UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 24, 2013

LICENSEE: Entergy Nuclear Operations, Inc.

FACILITY: Indian Point Nuclear Generating Unit No. 2

SUBJECT: SUMMARY OF AUGUST 26, 2013, MEETING WITH ENTERGY NUCLEAR OPERATIONS, INC. AND NETCO ON INDIAN POINT UNIT 2 SPENT FUEL POOL MANAGEMENT (TAC NOS. MF2450 AND MF2451)

On August 26, 2013, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) and representatives of Entergy Nuclear Operations, Inc. and their consultant, NETCO, at NRC Headquarters, Three White Flint North, 11601 Landsdown Street, Rockville, Maryland. The purpose of the meeting was to discuss the licensee's long-term spent fuel pool (SFP) management program. The meeting notice dated August 8, 2013, is available in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML13218A086. A list of attendees is provided as Enclosure 1.

The purpose of the meeting was to discuss the licensee's 4-year SFP management program that focuses on the Unit 2 SFP. The slides used for the licensee's presentation are included in Enclosure 2.

In November 2008, the fuel in the Unit 1 SFPs was moved to dry cask storage onsite and the Unit 1 pools were drained and are being maintained dry. The Unit 2 SFP, which has a capacity of 1,374 fuel assemblies, currently maintains 1,104 fuel assemblies and, as a result, is currently 80 percent full. The Unit 3 SFP, which has a capacity of 1,345 fuel assemblies, currently maintains 1,199 fuel assemblies and, similarly, is currently 89 percent full. In 2012, the NRC approved a newly-designed spent fuel transfer system at Indian Point that allowed the transfer of spent fuel assemblies from the Unit 3 SFP to the Unit 2 SFP for ultimate dry cask storage onsite.

The existing Unit 2 SFP criticality analysis of record, which takes credit for Boraflex inserts as neutron absorbers, was submitted by letter dated September 20, 2001 (ADAMS Accession No. ML012680336) and approved by the NRC staff by license amendment on May 29, 2002 (ADAMS Accession No. ML021230413). Subsequent operating experience has demonstrated the non-uniform physical degradation of Boraflex inserts. In addition, the licensee has acknowledged that the existing Technical Specifications (TSs) regarding the SFP criticality are non-conservative and compensatory measures have been implemented.

The licensee discussed their 4-year SFP management program that has already been initiated and runs through 2016. The new Unit 2 SFP criticality analysis is currently in progress and is scheduled for completion in November 2013. The new criticality analysis will not take credit for the existing Boraflex inserts but will take credit for newly designed neutron absorbing inserts. While the new neutron absorbing inserts have not reached final design, the new criticality analysis will be bounded by its design parameters. The licensee plans to choose a vendor and complete a new neutron absorber insert design by September 2014. A license amendment request proposing the new neutron absorbing inserts, the associated analyses, and revised TSs is planned for November 2014. Once approved by the NRC staff, installation will be planned over a two year period. The SFP management program will maintain full core offload capability for both Units 2 and 3 and will promote the transfer of spent fuel assemblies to dry cask storage.

The licensee described how the assumptions of the Unit 2 criticality analysis of record are being maintained through a combination of the computer program RACKLIFE and BADGER (Boron-10 Areal Density Gage for Evaluating Racks) testing. RACKLIFE is a FORTRAN computer model used to predict Boraflex degradation. Boraflex degradation is characterized by non-uniform thinning, cracking, and the development of localized holes which is difficult to model or predict. BADGER testing was performed at Indian Point in 2006 and 2010 and is used to measure Boraflex degradation in individual SFP storage cells. Additional BADGER testing is planned for November 2013. The licensee contends that BADGER test results have shown that RACKLIFE predictions of Boraflex degradation are conservative and bounded by the existing SFP criticality analysis of record. The NRC staff questioned the licensee's assumptions and findings. The licensee stated that they are currently in the process of updating the RACKLIFE model to better predict Boraflex degradation.

The new Unit 2 SFP criticality analysis, which is currently scheduled for completion by November 2013, is necessary to support the new neutron absorber insert design and the revised TSs. Although the neutron absorbing insert design, the associated analyses, and revised TSs are not scheduled for submittal until November 2014, the licensee indicated their desire to obtain NRC staff feedback of the criticality analysis in advance.

Members of the public were not in attendance but participated via a toll-free telephone conference bridge. Public Meeting Feedback forms were not received.

Please direct any inquiries to me at 301-415-1364, or Douglas.Pickett@nrc.gov.

Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosures:
1. List of Attendees
2. Licensee Slides

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LIST OF ATTENDEES

AUGUST 26, 2013, MEETING WITH ENTERGY NUCLEAR OPERATIONS, INC. AND NETCO

INDIAN POINT SPENT FUEL POOL MANAGEMENT PROGRAM

ROCKVILLE, MD

ENTERGY

Patric Conroy
Don Mayer
Joe DeFrancesco
Giancarlo Delfini
Roger Waters

NRC

Christopher Jackson
Robert Beall
Kent Wood
Ami Patel
Dan Hoang
Emma Wong
Doug Pickett

NETCO

Matt Harris

NUCLEAR CONSULTANTS.COM

Dale Lancaster

PARTICIPANTS VIA TELEPHONE CONFERENCE BRIDGE

STATE OF NEW YORK ATTORNEY GENERAL'S OFFICE

John Sipos
Janice Dean

Entergy Nuclear Operations

Indian Point Unit 2 Spent Fuel Pool Management
August 26, 2013

Entergy

Agenda

• Objectives	Don Mayer
• Spent Fuel Pool Management	Don Mayer
• Project Schedule	Joe DeFrancesco
• Interim Actions	Giancarlo Delfini
• Condition of Boraflex in the IP2 SFP	Matt Harris
• New Criticality Analysis for the IP2 SFP	Dale Lancaster
• License Amendment Request	Roger Waters
• Summary	Don Mayer
• NRC Feedback/Questions	

Objectives

- Provide the NRC with a summary of Entergy's spent fuel management strategy including associated modifications, analyses and license amendment request
- Provide the NRC preliminary project completion dates, analytical assumptions and results
- Obtain NRC feedback at this early stage in the project

Spent Fuel Pool Management

Don Mayer, Director of Special Projects

Entergy

Spent Fuel Pool (SFP) Management

- The Unit 2 and Unit 3 SFPs have storage capacities of 1374 and 1345 fuel assemblies and currently contain 1104 and 1199 assemblies, respectively.
- The spent fuel management project allows for continued full core offload capability during refueling outages as well as ongoing movement of spent fuel to dry storage.
- Unit 3 fuel is wet transferred from the Unit 3 SFP to the Unit 2 SFP. Current plans are to transfer:
  - 2014 – 96 assemblies
  - 2015 – 96 assemblies
  - 2016 – 36 assemblies
- Both Unit 3 and Unit 2 fuel is placed in dry cask storage from the Unit 2 SFP. Current plans are to transfer:
  - 2013 – 96 Unit 3 and 32 Unit 2 assemblies

Spent Fuel Pool (SFP) Management

- Both Unit 3 and Unit 2 fuel is placed in dry cask storage from the Unit 2 SFP. Current plans are to transfer:
  - 2013 – 96 Unit 3 and 32 Unit 2 assemblies
  - 2014 – 96 Unit 3 and 32 Unit 2 assemblies
  - 2015 – 96 Unit 3 and 32 Unit 2 assemblies
  - 2016 – 32 Unit 3 and 96 Unit 2 assemblies
- To increase the number of assemblies eligible for transfer and dry cask storage the "North Anna" type fuel assemblies in both the Unit 3 and Unit 2 SFPs must be repaired.

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Spent Fuel Pool (SFP) Management

- The Unit 2 SFP criticality analysis of record dates from 2002 and takes credit for Boraflex as a neutron absorber.
- Entergy plans to submit a new Unit 2 criticality analysis performed to the latest NRC and industry guidelines.
- The new criticality analysis will not take credit for Boraflex.
- The Unit 2 criticality analysis is in progress in support of the License Amendment Request planned for submittal to NRC in the fall of 2014.

