February 2, 1994

Docket No. 50-213

LICENSEE: CONNECTICUT YANKEE ATOMIC POWER COMPANY

FACILITY: HADDAM NECK PLANT

SUBJECT: SUMMARY OF JANUARY 27, 1994 MEETING REGARDING THE USE OF WESTINGHOUSE FUEL AT THE HADDAM NECK PLANT

On January 27, 1994, the NRC staff met with Connecticut Yankee Atomic Power Company (CYAPCO/licensee) to discuss their proposed use of Westinghouse fuel at the Haddam Neck Plant. The current fuel vendor is B&W and the licensee has decided to start using Standard Westinghouse 15X15 Vantage 5 fuel starting the next outage scheduled to start November 15, 1994. The licensee stated that several submittals have been submitted for staff review including the nuclear analyses methodology, thermal hydraulic methodology, and new and spent fuel storage Technical Specification change. In addition, a TS change relating to the use of the new fuel will be submitted in April 1994.

The licensee stated that they have attempted to minimize staff review by minimizing changes to the models and using previously approved Westinghouse methodologies when changes were required. The licensee noted that the most significant changes in fuel are the use of integral fuel burnable absorber and the use of a zircaloy skeleton instead of the stainless steel skeleton used by B&W. The staff stated that these changes were significant enough that we would also like to review the Technical Report Supporting Cycle Operation (TRSCO), which would include the fuel mechanical design prior to start-up. Normally the TRSCO is only submitted for staff information as it provides the cycle specific parameters removed from TS as allowed by GL 88-16. The staff also discussed the schedule and noted that most of the schedule appears to be reasonable except for the review of the fuel mechanical design and TRSCO. The staff needs CYAPCO to provide these reports as soon as possible to support a January 1994 start-up. The licensee stated that they will make every effort to support the staff in completing this review to support their outage currently planned for October/January 1994.

Enclosed is CYAPCO's handout for the meeting and the attendance list.

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Alan B. Wang, Project Manager
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Alan Wang

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Enclosure 1

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Michael Kai	Northeast Utilities
Tom Cleary	Northeast Utilities
Gerry Van Noordennen	Northeast Utilities
Bill Herwio	CYAPCO

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MEETING WITH THE NRC

ON

HADDAM NECK PLANT

CYCLE 19 RELOAD

Northeast Utilities Service Company
Connecticut Yankee Atomic Power Company

January 27, 1994

PARTICIPANTS

G. P. van Noordennen (Gerry)	Supervisor, Nuclear Licensing
J. J. Parillo (Joe)	Supervisor, Nuclear Analysis
M. S. Kai (Mike)	Supervisor, Safety Analysis
W. M. Herwig (Bill)	Supervisor, Reactor Engineering
T. G. Cleary (Tom)	Lead Licensing Engineer, Connecticut Yankee

AGENDA

Introduction	T. G. Cleary
Purpose of Meeting	T. G. Cleary
Background	W. M. Herwig
Cycle 19 Overview	
Mechanical Design Overview	W. M. Herwig
Cycle 19 Nuclear Design Fuel Storage	J. J. Parillo
Reload Analysis	M. S. Kai
Plant Modifications	W. M. Herwig
Remaining Activities	G. P. van Noordennen
Schedule	G. P. van Noordennen
Summary/Conclusions	G. P. van Noordennen
Discussion	All

PURPOSE OF MEETING

TO DISCUSS:

- Fuel Vendor and Design Changes
- Methodology Upgrades
- Planned Cycle 19 Licensing Activities
- Remaining Activities and Schedule

BACKGROUND

- Fuel Assembly Design Change for Cycle 19 is the Next Step in the Reload Related Technology and Hardware Upgrade That Began In the Mid 1980's

- Past Upgrades Include

  - Development of In-house Reload Capability

  - Revised Accident Analysis (Chapter 15 and Small and Large Break LOCA)

  - Use of Westinghouse Physics Technology

  - Zircaloy Cladding Conversion

- The Methodology Upgrades and Fuel Assembly Design Changes Discussed Today Establish the Foundation for 24 Month Cycles Beginning in Cycle 20

CYCLE 19 OVERVIEW

- Fuel Mechanical Design Changes
- Cycle 19 Nuclear Design
- Cycle 19 Reload Accident Analyses
- Fuel Storage
- Fuel Related Plant Modifications

