Paul

CONTENTS OF OFFICIAL RECORD FOLDERS FOR ACRS SUBCOMMITTEES

The Federal Advisory Committee Act requires retention of certain documents related to every advisory committee meeting. The ACRS has applied this requirement to all ACRS subcommittee meetings. The cognizant staff engineer is responsible for assembling an official record folder for each subcommittee meeting. The folder is retained on file by the Operations Support Branch (Michele Kelton). The following is a list of the documents that should be included in the official record folder.

- Original copy of the certified minutes,
- Signed Subcommittee Chairman certification sheet,
- Memorandum forwarding the certified minutes to the members,
- Memorandum forwarding the working draft of the minutes to the members,
- Marked-up agenda or proposed schedule,
- List of attendees
- Federal Register Notice, and
- Slides presented at the subcommittee meeting.

A copy of the certified minutes should be provided to the ACRS secretary.

Three copies of the certified minutes and an electronic copy of the certified minutes should be provided to the Operations Support Branch (Ethel Barnard) for further distribution.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 20, 2001

MEMORANDUM TO: ACRS MEMBERS

FROM: for Paul Boehnert, Senior Staff Engineer
Thermal-Hydraulic Phenomena Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING, OCTOBER 25-26, 2001 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on November 1, 2001, have been certified as the official record of the proceedings for that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: V. Schrock, ACRS Consultant

cc: via-email: J. Larkins
H. Larson
S. Bahadur
R. Savio
S. Duraiswamy
ACRS Fellows and Technical Staff

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 5, 2001

MEMORANDUM TO: Paul Boehnert, Senior Staff Engineer
ACRS/ACNW

FROM: Graham Wallis, Chairman
Thermal-Hydraulic Phenomena Subcommittee

SUBJECT: CERTIFICATION OF THE SUMMARY/MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON THERMAL-HYDRAULIC PHENOMENA - OCTOBER 25-26, 2001 - ROCKVILLE, MD

I hereby certify that, to the best of my knowledge and belief, the Minutes of the subject meeting issued November 1, 2001, are an accurate record of the proceedings for that meeting.

Graham Wallis 11/08/01
Graham Wallis, Chairman Date

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

November 1, 2001

MEMORANDUM FOR: G. Wallis, Chairman, Thermal-Hydraulic Phenomena Subcommittee

FROM: P. Boehnert, Senior Staff Engineer

SUBJECT: MINUTES OF THE ACRS THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING, OCTOBER 25-26, 2001 - ROCKVILLE, MARYLAND

A Working Copy of the subject meeting minutes is attached. I would appreciate your review and corrections as soon as possible. Copies are being sent to all ACRS members, and the Subcommittee Consultant for their information.

Attachment: As Stated

cc: ACRS Members
V. Schrock
R. Savio

cc via E-Mail:
J. Larkins
S. Bahadur
R. Savio
H. Larson
S. Duraiswamy
ACRS Staff Engineers
ACRS Fellows

DRAFT COPY - PREPARED FOR INTERNAL COMMITTEE USE

CERTIFIED **ISSUED** November 1, 2001
G. Wallis, Chairman
November 8, 2001

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING MINUTES:
DRESDEN/QUAD CITIES NUCLEAR PLANTS CORE POWER UPRATES
OCTOBER 26-27, 2001
ROCKVILLE, MARYLAND

INTRODUCTION:

The ACRS Subcommittee on Thermal-Hydraulic Phenomena held a meeting on October 25-26, 2001 with representatives of the NRC Office of Nuclear Reactor Regulation, the Exelon Generation Company, and GE Nuclear Energy. The purpose of this meeting was for the Subcommittee to review the license amendment request of the Exelon Generating Company for core power uprates for the Dresden Nuclear Power Station, Units 2&3, and the Quad Cities Nuclear Power Station, Units 1&2. The entire meeting was open to the public. Mr. P. Boehnert was the cognizant ACRS staff engineer and Designated Federal Official (DFO) for this meeting. The meeting was convened by the Chairman at 1:00 p.m., October 25, 2001, recessed at 5:40 p.m. that day; convened at 8:30 a.m, October 26, 2001 and adjourned at 11:45 p.m. that day.

ATTENDEES

ACRS Members/Staff:
G. Wallis, Chairman
F. Peter Ford, Member
T. Kress, Member
W. Shack, Member
V. Schrock, Consultant
P. Boehnert, DFO

NRC NRR Staff:
J. Zwolinski
L. Rossbach
QC LPM
R. Caruso
D. Harrson
K. Parchewski
S. Bailey
S. Bajwa
R. Architzel
K. Manoly
D. Harrison
G. Carpenter
K. Kavanagh

Exelon Generating Company
J. Benjamin
T. Hanley
H. Crockett
N. Hanley (S&W)
W. Burchill
J. Nosko
J. Freeman
M. Kluge
K. Moser

GE Nuclear Energy
J. Post
D. Pappone
I. Nir

A list of public attendees is attached to the Office Copy of these Minutes.

The presentation slides and handouts used during this meeting are attached to the Office Copy of these Minutes. The presentations to the Subcommittee are summarized below.

DRESDEN/QUAD CITIES CORE POWER UPRATE REQUESTS

Introduction

G. Wallis, Subcommittee Chairman, convened the meeting and said that given the volume of review material provided to the Subcommittee it was key that the speakers point the Subcommittee to the central issues of concern for consideration by the full ACRS.

F. Peter Ford declared himself in conflict of interest with GE Nuclear Energy, and, therefore, indicated that he will not advise the Committee on this matter.

Exelon Generating Company Presentation

Mr. J. Benjamin, Exelon, made some opening comments and introduced Mr. J. Nosko who introduced the presentation schedule for the Dresden/Quad Cities core power uprates. The following topics were discussed:

- Introduction
- Plant Modifications to Accommodate Power Uprate
- Compliance with Regulatory Requirements
- Selected Analyses and Evaluations:
  - Containment
  - Emergency Core Cooling System (ECCS)
  - Thermal-Hydraulic Stability
  - Anticipated Transients Without Scram (ATWS)/ATWS Instability
  - Piping
  - Reactor and Internals
  - Flow-Accelerated Corrosion (FAC)
- EPU Risk Evaluation
- Resolution of Open Items
  - ECCS NPSH Requirements
  - Ultimate Heat Sink
  - Standby Liquid Control System (SLCS)
  - Large Transient Tests
- Project Implementation - Training and Testing
- Concluding Remarks

Key points noted by Exelon included the following:

- Exelon has devoted two years to this project. These uprate submittals were performed in accordance with the GE Extended Power Uprate Licensing Topical

Reports, and utilized constant pressure methods. No significant impact on plant response or system integrity were found at EPU conditions. Plant risk is minimally impacted. Post-EPU, all four units will operate at the same power level (2957MWt). Exelon also plans to apply for license renewal for all four units.

- For the issues of thermal-hydraulic stability, ATWS, and ATWS instability were evaluated for the EPU conditions. For each issue, the results were found acceptable. Exelon will implement the Option III approach (automatic "detect and suppress" solution) for stability control, subsequent to resolution of the Part 21 Issue involving the DIVOM (**D**elta CPR/**I**nitial CPR **V**s **O**scillation **M**agnitude) curves.

- As a result of its piping analysis, modifications are required to torus-attached piping and the turbine stop valves for EPU conditions. For the reactor and internals, modifications to the steam dryer were needed to reduce moisture content and increase the margin to flow-induced vibration. Regarding materials inspection, Exelon utilizes the recommendations of the BWR Vessel Inspection Program. Use of "Noble Chem" and hydrogen water chemistry mitigate the effects of power uprate. Actions were taken to address FAC, including accelerating inspections of affected components. Evaluation of the reactor internals showed the need for modifications to some of the jet pump sensing lines to counter vibration stresses.

- Review of containment thermal-hydraulic and structural analyses show that all results are within ASME code allowables for EPU conditions.

- The SAFER/GESTR methodology will be used for the ECCS analyses on all four units, pursuant to SECY 83-472. Results of the PCT analyses show no significant impact on the safety or operating margin.

- A PRA evaluation was performed to identify potential vulnerabilities resulting from power uprate. A full-power internal events quantitative risk evaluation was performed. A qualitative risk evaluation was conducted to evaluate risk from internal flooding, fire, seismic and shutdown. The overall impact of the uprate on plant risk is minor (<10% increase in CDF and LERF) and conforms to the guidelines of Regulatory Guide 1.174. The risks from external events and shutdown conditions are also minor.

- Regarding resolution of open issues, Exelon will perform modifications to address concerns regarding the ultimate heat sink at Dresden and the SLCS systems at three of the four units. Exelon has taken the position that large transient testing is not needed. This issue is still unresolved (see below).

- A comprehensive training and start up testing program will be performed to implement the uprate program.

Subcommittee Comments

- In response to questions from J. Sieber, Exelon said they would look into the impact of loss of instantaneous overcurrent protection for the plants' auxiliary transformer, and the impact of a direct short on plant safety.

- Regarding the issue of conducting transient tests, the Subcommittee questioned the applicability of overseas test results to the Dresden/Quad Cities plants.

- The Subcommittee questioned whether the brackets securing the steam dryer to the upper vessel will be periodically inspected. Exelon said they would check on this matter and provide response.

- In response to questions from Mr. Sieber, Exelon said they would provide him with information on the details of the changes made to the setpoint methodology to accommodate EPU.

- Mr. Schrock questioned why the natural circulation lines on the power/flow curves are different for the two plants, given that the same methodology was used. GE said they would provide response.

NRC Staff Review

Mr. J. Zwolinski, NRR, commented on the staff's response to the Committee's concerns with the staff's SERs on power uprates, as noted in the ACRS letter commenting on the Duane Arnold uprate review. He said that the staff intends to revise the Duane Arnold SER and will provide it to the Committee for its inspection. His goal is to ensure that the Duane Arnold uprate SER is a robust document that can serve as a template for future reviews.

Mr. S. Bajwa, NRR, noted that the staff has not arrived at a decision with regard to the need for conduct of large transient tests. The issue will be addressed prior to the issuance of the license amendments allowing power uprate.

Mr. L. Rossbach, NRR, introduced the NRR staff presentation on the Dresden/Quad Cities uprate requests. The following topics were discussed:

- Scope and Method of Staff Review
- Reactor Systems Review
- Plant Systems Review
- Response to ACRS Questions
  - Material Degradation Issues
  - Need for Pipe and Support Modifications
  - Need for Electrical Modifications
  - PRA Analyses and Evaluation

- Conclusions

Key points of the staff's discussions included:

- The staff's review included three audits of: (1) Exelon's PRAs, (2) Plant Systems issues at the Dresden site, and (3) reactor core & fuel performance issues at GE's Global Nuclear Fuels site. These audits resulted in the need for GE to take corrective actions for some issues. This review paralleled the Duane Arnold uprate review.

- NRR's review of the areas of reactor core and fuel performance, ECCS function and performance, capability of the SLCS, and reactor safety performance features found sufficient technical bases to allow operation at the EPU level.

- The review of plant systems found the uprate application acceptable, with the exception of the open issue concerning the need for a water source from the ultimate heat sink to the isolation condenser that is seismically qualified at the Dresden site.

- Regarding the staff's review of the Exelon risk assessment for power uprate, NRR stated that while some issues were identified relevant to the fidelity of the PRA analysis vis-a-vis actual plant configurations, overall the quality of the PRA was very good. Some models were "bounding" in nature, because some plant systems are undergoing modifications in support of the uprate. The impact of the uprate on plant risk was shown to be minor.

- The staff indicated that Exelon's use of the EPRI CHECWORKS code for evaluation of erosion/corrosion is acceptable as the code data base bounds DAEC uprate conditions.

<u>Subcommittee Comments</u>

- Mr. Schrock, citing the issue with the differences in the "MELLLA" curves on the power/flow maps for the two plants, expressed concern that the staff SERs lacked adequate detail to sort out this issue.

- In response to Dr. Wallis, NRR said that one of the staff's review objectives is to verify that the GE and licensee's calculational methods are adequate.

- NRR said, in response to Dr. Kress, that GE demonstrated that the ECCS core spray distribution tests bound uprate conditions. GE also said that they would provide additional detail on this point.

- Mr. Schrock , in questioning the staff with regard to modifications made to one of the codes in the SAFER/GESTR methodology, said that specific information is not available to allow one to judge the acceptably of this particular code.

- NRR said, in response to questions from the Subcommittee, that they would verify that the licensee will be inspecting the brackets that support the steam dryer in the upper vessel.

Subcommittee Caucus (**Open**)

The following comments were noted by the Subcommittee Members/Consultant:

Dr. Shack - No comment.

Dr. Ford - Cited his conflict with GE for this review. He noted his personal satisfaction with the licensee's management of material degradation issues.

Mr. Sieber - Requested information relative to required changes to the setpoint methodology to accommodate EPU operation. [Note: this information was provided to Mr. Sieber via a E-Mail from Mr. S. Bailey, NRR]. He also noted his previous questions with regard to the planned modifications to the reserve/auxiliary transformers. The licensee said that they would provide response.

Dr. Kress - Requested licensee to verify that the LOCA codes are applicable for the uprated cores, given reliance on older test data, in particular the core spray tests. He also requested that the licensee clarify why the MELLLA curves are different for the two plants, given use of the same methods.