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Spent Fuel Pool (SFP) Management

- The overall project plan is to install neutron absorbing inserts which will be credited in the criticality analysis. Preliminary plans show insert installation complete as early as December 2016.
- Boraflex will continue to be monitored via BADGER testing and RACKLIFE.
  - Next BADGER test is Fall 2013
- The new Unit 2 criticality analysis will allow Unit 3 fuel to be stored in Regions 1 and 2 of the Unit 2 SFP and expand the population eligible for transfer currently limited to fuel discharged from cycles 1 through 11.

Project Schedule

Joe DeFrancesco, Project Manager

Entergy

Project Schedule

- The project has a preliminary 4 year implementation plan that runs from 2013 through 2016.
- The Criticality Analysis is currently in progress and is scheduled to complete in November, 2013.
- The critical path flows from the criticality analysis to the neutron absorber insert design to the license amendment request preparation to insert installation.

Project Schedule

- The criticality analysis will contain the bounding design parameters for the neutron absorber inserts.
- Request for Proposals will be sent out to insert vendors for the design and fabrication of the inserts.
- It is expected that once the vendor is chosen it will take 4 – 6 months to complete the design.
- Preliminary plans show that the insert design will be completed by September 2014.

Project Schedule

- The License Amendment Request (LAR) will include the neutron absorber insert design and analyses.
- The LAR is planned for submission to the NRC in November, 2014.
- Fabrication of the inserts is expected to take approximately 6 months from placing the order.
- Due to scheduled Refueling Outages, Wet Transfers and Dry Cask Loading activities we will be installing the inserts over a 2 year period in 2015 & 2016.

Project Schedule

- Insert installation in 2015 is currently planned to start after the completion of Dry Cask Loading and complete December 2015.
- Insert installation in 2016 is currently planned to start after the completion of Dry Cask Loading and complete December 2016.
- Our preliminary plans are to perform the installation under a 50.59 evaluation.

Interim Actions

Giancarlo Delfini, Supervisor, Reactor Engineering

Entergy

Interim Actions

- Analysis showed Region 2-2 of Unit 2 SFP has a Non-Conservative Tech Spec for the non-borated condition
  - –The analysis concluded that10CFR50.68 was not violated
- Administrative Letter 98-10 applied
  - Administrative controls are in place applying a burnup penalty to the region of the SFP which is non-conservative

Interim Actions

- In order to address the non-conservative Tech Spec a new Unit 2 SFP criticality analysis and license amendment are required
  - The new Unit 2 criticality analysis will not credit Boraflex

Interim Actions

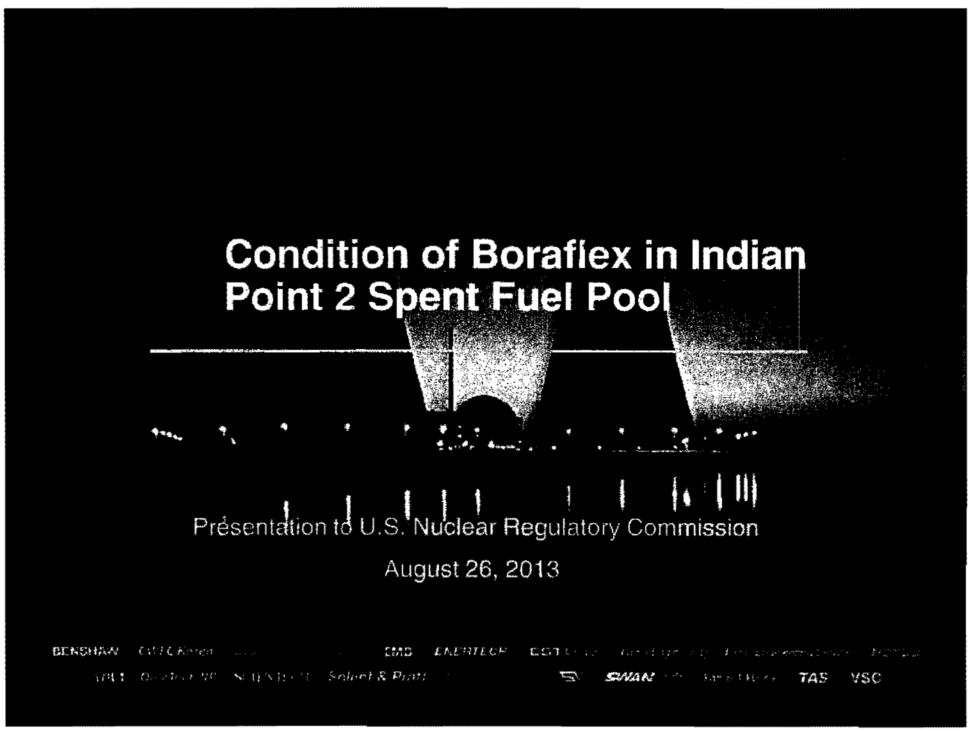
- BADGER testing is required to validate the RACKLIFE predictions of Boraflex degradation
- This assures the assumptions of the Unit 2 Criticality Analysis Of Record (CAOR) are met
- BADGER testing was performed in 2006 and 2010
  - BADGER testing showed assumptions of CAOR were met

Interim Actions

- RACKLIFE , with consideration of prior BADGER testing results, was used to predict degradation to December 2013
- Analysis concluded all cells in SFP will be within assumptions of current CAOR, with the exception of 12 panels which affected 15 fuel cell locations
  - The fuel assemblies in these locations have been moved
  - These 15 cells will not be used to store fuel unless confirmatory BADGER measurements show degradation within assumptions of CAOR

Interim Actions

- Currently in the process of updating the RACKLIFE model to better predict BORAFLEX degradation
- Historically BADGER results have shown the RACKLIFE predictions to be conservative
- Analysis is expected to ensure degradation of BORAFLEX remains bounded with respect to the CAOR



Key Discussion Topics

- Overview of Method Used to Incorporate Boraflex Degradation into IP2 SFP Criticality Analysis of Record (CAOR)
- Review of Most Recent BADGER Data (2006 and 2010 Test Campaigns)
- Comparison of Most Recent BADGER Test Data on CAOR
- Preliminary Conclusions
- Current and Future Activities
  - RACKLIFE Model Assessment/Improved Projections
  - Future BADGER Test – November 2013

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BADGER and RACKLIFE

- RACKLIFE
  - Fortran computer program that performs a time-dependent silica mass balance on the spent fuel pool, via Euler's Method
  - Provides a convenient method of predicting the boron carbide loss of Boraflex panels.
- B.A.D.G.E.R. – **B**oron-10 **A**real **D**ensity **G**age for **E**valuating **R**acks
  - In-situ method for estimation of the Boraflex panel areal density.
  - Provides a means to validate CAOR assumptions and calibrate the RACKLIFE model

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Overview of Method to Incorporate Boraflex Degradation

- Criticality Analysis of Record(NET-173-01) Boraflex Degradation is based on Reactivity Equivalent Uniform Thinning from Minimum Certified Areal Density:
  - 50% for Region 1-2 and 30% for Region 2-2
- Determination of uniform equivalent thinning for CAOR was based on conservative projections of uniform and local measured dissolution.
- Elements of the panels included are as follows:
  - 14 panels measured during 2000 BADGER test were selected
  - RACKLIFE Boraflex Projections made to 2006
  - BADGER losses in 2000 scaled to match 2006 RACKLIFE projections based on worst loss panel in Region 2-2

  (Will discuss later how these panels compare to more recent BADGER measurements)

  - 128 Boraflex panels with random placement of scaled degradation features were created

Method to Determine Equivalent Loss

- Scaled panels randomly placed into a 8x8 array of cells
  - Random placement repeated 50 times to create a set of 50 modules for which 50 values of k_{eff} were calculated.
  - Distribution of k_{eff} values tested for normality to determine 95/95 reactivity effects due to Boraflex degradation
  - 95/95 Δk_{eff} calculated based on 3-D simulations of panels (Table 4-3 of NET-173-01)
  - For a range of amounts of thinning(5% to 50%) Δk_{eff} due to thinning was determined (shown in Table 4-2 of NET-173-01). The reactivity equivalent uniform thinning corresponding to the reactivity effects of Boraflex degradation predicted to 2006 determined (44.2% for Region 1-2 and 21.6% for Region 2-2).
  - For added conservatism, the equivalent thinning values were adjusted upward (50% for Region 1-2 and 30% for Region 2-2).
  - Balance of criticality analysis performed at 50% and 30% uniform thinning
- Percent (%) degradation used in projecting BADGER panels was relative to nominal (results in a larger relative loss), however CAOR assumed the values were relative to the minimum certified.