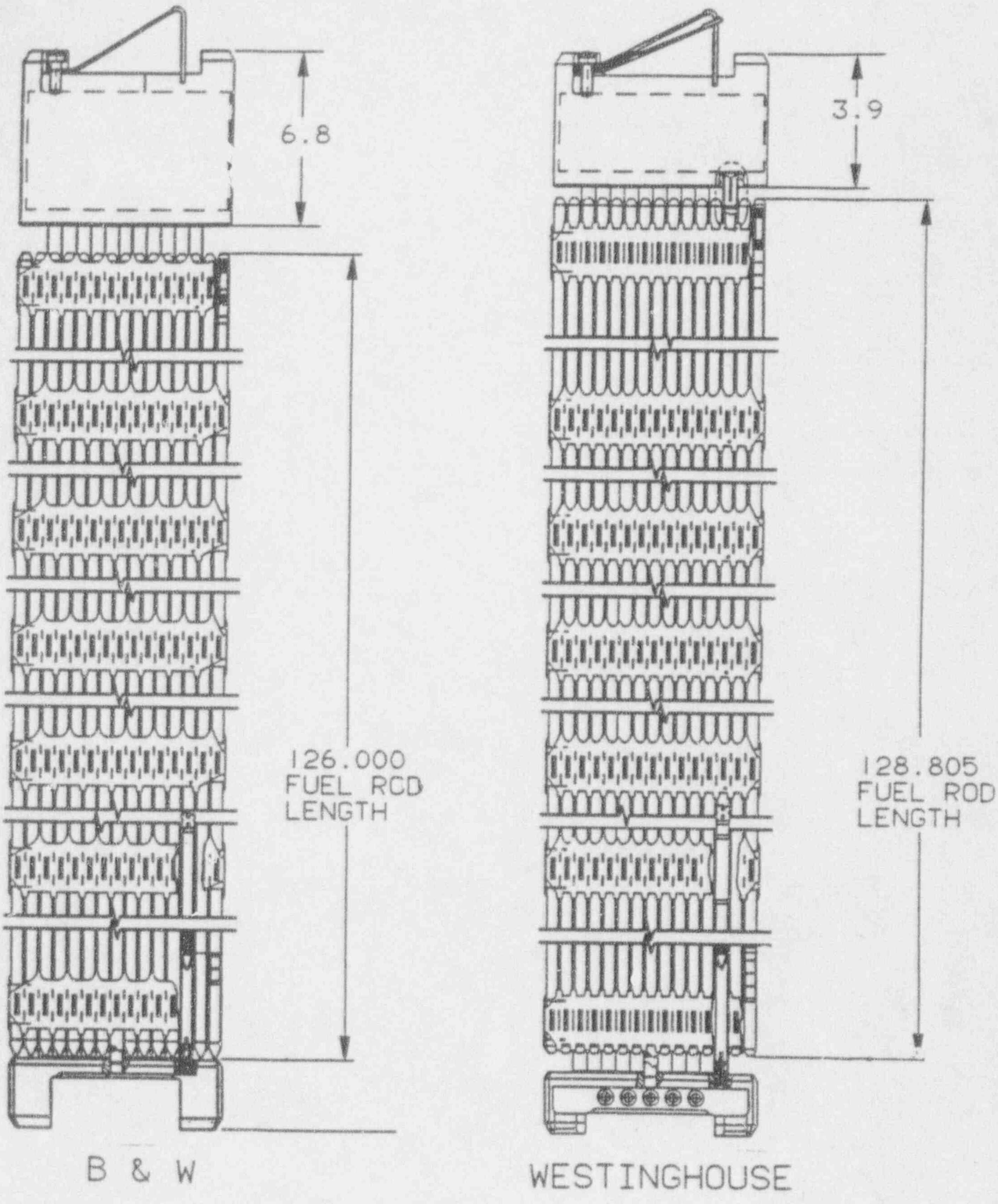
CYCLE 19 FUEL MECHANICAL DESIGN CHANGES

- Fuel Vendor Change From BWFC to Westinghouse
- Standard Westinghouse Fuel Mechanical Design Methodology
- Design Improvements

<u>BWFC</u>	<u>Westinghouse</u>
SS Guide Tubes, Instrument Tube	Zircaloy Guide Tubes, Instrument Tube
Inconel Spacer Grids	Zircaloy Spacer Grids
No Burnable Poison	Integral Fuel Burnable Absorber

FUEL ASSEMBLY DESIGN FEATURES

- Design Based on Standard Westinghouse Vantage 5
  - Improved Zr-4 Cladding
  - High Thermal Performance Zr-4 Grids
  - Debris Resistant Bottom Nozzle
  - Debris Resistant Fuel Rod Bottom End Plug
  - Removable Top Nozzle
  - Integral Fuel Burnable Absorber (IFBA)
  - Low Profile Top Nozzle
- Deletion of Thimble Plugging Devices
- New Secondary Source Assemblies



B & W

WESTINGHOUSE

CYCLE 19 NUCLEAR DESIGN

- NUSCO has performed core reload design for Haddam Neck Cycles 15, 16, 17, and 18 using Westinghouse's ARK/TORTIS software.

- Agreement between plant measurements and physics predictions were always very good. This good agreement includes mixed cores of zircaloy clad fuel and stainless steel clad fuel in Cycles 17 and 18.

- To remain technically current, NUSCO has upgraded to the PHOENIX-P/ANC Westinghouse core reload design software.

- The newer PHOENIX-P/ANC reload physics methodology is the standard methodology currently used by Westinghouse for all of their reload designs. Topical Reports on the PHOENIX-P/ANC code packages approved by the NRC.

- Haddam Neck is currently operating in Cycle 18. NUSCO has designed Cycle 19 using the new PHOENIX-P/ANC software.

- Cycle 19 is planned as a 1/3 core reload, 490 EFPD. Fuel enriched to 4.2 w/o and 4.6 w/o. Most of the 4.2 w/o fuel has 48 IFBA rods per assembly. Each IFBA is "1.5X".

- In 1986, NUSCO submitted the topical "Physics Methodology for PWR Reload Design, NUSCO-152." This topical used ARK/TORTIS.

CYCLE 19 NUCLEAR DESIGN (cont'd.)