Mr. Schrock - Doesn't understand why the staff is struggling with a decision on the need for conduct of large transient tests. While the staff's review is probably acceptable, he is concerned with the staff's reliance on previously approved methods, absent a reevaluation.

The Subcommittee agreed that this matter should be brought to the full Committee for review. The licensee and the staff were given direction regarding the scope and content of their presentations before the ACRS.

FOLLOW-ON ACTIONS

In response to Mr. Sieber, the staff was to provide material pertaining to the setpoint methodology to be employed for these uprate plants. [This material was provided to Mr. Sieber.] The licensee is to determine if the modifications made to the plants' auxiliary transformers results in a vulnerability to overcurrent conditions.

In response to Mr. Schrock, the licensee will investigate the reason for the differences in the natural circulation curve on the power/flow maps for the two plant sites, and will address the use of the ORIGEN code, vis-a-vis modeling of spatial distributions.

The licensee and staff will verify that the brackets supporting the steam dryer are to be inspected on a routine basis.

The licensee will address the applicability of the core spray test data to EPU conditions.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

Memorandum, dated October 12, 2001, from P. Boehnert, ACRS, to T/H Phenomena Subcommittee containing:

- Memorandum, J. Zwolinski, NRR, to J. Larkins, ACRS, dated October 10, 2001, transmitting NRR (draft) Safety Evaluation Reports by the Office of Nuclear Reactor Regulation for the Dresden Nuclear Power Station, Units 2&3, and the Quad Cities Nuclear Power Station, Units 1&2 Core Power Uprates

- GE Topical Report, NEDC-32962P, "Safety Analysis Report for Dresden 2&3 Extended Power Uprate", dated December 2000 (**Proprietary**)

- GE Topical Report, NEDC-32961P, "Safety Analysis Report for Quad Cities 1&2 Extended Power Uprate", dated December 2000 (**Proprietary**)

- Exelon Generating Company Memoranda: Response to Requests for Additional Information Supporting License Amendment Requests to Permit Uprated Power Operation, Dresden Nuclear Power Station and Quad Cities Nuclear Power Station, August 31, August 31, September 5, September 5, September 14, September 19, September 25, September 26, September 27 (contains proprietary information), and, September 27, 2001.

••

NOTE: Additional details of the open portions of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW", or can be purchased from Neal R. Gross & Co., Inc., 1323 Rhode Island Ave., NW, Washington, D.C., 20005, (202) 234-4433 (Voice), 387-7330 (Fax), E-Mail: "nrgross@nealrgross.com".

R

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING:
DRESDEN/QUAD CITIES NUCLEAR POWER STATION UPRATE REQUESTS
OCTOBER 25-26, 2001
ROCKVILLE, MARYLAND

PRESENTATION SCHEDULE

Contact: P. Boehnert (301/415-8065)
("pab2@nrc.gov")

October 25, 2001

TOPIC	PRESENTER	TIME
I. Opening Remarks	G. Wallis, Chairman,	1:00 p.m.
II. Dresden/Quad Cities Power Uprates		
A. Exelon Presentations (**Closed as Necessary**)		1:15 p.m.
1. Introduction	J. Nosko, Exelon	
2. Plant Changes to Accommodate Power Uprate[1]	J. Nosko	
3. Compliance with Regulatory Requirements - Hardware Modifications - Analyses Performed - Impact on Plant Margins	J. Nosko	
4. Review Issues:		
- Plant Operator Training/ Operator Action Times	T. Hanley, Exelon	
- Stability Monitor/ Instability Avoidance	J. Post, GE/ T. Hanley	
- ATWS Event Response for Uprate Conditions	J. Post/ J. Freeman, Exelon	
- Fuel response for ATWS/ Power Excursions	J. Post/ J. Freeman	

[1] For Discussion Items 2-7, note differences in plant designs as necessary.

TOPIC	PRESENTER	TIME
- Material Degradation Issues (e.g.,IASCC/FAC)	H. Crockett, Exelon	
- Impact on Containment Response/Load Limits	M. Kluge, Exelon	
- Piping Stress Limits	N. Hanley, Stone& Webster	
- Inspection Program	K. Moser, Exelon	
- Impact of Uprate on Steam Separator/Dryer System	K. Moser	
5. ECCS Analyses	D. Pappone, GE/	
- Results	J. Freeman	
6. Resolution of Open Issues		
- NPSH Requirements	M. Kluge	
- Ultimate Heat Sink	M. Kluge	
- ATWS Audit	J. Freeman	
- Transient Tests	T. Hanley	
7. PRA Analyses	W. Burchill, Exelon	
- Analyses Performed/ Results (CDF/LERF, Δ CDFs & LERFs, Associated Uncertainties[2])		
- Evaluation of Missing PRA Elements[3] (e.g., fire, seismic, shutdown risk)		
8. Concluding Remarks	J. Benjamin, Exelon	
III. Recess		6:00 p.m.

[2] If uncertainty evaluations not performed, provide a qualitative discussion of this matter.

[3] If quantitative analyses not available, provide rigorous qualitative arguments.

October 26, 2001

TOPIC	PRESENTER	TIME
IV. Reconvene	G. Wallis, Chairman	8:30 a.m.
V. Dresden/Quad Cities Uprate (Cont.)		
B. NRR Presentations (**Open**)		8:40 a.m.
1. Introduction - Staff Evaluation - Response to Duane Arnold Review - Major Transient Testing	S. Bajwa/L. Rossbach	
2. Reactor Systems Review	R. Caruso	
3. Plant Systems Review	R. Architzel	
4. Response to ACRS Questions		
- Material Degradation Issues	E. Andruszkiewicz/ K. Parczewski	
- Need for Pipe and Support Modifications	K. Manoly	
- Need for Electrical Modifications	A. Pal	
- PRA Analyses/Evaluation Of Licensee's PRA Examinations	D. Harrison	
5. Concluding Remarks		
C. Subcommittee Caucus (**Open**)		12:00 p.m.
1. Comments on Meeting Presentations		
2. Follow-on Actions		
3. Decision to Bring Review to ACRS/ Instructions to Presenters		
VI. Adjourn		12:30 p.m.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON THERMAL-HYDRAULIC PHENOMENA

OCTOBER 25, 2001

ATTENDEES PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	AFFILIATION
Bob Kitchen	CP&L - Progress Energy
Bill Bohlke	Exelon
Bill Burchill	Exelon
MARK KLUGE	EXELON
LAWRENCE LEE	ERIN
HAROLD CROCKETT	EXELON
DAVID C. TUBBS	MidAmerican Energy
Norm Hamla	Stone & Webster
John Nosko	Exelon
Ted Madden	Stone & Webster
Henry L. Jones	TVA
Jack Benjamin	Exelon
Jens Anderson	GNF
GEORGE NELSON	GENE
George Strombach	GENE
JOHN FREEMAN	EXELON
Israel Nir	GE
Ed Hartwig	TVA - Browns Ferry
DAN PAPPONE	GENE
Dale Spencer	Exelon

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON THERMAL-HYDRAULIC PHENOMENA

OCTOBER 25, 2001

ATTENDEES PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	AFFILIATION
G. Tietz	S&L
Jason Post	GENE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON THERMAL-HYDRAULIC PHENOMENA

OCTOBER 25, 2001

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Jim Wiggington	DIPM, NRR
John Hannon	DSSA/NRR
RALAT CHEMO	NRR/SRXB
Donnie Harrison	NRR/SPSB
Larry Rossbach	NRR/DLPM
Rob Elliott	NRR/SPLB
Steven Jones	NRR/SPLB
Stewart Bailey	NRR/DLPM
Cheng-In Wu	NRR/EMEB
Ed Andruszkiewicz	NRR/DE/EMCB
Ed Kendrick	NRR DSSA SRXB
S. Singh Bajwa	NRR/DLPM/PD III
R. Architzel	NRR/DSSA/SPLB
R. Eckenrode	NRR/DIPM/IQMB
K. Parczewski	NRR/DE/EMCB
George Georgiev	NRR/DE/EMCB
Juan Peralta	NRR/DIPM
George Hubbard	NRR/SPLB
Joseph Staudenmeier	RES/DSARE/SMSAB
Tai Huang	NRR/DSSA/SRXB
Mohammed Shuaibi	NRR/DSSA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON THERMAL-HYDRAULIC PHENOMENA

OCTOBER 26, 2001

ATTENDEES PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	AFFILIATION
~~Don~~ AL HAEGER	Exelon
JOHN NOSKO	EXELON
Tim Hanley	Exelon
Ed Hartwig	TVA-Browns Ferry
Henry L. Jones	TVA
Jason Post	GE
Bob Kitchen	CP&L - Progress Energy
DAVID C. TUBBS	MidAmerican Energy
John Hannon	NRR/DSSA
Israel Nir	GE
TED MADDEN	Stone & Webster
George Strambach	GE NE
Norm Hanley	Stone & Webster
DAN PAPPONE	GE NE
Jens Anderson	GAF (GE)
LAWRENCE LEE	ERIN
GEORGE NELSON	GE
JOHN FREEMAN	EXELON
Bill Byrdsill	Exelon
~~GEE~~ Jerry Tietz	Sargent & Lundy
Dale Spencer	Exelon

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON THERMAL-HYDRAULIC PHENOMENA

OCTOBER 26, 2001

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
George GEORGIEV	NRR/DE/EMCB
C. Hoxden	NRR/DE/EEEB
C E Carpenter	NRR/DE/EMCB
Richard Eckenrode	NRR/DIPM/IQMB
K. Parczewski	NRR/DE/EMCB
Steven Jones	NRR/DSSA/SPLB
Tai Huang	NRR/DSSA/SRXB
Frank Akstulewicz	NRR/DSSA/SRXB
Paul Gill	NRR/DE/EELB
Ralph Architzel	NRR/DSSA/SPLB
Camal Manoly	NRR/DE/EMEB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON THERMAL-HYDRAULIC PHENOMENA

OCTOBER 26, 2001

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Donnie Harrison	NRR/DSSA/SPSB
JIM WIGGINTON	NRR/DIPM/IRSP
RALPH CARUSO	NRR/SRXB
Stewart Bailey	NRR/DLPM
Edward Kendrick	NRR/DSSA/SRXB
Cheng-Ih Wu	NRR/DE/EMEB
Jared Wermiel	NRR/DSSA/SRXB
RICH BARRETT	NRR/DSSA/SPSB
Juan Peralta	NRR/DIPM
Zena Abdullahi	NRR/SRXB
EDWARD ANDRUSZKIEWICZ	NRR/DE/EMCB
BARRY ELLIOT	NRR/DE/EMCB
BRIAN THOMAS	NRR/DSSA/SPLB
TED QUAY	NRR/DIPM/IQPB
AMAR PAL	NRR/DE/EEIB
George Hubbard	NRR/SPLB
Mark Rubin	NRC
J H. Raval	NRC/SPLB/DSSA
HAROLD WALKER	NRC/NRR/SPLB
Brenda Mozafari	NRC/NRR/PDIII-1

INTRODUCTION

DRESDEN 2 & 3 AND QUAD CITIES 1 & 2
EXTENDED POWER UPRATES

ACRS THERMAL HYDRAULIC PHENOMENA
SUBCOMMITTEE MEETING
OCTOBER 26, 2001

S. SINGH BAJWA
DIRECTOR, PROJECT DIRECTORATE III
DIVISION OF LICENSING PROJECT
MANAGEMENT

ACRS FEEDBACK ON DUANE ARNOLD REVIEW

- The Safety Evaluation (SE) should be revised to document adequately the technical resolution of the issues raised by the staff
- The staff should develop improved guidance on the detail to be provided in SEs
- The staff should develop criteria for when independent assessments should be performed to complement its reviews of applicant submittals.

