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

September 10, 2001

MEMORANDUM TO: ACRS Members

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

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON THE THERMAL-HYDRAULIC PHENOMENA, JUNE 12, 2001, ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on June 25, 2001, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc via e-mail:

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

August 15, 2001

MEMORANDUM TO: Paul Boehnert, Senior Staff Engineer
Technical Support Staff

FROM: G. Wallis, Chairman
Thermal-Hydraulic Phenomena Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON THERMAL-HYDRAULIC PHENOMENA, JUNE 12, 2001, - ROCKVILLE, MARYLAND

I hereby certify that, to the best of my knowledge and belief, the Minutes of the subject meeting issued June 25, 2001, are an accurate record of the proceedings for that meeting.

G. Wallis, Chairman

9/07/01
Date

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

GT350

June 25, 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, JUNE 12, 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
R. Savio
S. Duraiswamy
ACRS Staff Engineers
ACRS Fellows

DRAFT COPY - PREPARED FOR INTERNAL COMMITTEE USE

CERTIFIED BY: G. Wallis, September 7, 2001
ISSUED: June 25, 2001

CERTIFIED

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING MINUTES:
ISSUES PERTAINING TO CORE POWER UPRATES
JUNE 12, 2001
ROCKVILLE, MARYLAND

INTRODUCTION:

The ACRS Subcommittee on Thermal-Hydraulic Phenomena held a meeting on June 12, 2001 with representatives of the NRC staff, the ACRS staff, and General Electric (GE) Nuclear Energy. The purpose of this meeting was for the Subcommittee to discuss potential issues for consideration by the NRC staff pertaining to its review of applications for core power uprates. A portion of the meeting was closed to the public to discuss information proprietary to General Electric Nuclear Energy. Mr. P. Boehnert was the cognizant ACRS staff engineer and Designated Federal Official (DFO) for this meeting. The meeting was convened by the Subcommittee Chairman at 8:30 a.m., June12, 2001, and adjourned at 5:05 p.m. that day.

ATTENDEES

ACRS Members/Staff:

G. Wallis, Chairman
F. Peter Ford, Member
T. Kress, Member
P. Boehnert, DFO
G. Leitch, Member
R. Uhrig, Member
V. Schrock, Consultant
A. Cronenberg, Senior Fellow

NRC Staff:

J. Hopkins, NRR
R. Caruso, NRR
D. Harrison, NRR
J. Rosenthal, RES

GE Nuclear Energy

J. Klapproth
E. Eckert
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 slides and handouts used during the open portions of this meeting are attached to the portion of the transcript available to the public. The presentations to the Subcommittee are summarized below.

CHAIRMAN'S COMMENTS (**Open**)

Dr. G. Wallis, Subcommittee Chairman, convened the meeting. He noted that the on-going program to uprate plant power is now a major impact on nuclear generation in the United States and that as much as 10,000 MWe of additional generation may be obtained by uprates.

NRC STAFF PRESENTATIONS **(Open)**

Introduction

Mr. J. Hopkins, NRR, provided introductory remarks. He said that the staff's presentations will focus on the so-called Extended Power Uprate Program now underway for GE plants. He noted that the staff is performing a survey of all BWR plant licensees to determine how many plan to make extended power uprate (>15-20% of nominal) submittals.

Duane Arnold Power Uprate - Plant Audit Results

The results of NRR's audit of various licensing calculations performed in support of the Duane Arnold plant power uprate were provided by Mr. R. Caruso, NRR. The Background, Scope and Findings were noted. Significant findings were found in two areas:

- For the issue of core stability, NRR found that the uprate and corresponding change of licensing methodology will require plant operators to rely more on the use of the plant stability monitor (SOLOMON Monitoring System). More operator training will be needed.

- GE and the staff need to reach resolution on the issue of the use of the COBRAG code to generate data for use in the GEXL 14 database.

NRR plans to continue to perform similar audits of analysis methods and results for upcoming BWR Extended Power uprates (e.g., Dresden/Quad Cities, Clinton).

In response to Subcommittee questions, NRR noted that the results of the plant audits will be included in the NRC SER's supporting power uprate. In response to Dr. Ford, NRR said that the BWR Vessel Improvement Program (VIP) accounts for the impact of the increase in fluence seen for power uprates/plant life extension. Dr. Ford said that the staff needs to investigate the issue of time-dependent material degradation with regard to power uprates. Professor Schrock requested detailed information relative to the GEXL 14 correlation.