- In January 1994, NUSCO submitted Addendum 3 to NUSCO–152. This topical re-analyzed Haddam neck Cycles 15, 16, 17, and 18 with PHOENIX-P/ANC. This Addendum compares physics parameters calculated by PHOENIX-P/ANC to plant measurements and also, where relevant, to the predictions from the older ARK/TORTIS software.

- As shown in NUSCO-152, Addendum 3, agreement between plant measurements and PHOENIX-P/ANC predictions is very good. The agreement is similar or slightly better than the agreement between plant measurements and the older ARK/TORTIS predictions. Typical agreement between measurements and PHOENIX-P/ANC predictions are:

— Critical Boron Concentration at Startup	+/- 25 ppm
— Isothermal Temperature Coefficient at Startup	+/- 1 pcm/°F
— Control Rod Worths at Startup	+/- 3 %
— Axial Offset at HFP	+/- 1 % AO
— Peak FDH at HFP	+/- 1 %
— Peak FQ at HFP	+/- 2 %
— Average Radial Power Distribution for Individual Fuel Assemblies at HFP	+/- 1 %
— End of Cycle Burnup	+/- 100 mwd/mtu

- Summary

CYCLE 19 ACCIDENT ANALYSIS

CURRENT METHODOLOGY

- Approved by the NRC
- RETRAN for System Analysis
- VIPRE With W3-L For Thermal Hydraulic Analysis

IMPACT OF FUEL DESIGN CHANGE

- The Grid Design Change Impacts the Thermal Hydraulic Analysis Methodology
- RETRAN Analysis Unaffected
- VIPRE Needs to be Benchmarked for the New Grid

ANALYSIS PLAN

- Minimize Changes to Models As Much As Possible to Minimize Required Review

- Where Changes Are Required, Use Westinghouse Methodology That Has Been Approved By The NRC

PLANNED CHANGES

- Addition of WRB-1 DNBR Correlation as a User Added Subroutine to VIPRE with no Changes to VIPRE Itself
- Maintain Current VIPRE Model With Minor Changes to Reflect the New Fuel Design and Consistency with Standard Westinghouse Methodology
- Implement the Westinghouse Methodology for Transition Core Evaluations
- Implement the Westinghouse Methodology for Thermal Design Procedure for Margin

THERMAL DESIGN PROCEDURE

- Standard Westinghouse Methodology Previously Approved by NRC
  - Improved Thermal Design Procedure WCAP 8567-P-A
  - Mini-Revised Thermal Design Procedure WCAP 12178-P-A
- Only Uncertainties Associated with Peaking Factors Will Be Statistically Combined
- All Other Uncertainties Will Be Treated in the Standard Conservative Method
- Methodology Achieves Required Margin While Minimizing the Required Review
- System Uncertainties Still Treated in the Standard Conservative Method — No Changes to RETRAN Analysis Necessary

CHANGES SUBMITTED FOR APPROVAL

- Topical Report Benchmarking VIPRE With WRB-1
- Technical Specification Bases Change Reflecting the Use of WRB-1 and the Thermal Design Procedure

FUEL STORAGE

- PURPOSE

  — Allow up to 5.0 w/o U-235 enriched fuel to be stored in the spent fuel pool and in the new fuel storage racks.

  — Proposed revision to Technical Specifications submitted. Criticality analysis performed by Westinghouse.

- BACKGROUND

  — The existing spent fuel storage racks and new fuel storage racks are allowed to store up to 4.0 w/o U-235 enriched fuel assemblies with stainless steel clad fuel rods and up to 3.9 w/o U-235 enriched fuel assemblies with Zircaloy clad fuel rods.

  — No credit is currently taken for fuel burnup or soluble boron in the spent fuel pool.

  — Review current spent fuel storage racks.

  — Review current new fuel storage racks. Minimum center-to-center spacing of 18.625". Polyvinyl-Chloride (PVC) tubes 12.75" diameter used in locations which will store new fuel.

NEW FUEL STORAGE RACK ANALYSIS AND RESULTS

- The limiting events involve the inadvertent introduction of water to the normally dry new fuel racks. The criticality analysis shows K_{eff} is less than .95 for the accidental full water density flooding scenario and less than .98 for the accidental optimum moderation flooding scenario.

- The accidental full water density flooding scenario results in a K_{eff} equal to .9477, including all biases and uncertainties.

- The accidental optimum moderation flooding scenario results in a K_{eff} equal to .9237, including all biases and uncertainties.

- Fresh fuel of up to 5.0 w/o U-235 enrichment may be stored in the existing new fuel racks without any physical modifications, with 2 limitations:

  — New fuel may be stored in only certain locations within the new fuel storage racks.

  — Fuel enriched to greater than 4.6 w/o U-235 must have a minimum amount of IFBA fuel rods per fuel assembly.

NEW FUEL STORAGE RACK ANALYSIS AND RESULTS (cont'd.)

- Both limitations have been added to the proposed TSs.
- The acceptability of the criticality analysis crediting that not all new fuel storage locations may be used is based on:
  - Procedural guidance which will explicitly exclude fuel in the prohibited locations.
  - The PVC liners are an unmistakable visual reference. The PVC liners are present only in allowed new fuel storage locations. Locations not allowed to store new fuel do not have PVC liners.
  - In the unlikely case where a new fuel assembly is stored in a prohibited storage location, that by itself does not cause the rack K_{eff} to exceed its limit. The addition of an optimum moderation accident would be required along with the mislocated fuel to cause rack K_{eff} to exceed its limit.