REVIEW METHODOLOGY

- Template Reviews
- Standard Review Plan
- Extent of Staff Review
- Staff Expertise
- Reliance on Technical Reviews Performed on Generic GE Topical Reports
- Management Oversight and Involvement

NRC STAFF REVIEW APPROACH

- ELTR1 AND ELTR2
- Standard Review Plan
- Monticello EPU Safety Evaluation

NRC STAFF REVIEW APPROACH

- Paralleled Duane Arnold EPU Review
- Requests for Additional Information
- Three audits

PRINCIPLE AREAS OF REVIEW

- REACTOR CORE AND FUEL PERFORMANCE
- REACTOR COOLANT SYSTEM
- CONTAINMENT ANALYSES
- EMERGENCY CORE COOLING SYSTEM PERFORMANCE EVALUATION

PRINCIPLE AREAS OF REVIEW

- INSTRUMENTATION AND CONTROLS
- ELECTRICAL POWER AND POWER CONVERSION SYSTEMS
- AUXILIARY SYSTEMS
- RADIOLOGICAL CONSEQUENCES

PRINCIPLE AREAS OF REVIEW

- SPECIAL EVENTS/LIMITING OPERATIONAL TRANSIENTS
- PRA
- HUMAN PERFORMANCE
- ENVIRONMENTAL ASSESSMENT

ORDER OF NRR PRESENTATION

- Introduction
- Reactor Systems Review
- Plant Systems Review

- Response to ACRS questions
  - Material Degradation Issues
  - Need for Pipe and Support Modifications
  - Need for Electrical Modifications
  - PRA Analyses and Evaluation

Dresden/Quad Cities EPU

Fuel and Reactor Systems

Ralph Caruso
Section Chief
BWR Systems and Nuclear Performance

Background

- Extended Power Uprate (EPU) includes
  - implementation of MELLLA and ARTS
  - introduction of GE14 fuel into mixed vendor core
  - 17 (17.8) % increase from the original rated thermal power (ORTP)
- SRXB staff performed
  - On-Site Audit at GNF-Wilmington
    - reviewed compliance with NRC-approved analytical methods and codes
    - performed audits of Dresden/Quad Cities EPU system performance and design bases safety analyses
  - Review of the EPU safety analysis reports (PUSAR), NEDC-32961P and NEDC-32962 (EPU submittal)

REVIEW SCOPE

- Staff reviewed the impact of the extended power uprate (EPU) on
  - Reactor Core and Fuel Performance
    - fuel design and operation, thermal limits assessment, operation along the MELLLA operating domain, stability detect and suppress capability and control rod drive system integrity.
  - Reactor Coolant System and Connected Systems
    - Overpressure protection, capability of the Recirculation system, RCIC/IC, and SDC to perform their safety functions.
  - ECCS system function and performance during LOCA (HPCI, LPCI, CS, ADS and LOCA limits
  - Capability of the SLC system
  - Reactor Safety Performance Features
    - Design basis safety analyses, ATWS, ATWS instability and SBO

CONCLUSION

- Based on the staff's on-site audit and review of the safety analysis report supporting operation at the higher power level, the staff believes:
  - The licensee has demonstrated that Dresden/Quad Cities can operate safely with EPU during steady state, AOOs and accident conditions as designed and licensed
  - The design basis analysis and the fuel design would be performed according to the NRC-approved methodology (process), analytical methods and codes.
  - The cycle-specific analyses for subsequent reloads would ensure plant operation within the thermal limits
  - The licensee would operate the plants in compliance with the applicable NRC regulations (i.e. GDC, ASME overpressure, Stability detect and suppress, ATWS rule, SBO).

Power Uprate Safety Analyses

- Scope and EPU evaluations followed NRC-accepted, generic EPU guidelines and evaluations (ELTR1, ELTR2, and Supplement 1 to ELTR2)

- Analyses and evaluations are based on NRC-approved methodologies, analytical methods and codes

SRXB Review

- On-site audit
  - safety analyses and system and component performance evaluations.
  - compliance with the NRC-approved analytical methods and codes

PUSAR review
- Reactor Core and Fuel Performance
- Reactor Coolant System
- Engineered Safety Features
- Instrumentation and Control
- Standby Liquid Control System
- Reactor Safety Performance Features

Reactor Core and Fuel Performance

2.1 FUEL DESIGN AND OPERATION

- Fuel system design criteria
  - Discussed in SRP Section 4.2
    - ensure fuel bundles not damaged during normal operation and AOOs
    - any damage would not prevent CR insertion
    - number of fuel rod failures during accident are not underestimated
    - coolability always ensured
  - NRC-approved the fuel design methodology, the analytical methods, and the codes used to ensure that the fuel design meet the fuel design acceptance criteria (based on regulatory guide, GDC, regulation and industry standards.)

Reactor Core and Fuel Performance

2.1 FUEL DESIGN AND OPERATION

- Fuel vendor (GNF) performs thermal-mechanical, thermal-hydraulic, neutronic and material analysis to ensure that the fuel system design meets the applicable fuel design acceptance criteria during steady state, AOO or accident conditions.

- New fuel design (i.e GE14 fuel) must conform with NRC-approved fuel design methodology (GESTAR-II).

Reactor Core and Fuel Performance

2.6 SRXB EPU AUDIT

- Onsite review covered compliance with the NRC-approved methodology in performing the fuel system design.
  - Staff determined insufficient experimental data to qualify GE14 fuel design CPR correlation for up-skew and down-skew power profiles
  - GE has recorrelated CPR and submitted for staff review
  - Changes to SL MCPR would require TS submittal and staff review and approval

Reactor Core and Fuel Performance

2.2 THERMAL LIMITS ASSESSMENT

- Licensee performed thermal limits evaluation based on
  - representative "bounding unit" GE 14 equilibrium core
  - Operation along the MELLLA rod line at rated and offrated conditions
  - 18 % increase from the ORTP

Reactor Core and Fuel Performance

2.2 THERMAL LIMITS ASSESSMENT

- Thermal limits are established or confirmed every reload
- COLR provides the cycle-specific thermal limits.
- SLMCPR changes require TS amendment request and staff review and approval
- No change in licensed burnup limits

Reactor Core and Fuel Performance

- Impact on Margin
  - Plant operation must ensure adherence to the cycle-specific thermal limits (SL MCPR, OL MCPR, MAPLHGR, LHGR, LOCA)
  - Margin to licensing limits is available to licensee
- Conclusions
  - Equilibrium "bounding unit" core EPU thermal limit evaluations show that Dresden/Quad Cities reload cores can be designed to operate within the fuel design limits during steady state, AOOs and accident conditions.

Reactor Core and Fuel Performance

2.4 STABILITY

- Dresden/Quad Cities are Option III Long Term Solution plants
  - Currently operating under requirements of Reactor Stability Interim Corrective Actions (ICA)
  - ABB Option III Oscillation Power range Monitor (OPRM) instrumentation
  - OPRM installed, not arming prior to EPU

- Option III (LPRM-based detect and suppress) includes
  - Four independent OPRM channels (one per RPS channel)
  - Each OPRM channel consists of two modules (either can generate a channel trip signal)
  - Each OPRM module receives signal from dedicated LPRMs and provides LPRM signals to the other module in its channel through a fiber optic data link)
  - OPRM module combines the locally wired LPRM signals with the shared LPRM signals to create LPRM cells that represent the neutron flux distribution in the reactor core
  - Microprocessor in each module uses the cells of LPRM signals to calculate the trip function values with the oscillation detection algorithm (ODA)

Reactor Core and Fuel Performance

2.4 STABILITY

- Stability Detect and Suppress
  - Trip Enabled Region is core and fuel dependent and is also affected by the rated core power and the corresponding operating conditions.
  - The reload methodology defines the setpoints used to limit the oscillation size in order to provide adequate MCPR safety limit protection.

Reactor Core and Fuel Performance

2.4 STABILITY

- Stability - Part 21 Reporting
  - Divom (**D**elta CPR/**I**nitial **V**s. **O**scillation **M**agnitude) curve
    - normalized CPR change vs. Oscillation size
    - Generic regional mode and core-wide DIVOM curves used in the stability licensing methodology
    - Option III uses the regional mode DIVOM curve
    - BWROG reconvened the Detect and Suppress Committee to resolve stability detect and suppress reload licensing calculation issue by 2Q02

Reactor Core and Fuel Performance

2.4 STABILITY

- Staff On-Site Audit
  - Discussed Dresden/Quad Cities implementation plan for the Option III stability solution for EPU operation.
  - Reviewed the status of the generic DIVOM curve issue for EPU operation.
  - Followup on GE response to internal GE document questioning the applicability of the DIVOM curves for the EPU operation using GE 14 fuel with higher bundle powers.
  - Staff concluded from the audit that Option III was still applicable to Dresden/ Quad Cities EPU operation, subject to resolution of Part 21 DIVOM curve issue

Reactor Core and Fuel Performance

2.4 STABILITY

- Part 21 Report
  - GE determined
    - generic regional mode DIVOM curve is strongly affected by the peak bundle power
    - effect of the peak bundle power was not adequately modeled in the original development of the generic curves.
    - Plant reloads could be designed with operating peak bundle power above the peak bundle power considered during the development of the DIVOM curves.

Reactor Core and Fuel Performance

2.4 STABILITY

- Part 21 Response
  - GE recommended that licensees use a stability figure of merit for each DIVOM curve to determine the applicability of the existing generic curves
- CONCLUSION
  - The licensee has chosen to postpone implementation of Option III until issue is resolved and will retain current approved ICA restrictions

System and Component Performance

General Discussions

- The EPU review included performance of safety related systems and engineering safety features to perform their design functions and ensure operation within the thermal limits.

- Some of the design basis analyses were based on the representative "bounding unit" equilibrium core and will be reanalyzed or reconfirmed in the cycle-specific EPU reload analysis. (i.e. ASME overpressure - nuclear steam pressure relief)

ECCS Performance

- Licensee performed the EPU LOCA analysis for "bounding unit" equilibrium GE14 core

- Staff evaluated Dresden pre-EPU and EPU analyses for LOCA analysis.
  - LOCA analyses based on NRC-approved SAFER/GESTR methodology
  - determined that GENE followed NRC-approved process in their ECCS-LOCA analysis

- Conclusion
  - EPU ECCS-LOCA performance complies with 10 CFR 50.46 and Appendix K requirements and the analyses conform to NRC-approved methods and codes.

Reactor Performance Evaluation

SRXB EPU ON-SITE AUDIT REVIEWS

- 2 Reactor Core and Fuel Performance
- 2.1 Fuel Design and Operation
- 2.2 Thermal Limits Assessment
- 2.3 Reactivity Characteristics
- 2.4 Stability

- 9 Reactor Safety Performance Evaluations
- 9.1 Reactor Transients
- 9.2 Design Basis Accidents
- 9.4 Special Events

Reactor Safety Performance Features

9.1 REACTOR TRANSIENTS

- Licensee analyzed the limiting transients based on representative "bounding unit" equilibrium core for the EPU operation
- The transients analyzed are consistent with Appendix E of ELTR1 and are based on NRC-approved methodology and codes
- Limiting transients would be reanalyzed or reconfirmed based on actual core design and the uprated power level for each transition reload EPU as well as in the subsequent cycles
- Conclusion
  - Licensee demonstrated that the EPU transient analyses do not identify any major changes to the basic characteristics of any of the limiting transients

Special Events

9.4.1 ATWS

- Dresden/Quad Cities meets the ATWS mitigation features specified in 10 CFR 50.62 (ARI, ATWS-RPT, SLC-boron injection capability)

- Licensee analyzed four ATWS events for the EPU operating conditions (MSIVC, PRFO, LOOP and IORV)

- Conclusion

  - Licensee meets the requirements in 10 CFR 50.62 and the ATWS acceptance criteria for the EPU operation (i.e. limiting peak vessel bottom pressure and peak cladding temperature)
  - Future reload analyses would confirm that the plant's response to an ATWS event will meet ATWS acceptance criteria.

Conclusions

- The staff safety evaluation was based on reviews establishing that the:
  - licensing analyses are based on NRC-approved acceptance criteria, methodology (process), analytical methods and codes (i.e. fuel design acceptance criteria)
  - review included on-site audit of compliance to approved methodology
  - EPU SAR is consistent with NRC-accepted guidelines and generic evaluations for evaluating the impact of extended power uprate
  - thermal limits and the applicable safety analyses would be reanalyzed or reconfirmed based on cycle specific core configuration and conditions.
  - EPU analysis results and justifications support the proposed power uprate.
- In SRXB areas of review, the staff finds that the licensee provided sufficient technical bases to support operation of Dresden/Quad Cities at the proposed power level of 2957 MWt

Standby Liquid Control System

6.5 SLC SYSTEM

- The SLC system is a manually initiated system
- Minimum reactor boron concentration and tank inventory confirmed to be adequate for introduction of new fuel design and for EPU operation.
- SLCS shutdown capability is reconfirmed for every reload
- Licensee requested to confirm SLCS will be able to inject the required flow rate at the injection time assumed in the ATWS analyses
- Licensee plant/cycle-specific analyses for Dresden 2, Cycle 18 operation at EPU conditions confims adequate margin to inject against system pressure at the limiting transient conditions

R

Exelon Nuclear

Extended Power Uprate

Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2

Exelon Nuclear
presentation to
Thermal-Hydraulic Phenomena Subcommittee
Advisory Committee on Reactor Safeguards
October 25, 2001

Agenda

- Introduction
- Plant modifications
- Selected analyses and evaluations
- EPU risk evaluation
- Resolution of open items
- Project implementation
- Conclusion