Review of BADGER Data

- For the 2006 and 2010 BADGER Tests:
  - Data was summarized for each panel to include:
    - Panel Uniform Areal Density
    - Lowest areal density of any given detector in local dissolution region
    - Allows calculation of % loss relative to minimum certified for comparison to bounding panels modeled in NET-170-02
- Gaps are taken directly from BADGER reports.
- Panel characterizations permitted comparisons to panel degradation patterns used in modeling

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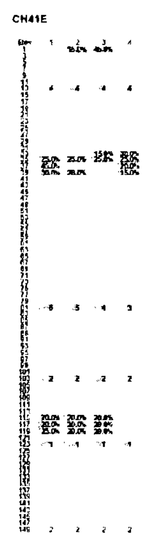
Panel Model Features - Example

Key:

- blank(empty) cells indicate cells at the uniform thinning loss
- Percentage values indicate local dissolution beyond the uniform thinning loss
- Integers indicate gaps in 1/3rds of an inch



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Local Dissolution Patterns Used in CAOR

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Impact of 2006 BADGER Data on CAOR

Region 2-2 - 2006

	Uniform Loss			Local Dissolution Minimums		Bounding
Panel	Areal Density	% loss from Nominal	% Loss from Minimum	Areal Den	% loss from Minimum	Panel
BH75-E	0.0247	4.3%	-12.3%	0.0087	60%	CJ42N
BH75-N	0.0302	-17.1%	-37.3%	0.0241	-10%	Any
BH75-W	0.0257	0.4%	-16.8%	0.0199	10%	any
BJ74-E	0.0241	6.6%	-9.5%	0.0164	25%	any
BJ74-S	0.0243	5.8%	-10.5%	0.0163	26%	any
BK48-E	0.0226	12.4%	-2.7%	0.0155	30%	any
BK48-S	0.0217	15.9%	1.4%	0.0151	31%	CL42W
BK50-N	0.0191	26.0%	13.2%	0.0138	37%	CL46W
BK75-E	0.0195	24.4%	11.4%	0.0153	30%	CL45E
BK75-N	0.0269	-4.3%	-22.3%	0.0189	14%	Any
BK75-NR	0.0250	3.1%	-13.6%	0.0167	24%	Any
BK75-W	0.0265	-2.7%	-20.5%	0.0200	9%	Any
BL49-W	0.0268	-3.9%	-21.8%	0.0195	11%	Any
BL74-E	0.0265	-2.7%	-20.5%	0.0163	26%	Any
BL74-N	0.0303	-17.4%	-37.7%	0.0211	4%	Any
BL74-S	0.0267	-3.5%	-21.4%	0.0161	27%	Any
BL74-W	0.0321	-24.4%	-45.9%	0.0166	25%	Any
BM73-E	0.0296	-14.7%	-34.5%	0.0193	12%	Any
BM73-N	0.0267	-3.5%	-21.4%	0.0184	16%	Any
BM73-S	0.0245	5.0%	-11.4%	0.0127	42%	Any
BM73-W	0.0327	-26.7%	-48.6%	0.0202	8%	Any
BM75-E	0.0207	19.8%	5.9%	0.0151	31%	CH45W
BM75-N	0.0261	-1.2%	-18.6%	0.0241	-10%	any
BM75-W	0.0195	24.4%	11.4%	0.0151	31%	CH45N
BN74-W	0.0223	13.6%	-1.4%	0.0143	35%	Any
DJ73-E	0.0249	3.5%	-13.2%	0.0195	11%	Any
DK74-E	0.0267	-3.5%	-21.4%	0.0195	11%	Any
DK74-N	0.0281	-8.9%	-27.7%	0.0105	52%	CL46W
DK74-W	0.0247	4.3%	-12.3%	0.0175	20%	Any
Average Uniform Loss		1.2%	-15.9%			
St. Dev.(all panels)		13.7%	16.1%	MAX LOSS	60%	
Average from panels w/loss		8.2%	5.6%			
St. Dev. (loss panels)		10.0%	4.9%			

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Impact of 2010 BADGER Data on CAOR

Region 2-2 - 2010

	Uniform Loss			Local Dissolution Minimums		Bounding
Panel	Areal Density	% loss from Nominal	% Loss from Minimum	Areal Density	% loss from Minimum	Panel
BJ68-E	0.0284	-10%	-29%	0.0257	-17%	ANY
BJ68-N	0.0242	6%	-10%	0.0199	10%	CL46E
BJ68-S	0.0236	9%	-7%	0.0223	-1%	CL46W
BJ68-W	0.0273	-6%	-24%	0.0261	-19%	ANY
BJ70-E	0.0241	7%	-10%	0.0220	0%	CH45S
BJ70-N	0.0246	5%	-12%	0.0231	-5%	CH45N
BJ70-S	0.0237	8%	-8%	0.0202	8%	CG50N
BJ70-W	0.0268	-4%	-22%	0.0255	-16%	ANY
BJ72-E	0.0253	2%	-15%	0.0241	-10%	CH51N
BJ72-N	0.0287	-11%	-30%	0.0266	-21%	ANY
BJ72-S	0.0295	-14%	-34%	0.0283	-29%	ANY
BJ72-W	0.0250	3%	-14%	0.0236	-7%	CJ42N
BK69-E	0.0290	-12%	-32%	0.0277	-26%	ANY
BK69-N	0.0275	-7%	-25%	0.0211	4%	ANY
BK69-S	0.0286	-11%	-30%	0.0273	-24%	ANY
BK69-W	0.0290	-12%	-32%	0.0282	-28%	ANY
DH74-E	0.0258	0%	-17%	0.0248	-13%	CL42W
DH74-W	0.0257	0%	-17%	0.0244	-11%	CH51S
Average Uniform Loss		-2.7%	-20.4%			
St. Dev.(all panels)		8.0%	9.3%	MAX LOSS	9.5%	
Average from panels w/loss		5.6%				
St. Dev. (loss panels)		2.5%				

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Preliminary Conclusions

Region 2-2

- BADGER results indicate that the measured uniform and local degradation of the Region 2-2 panels tested to date, are bounded by the 128 panel projections incorporated in the current CAOR
- Latest measurements indicate additional margin is likely available
  - Possibly due to projected panels based on Region 2-1 panels measured in 2000
- RACKLIFE has continued to over-predict BADGER measurements

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Preliminary Conclusions

Region 2-2

- RACKLIFE projections in 2010 exceeded the BADGER(total loss) measurements by 14% on average and 12% for the maximum panel.
- RACKLIFE projections in 2006 exceeded the BADGER Measurements by 3% on average. Panel-to-Panel Variations were larger.
- RACKLIFE has continued to, on average, over-predict BADGER measurements

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Current Activities

- RACKLIFE Model Assessment
  - Identify Inputs that may require updating based on current operation (e.g., cleanup systems, temperatures, escape coefficients)
- Assessment will improve accuracy of future RACKLIFE projections of boron carbide loss
- Analysis is expected show that Region 1-2 and 2-2 BORAFLEX degradation remains bounded with respect to the CAOR