NEW FUEL STORAGE RACK ANALYSIS AND RESULTS (cont'd)

- Westinghouse criticality analysis uses KENO Va for reactivity determination. Calculations for crediting IFBA use of PHOENIX code. Benchmark calculations are discussed in the Westinghouse criticality analysis.

- The most reactive fuel design planned for Haddam Neck is used in the analysis to ensure conservative results. The fuel assembly is modeled at its most reactive point in life.

- No reactivity credit is taken for fuel assembly spacer grids or sleeves.

- No reactivity credit is taken for the new fuel storage rack structural materials or the PVC liners.

SPENT FUEL POOL ANALYSIS AND RESULTS

- Westinghouse criticality analysis for the Spent Fuel Pool (SFP) shows that the SFP is maintained with a K_{eff} less than or equal to .95 under normal and accident conditions.

- The results of the spent fuel pool criticality analysis under nonaccident conditions is K_{eff} = .9457 including all biases and uncertainties.

- The limiting accident is a fuel assembly drop between the pool wall and racks. The use of boron in the SFP water is used to mitigate the reactivity consequences of this event.

- Fresh fuel of up to 5.0 w/o U-235 enrichment may be stored in the existing spent fuel pool without any physical modifications, with 2 limitations:

  - Credit for fuel burnup would be required. An alternating row pattern was selected. A given row may contain fuel up to 5.0 w/o U-235 fresh fuel. The adjacent row has a burnup requirement that is a function of initial enrichment.

  - A minimum o 500 ppm boron in the spent fuel pool is required during fuel handling to mitigate the potential criticality consequences of certain fuel handling accidents. A value of 800 ppm boron was chosen for the proposed LCO to provide margin.

SPENT FUEL POOL ANALYSIS AND RESULTS (cont'd.)

- An alternating row pattern was selected to make administrative controls of the SFP as easy as possible. With an alternating row pattern, new fuel will be lined up in rows and, therefore, the fuel handlers would have an obvious visual cue that would prevent new fuel from being mislocated in the spent fuel pool.

- The burnups to be credited are relatively small. Currently all fuel in the SFP meet the proposed burnup requirements. Therefore, a misloading event in the SFP is not currently even possible.

- Loading of new fuel into the SFP will have special controls. Procedures will require that prior to moving new fuel to the SFP, a region of the SFP will be designated for storage of the new fuel. Within this designated region, all restricted fuel rows must be completely full with qualified fuel, leaving no possibility for misloading new fuel.

- Addition of form to document qualification of fuel assemblies that meet the requirements of TYPE I FUEL.

- All fuel movement in the SFP will require independent verification that the fuel assembly is going into (or is being removed from) the correct SFP location.

SPENT FUEL POOL ANALYSIS AND RESULTS (cont'd.)

- Should the unlikely event occur that fuel is misloaded, the effect on the SFP K_{eff} is addressed in the criticality analysis. The misloading of a 5 w/o fresh fuel assembly in <u>every other</u> location of the restricted rows results in an increase of .026 K_{eff}. The limiting accident condition is not a fuel misloading event, but the dropping of a fresh fuel assembly between the SFP wall and adjacent rack, which requires 500 ppm of boron in the SFP water. The proposed surveillance requirement is 800 ppm Boron.

- About 2000 ppm of Boron is kept in the SFP at all times.

SPENT FUEL POOL ANALYSIS AND RESULTS (cont'd.)

- Westinghouse criticality analysis uses KENO Va for reactivity determination. Calculations for crediting IFBA and fuel burnup use the PHOENIX code. Benchmark calculations are discussed in the Westinghouse criticality analysis.

- The most reactive fuel design planned for Haddam Neck is used in the analysis to ensure conservative results.

- No reactivity credit is taken for fuel assembly spacer grids or sleeves.

- The most reactive temperature condition for the spent fuel pool is the lowest temperatures, and temperature down to 32°F were evaluated.