Introduction

John Nosko, Exelon Nuclear

EPU Project Manager

Introduction

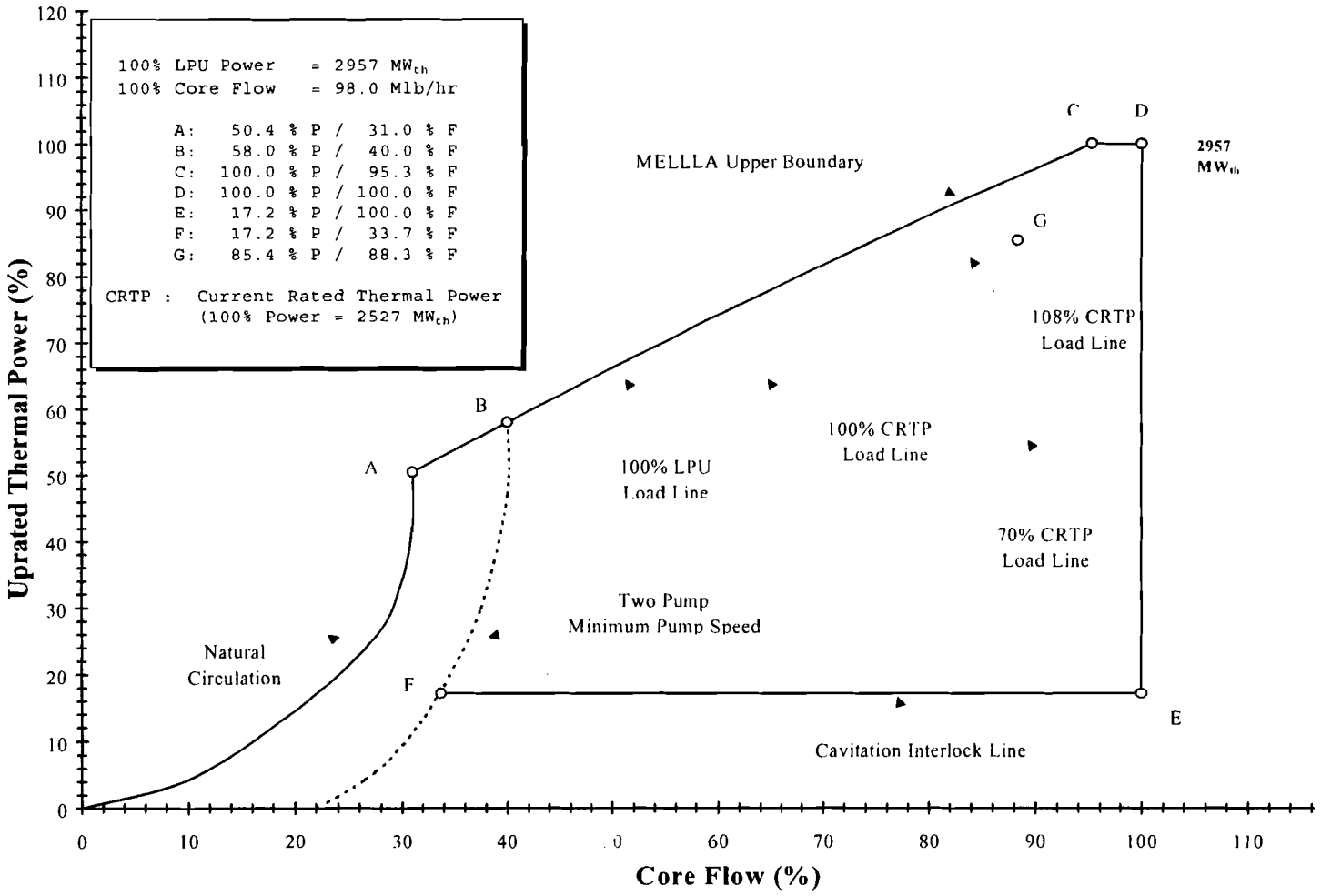
- Project goals
  - Safely use installed excess capacity at stations
  - Use accepted methodology to leverage industry experience
  - Maintain constant reactor dome pressure to minimize impact of the uprate
  - Drive for commonality between stations to simplify analyses, designs, reviews, and configuration management

Introduction

- Compliance with regulatory requirements
  - Submittal prepared in accordance with ELTR-1 and ELTR-2 (with one exception taken)
    - Exception for large transient testing
  - Demonstrating compliance with applicable regulations and safety limits
  - Analyses performed
    - Reactor operating conditions, accidents, and transients
    - Radiological consequences
    - Environmental impact
    - Probabilistic risk
    - Programmatic evaluations

Exelon Nuclear

MELLLA Domain



Exelon Nuclear

Current and EPU Operating Conditions

Parameter	Current Rated Power Value	Extended Power Uprate Value
Thermal Power (MWth)	2527 (DNPS)	2957
Vessel Steam Flow (Mlb/hr)	9.81	11.71
Full Power Core Flow Range		
Mlb/hr	85.3 to 98	93.4 to 98
% Rated	87 to 100	95.3 to 100
Dome Pressure (psig)	1005	No change
Dome Temperature (°F)	547.0	No change
Turbine Inlet Pressure (psig)	935.0	906.0
Full Power Feedwater		
Flow (Mlb/hr)	9.78	11.68
Temperature Range (°F)	350 to 250	356 to 256

Introduction

- Project overview
  - Using installed spare equipment
  - Uprates to be accomplished after one outage
  - Units will be generator-limited following uprate
- Implementation schedule
  - Dresden 2 (outage now in progress), uprate on-line
  - Quad Cities 2 (February, 2002), uprate following outage
  - Dresden 3 (October, 2002), uprate following outage
  - Quad Cities 1 (November, 2002), uprate following outage

Exelon Nuclear

Plant Modifications

John Nosko, Exelon Nuclear

EPU Project Manager

Plant Modifications

- Generally require the same modifications at both stations
- Relatively few modifications to safety-related structures, systems, and components
- Majority of changes being made to balance of plant systems

Plant Modifications

Reactor

- Transition to GE14 fuel
- Steam dryer
- Jet pump sensing line clamps
- Reactor recirculation system runback
- Low reactor water level scram setpoint change
- Low pressure coolant injection swing bus timer changes
- Isolation condenser time delay relay change (DNPS only)
- Nuclear instrumentation setpoint changes

Plant Modifications

Turbine-Generator Systems

- New high pressure turbine
- Cross-around relief valve setpoints
- Steam line resonance compensator card
- Isolated phase bus duct cooling (DNPS only)

Exelon Nuclear

Plant Modifications

Condensate/Feedwater Systems

- Condenser tube staking (DNPS only)
- Condensate demineralizer system changes
- Feedwater heater rerating
- Heater drain system valves (DNPS only)
- Staggered reactor feed pump low suction pressure trips (QCNPS only)
- New digital feedwater control system being installed (QCNPS only)

Plant Modifications

Other

- Main steam and torus-attached piping supports
- Drywell steel beams and connections
- Non-safety related 4kV switchgear
- Trip 4th condensate pump on loss of coolant accident (LOCA) signal
- New cooling towers (DNPS only)

Selected Analyses and Evaluations

- A full scope of evaluations was performed in accordance with ELTR-1 and ELTR-2
- Analyses used NRC-approved methods within previously-accepted ranges
- In all cases, results were within acceptance criteria

Selected Analyses and Evaluations

- Analyses to be discussed
  - Containment
  - Emergency core cooling system (ECCS)
  - Thermal-hydraulic stability
  - Anticipated transient without scram (ATWS)
  - Piping
  - Reactor and internals
  - Flow accelerated corrosion (FAC)

Containment Analysis

Mark Kluge, Exelon Nuclear
Dan Pappone, GE

Introduction

- Methodology
- Results
  - Design basis accident (DBA)
  - Mark I loads
- Conclusion

Containment Analysis

Methodology

- Methods used
  - Short-term: M3CPT and LAMB
    - Used to calculate peak drywell pressure and temperature
  - Long-term: SHEX
    - Used to calculate wetwell pressure, pool temperature
  - Mark I loads: Mark I Long-Term Program
- EPU power within original code range
- Full spectrum of break sizes, locations evaluated
- Analysis used limiting input parameters

Exelon Nuclear

Containment Analysis

Results for Design Basis Accident

Parameter	Acceptance limit	Pre-EPU	EPU
Drywell pressure	62 psig	42.8 psig	43.9 psig
Drywell air temperature	340°F	289°F	291°F

Exelon Nuclear

Containment Analysis

Results for Design Basis Accident

Parameter	Acceptance limit	Pre-EPU	EPU
Pool temperature	281°F (structural limit)	193°F	202°F (bounding) 196°F (DNPS) 199°F (QCNPS)
Wetwell pressure	62 psig	36.6 psig	36.7 psig

Exelon Nuclear

Containment Analysis

Results for Mark I Loads

- Mark I loads
  - Pool Swell
  - Vent Thrust
  - Condensation Oscillation
  - Chugging
  - Safety/Relief valve
- Since reactor pressure is unchanged, the current Mark I loads remain bounding for EPU

Exelon Nuclear

Containment Analysis
Conclusion

- Summary
  - Used accepted methods within code development range
  - Used conservative input parameters
  - Results within acceptance criteria

Containment performance is acceptable for EPU conditions

ECCS Analysis

John Freeman, Exelon Nuclear

Dan Pappone, GE

Introduction

- Methodology and acceptance criteria
- EPU approach and results
- Conclusion

ECCS Analysis

Methodology

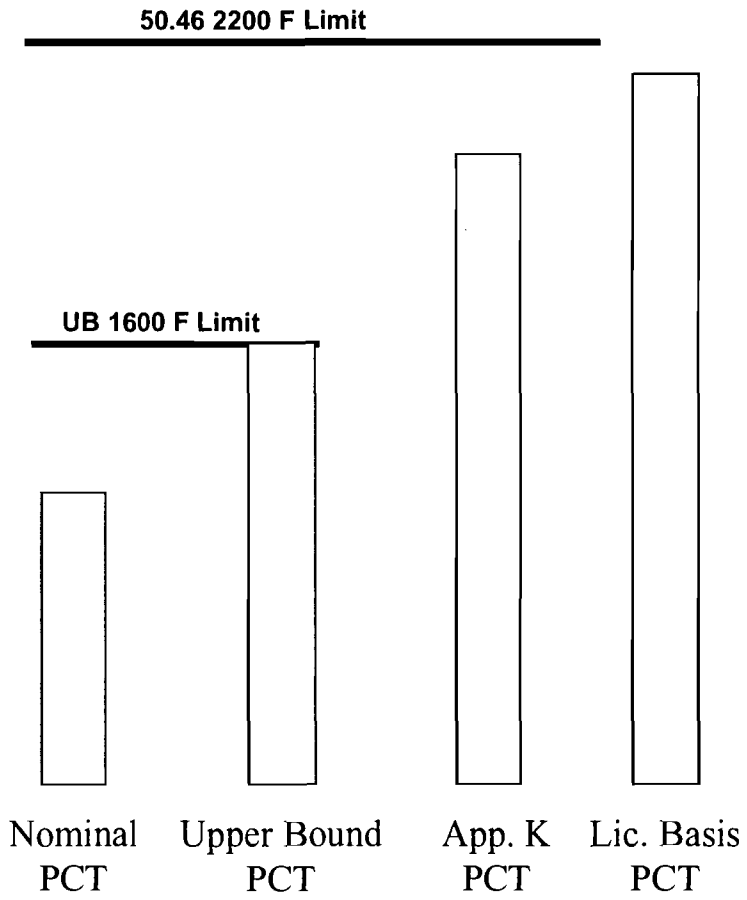
- **SAFER/GESTR-LOCA is NRC approved**
  - Applied as outlined in the ELTR
  - Realistic analysis for break spectrum using nominal models and inputs
  - Licensing basis peak cladding temperature (PCT) calculated for limiting nominal case using required Appendix K models
  - Upper bound PCT calculated for limiting nominal case
    - This approach demonstrates that the licensing basis PCT is conservative

Exelon Nuclear

ECCS Analysis

Acceptance Criteria

- Meet 10 CFR 50.46 limits
- Licensing basis PCT < 2200°F
- Licensing basis PCT > Upper bound PCT for SAFER/GESTR
- Upper bound PCT < 1600°F for SAFER/GESTR



Exelon Nuclear

ECCS Analysis

EPU Approach

- Full scope SAFER analysis
  - Update Quad Cities to current code version
  - Apply SAFER to Dresden
- Analyses for GE14 fuel at EPU conditions
- Calculated limiting PCT for legacy fuel types
- Compared pre- and post-EPU PCTs

Exelon Nuclear

ECCS Analysis

EPU Approach

- Full break spectrum and single failure analysis
- Breaks evaluated
  - Recirculation suction line break (DBA - limiting)
  - Large break and selected small breaks per App. K
  - Steam line breaks inside and outside containment
  - Core spray line break
  - Feedwater line break
- Single failure evaluation concluded DG failure is the most limiting

Exelon Nuclear

ECCS Analysis

Results

- Licensing basis PCT = 2110 °F (< 2200 °F)
- Upper bound PCT = 1600 °F (≤ 1600 °F)
- Maximum local oxidation = 6% (< 17%)
- Core wide metal-water reaction = 0.1% (< 1%)
- Coolable geometry, Long-term cooling demonstrated generically
  - Not affected by EPU
- EPU effect on PCT <10 °F
  - Small effect (consistent with ELTR, other EPUs)

Exelon Nuclear

ECCS Analysis

Results

- EPU effect on large break LOCA
  - A setpoint change will be performed to shorten the LPCI swing bus delay
- EPU effect on small break LOCA
  - EPU increases PCT for small break LOCA
    - Higher decay heat → longer vessel depressurization
  - Pre-EPU LOCA analyses assume 1 automatic depressurization system (ADS) valve out-of-service
  - EPU analysis shows that all 5 ADS valves are required to keep upper bound PCT below 1600°F

Exelon Nuclear

ECCS Analysis

Conclusion

- Methodology has conservative basis
  - Licensing basis PCT based on conservative Appendix K models
  - Upper bound PCT acceptance criteria are conservative
- Results indicate that all 10 CFR 50.46 criteria met