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Future BADGER Testing

- Perform BADGER test in November 2013
- Utilizing 2nd Generation BADGER equipment
- New calibration cell will be built to IP2 spent fuel rack specifications

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New Criticality Analysis For Indian Point Unit 2 Spent Fuel Pool

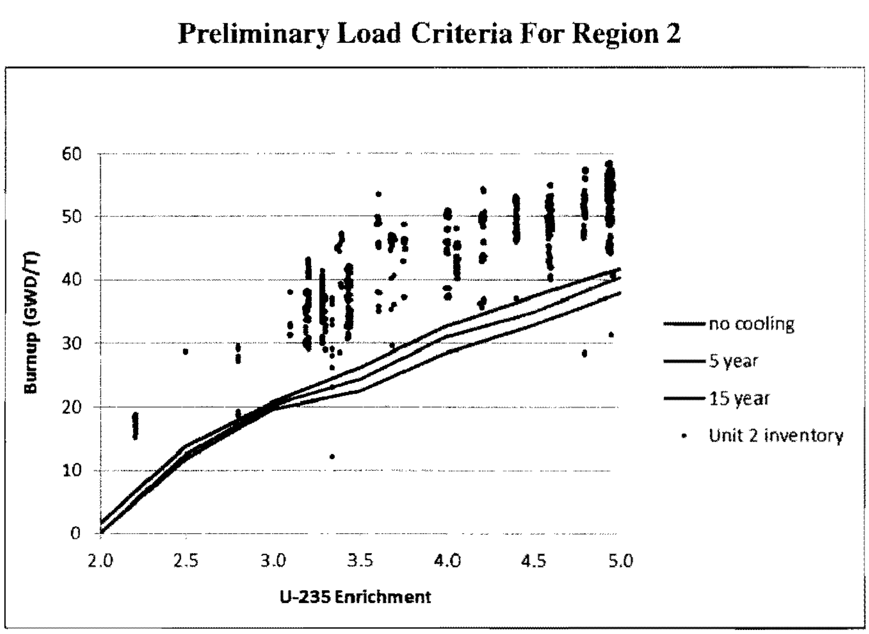
Dale Lancaster
NETCO/NuclearConsultants.com

Philosophy

- Put enough new absorber panels in the pool to make:
  1. Operation simple
  2. Flexibility for the future
  3. Straight forward to license
- An absorber L will be placed in **every cell** (both regions)
- The B-10 areal density of the absorbers is sufficient to support the philosophy.
- The panels are permanently placed (removable for testing or replacing but not as part of normal operations)

Simple Operations

- Region 1 allows all fuel in all cells.
- **One loading curve** for Region 2.
- Fresh fuel not allowed in Region 2.



Flexibility

- Three solutions for any fuel that does not meet Region 1 loading criteria:
  1. Put the fuel in Region 1 (Only 5 assemblies currently do not meet the projected loading criteria).
  2. Put the assembly on the periphery. A ~7 GWd/T reduction in the requirement will be shown for the periphery.
  3. Put a control rod in the assembly. (a control rod is sufficient for even 5 wt% U-235 and no burnup)

Flexibility

- Analysis parameters selected to maximize flexibility for future fuel management.
  - High pellet theoretical density
  - High radial peaking factor (Mod and fuel temp)
  - Limiting fuel dimensions.
- For absorber panel testing, it has been shown that an empty cell provides more negative reactivity than an insert.

Straightforward Licensing

- **1% in k_{eff} licensing margin is used.** Where 1.0 is the regulatory limit the Indian Point loading curve limit is 0.99 (0.94 for 0.95 limits).
- Large margin to k limits for borated cases by better use of boron dilution analysis.
- Following NEI Guidance as well as recent NRC positions.

Straightforward Licensing Reduced Dependence on Administrative Controls

- 100% misload of Region 2 with fresh fuel does not go above 0.95 (using Tech Spec ppm).
- Please remember that no fresh fuel should be in Region 2 and that fresh fuel is visually different.
- Since Region 1 can store fresh fuel, a misload in Region 1 is not possible.

Straightforward Licensing Reduced Dependence on Administrative Controls

- Selected limiting conditions that are very generous
  - No credit for Axial Blankets
  - Design limit radial peaking factor
  - Maximum expected burnable absorber loading
  - Some control rod operation is included (up to 2 GWd/T)

Real Margin

- As shown previously, most assemblies significantly exceed the burnup requirements.
- The average fuel assembly exceeds the loading requirements by about 10 GWd/T.
- Licensing based on limiting not average assemblies.
- Review has shown that the assemblies close to the loading curve all have had significant margin when analyzed with knowledge of actual depletion conditions.

Overview - Summary

- New pool criticality analysis uses Simplicity, with Flexibility and has significant Margin.
- The new pool criticality analysis depends on physical properties (high quantity of absorber panels) rather than administrative controls.

Details

- Will Follow the NEI Spent Fuel Pool Criticality Guidance (NEI 12-16)
- Section 2 of the NEI Guidance (analytical techniques) is followed but with the addition of 1% licensing margin (0.99 limit on k for unborated conditions.)

Computer Codes (NEI Section 3)

- SCALE 6.1.2 with the ENDF/B-VII library will be used for all calculations. (most recent version of the code with the most recent cross section library (238 energy groups))
- Depletion analysis is performed using TRITON module t5-depl which calls KENO Va for the flux calculations. Sufficient neutron sampling shown by doubling the population producing the same results.

Computer Codes (Section 3)

- Pool criticality calculations use CSAS5 which calls CENTRM, BONAMI, and KENO.Va.
- Final calculations done using at least 1500 generations and 6000 neutrons per generation.

Fresh Fuel Computer Code Validation (NEI Section 3.2.1)

- 236 Fresh UO_2 Fuel Critical Experiments are used in the fresh fuel validation of SCALE 6.1.2 and the ENDF/B-VII 238 group cross section library.
- The validation generally follows NUREG/CR-6698.

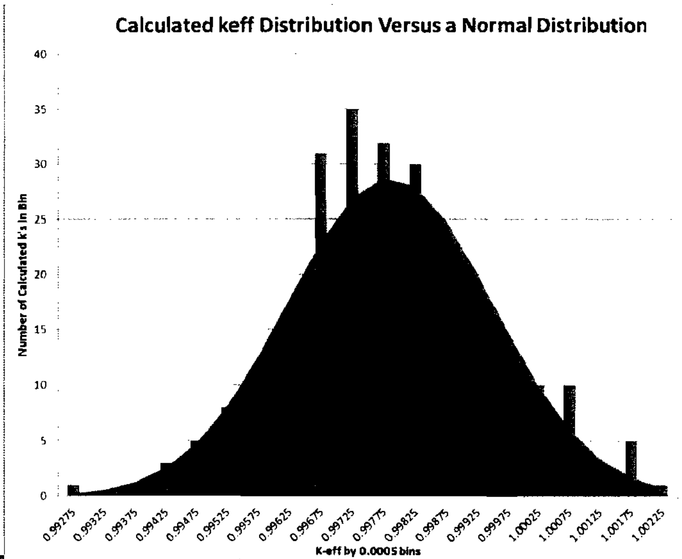
Selection Criteria

- Low enriched (5 wt% U-235 or less) UO_2 to cover the principle isotopes of concern.
- Fuel in rods to assure that the heterogeneous analysis is correct.
- Square lattices to assure the lattice features of SCALE used in the rack analysis are verified.
- Presence of soluble boron, borated steel, boron bearing rods, sheets of aluminum with boron, Boraflex™, and Ag-In-Cd.
- No emphasis on a feature or material not of importance to the rack analysis (e.g. lead reflectors).
- All data from OECD/NEA criticality benchmark handbook
- From diverse evaluations (31 different evaluations used)
- From diverse labs (from 6 different critical facilities).