- Summary

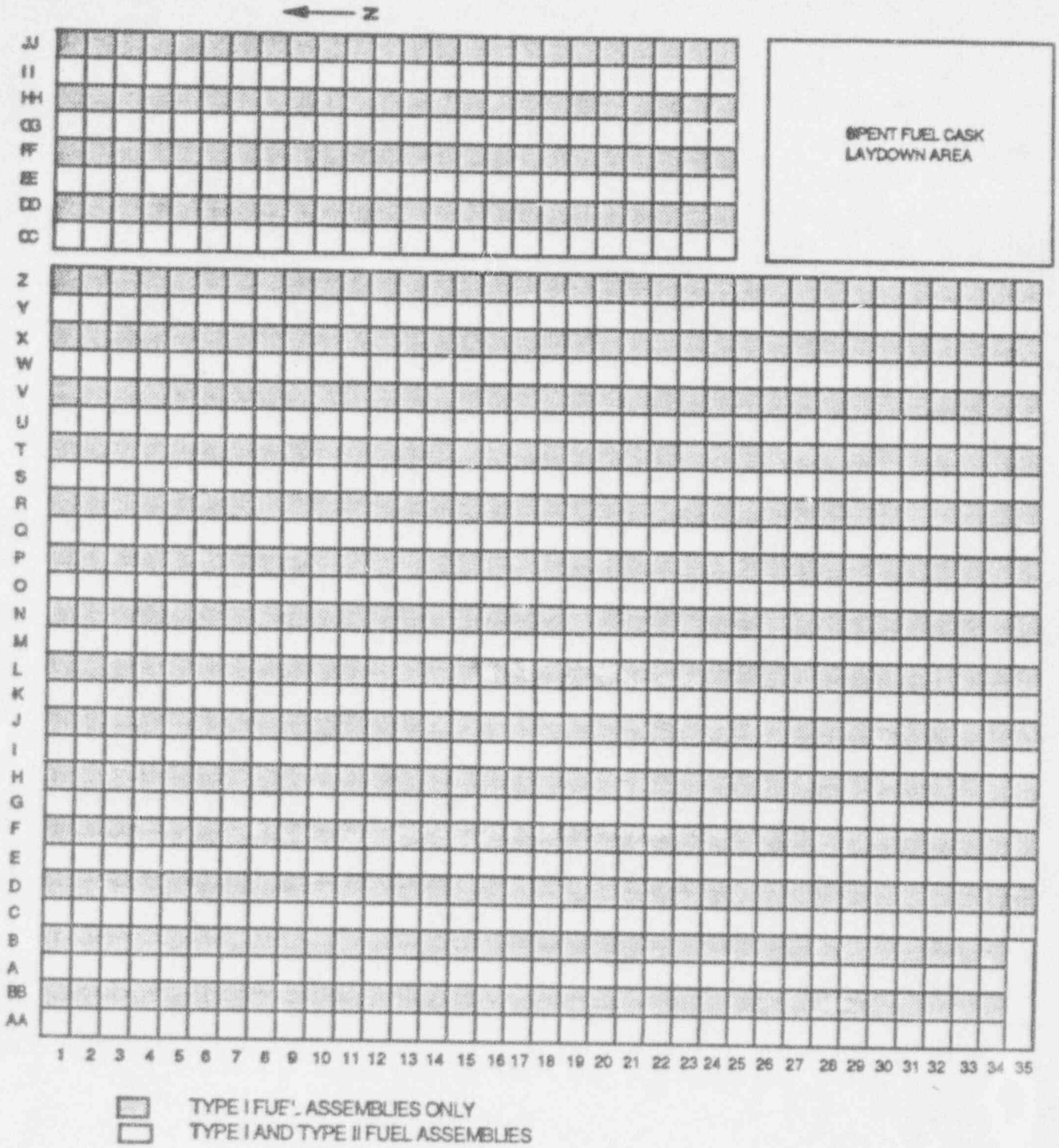


FIGURE 3.9-2 SPENT FUEL POOL RACK ALTERNATING ROW STORAGE CONFIGURATION

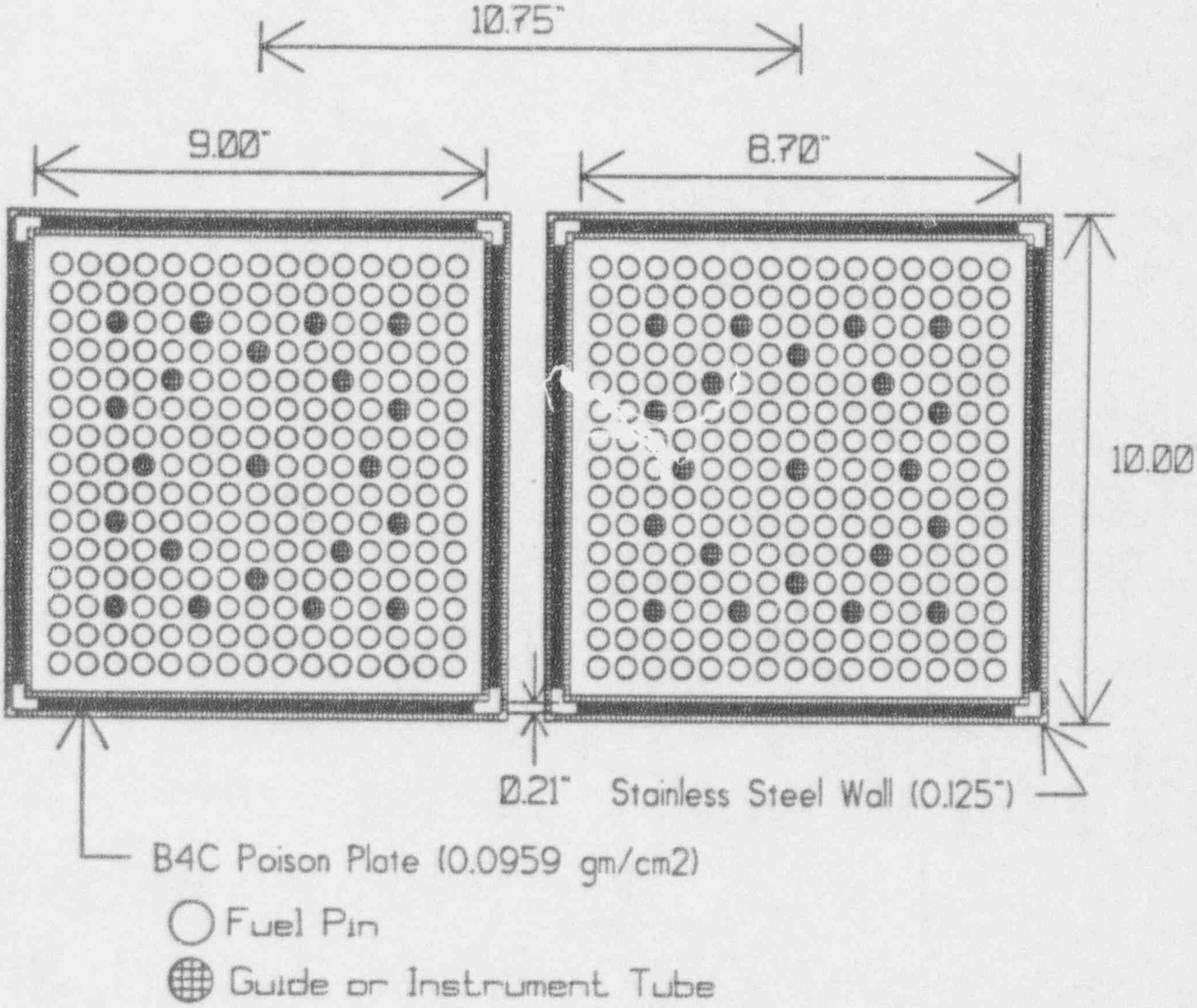


Figure 3 Connecticut Yankee Spent Fuel Storage Cell Nominal Dimensions

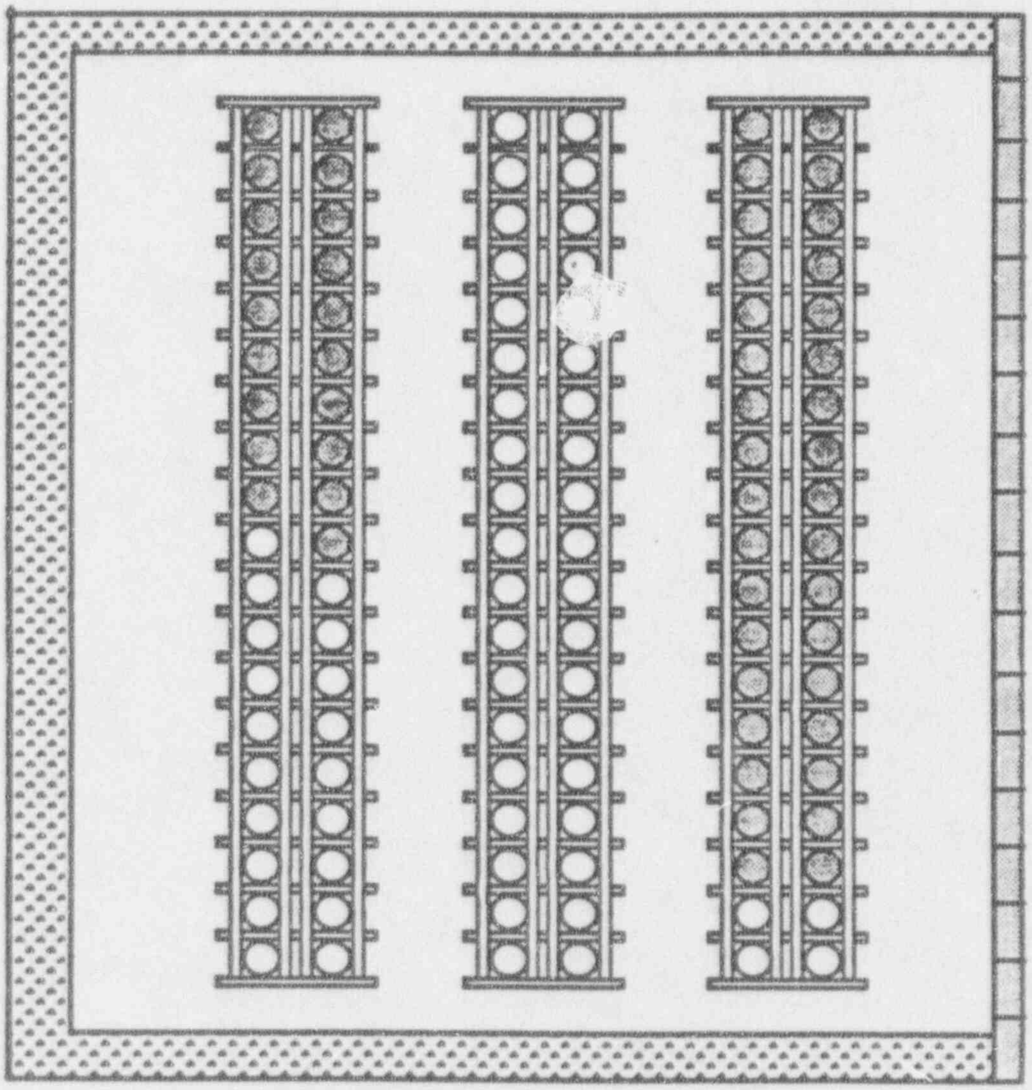


FIGURE 5.6-2 NEW FUEL STORAGE RACK ARRAY LAYOUT

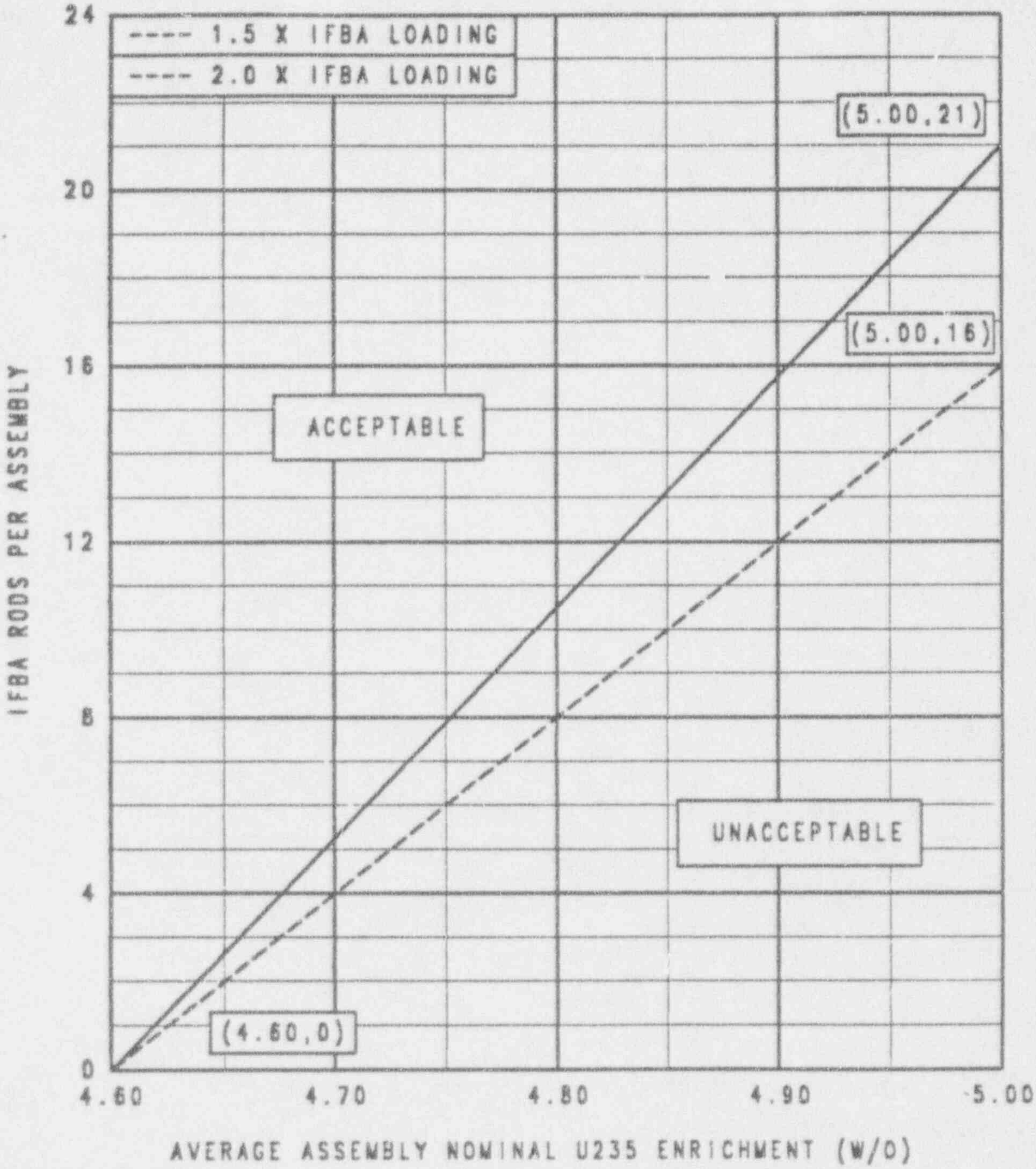


FIGURE 5.6-1 NEW FUEL STORAGE RACK MINIMUM IFBA REQUIREMENTS

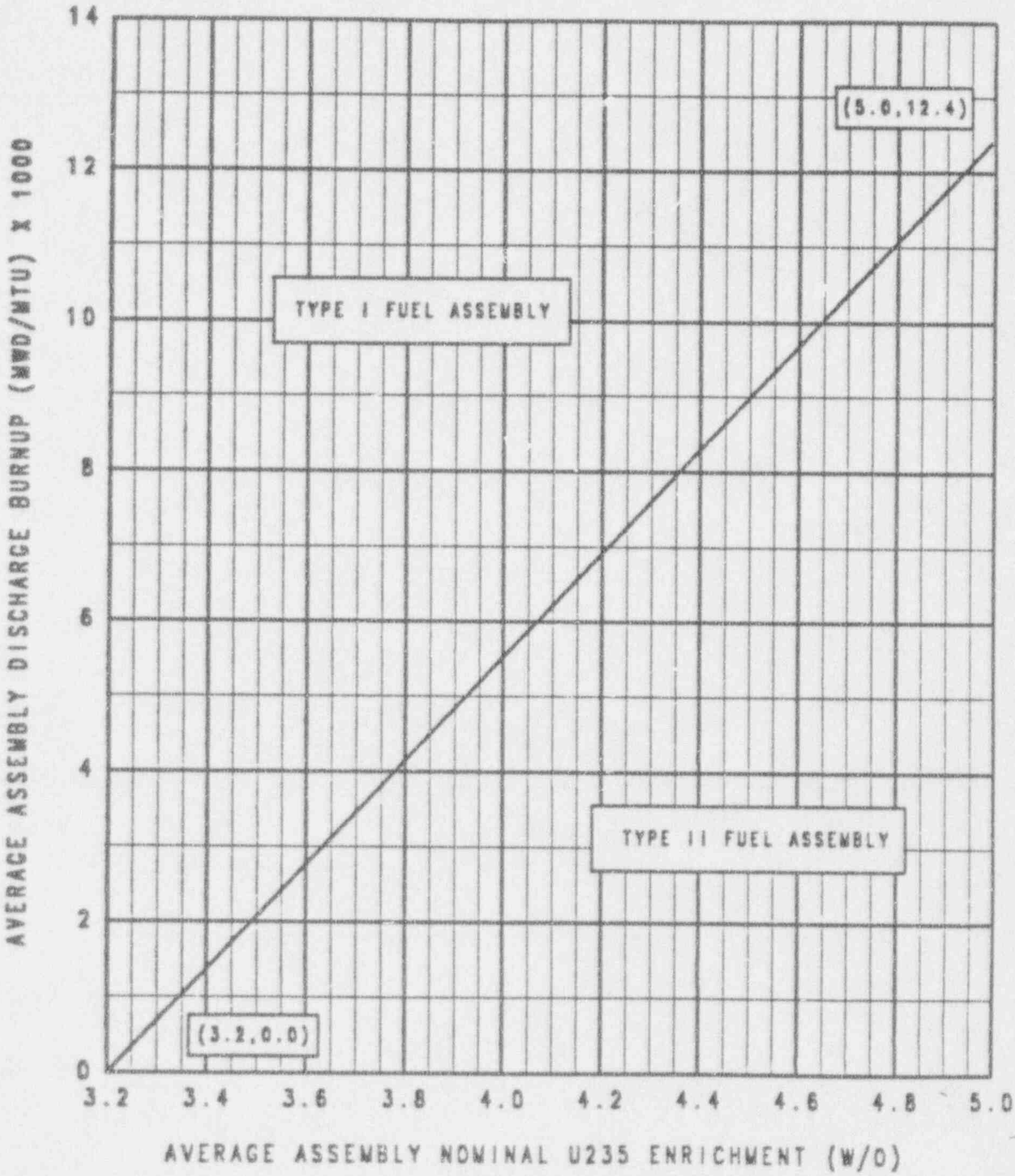


FIGURE 3.9-1 SPENT FUEL POOL RACK MINIMUM BURNUP REQUIREMENTS FOR ALTERNATING ROWS STORAGE CONFIGURATION

OTHER CYCLE 19 RELOAD TECHNICAL SPECIFICATION CHANGES

- Additional proposed technical specification changes will be submitted for the Haddam Neck Cycle 19 reload.