ECCS system performance is acceptable at EPU conditions

Exelon Nuclear

Thermal-Hydraulic Stability

Tim Hanley, Exelon Nuclear

Jason Post, GE

Introduction

- Background
- Methodology
- Analysis results
- Operational aspects
- Conclusion

Stability

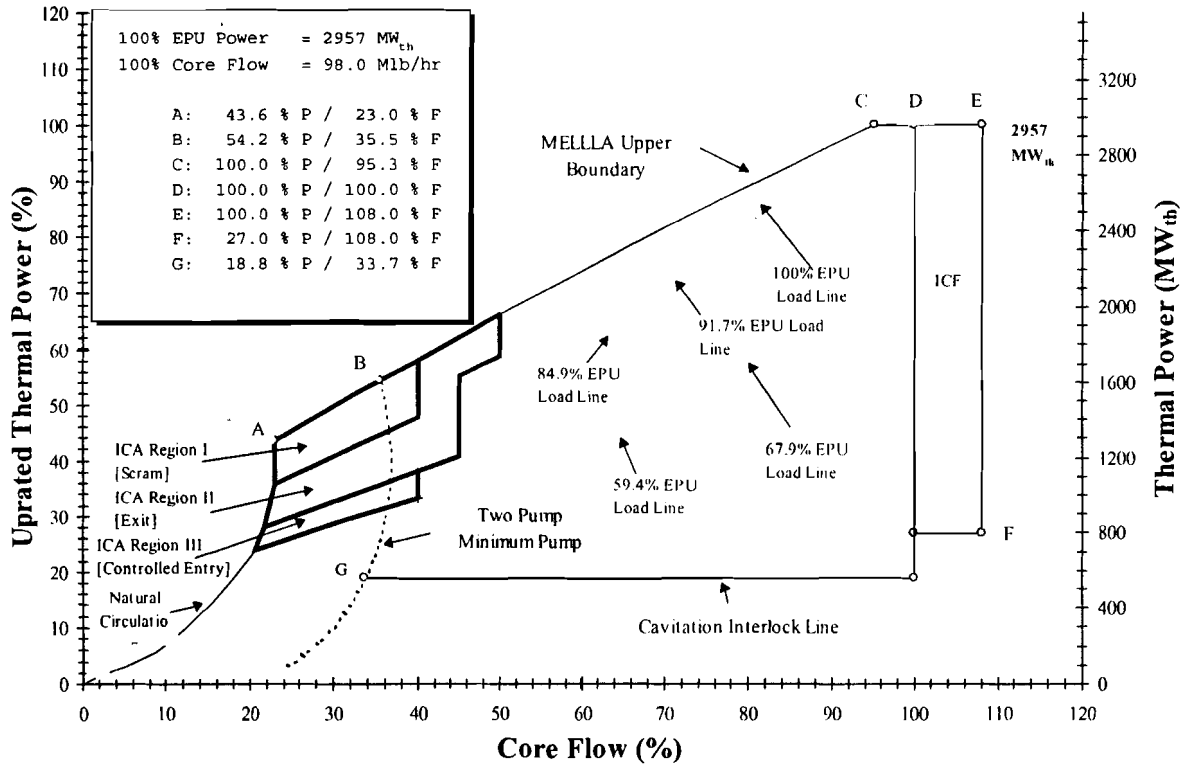
Background

- BWR Owner's Group interim corrective actions (ICAs) in place
  - ICAs provide manual prevention and suppression
- DNPS/QCNPS implementing stability solution Option III
  - Robust automatic "detect and suppress" solution
  - Detection algorithm implemented with new hardware: Oscillation Power Range Monitor (OPRM)
  - OPRM initiates reactor scram in the trip enabled region if oscillations reach OPRM count and amplitude setpoints

Exelon Nuclear

Stability

Methodology - ICA Power Flow Map

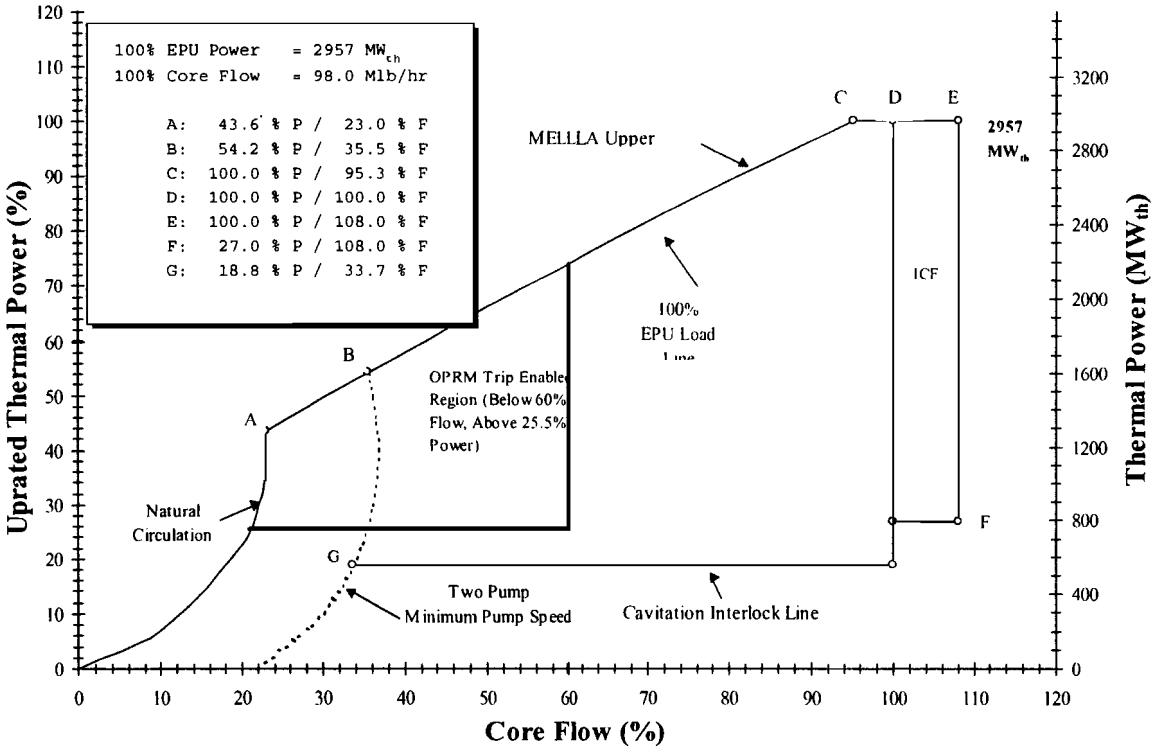


For EPU, ICA boundaries are rescaled to maintain the same absolute power and flow

Exelon Nuclear

Stability

Methodology - Option III Power-Flow Map



For EPU, trip enabled region boundaries are rescaled to maintain the same absolute power and flow

Stability

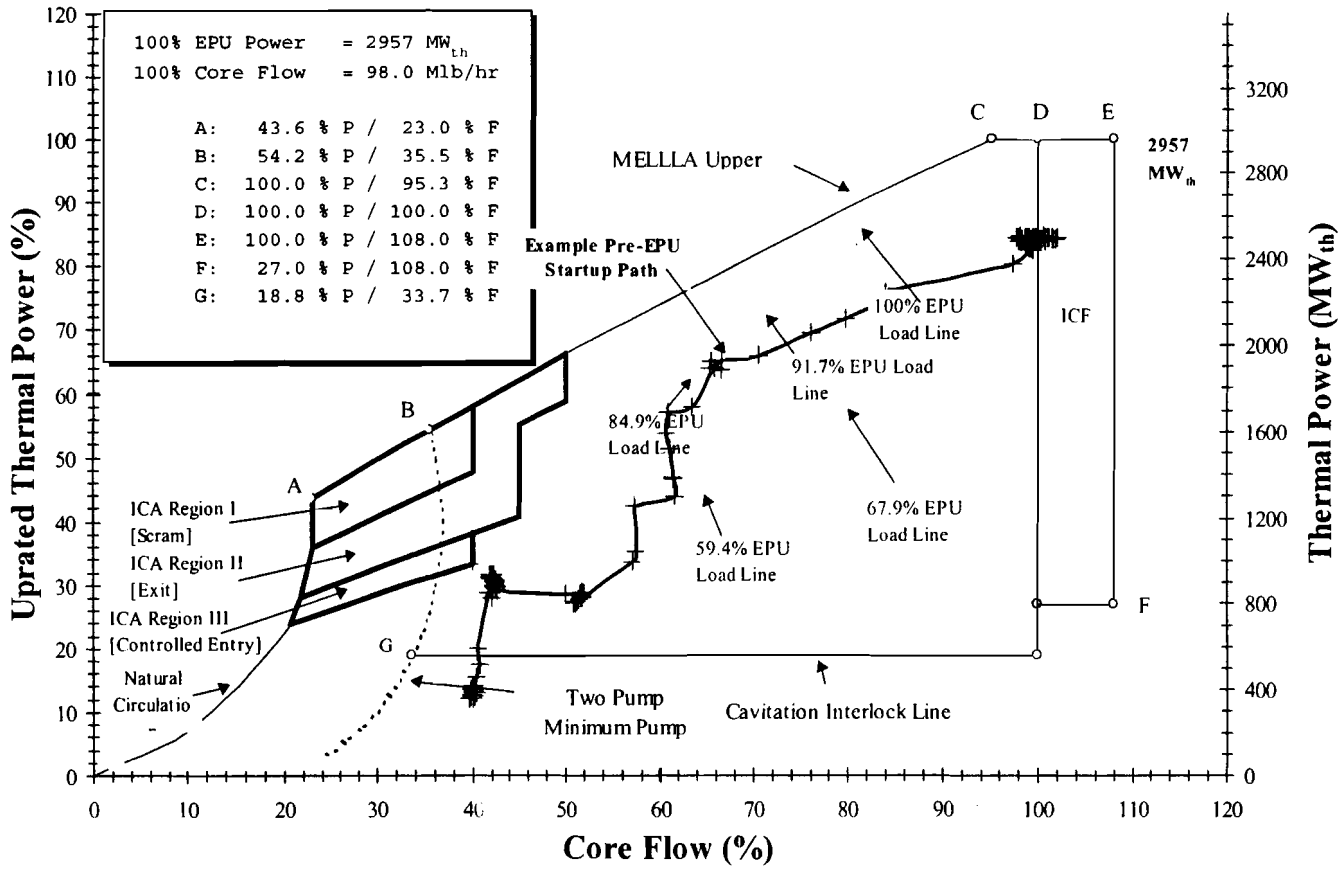
Analysis Results

- Cycle-specific OPRM setpoint calculation to ensure MCPR Safety Limit protection maintained
  - Hot bundle oscillation magnitude depends upon OPRM hardware; unaffected by EPU, MELLLA, GE14
  - CPR change vs. oscillation magnitude (DIVOM curve) currently being revised by BWROG and GE
  - Fuel specific CPR performance and limits addressed in cycle-specific analysis

Exelon Nuclear

Stability

Operational Aspects



Stability

Conclusion

- Expect DNPS/QCNPS EPU startup under ICAs
- ICAs and OPRM boundaries re-scaled to maintain pre-EPU power and flow conditions
- ICAs provide adequate protection to prevent and terminate oscillations
- EPU configuration and fuel loading does not affect operator actions

Thermal-hydraulic stability is acceptable at EPU conditions

Exelon Nuclear

Anticipated Transient Without Scram

John Freeman, Exelon Nuclear

Jason Post, GE

Introduction

- Methodology and assumptions
- ATWS instability
- Analysis results
- Conclusion

ATWS Analysis
Methodology

- ODYN code used for ATWS analysis
  - ODYN code is NRC-approved and was used within the restrictions of the approval for ATWS applications
  - Key conservatism: boron mixing efficiency in reactor
  - ODYN code bounds best estimate model (TRACG)

ATWS Analysis

Assumptions

- Operator action times
  - Initiate SLCS 2 minutes after ATWS indications
  - Pool cooling in service 10 minutes after event begins
  - Times unchanged for EPU
- Reactor relief and safety valve setpoints at maximum allowed by Technical Specifications

ATWS Instability

Background

- NEDO-32047-A
  - Response to ATWS instability with no mitigation
  - Portion of high power bundles experience extended dryout
  - 0.5% of fuel experiences damage
- NEDO-32614-A
  - Response to ATWS instability with mitigation
  - Water level reduction prevents fuel damage
  - Boron injection terminates instability
- Previous argument: generic studies applicable to EPU/MELLLA

ATWS Instability

Sensitivity Study

- ATWS instability sensitivity study performed
  - Rod line above MELLLA
  - EPU
  - GE14
- No mitigation case showed less severe fuel response
- Confirms expectation that
  - Generic studies are valid for EPU/MELLLA
  - Mitigation actions will be effective for EPU/MELLLA

Exelon Nuclear

ATWS Analysis

Results

Parameter	Acceptance Criteria	Pre-EPU results	EPU results	Limiting event
Peak Vessel Pressure	1500 psig	1402 psig	1492 psig (GE14) 1499 psig (transition)	PRFO
Peak Suppression Pool Temperature	202 °F	190 °F	201 °F	PRFO
Peak Containment Pressure	62 psig	13.2 psig	16.5 psig	PRFO
Peak Fuel Cladding Temperature	2200 °F	1478 °F	1418 °F	MSIVC
Maximum Local Oxidation	17%	Negligible	Negligible	MSIVC