Normality

- Data is slightly non-normal but due to the large sample size (236 experiments), the normality assumption is appropriate (references given) and used so long as plots show this assumption is reasonable.
- Normality is conservative in this case since results are clustered closer to the mean.

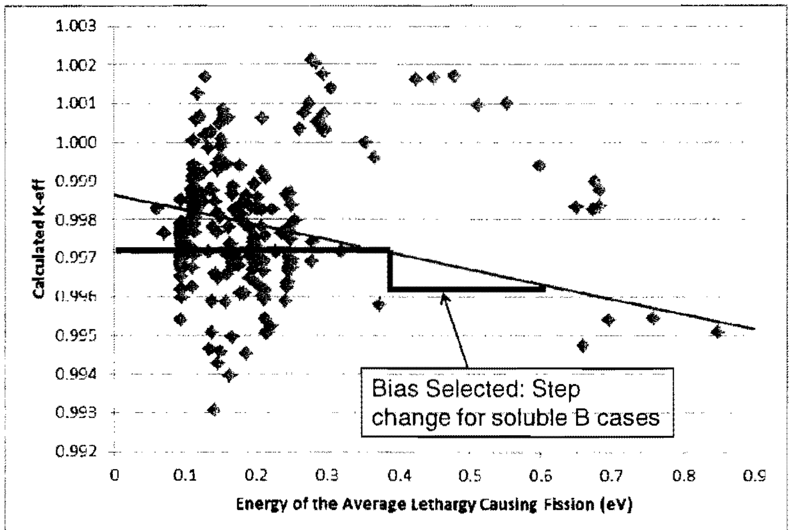
Very Close to Normal Where It Counts



Trending

- Trends analyzed for:
  - Neutron spectrum (EALF)
  - Pin Diameter
  - Lattice Pitch
  - Enrichment
  - Boron areal density, and
  - Soluble boron ppm.
- Although tests for statistical significance are performed, **all trends are used in the final analysis**.
- Range of applicability is clearly defined.
- **Largest bias and uncertainty from all the trends is used.**

EALF (Limiting Trend)



Limiting Bias and Uncertainty

- Over the range of applicability the largest bias considering all trends was 0.0029 (EALF less than 0.4) and 0.0037 (EALF between 0.4 and 0.6 – heavily borated).
- The uncertainty is 0.0050 which comes from the most limiting of all trend analysis.

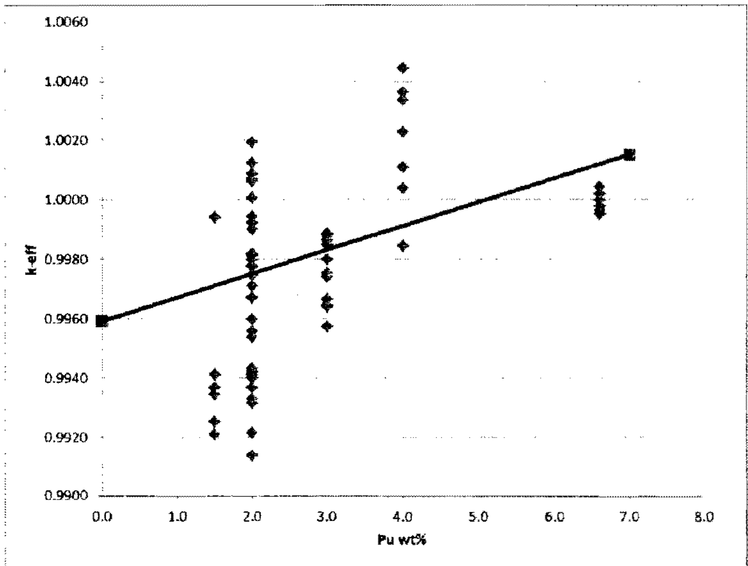
HTC Critical Experiments

- 117 HTC critical experiments are also evaluated.
- These critical experiments cover the reactivity effect of Pu in spent fuel.
- Spent fuel can range from very little Pu to the HTC levels and beyond the analysis
- The bias and uncertainty for the HTC critical experiments are determined separately from the fresh fuel criticals and the most limiting bias and uncertainty of the sets is used for all the analysis (The fresh fuel bias and uncertainty is more limiting.)
- Assume HTC crits (4.5 wt % 37.5 GWd/T) covers the entire range of spent fuel but this assumption is augmented by the analysis of the EPRI benchmarks.

MOX Critical Experiments

- 63 MOX criticals were analyzed (All low enriched Pu lattice experiments in OECD Handbook)
- ENDF/B-VII predicts higher k for Pu containing critical experiments than UO_2 critical experiments. Therefore the UO_2 data sets the limits.
- Mean k
  - 0.9978 for UO_2 criticals
  - 0.9988 for HTC criticals
  - 0.9984 for MOX criticals

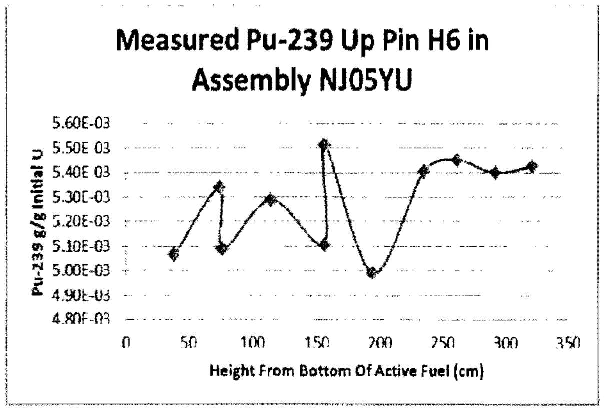
MOX results



Used Fuel Criticality Validation (NEI Section 3.2.2)

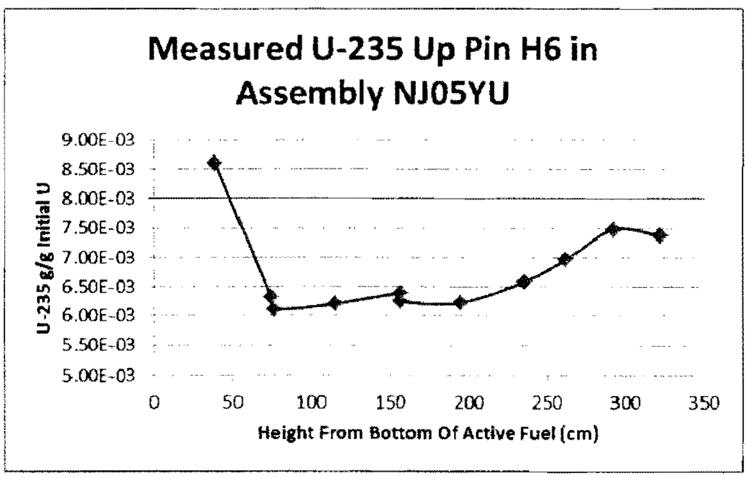
- Will use most limiting of methods from sections 3.2.2.1 (EPRI benchmarks) and 3.2.2.3 (Chemical Assays).
- Used Direct Difference approach from NUREG/CR-7108.
- By **inspection of experimental data,** removed 12 TMI chemical assays and one HB Robinson assay (taken from under an Inconel grid. Would require 3D methods.)

