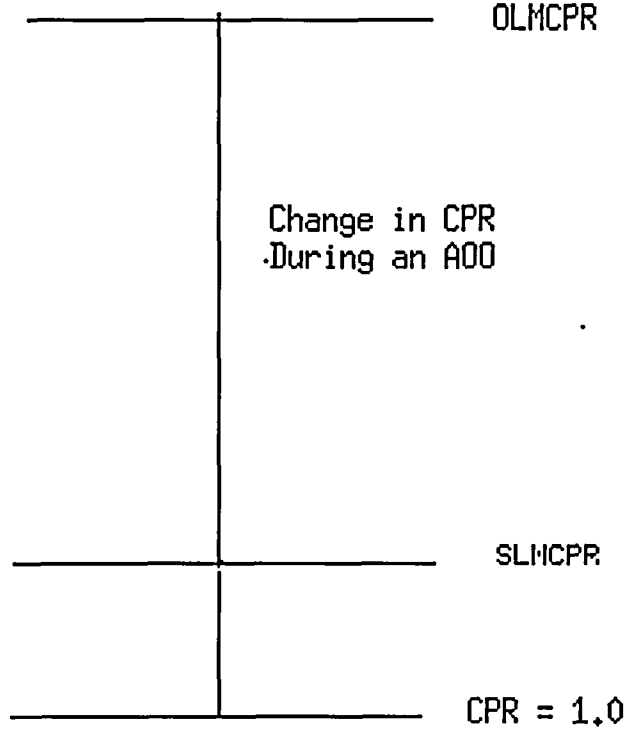
Westinghouse SLMCPR Correspondence

- CENPD-300-P-A: Reference Safety Report for Boiling Water Reactor Reload Fuel, July 1996
  - –Section 5.3.1 and RAI F11 and F13
- CENPD-389-P-A: 10x10 SVEA Fuel Critical Power Experiments and CPR Correlations: SVEA-96+, September 1999
  - –RAI D-13
- CGS Cycle 14 and 15 Licensing Amendment Requests
- HCGS Cycle 10 Licensing Amendment Request

Westinghouse SLMCPR Methodology

SLMCPR Safety Limit MCPR Definition

- **BWR Approach to showing 99.9% of fuel rods are not expected to experience Boiling Transition**



Westinghouse SLMCPR Methodology

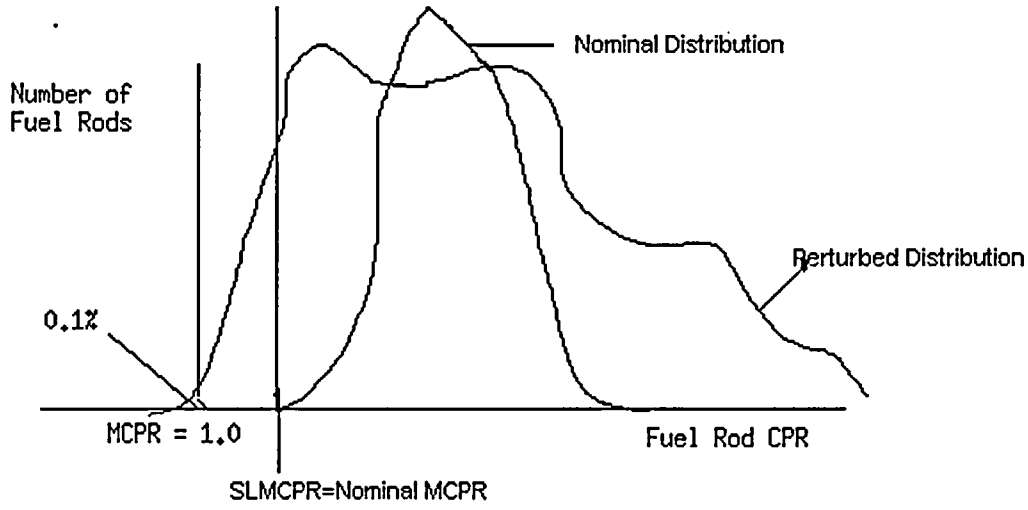
SLMCPR Calculation

- SLMCPR Calculation = Convolution of Uncertainties for nominal fuel rod CPR distribution with MCPR value = $MCPR_0$ to establish CPR distribution. Then $MCPR_0$ established to assure 99.9% of the rods avoid dryout
- Convolution done during cycle to establish most limiting point
- Convolution done using a Monte Carlo choice of perturbed parameters
  - –Core process parameters
  - –Assembly design parameters
  - –Calculational methods

Westinghouse SLMCPR Methodology

SLMCPR General Monte Carlo Method

- Conceptual SLMCPR Application
  - Starting with nominal fuel rod CPR distribution, convolute uncertainties which significantly affect fuel rod CPRs to establish distribution of fuel rod CPRs
  - SLMCPR for that nominal fuel rod CPR distribution is the nominal MCPR such that 99.9% of the fuel rods avoid boiling transition



Westinghouse SLMCPR Methodology

Westinghouse Strategy for Cycle SLMCPR

- Perform SLMCPR Calculation Throughout Cycle
  - Systematically finds limiting combination of bounding assembly CPR and relative fuel rod CPR distributions
- Use Conservative Nominal Assembly CPR Distribution at Each Statepoint
  - Use design assembly power distribution with margin to OLMCPR

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Westinghouse SLMCPR Methodology

Westinghouse Strategies for Mixed Core SLMCPR

- Methodology allows treatment of any number of different fuel types in SLMCPR calculation
- Approved CENPD-300-P-A Strategy
  - Use SLMCPR from previous cycles for non-Westinghouse resident fuel a, c

Westinghouse SLMCPR Methodology

Uncertainties

- Rod CPR distribution for a MC Trial - convolute uncertainties

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Westinghouse SLMCPR Methodology

Westinghouse Strategies for Mixed Core SLMCPR

- SLMCPR for non-Westinghouse Fuel

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Westinghouse SLMCPR Methodology

Sample Analysis

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Westinghouse SLMCPR Methodology

Sample Analysis

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Westinghouse SLMCPR Methodology

Sample Analysis

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Westinghouse SLMCPR Methodology

Sample Analysis

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Westinghouse SLMCPR Methodology

Sample Analysis

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Backup

Westinghouse SLMCPR Methodology

SLMCPR General Monte Carlo Method

- Want uncertainty in $f(x_1, x_2, \ldots x_i \ldots)$
- f calculated many times. Each calculation based on perturbed values of x_i about their nominal value. Perturbed values selected randomly such that individual probability distribution functions of each x_i ($p(x_i)$) are reproduced after many trials
- f is calculated a sufficient number of times to establish its distribution

Westinghouse SLMCPR Methodology

SLMCPR General Monte Carlo Method

- Typical Monte Carlo Strategy for parameter selection:
  - n = Random Number
  - x_i = corresponding value of x_i

$$n = \int_{-\infty}^{x_i} p(x)dx$$

Westinghouse SLMCPR Methodology

SLMCPR Overview

- Nominal CPR Distribution

Westinghouse SLMCPR Methodology

Best Estimate Mixed Core Confirmation

- Confirmation that separate, bounding SLMCPRs for W and non-W protect the actual mixed core SLMCPR

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Westinghouse SLMCPR Methodology

Best Estimate Mixed Core Confirmation

- Determination of Effective Legacy Fuel Correlation Uncertainty

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Westinghouse SLMCPR Methodology

Best Estimate Mixed Core Confirmation – Sample Analysis

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