PRFO - pressure regulator failure - open

MSIVC - main steam isolation valve closure

Exelon Nuclear

ATWS Analysis

Conclusion

- Methodology is appropriate for this application
- All ATWS criteria met

ATWS response is acceptable at EPU conditions

Piping Analysis

Norm Hanley, Stone and Webster

Introduction

- Methodology
- EPU impacts
- Conclusion

Piping Analysis
Methodology

- Determined the design and analytical basis
- Used original code of record, code allowables, and analytical techniques
  - Exception: evaluation of the main steam piping
- No changes in the analysis parameters for the majority of systems

Piping Analysis
Methodology

- EPU evaluations performed by determining "change factors" for the changes in thermal, pressure, flow rate, and total design load conditions
  - If change factor is less than 1.0, the pre-EPU (existing) conditions envelop or equal the power uprate conditions
  - For minor changes resulting in a change factor between 1.0 and 1.05 (or 5%), the increase was considered acceptable
    - Small increase is offset by conservatism inherent in the analytical methods used to calculate the existing stresses and loads

Piping Analysis

Methodology

- For change factors greater than 1.05, simple and conservative evaluations were performed to address the increase in stress and load values
- If the evaluated loads did not meet the acceptance criteria, a modification was designed

Piping Analysis

EPU Impact

- Torus attached piping will experience temperature increases due to long term post-LOCA conditions in which water is being drawn from the torus
  - Modifications address these increased stresses
- Main steam dynamic loads will increase due to increased steam velocity
  - Dynamic loads due to turbine stop valve closure were not part of original design basis
  - Modifications address these increased stresses

Piping Analysis

Conclusion

- Piping loads were comprehensively evaluated using standard methods
- EPU plant parameter changes are restricted to temperature and flow changes in a few systems
- Modifications resolve any stresses that increased beyond acceptance limits

Piping systems are acceptable at EPU conditions

Reactor and Internals

Keith Moser, Exelon Nuclear
Sam Ranganath, GE

Introduction

- Scope and methods
- EPU effects
- Modifications
- Conclusion

Reactor and Internals
Scope and Methods

- Exelon reactor internals asset management strategy involves a systematic review of components
  - Inspection, evaluation, mitigation and selective repair
  - Degradation modes – SCC and Fatigue addressed
- Component by component review for EPU effects
  - Current P-T curves conservative for one cycle of operation
  - All reactor internals evaluated for Flow Induced Vibration (FIV) due to increased steam flow and recirc. drive flow
    - Modifications needed for the steam dryer and selected jet pump sensing lines
    - All other components have required FIV margins for EPU flow

Exelon Nuclear

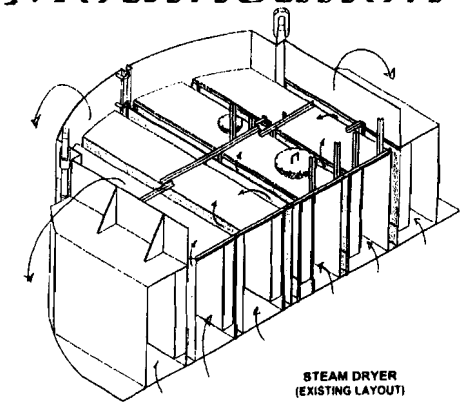
Reactor and Internals
EPU effects

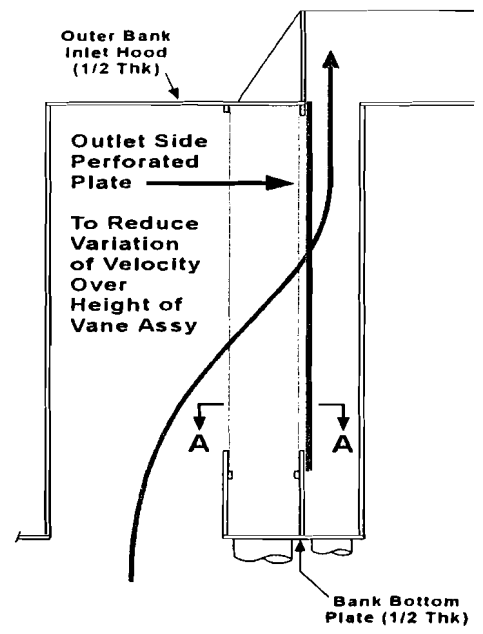
- Vibration due to increased recirculation drive flow analyzed
  - Vibration startup data extrapolated to EPU conditions
  - Except for 8 jet pump sensing lines, vibration stresses evaluated and shown to be acceptable (less than endurance limit)
    - Modifications to jet pump sensing line
- Need for steam dryer modification identified

Reactor and Internals

Steam Dryer Modification

- Increased steam flow leads to higher moisture content
  - Modification – addition of perforated plates
  - Reduces velocity variation
  - Moisture carryover $\leq$ 0.2%
- Provides additional stiffness for the dryer
  - Adds FIV margin
  - Field experience shows that higher stiffness improves FIV resistance





Reactor and Internals
Steam Dryer Experience

- Fatigue cracking has been observed in steam dryer
  - Drain channel and tie bar
- Not a safety concern; can be managed by inspections
  - Dryer cracking consequences addressed in BWRVIP-06
    - Steam dryer is not a safety related component
  - Lost part consequences shown to be acceptable
    - Loose parts expected to be large
    - Core shroud, shroud head form protective boundary
    - Lost dryer parts pose no threat to core and fuel integrity
    - MSIV closure is assured even with lost dryer parts
  - Pre- and post-EPU inspection to be performed on dryer
    - Steam dryer removed during each refueling outage. Cracks readily seen and corrective actions taken

Reactor and Internals
Conclusion

- Comprehensive evaluation of reactor and internals using approved methods
- Used recent industry experience
- Modifications and repairs performed

Reactor and internals
are acceptable at EPU conditions

Exelon Nuclear

Flow Accelerated Corrosion (FAC)

Harold Crockett, Exelon Nuclear

Introduction

- Predictive analysis methodology
- EPU impact
- Programmatic controls
- Conclusion

Exelon Nuclear

Flow Accelerated Corrosion

Predictive Analysis Methodology

- FAC affects carbon steel components in the steam cycle where process temperature exceeds 200° F
- DNPS and QCNPS use standardized programs to predict, detect, and monitor for FAC based on EPRI Guidelines
- EPRI CHECWORKS code used to evaluate EPU changes
- EPU conditions are bounded by CHECWORKS parameter ranges

Exelon Nuclear

Flow Accelerated Corrosion

Predictive Analysis Methodology

CHECWORKS Parameter	Allowable Input	Actual Values (pre-EPU)	Actual Values (post-EPU)	Comments
Steam rate	0 to 1E8 lb/hr	9.46E6 lb/hr	11.66E6 lb/hr	BWRs @ 14.1E6 PWRs @15.3E6
Velocity	Calc. in program	22.0 ft/sec FW	23.1 ft/sec FW	FW survey – 24.5 fps
Steam quality	0% to 100%	>99.8%	>99.8%	Water = 0% Dry steam = 100%
Op. temperature	0 to 750°F	340°F (FW)	356°F (FW)	BWRs @ 420°F PWRs @ 446°F

CONCLUSION: All values within parameters

Flow Accelerated Corrosion

EPU Impact

- Wear rate increased in some lines, based on predictive analysis
  - Largest percentage increase is 33% (from 1 mil/yr to 1.3 mil/yr)
  - Feedwater increase from 19 mil/yr to 21 mil/yr is representative of the more significant increases
- Most pre-uprate wear rates are low and therefore the small increases predicted are not significant
- Steam dryer modification to ensure steam line moisture has an insignificant effect on FAC

Exelon Nuclear

Flow Accelerated Corrosion

Programmatic Controls

- For lines with increased wear rates, inspections have been accelerated
- Additional margins exist
  - 1.1 factor of safety on wear rates.
  - Re-inspection interval ensures inspection at least 1 cycle before minimum allowable wall thickness reached
- Will continue to perform inspections on susceptible lines and compare to predictions
- Sample expansion will ensure extent of condition is assessed if unanticipated wear rates occur

Exelon Nuclear

Flow Accelerated Corrosion

Conclusion

- No new systems are affected by EPU
- Predictive analysis has been revised to determine potential impacts
- Inspections for affected components have been accelerated
- Inspection data is incorporated into the program

EPU conditions do not significantly affect FAC

Exelon Nuclear

EPU Risk Evaluation

Bill Burchill, Exelon Nuclear

Larry Lee, ERIN Engineering

Introduction

- Full power internal events (FPIE) quantitative risk evaluation
  - Evaluation methods
  - Dominant effects of EPU
  - Quantitative results
  - Uncertainties
- Qualitative risk evaluation
  - Internal flooding risk
  - Fire risk
  - Seismic risk
  - Shutdown risk
- Summary of EPU risk impacts

Purpose of EPU Risk Evaluation

- Estimate change in FPIE core damage frequency (CDF) produced by EPU
- Estimate change in FPIE large early release frequency (LERF) produced by EPU
- Identify qualitatively changes in risk from other sources, e.g. external events or shutdown state, produced by EPU
- Identify PRA revisions required to represent plants following EPU

FPIE Risk Evaluation

Methods

- Identify plant configuration changes due to EPU
- Use recently-updated PRA models
- Identify those PRA elements affected by plant configuration changes
- Use realistic models
- Compare with realistic success criteria and limits
- Model hardware and procedure changes using sensitivity cases with PRA model
- Compare results with Reg. Guide 1.174 guidelines

Exelon Nuclear

FPIE Risk Evaluation

Methods

- Reviewed PRA technical elements

-Initiating events	-Operator responses
-Success criteria	-Structural analysis
-Systems	-Quantification
-Data	-Event tree sequences

- Evaluated the impact on thermal-hydraulic parameters (e.g., time to boildown, core damage)

FPIE Risk Evaluation

EPU Effects - Operating Conditions

- Increased decay heat load reduces times to boil down in core, pool temperature limits, and core damage
  - More limiting success criteria and reduced time for equipment response and operator actions
- Increased ATWS power levels and peak pressure
  - More limiting success criteria and reduced time for operator actions
- Increased required number of feedwater and condensate pumps
  - Increases turbine trip initiating event frequency (IEF)

FPIE Risk Evaluation

EPU Effects - Systems

- Increased number of reactor safety and relief valves required for overpressure protection
  - Modifies ATWS success criteria
- Increased number of reactor relief valves required for emergency depressurization
  - Modifies success criteria for transients, small and medium LOCAs, and ATWS
- Numerous BOP changes, setpoint changes, and logic changes produce negligible risk impact

FPIE Risk Evaluation

Dominant PRA Model Changes to Represent EPU

PRA Technical Element	PRA Model Change	Contribution to CDF Increase
Turbine Trip Initiating Event Frequency	TT IEF increases 2.5 - 18% due to increased number of operating FW & Cond Pumps	<1 – 2.5%
Op. Fails to Initiate RPV Depressurization (Medium LOCA)	Time available decreases 20%	0 - 1.4%
Op. Fails to Initiate SLC Injection During ATWS (Late)	Time available decreases 20%	1%
Op. Fails to Control RPV/Power Level During ATWS (Late)	Time available decreases 20%	1%
Op. Fails to Initiate SPC During ATWS	Time available decreases 20%	0 - 1%
Op. Fails to Initiate Isolation Condenser Shell Side Makeup (Dresden only)	Time available decrease 20%	<2%
Change RPV Depressurization Success Criteria From 1 to 2 ERVs/SRVs	Modified IORV/SORV accident sequences & increased ERV/SRV CCF	1 – 2%

Exelon Nuclear

FPIE Risk Evaluation

Level 2 Risk

- EPU effects
  - LERF is calculated using NUREG/CR-6595 method
    - Release category binning is unaffected by EPU
    - Release frequency in each bin is proportional to Level 1 result
    - Impact of EPU is specific to each bin
    - Minor changes in Level 2 HEPs
    - Minor changes in accident progression timing & decay heat load
    - Negligible change due to timing of containment failure (estimate 5-30 minutes over 6-30 hour accident times)
- EPU has minor impact on Level 2 event trees
- EPU impact on LERF is similar to Level 1 impact

Exelon Nuclear

FPIE Risk Evaluation

Results

- Pre-EPU PRA results

Plant	CDF, yr^{-1}	LERF, yr^{-1}
Quad Cities	4.6 E-6	3.3 E-6
Dresden	2.6 E-6	1.4 E-6

- EPU has small impact on CDF and LERF
  - Quad Cities: CDF (+5%), LERF (+4%)
  - Dresden: CDF (+9%), LERF (+10%)

- Risk changes conform to Reg. Guide 1.174 Guidelines
  - ΔCDF is in Region III (very small risk change)
  - ΔLERF is in Region II (small risk change)

FPIE Risk Evaluation

Uncertainties

- Uncertainty in FPIE PRAs was examined using
  - Risk importance measures
  - Sensitivity studies
  - Comparison to NUREG-1150 uncertainty results
- No uncertainty sources beyond those identified by NUREG-1150 were found
- No quantitative uncertainty analysis of EPU risk evaluation was performed
- An uncertainty in EPU risk evaluation in the range estimated by NUREG-1150 would only change the ΔCDF results from "very small risk" to "small risk" (per RG 1.174)

Qualitative Risk Evaluation

Internal Flooding Risk

- EPU effects
  - Dominant FPIE PRA model changes apply
    - Negligible impact because internal flooding risk is very small fraction (~ 1%) of FPIE risk
  - No new initiating events or increased IEFs
- EPU has negligible impact on internal flooding risk