TMI Pu-239 Assays Done By ANL

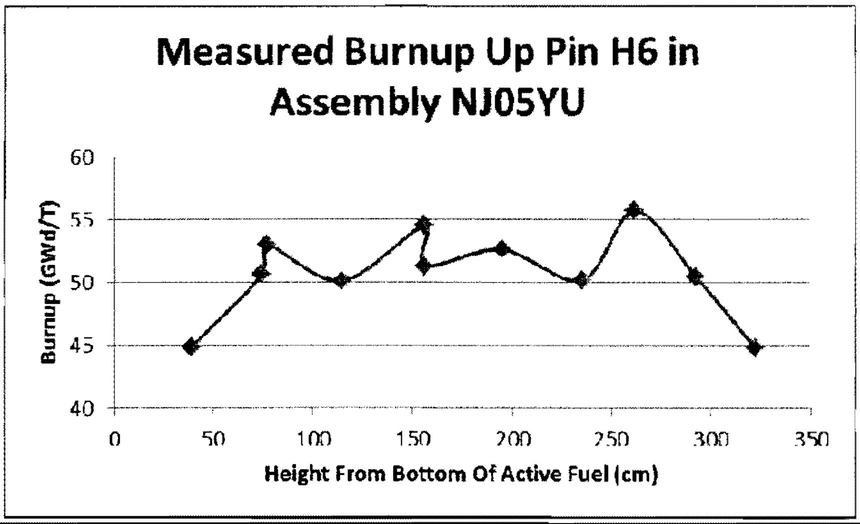


Measured Pu-239 Content Up Pin H6 in Assembly NJ05YU

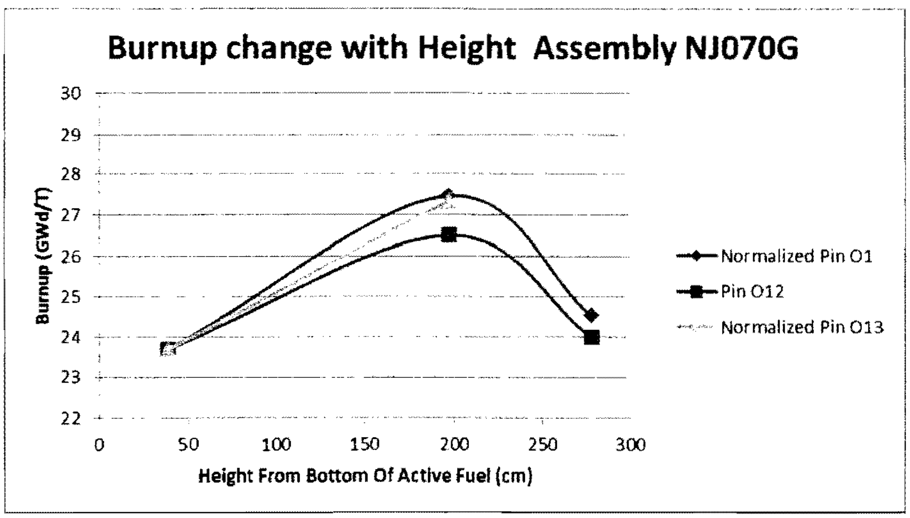
TMI U-235 Assays Done By ANL



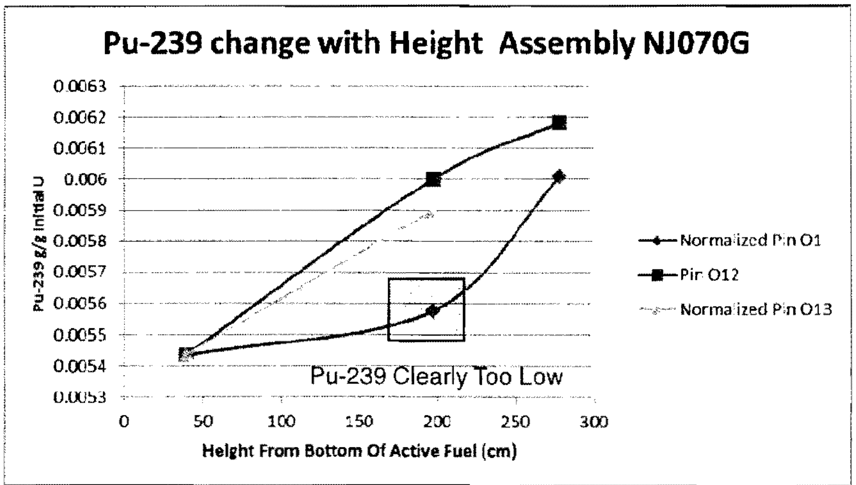
TMI Burnup For Assays Done By ANL



TMI Burnup Up Assembly NJ070G



TMI Pu-239 Up Assembly NJ070G



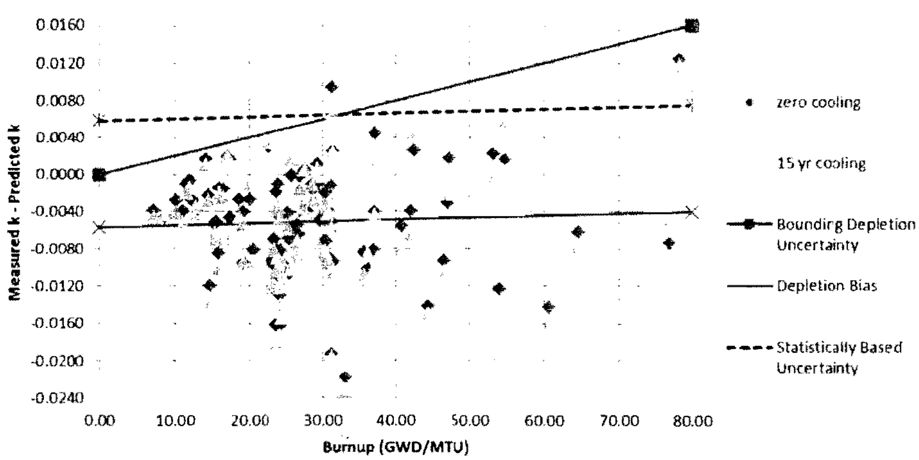
Chemical Assay Approach

- Added 5 chemical assays from Vandellós II which were not included in NUREG/CR-7108.
- A total of 92 chemical assays are used (NUREG/CR-7108 used 100).
- Much better results due to removing bad experimental data and correcting NUREG/CR-7108 errors.
- Since the chemical assays cover only 28 isotopes and the criticality analysis will be performed using 185 isotopes the EPRI benchmarks are used to confirm none of the added isotopes produces a large unexpected reactivity.

Chemical Assay Approach

- The direct difference analysis produces 92 delta ks.
- Analysis is performed for Region 2 racks.
- Used trend in delta k as a function of burnup.
- Negative bias ignored.
- Statistical uncertainty did not show physical knowledge of 0 bias and uncertainty at 0 burnup.
- Applied engineering uncertainty. See graph.

Direct-Difference Bias and Uncertainty



Minor Actinide and FP Worth Bias

- UO_2 (most limiting), HTC and MOX criticals cover the major actinides.
- Added a bias of 1.5% of the minor actinides and fission product worth to cover possible bias and uncertainty in the reactivity worth of the minor actinides and fission products (See NUREG/CR-7109 and ISG 8 Rev. 3).

EPRI Benchmarks

- The full set of EPRI benchmarks have been analyzed using SCALE 6.1 and ENDF/B-VII and published as the EPRI Technical Report Number 1025203.
- Analysis redone with SCALE 6.1.2.