- The proposed technical specification changes to be submitted are expected to be:

  - Section 4.2.2.1.2, Linear Heat Rate uncertainties currently shown in the technical specifications are for B&W fuel. The standard Westinghouse Linear Heat Rate uncertainties need to be added for Westinghouse fuel.

  - Section 5.3.1 Design needs to be updated to reflect the new fuel design and assembly weights.

  - Section 6.9.1.9.b references need to be updated to reflect the NRC approval of the reload topicals.

  - Section 2 Bases need to be updated to reflect use of the new WRB-1 DNB correlation.

FUEL RELATED PLANT MODIFICATIONS

- Fuel/Plant Interface Affected by Zircaloy Skeleton and Low profile Top Nozzle Design Changes
- Manipulator Crane Grapple
  - Design and Fabrication Performed by Original Equipment Supplier
- Fuel Assembly Transfer Cart
  - Modify Fuel Assembly Insert Restraining Plate
- Control Rod Change Fixture in Containment
  - New Lower Section of Guide Tube
- New Handling Tools for New and Spent Fuel

REMAINING ACTIVITIES

- Ongoing — Support NRC Review of Submitted Documents
  - VIPRE Thermal Hydraulic Methodology
  - Nuclear Analysis Methodology
  - Fresh and Spent Fuel Storage Tech Spec Change
- Planned Submittals
  - April 1994 — Tech Spec Change Request
  - Summer 1994 — Fuel Mechanical Design Report
  - September 1994 — Technical Report Supporting Cycle Operation

PROPOSED NRC REVIEW SCHEDULE

• Nuclear Analysis Methodology Upgrade	July 1, 1994
• Fuel Storage Tech Spec	August 1, 1994
• Remaining Approvals	October 15, 1994

— Related Tech Spec Changes

— VIPRE Topical Report

SUMMARY AND CONCLUSIONS

- The Technology and Hardware Upgrades Planned for Cycle 19 are the Next Logical Step in a Process That Began Almost 10 Years Ago
- The Upgrades Will Be Accomplished Using the Application of Previously Approved Westinghouse Methodology
- Normal Reload Analyses are Utilized
- Revised Spent Fuel Pool Storage Method
- Respectfully Request NRC To Perform Necessary Reviews to Support Cycle 19 Startup in January 1995

DISCUSSION