Qualitative Risk Evaluation

Fire Risk

- EPU effects
  - Dominant FPIE PRA model changes apply
    - Examined top ten scenarios of recently updated fire PRAs
    - Control room scenario HEP increases are subsumed in existing conservatism (used CCDP = 0.5)
    - Minor impact on decay heat removal scenario HEPs because of long times available for response/recovery actions
    - Changes in success criteria, e.g. number of relief valves required for depressurization, have negligible impact
    - ATWS-related impact negligible due to low probability of fire-induced ATWS
  - No new fire initiating events or increased fire IEFs
- EPU has negligible impact on fire risk

Qualitative Risk Evaluation

Seismic Risk

- EPU effects
  - No change in Seismic Margins Analysis (SMA)
    - Little or no impact on seismic qualifications of SSCs
    - Negligible impact of increased stored energy on blowdown loads on reactor vessel and containment
  - EPU impact on DNPS IPEEE ultimate heat sink issues determined to be minor using scenario specific event trees and standard seismic risk methods
- EPU has minor impact on seismic risk

Qualitative Risk Evaluation

Shutdown Risk

- EPU effects
  - Dominant FPIE PRA model changes do not apply
  - No new initiating events or increased IEFs
  - Time to boiling and boildown times decrease
    - Minor impact on human error probabilities (HEPs) because of long times available for response/recovery actions
    - Changes decay heat removal (DHR) success criteria (delays time when alternate, low capacity systems can be used)
  - EPU risk impact is minimized using ORAM (Outage Risk Assessment and Management) configuration risk mgmt., e.g., protected equipment and contingency plans
- EPU has negligible impact on shutdown risk

Summary of EPU Risk Impact

- Risk impact was evaluated using standard PRA methods (quantitative and qualitative)
- Quantified risk impact is a small percentage of current plant risk
- ΔCDF is a very small risk change per Reg. Guide 1.174
- ΔLERF is a small risk change per Reg. Guide 1.174
- Risk impacts from external events and shutdown conditions are either negligible or minor

The EPU risk impact is acceptable

Exelon Nuclear

Resolution of Open Items

Mark Kluge, Exelon Nuclear

John Freeman, Exelon Nuclear

Tim Hanley, Exelon Nuclear

Introduction

- ECCS net positive suction head (NPSH) requirements
- Ultimate heat sink
- Standby liquid control system (SLCS)
- Large transient tests

ECCS Pump NPSH

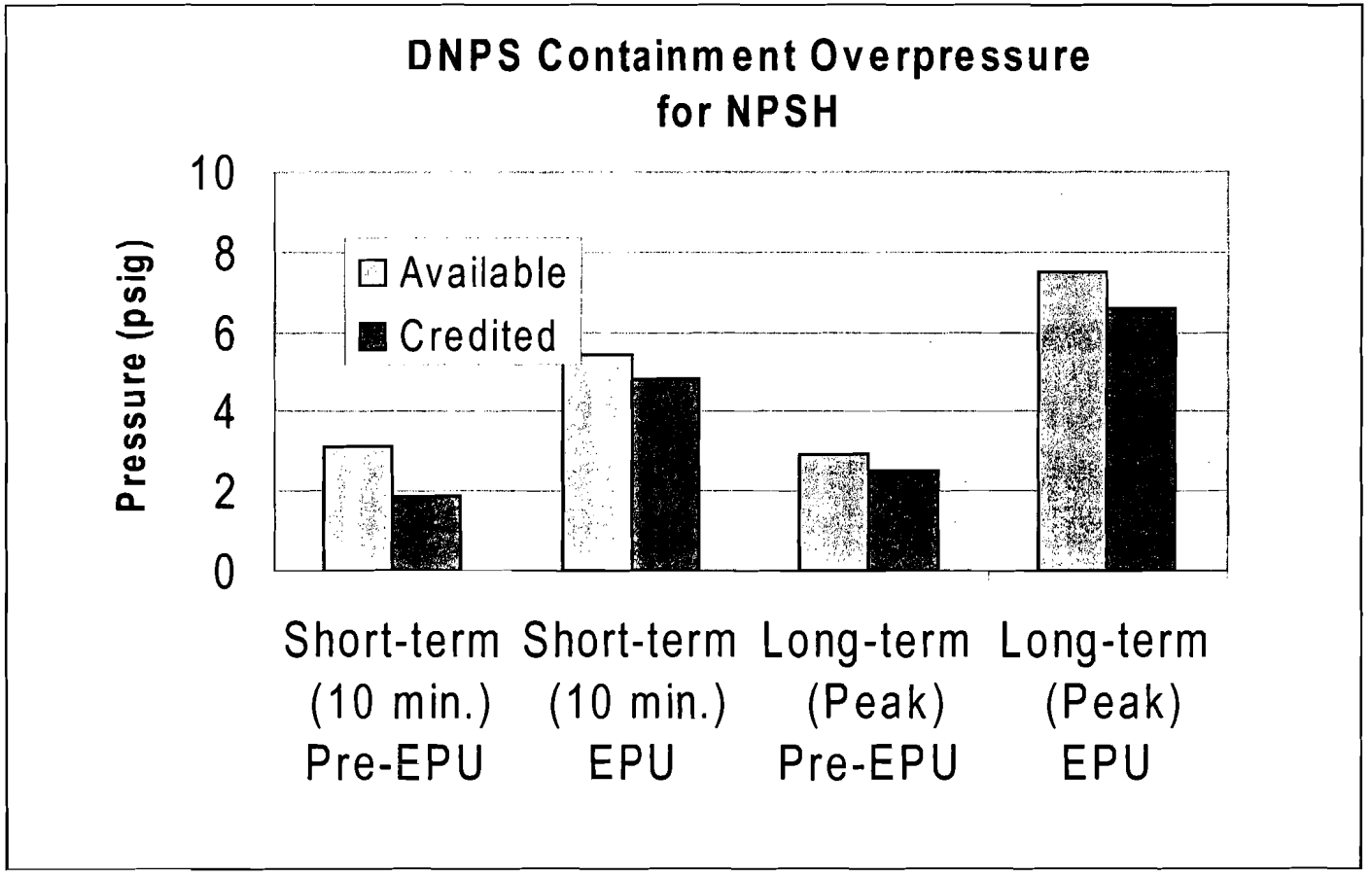
- Pre-EPU licensing basis
  - Credit for containment overpressure in short term and long term
  - Operating procedures, precautions and training to maintain long-term NPSH for ECCS pumps
- EPU impacts
  - Higher torus temperature reduces NPSH available
  - Higher torus pressure increases NPSH available
  - Overall impact: increase in containment overpressure credit

ECCS Pump NPSH

- ECCS suction strainer head loss calculation methodology revised to address concerns with previous methods
  - Results in increased head loss at same ECCS flow
- Overall effect from EPU
  - Shorter period of pump cavitation in short term
    - Cavitation previously evaluated as acceptable for short term
  - Reduced pump flow in long term compared to previous licensing basis
    - All flow requirements continue to be met

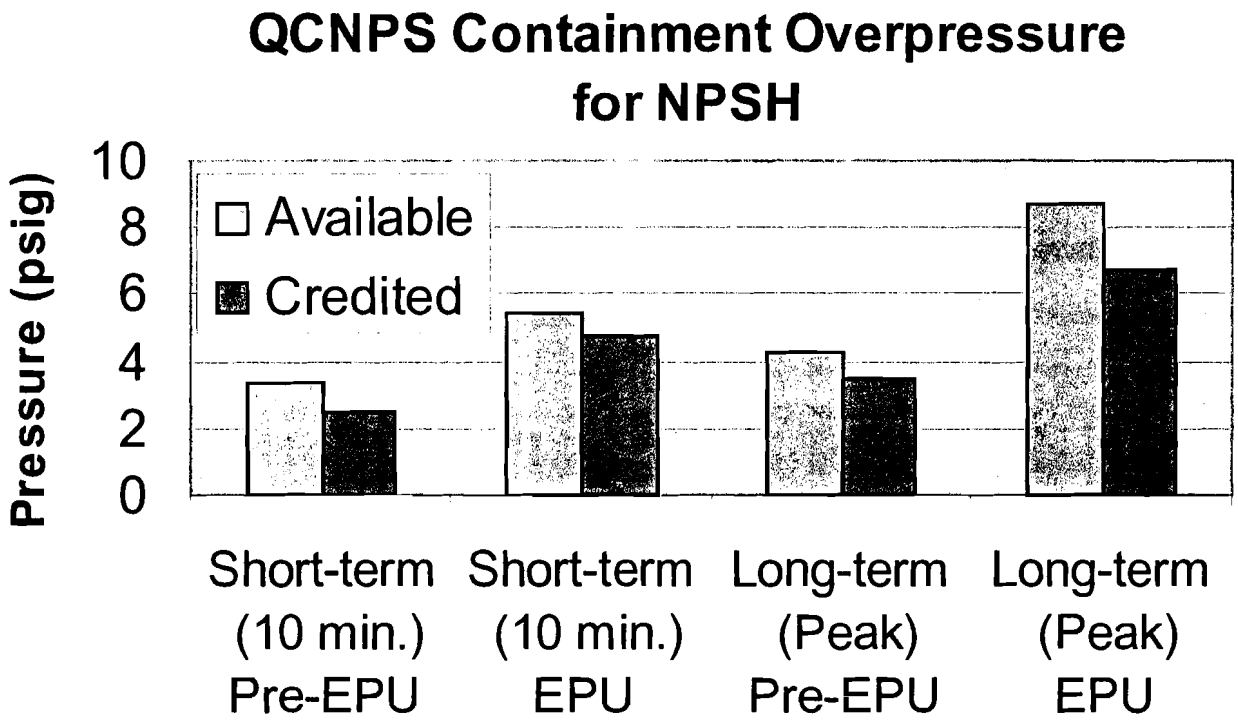
Exelon Nuclear

ECCS Pump NPSH



Exelon Nuclear

ECCS Pump NPSH



ECCS Pump NPSH

- Summary
  - Used acceptable methods to determine suction strainer head loss and NPSH requirements
  - Short-term pump cavitation has been evaluated and has no detrimental effect on pump operability or available flow
  - Long-term flow is acceptable and operators will maintain long term NPSH

ECCS Pump NPSH is acceptable for EPU conditions

Ultimate Heat Sink
DNPS

- Intake and discharge canal contour traps water inventory in the event of dam failure
  - Inventory used as makeup to isolation condensers and for diesel generator cooling
  - Canal replenished by means of portable pumps for long term safe shutdown
  - All required actions are currently in procedures
- EPU changes minimum available time for replenishing canal from 5.5 days to 4 days

Ultimate Heat Sink

Dresden Lock and Dam Failure Scenario

- Scenario of concern is catastrophic failure of Dresden Lock and Dam
  - Initial isolation condenser makeup from onsite tanks
  - Stop logs are installed and pumps used to reflood crib house intake bay within 2 hours
  - Diesel driven fire pump provides long-term isolation condenser makeup

Ultimate Heat Sink

Dresden Lock and Dam Failure Scenario

- UHS also supplies Diesel Generator Cooling Water pumps
  - DGCW suction elevation limits usable inventory in intake canal
  - DGCW return flows into discharge canal, recirculates to intake canal by any of 3 paths

Ultimate Heat Sink

Dresden Lock and Dam Failure Scenario

- Lack of seismic makeup path to isolation condensers was identified as outlier during seismic margins analysis
  - Original FSAR coping analysis credited non-seismic equipment, relied on diversity of makeup sources
  - Modification planned to provide seismic makeup path
  - EPU has insignificant impact on plant risk for this scenario

Exelon Nuclear

Ultimate Heat Sink

Dresden Lock and Dam Failure Scenario

- Seismic margins success path must also address small (equivalent 1-inch) LOCA
  - Analysis indicates isolation condenser and ECCS will mitigate for at least 24 hours
  - Modification planned to provide long-term containment cooling capability via portable pumps
  - All operator actions will be in procedures
  - EPU has no impact on plant risk for this scenario

Ultimate Heat Sink

Dresden Lock and Dam Failure Scenario

- Standard seismic risk assessment techniques used to estimate risk
  - Dresden site seismic hazard curve (NUREG-1488)
  - Entire hazard curve analyzed, divided into discrete 0.1g intervals
  - Seismic frequency calculated at beginning of interval, and seismic fragilities calculated at midpoint of interval
  - Seismic-induced failures (fragilities) quantified for limiting components
  - HEPs calculated using DNPS HRA approach (only proceduralized make-up actions credited)
  - Event tree approach used to quantify sequences

Ultimate Heat Sink

Dresden Lock and Dam Failure Scenario

- Safe shutdown with IC - intact primary system
  - deltaCDF due to EPU ~1E-8/yr
- Case with coincident SLOCA
  - deltaCDF due to EPU *negligible*

Ultimate Heat Sink
Conclusion

- Summary
  - EPU has minimal impact on ultimate heat sink capability
  - DNPS modifications will be completed on same schedule as committed to for IPEEE resolutions
  - Risk impact and increase in risk is small for these scenarios

Ultimate heat sink is acceptable for EPU conditions

Standby Liquid Control System

- Evaluated SLCS relief valve margin in response to Information Notice 2001-13
- No interruption of SLCS injection on DNPS 2
  - SLCS relief valves will reclose if operator injects early
- Potential modifications to raise SLCS relief valve setpoint on remaining units