EPRI Benchmark Biases

	Lattice Description	Bias (Calculated Reactivity Decrement - Measured Reactivity Decrement) for 100 Hour Cooling					
	3.25% enrichment depletion	-0.0008	-0.0017	-0.0024	-0.0037	-0.0040	-0.0044
	5.00% enrichment depletion	-0.0001	-0.0003	-0.0005	-0.0012	-0.0014	-0.0018
	4.25% enrichment depletion	0.0002	-0.0004	-0.0010	-0.0018	-0.0026	-0.0029
	off-nominal pin depletion	-0.0008	-0.0016	-0.0023	-0.0029	-0.0037	-0.0046
	20 WABA depletion	0.0000	0.0003	-0.0005	-0.0014	-0.0018	-0.0025
	104 IFBA depletion	0.0009	0.0005	-0.0003	-0.0016	-0.0024	-0.0036
	104 IFBA, 20 WABA depletion	0.0007	0.0011	0.0000	-0.0008	-0.0019	-0.0031
	high boron depletion = 1500 ppm	-0.0003	-0.0006	-0.0011	-0.0017	-0.0018	-0.0024
	branch to hot rack = 338.7K	-0.0004	-0.0007	-0.0008	-0.0017	-0.0019	-0.0025
	branch to rack boron = 1500 ppm	-0.0009	-0.0019	-0.0027	-0.0036	-0.0044	-0.0049
	high power density depletion	-0.0002	-0.0012	-0.0016	-0.0022	-0.0026	-0.0032

EPRI Benchmark Bias and Uncertainty

- Negative biases are conservative
- Maximum bias from all the cooling times is 0.0026
- Pin Diameter of Indian Point is larger than the 17X17 base pin diameter in EPRI Benchmarks – Used Difference between Cases 3 and 4 to correct for this.
- **Final Bias is 0.003. The uncertainty is 0.0064**

Limiting Bias and Uncertainty for Burned Fuel

- The chemical assay bias is zero. And the worth bias is about 0.0015. The EPRI benchmark bias is 0.003.
- The EPRI uncertainty is 0.0064. The chemical assay uncertainty does not reach this until 32 GWd/T.
- The net effect is the **EPRI benchmark approach is more limiting** to about 45 GWd/T (almost all of the loading curve).
- There is significant agreement between the two methods now that the bad data and errors are removed.

Rack and Fresh Fuel Modeling NEI Section 4

- Indian Point (both units) has used the same fuel supplier (Westinghouse) for all its fuel. The fuel pin design has remained the same and is expected to remain the same.
- The criticality analysis will be performed with limiting dimensions since the reactivity impact of fuel assembly tolerances are small. (The exception to this is the fuel pin pitch which is set by plant design and will be done as nominal and tolerance uncertainty.)
- The limiting dimensions use slightly enlarged tolerances to be able to accommodate possible changes in fuel manufacturing.

Rack and Fresh Fuel Modeling
NEI Section 4

- Statistical combination of uncertainties will be used for:
  - Pitch (tolerance is base on plant assembly pitch)
  - Rack cell dimensions
  - Fresh fuel validation uncertainty
  - Depletion uncertainty
  - Reactor Record Burnup Uncertainty (5% of the burnup)
  - Eccentric Positioning of Fuel Assemblies

Rack and Fresh Fuel Modeling
NEI Section 4

- Rack criticality models will be 2X2 models of the rack with periodic boundary conditions on the sides and 30 cm of water reflector on top and bottom
- The 30 cm water reflector has long been established as optimum reflection and no proof of this is anticipated to be needed.

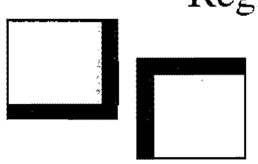
Rack Neutron Absorbers
NEI Section 4.4

- The absorber is modeled with a minimum specified areal density.
- Indian Point will order a minimum areal density in excess of this to provide margin for future degradation issues.
- No reduction of the minimum areal density is needed or taken for grain size, etc.
- A minimum or maximum (region dependent) absorber thickness is assumed so that the criticality analysis is independent of absorber vendor.
- The absorber width is slightly less than the full width of the cell and is a minimum requirement of the manufacturer.

Configuration Modeling
NEI Section 5

- Interfaces: The absorber panel placement is such that the interface has two absorbers and is never limiting. Full pool models confirm the interface is non-limiting.

Region 1 cells Region 2 cells



Configuration Modeling
NEI Section 5

- Interfaces: Separate (less restrictive) loading criteria will be provided for assemblies on the outside periphery of the rack. A full pool analysis will be used to verify modeling.

[redacted]

Multiple Assembly Misloads
NEI Section 5.3.3

- 100% misload with fresh 5 wt% fuel with 48 IFBA meets safety requirements (less than 0.95) with Tech Spec ppm
- 100% misload with 5 wt% fuel with 10 GWd/T burnup meets safety requirements with Tech Spec ppm
- Multiple misload of this type and boron dilution below Tech Spec ppm are considered two independent unlikely events.

Soluble Boron Credit NEI Section 6

- Boron Dilution Analysis will be performed.
- This will establish a boron level for which further boron dilution is not credible.
- That boron level minus some margin will be used to show the critical condition is less than 0.95.
- Since this will show a large margin, only a few cases will be presented and no sensitivity analysis is needed or presented.

Reactivity Effects of Depletion NEI Section 7.1.1

- Moderator Temperature/density and Fuel Temperature:
  - Will use the design assembly average peaking factor (1.40)
  - Moderator Temperature and density taken using the outlet temperature of the assembly with the 1.40 peaking factor
  - Outlet conditions are assumed over the full length of the fuel.

Reactivity Effects of Depletion NEI Section 7.1.1

- Fuel Temperature from the limiting peaking factor (1.40) is assumed.
- Highest burnup averaged soluble boron is assumed. The assumed value (1000 ppm) is confirmed during design as part of the reload safety analysis check (RSAC).

Reactivity Effects of Depletion NEI Section 7.1.1

- Burnable Absorbers: The depletion analysis will be done using 20 fingered (maximum) WABAs plus 148 IFBA rods (1.5X loading) (Current max for Westinghouse 15X15 fuel).
- The WABAs will be removed at 35 GWd/T.
- Gad fuel has not been used in the past but this analysis covers any future application of Gd.

Reactivity Effects of Depletion NEI Section 7.1.1

- Rodded Operation:
  - All analysis will start with 2 GWd/T of depletion with fully inserted control rods (WABAs between 2 GWd/T and 35 GWd/T) to cover the spectral impact of control rods.
  - Indian Point does not operate with control rods inserted during depletion but this allows some margin for possible future operational flexibility.

Reactivity Effects of Depletion NEI Section 7.1.1

- Rodded Operation – "Bite Position"
  - Indian Point Unit 2 operated using control rods at the "bite position" for cycles 1 through 16.
  - The average bite position for a cycle was always less than 8 inches (size of the top node in crit models)
  - For all fuel less than 4.5 wt% U-235 the top node is depleted for all life with control rods inserted.
  - For fuel 4.5 wt% and greater (current reloads) the top node has unborated WABAs for all life.
  - The few assemblies from cycle 16 and before that are greater than 4.5 wt% were evaluated to show assembly specific margin to cover rodded operation.

Axial Burnup Distribution

- Assume full length fuel and limiting axial burnup distributions from NUREG/CR-6801 (DOE shapes). No verification of the axial burnup distribution is required since this is very conservative for axially blanketed fuel.
- No unusual operation in the pre-blanket fuel was used to invalidate the DOE shapes.

Other Credits
NEI Section 8

- Decay time will be credited for the region 2 loading curve.
- Credit will be demonstrated for control rods in an assembly. This will be used to store a few underburned assemblies in region 2.

Licensee Controls
NEI Section 9

- Administrative Controls:
  - Controls on maximum average peaking factors and maximum average ppm will be applied as part of the RSAC.
  - The maximum average peaking factor and ppm will not be in the Tech Specs since they are controlled via the RSAC procedure.

Licensee Controls
NEI Section 9

- Simplicity:
  - Only one fuel type.
  - One loading curve for normal operation
  - Burnup reduction for assemblies at the pool edge.

Licensee Controls
NEI Section 9.3 Future Fuel Types

- Analysis has been done with limiting fuel dimensions (not nominal and tolerances) so if the new fuel maintains the nominal fuel pin diameter, new fuel types do not need analysis assuming they meet the limiting dimensions.

Licensee Controls
NEI Section 9.5
Neutron Absorber Surveillance Program

- The neutron absorber surveillance program will be handled separately from the criticality safety analysis report.