SLCS is acceptable at EPU conditions

Large Transient Tests

- ELTR-1 specified large transient tests
  - MSIV closure of all valves at 110 % power
  - Generator load rejection at 115 % power
- Exelon Nuclear took exception to these tests in the submitted PUSARs
- EPU reactor pressure remains the same
- GE has concluded these tests are no longer necessary when reactor pressure is unchanged

Large Transient Tests

- Tests will not provide new information about transient modeling
  - Safety analysis used NRC-approved ODYN modeling code
  - ODYN has been benchmarked against BWR test data, and incorporates industry experience
  - ODYN uses plant specific inputs to predict integrated plant response
  - ODYN simulations have concluded significant changes to analyzed transients will not occur

Large Transient Tests

- Power uprate industry experience
  - Data collected from uprated plants have shown the response to be within the expected code predictions
  - Unplanned transients at uprated power levels showed parameters were acceptably predicted
  - DNPS and QCNPS possess adequate data collection capability should these transients occur

Large Transient Tests

- Component analysis
  - Exelon Nuclear and GE have analyzed the major components that affect large transient tests
    - MSIVs, main steam piping, scram signals, safety/relief valves and turbine valves
  - Years of operational experience provides well documented responses to transients
  - Transient testing is not needed to show these components will respond as designed

Large Transient Tests

- Conclusion
  - No new significant information will be gained by performing these tests
    - ODYN code has been shown to be acceptable
    - Plant response to these transients as a result of EPU will not change significantly
    - There is operating experience on uprated plants
    - Plant components will perform as designed
  - From a safety and risk perspective, should not induce unnecessary plant transients without commensurate benefit

Exelon Nuclear

Project Implementation - Training and Testing

Tim Hanley, Exelon Nuclear

Shift Operations Superintendent

QCNPS

Introduction

- Training
  - Classroom training
  - Simulator training
- Testing
  - Overview
  - Test summary
- Conclusion

Project Implementation

Classroom Training

- Classroom material presented
  - Technical Specifications and Updated Final Safety Analysis Report changes
  - Plant limits and operating condition changes
  - Design changes for EPU
  - MELLLA - new power/flow map
  - Operating procedure revisions
  - Uprate operating experience

Exelon Nuclear

Project Implementation

Simulator Training

- Static walkthrough EPU full power conditions
- Normal operations scenarios
- Dynamic scenarios selected to highlight both similarities and differences in plant response at EPU vs current power level to transients and accidents

Project Implementation

Start-up Test Overview

- 3% incremental power test program
- Power increases along constant FCL to maximum achievable power level
- Increasing power once a day
- Steady state data collection and testing beginning at 90% of current power

Project Implementation

Test Summary

- Demonstration of signal and system response
  - Reactor pressure control system
    - Incremental pressure changes
    - Pressure regulator fail-over tests
  - Feedwater level control system
    - Incremental level change
    - FW Reg valve position changes
  - These tests are performed at graduated power levels

Exelon Nuclear

Project Implementation

Test Summary

- System/equipment performance monitoring
  - Power dependent BOP systems monitored
  - Each parameter has a predetermined acceptance criterion selected by plant engineers
  - Performance parameters are compared to the acceptance criteria prior to increasing to the next power level

Project Implementation

Test Summary

- System/equipment performance monitoring
  - Power dependent BOP systems monitored
  - Each parameter has a predetermined acceptance criterion selected by plant engineers
  - Performance parameters are compared to the acceptance criterion prior to increasing to the next power level

Project Implementation

Test Summary

- Piping vibration monitoring
  - Monitoring FW and MS piping inside and outside the containment
  - Low power vibration data collection used for comparison to uprated power vibration data
  - Acceptance criteria established from ASME stress analysis limits

Exelon Nuclear

Project Implementation

Test Summary

- Turbine valve stroking
- Reactor FW pump curve performance monitoring
- FW flow element comparison check
- MS flow element comparison check
- Steam dryer carryover performance

Project Implementation

Feedwater Test Program

- Instrument calibrations and checks
- Staggered reactor feed pump trips - mod test
  - Calibrations and test signals to verify logic
- Feddwater level control system tuning
  - Verify stable control system response
- Reactor recirculation pump runback/feedwater pump trip
  - Simulated logic functional test
  - Overlap testing demonstrating overall performance

Exelon Nuclear

Project Implementation

Conclusion

- Operator training is extensive
- Testing plan is incremental and comprehensive

Project implementation will ensure that EPU is implemented as designed and plant response is as expected

Conclusion

Jeff Benjamin, Exelon Nuclear

Vice President - Licensing and Regulatory Affairs

Conclusion

- Followed standard approach for extended uprates for GE BWRs using constant pressure
- Extensive analyses using accepted methodology and conservative inputs
- No significant impacts on plant response or system integrity
- Minimal changes in plant risk

Plant operation is acceptable at EPU conditions

Dresden /Quad Cities EPU Safety Evaluation Report

Ralph Architzel, Ron Young,
Steven Jones, Robert Elliott
Plant Systems Branch
NRR

PLANT SYSTEMS BRANCH

Responsibilities

- Reviewed design/operation requirements for systems using UFSAR
- Examined application for conformance with approved topical report and statements in associated Safety Evaluation
- Assured agency regulations/guidelines met under EPU conditions (Regulatory Guides, Standard Review Plan, Regulations)

PLANT SYSTEMS BRANCH

Responsibilities

- Telephone conferences to clarify application, systems design and operation, and responses
- Issued RAIs to resolve questions regarding licensee's EPU evaluation results and supplement docketed information

PLANT SYSTEMS BRANCH

Responsibilities

- Coordination with other branches with lead/project responsibility, including PRA impacts on affected systems, startup testing issues, station blackout input, radiological source
- Concluded EPU does not adversely affect operation or basis of responsible areas

PLANT SYSTEMS BRANCH

Areas of Input in EPU

- 3.3.4 Steam Separator and Dryer Performance
- 3.7 Main Steam Isolation Valves

- 3.9 Residual Heat Removal/LPCI/Containment Cooling and Shutdown Cooling Systems:
  - Suppression Pool Cooling Mode
  - Containment Spray Cooling Mode
  - Fuel Pool Cooling Assist Mode

PLANT SYSTEMS BRANCH

Areas of Input in EPU

- 4.1Containment System Performance:
  - Containment Pressure and Temperature Response
  - Suppression Pool Temperature Response
  - Containment Airspace Temperature Response
  - Containment Pressure Response
  - Containment Dynamic Loads
  - Loss-of-Coolant Accident Containment Dynamic Loads
  - Relief Plus Safety Relief Valve Discharge Loads
  - Subcompartment Pressurization

PLANT SYSTEMS BRANCH

Areas of Input in EPU

- 4.2.5 Net Positive Suction Head
- 4.4 Standby Gas Treatment System
- 4.5.1 Post-LOCA Combustible Gas Control System
- 4.5.2 Main Control Room Atmosphere Control System

PLANT SYSTEMS BRANCH

Areas of Input in EPU

- 6.3 Fuel Pool Cooling
- 6.4 Water Systems:
  - Service Water Systems
  - Main Condenser, Circulating Water
  - Reactor Building Closed Cooling Water
  - Turbine Building Closed Cooling Water
- 6.4.5 Ultimate Heat Sink (UHS)

PLANT SYSTEMS BRANCH

Areas of Input in EPU

- 6.6 Power Dependent Heating, Ventilation, and Air Conditioning
- 6.7 Fire Protection Program
- 6.8 Systems Not Impacted or Insignificantly Impacted by Power Uprate

PLANT SYSTEMS BRANCH

Areas of Input in EPU

- 7.1 Turbine--Generator
- 7.2 Condenser and Steam Jet Air Ejectors
- 7.3 Turbine Steam Bypass
- 7.4 Feedwater and Condensate Systems

PLANT SYSTEMS BRANCH

Areas of Input in EPU

- 8.0 Liquid Waste, Gaseous Waste and Offgas Systems
- 9.4.2 Station Blackout
- 10.1 High Energy Line Breaks
- 10.2 Moderate Energy Line Break
- 10.3 Equipment Qualification of Mechanical Components

PLANT SYSTEMS BRANCH

Areas with Substantive RAI's/Clarification

- 4.1Containment System Performance:
  - Containment Pressure and Temperature Response
- 4.2.5 Net Positive Suction Head
- 6.3 Fuel Pool Cooling
- 6.4.5 Ultimate Heat Sink (UHS)
- 7.4 Feedwater and Condensate Systems

R

DRESDEN 2 & 3
QUAD CITIES 1 & 2
EXTENDED POWER UPRATE

STAFF RISK ASSESSMENT REVIEW

OCTOBER 26, 2001

DRESDEN/QUAD CITIES EPU
Staff Risk Assessment Review

Review Information

- **LICENSEE SUBMITTED RISK INFORMATION**
  - Original Submittal Contained Only Overall Risk Impact Results
  - Supplemental Submittal Addressed Risk Areas at High Level
  - Responses to Staff Requests for Additional Information Addressed and/or Clarified Risk Impacts in All Key Risk Review Areas
    - Internal Events, External Events, Shutdown Operations, and PRA Quality
- **STAFF RISK ASSESSMENT REVIEW**
  - Licensee EPU Risk Information
  - Licensee Information in Support of Other Staff EPU Review Areas
  - Staff SEs on IPEs and IPEEEs
  - Licensee PRA Maintenance/Update Procedures and Processes

DRESDEN/QUAD CITIES EPU
Staff Risk Assessment Review

Results

- **INTERNAL EVENTS**
  - Potential Impacts Identified to Initiating Events, Component Reliability, Success Critieria, and Operator Actions
    - Potential Risk Impacts to Initiating Events and Component Reliability are Mostly Mitigated or Offset by Plant Modifications
    - Majority of Risk Impacts Due to Success Criteria and Operator Actions
    - Increase CDF DNPS (8%), QCNPS (5%) - Very Small Risk Increase
    - Increase LERF DNPS (10%), QCNPS (4%) - Small Risk Increase
  - Licensee Used Simplified Models and Simplistic Calculations to Determine Some Risk Impacts
  - Licensee Assumed UAT/RAT, and Associated Switchgear and Breakers, Not Affected by EPU Due to Mitigative Operator Actions and/or Testing with Field Modifications
  - Licensee Thermal Hydraulic Analysis, Used to Determine Success Criteria and Available Operator Response Times, Performed at Power Level ~ 2% Less Than EPU Level of 2957 MWt

DRESDEN/QUAD CITIES EPU
Staff Risk Assessment Review

Results (continued)

- **EXTERNAL EVENTS**
  - Seismic Outliers Identified, with Modifications Credited in IPEEE SMA
    - Licensee Performed Simplified Seismic Risk Evaluation of Remaining Outliers to Determine Risk Impacts of EPU were Negligibly Small
  - Fire Analysis
    - Negligibly Small Risk Increase
- **SHUTDOWN OPERATIONS**
  - Typically Long Times to Boil
  - Shutdown Risk Management Approach, ORAM, & Short Refuelings
  - Negligibly Small Increase in Risk Expected
- **PRA QUALITY**
  - BWROG Peer/Certification of Internal Events PRA
  - Simplified Models and Simplistic Calculations May Not Reflect Actual Installed Equipment or Proceduralized Operator Actions
  - IPEEE SMA Does Not Reflect Current or EPU Plant Conditions

DRESDEN/QUAD CITIES EPU
Staff Risk Assessment Review

Overall EPU Risk Conclusions

- **STAFF IDENTIFIED A NUMBER OF ISSUES**
  - Recirculation Pump Runback Circuit Simplified PRA Model
  - UAT/RAT Overload LOOP Simplistic Calculation
  - UAT/RAT Switchgear and Breaker Tests/Field Implementation
  - Thermal Hydraulic Analysis at Less Than EPU Level of 2957 MWt
  - Current and EPU Seismic Capacity vs. IPEEE SMA Criteria
- **IF RISK-INFORMED, WOULD REQUIRE CONFIRMATION OF ACCEPTABLE RESOLUTION OF ISSUES**
  - Analyze Actual Installed Components, New Procedularized Operator Actions, & EPU Thermal Hydraulic Analysis (i.e., EPU PRA Update)
  - Modifications to Achieve IPEEE SMA Criteria
- **SUBMITTAL IS NOT RISK-INFORMED**
  - Meets Deterministic Requirements
  - Information Does Not Rebut Presumption of Adequate Protection
    - Do Not Believe Risk Increase Would Warrant Denial of Application

STRUCTURAL INTEGRITY OF STEAM DRYER

- Insignificant increase in design parameters affecting structural integrity:

  - No increase in Reactor Pressure
  - No increase in Reactor Temperature
  - No increase in Core Flow
  - Very little increase (less than 2.2%) in the drive flow
  - Approximately 20 % increase in steam flow

- Per BWRVIP-06, the dryers are visually inspected during removal in each refueling outage, and any significant cracking will be identified and assessed.

- Stresses in the dryers during the plant operation are small in relation to allowable endurance limit of 10 ksi. Significant cracking of the dryer during one operation cycle is considered very unlikely.

- The steam dryer is designed to maintain its structural integrity during a steam line break.