Summary

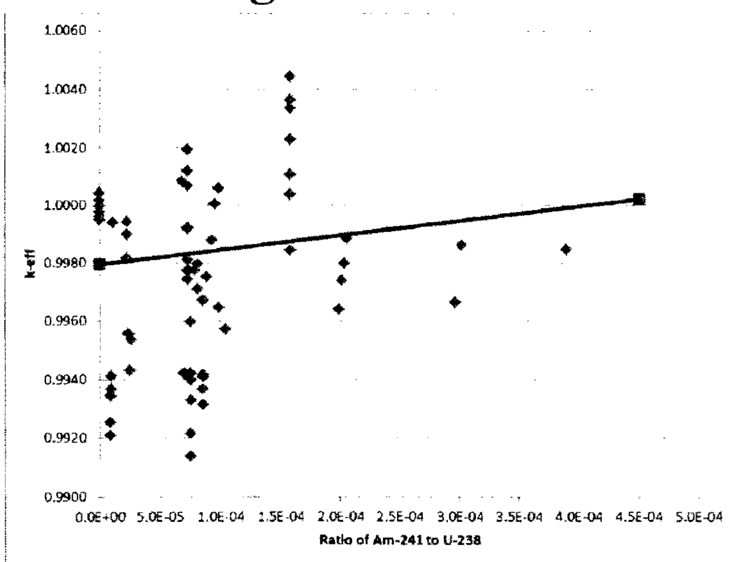
- A predictable, licensable, criticality analysis method has been described.
- Extra margin of 1% in k is added to make the licensing easier.
- Feedback is appreciated to reduce the need for RAIs.
- This criticality effort removes the reliance on Boraflex absorbers.

Backup Slides (not part of handout)

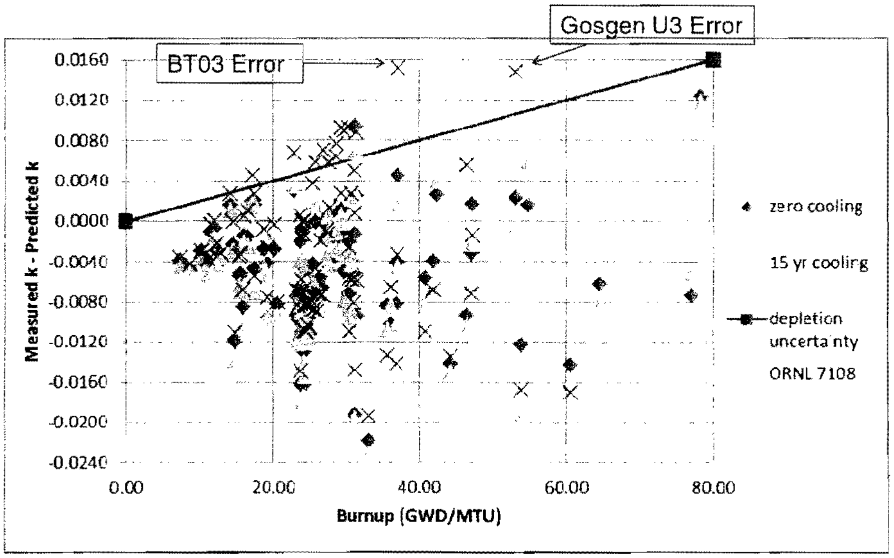
Am-241 – Cooling Time

- Decay of Pu-241 to Am-241 dominates the reactivity change with cooling time
- MOX critical rods have various amounts of Am-241 due to decay from the time of reprocessing.
- MOX criticals show that with increased cooling time (more Am-241) the predicted k of critical experiments increases.
- Therefore it is conservative to use a zero cooling time bias.

Cooling Time Validation



Comparison with NUREG/CR-7108



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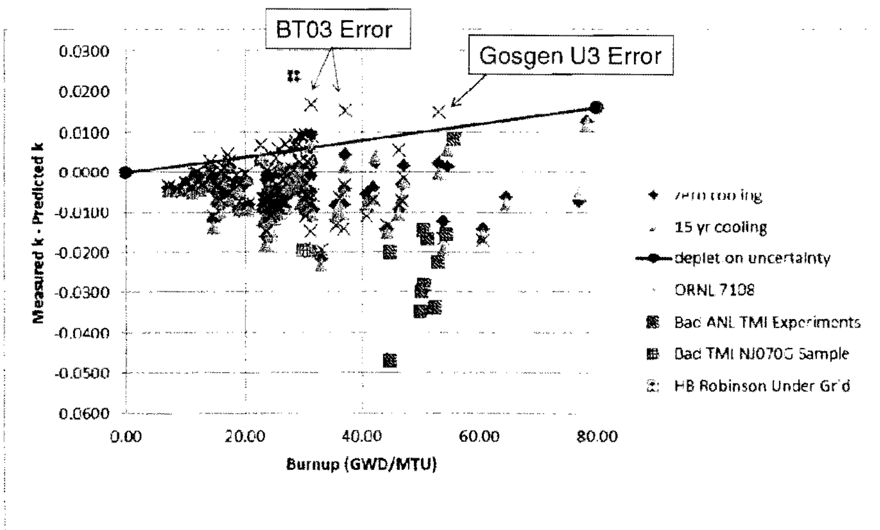

License Amendment Request

Roger Waters, Licensing

Entergy

License Amendment Request

- A LAR will be submitted to address:
  - The non-conservative Technical Specification associated with the currently approved SFP criticality analysis methodology
    - Mark-up of proposed Technical Specification changes
  - Boraflex degradation
  - Installation of new neutron absorbing inserts
    - Criticality
    - Seismic
    - Structural
    - Thermal
    - Insert design details
    - Insert qualification
    - Insert surveillance

License Amendment

- The interim configuration/no credit for inserts
- The final configuration/credit for inserts

- Entergy is considering submitting the new criticality analysis in the fall of 2013.
- Entergy plans to submit LAR the fall of 2014.

Summary

Don Mayer, Director of Special Projects

Entergy

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Summary

- Entergy's objective was to engage the NRC at this early stage of our overall project plan to license a new criticality analysis that will resolve a non-conservative Technical Specification and eliminate credit for the installed boraflex in the Unit 2 spent fuel pool.
- Engagement of the NRC at this time, and throughout the project, is intended to facilitate the NRC license amendment request review and approval process.

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Open Discussion

complete a new neutron absorber insert design by September 2014. A license amendment request proposing the new neutron absorbing inserts, the associated analyses, and revised TSs is planned for November 2014. Once approved by the NRC staff, installation will be planned over a two year period. The SFP management program will maintain full core offload capability for both Units 2 and 3 and will promote the transfer of spent fuel assemblies to dry cask storage.

The licensee described how the assumptions of the Unit 2 criticality analysis of record are being maintained through a combination of the computer program RACKLIFE and BADGER (Boron-10 Areal Density Gage for Evaluating Racks) testing. RACKLIFE is a FORTRAN computer model used to predict Boraflex degradation. Boraflex degradation is characterized by non-uniform thinning, cracking, and the development of localized holes which is difficult to model or predict. BADGER testing was performed at Indian Point in 2006 and 2010 and is used to measure Boraflex degradation in individual SFP storage cells. Additional BADGER testing is planned for November 2013. The licensee contends that BADGER test results have shown that RACKLIFE predictions of Boraflex degradation are conservative and bounded by the existing SFP criticality analysis of record. The NRC staff questioned the licensee's assumptions and findings. The licensee stated that they are currently in the process of updating the RACKLIFE model to better predict Boraflex degradation.

The new Unit 2 SFP criticality analysis, which is currently scheduled for completion by November 2013, is necessary to support the new neutron absorber insert design and the revised TSs. Although the neutron absorbing insert design, the associated analyses, and revised TSs are not scheduled for submittal until November 2014, the licensee indicated their desire to obtain NRC staff feedback of the criticality analysis in advance.

Members of the public were not in attendance but participated via a toll-free telephone conference bridge. Public Meeting Feedback forms were not received.

Please direct any inquiries to me at 301-415-1364, or Douglas.Pickett@nrc.gov.

/ra/

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosures:
1. List of Attendees
2. Licensee Slides

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Meeting Notice: ML13218A086 **Meeting Summary: ML13256A086**

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