Risk-Informed Review Considerations for Uprates

Messrs D. Harrison and M. Rubin, NRR, provided discussion relative to risk-informed review considerations, based on a review of the risk for the Duane Arnold plant, and responded to a set of questions submitted by the Committee. Key points noted relative to the risk impact of the Duane Arnold uprate were:

- The uprate application was not submitted as a risk-informed licensing action. NRR is considering performing a review of the peer review of the licensee's PRA.

- Evaluation of operator success criteria shows an increase in the CDF and LERF of 9% & 16%, respectively - mostly driven by the decreased response time to ATWS events.

- Component reliability is not expected to change due to monitoring programs in place (e.g., Maintenance Rule, EQ, etc.). No significant impact from shutdown risk is anticipated (the NUMARC 91-06 plant guidelines are being followed).

- The staff is following the framework of Regulatory Guide 1.174 to ensure no significant risk changes or new vulnerabilities are introduced.

Regarding the Committee's risk-informed review questions (Figure 1), the staff noted the following points:

- The strictures of Regulatory Guide 1.174 are adequate to ensure that no additional risk metrics are necessary for evaluation of fission product release rates.

- Licensees and vendors control the use of margins up to NRC approved limits.

- Regarding the impact of uprates on the DBA Source Terms, the factor of 500 multiplier will continue to compensate for the uncertainty in the iodine spike rate.

- The shorter operator response time available is one of the main areas impacted by power uprates (e.g., for Duane Arnold, response time for an ATWS decreases from 6 to 4 minutes).

- NRC is considering how plant monitoring programs, Performance Indicators, etc., could be used to provide early indication of power uprate impacts.

- Evaluation of the Leibstadt Study on power uprates shows that the overall risk impact remains small.

In response to Dr. Ford, NRR noted that the Duane Arnold licensee is installing equipment to monitor for flow-induced vibration on the main steamlines, given the 24% increase in steam flow seen for the uprate. Regarding the results of the Leibstadt Study noted above, NRR said, in response to Dr. Kress, that the Swiss regulatory authorities required that the PRA Success Criteria be held constant as a condition for granting the power uprate. A Duane Arnold licensee representative noted that a daily PRA run is made to determine the plant risk profile; the PRA will be upgraded to account for the impact of the power uprate. Following further discussion, Dr. Kress observed that for the BWR uprates, the GE topical reports on the Extended Uprate Program apparently functions as a "surrogate Standard Review Plan".

RES Program to Assess Synergistic Effects of Power Uprates

Mr. J. Rosenthal, Office of Nuclear Regulatory Research, (RES), discussed a planned program to consider potential synergies, systems interactions, and phenomenological interactions pertaining to power uprates. The program will use the RES TRAC-M code, with the PARC neutronics package, to provide quantitative assessment. BWR "extended" uprates will be evaluated initially. The focus will be on PRA Success Criteria and severe accident issues. The work will commence in FY-02 and the effort will comprise 2-3 FTEs and $850K. Issues to be explored include review of: PRA Success Criteria, two generic issues, severe accident/accident management issues, and various power plant systems (containment, electrical, control, etc.). Mr. Rosenthal suggested that RES report the results of this study to the ACRS in the fall of 2003.

ACRS SENIOR FELLOW PRESENTATION **(Open)**

Effects of Margin Reduction/Potential Synergistic Effects for Power Uprate/License Renewal

Dr. A. Cronenberg discussed the results of his investigation into the issues of the effects of margin reduction and potential synergistic effects for power uprates. Issues discussed included:

- Margins in the Regulatory Process
- Margin Estimates for Power Uprates and License Renewal Plants
- An Overview of Power Uprates
- Events Noted for Uprate Plants
- Potential Synergistic Safety Issues

Dr. Cronenberg concluded that NRC should develop a Standard Review Plan for power uprate applications and that Legacy Tables should be developed to track plant conditions impacted by such cumulative licensing actions as power uprates, license renewal, use of

high-burnup fuel, etc. Risk assessments should be performed for significant (15-20%) power uprates, and NRC should perform verifications of licensee analyses.

In response to Dr. Ford, Dr. Cronenberg noted that use of Cumulative Usage Factor methodology for determining the impact of uprates is a new NRC staff requirement and is welcome. Dr. Wallis asked why the staff has not developed a SRP for power uprates. Mr. Shuaibi, NRR, said that the staff looks upon the Duane Arnold, Dresden, and Quad Cities extended reviews as "first of a kind", and will be issuing guidance to the industry regarding submittal content for future uprates via a yet unidentified platform.

GE NUCLEAR ENERGY PRESENTATION (**Open/Closed**)

Note: Dr. F. Peter Ford declared himself in conflict of interest with GE Nuclear Energy and refrained from advising the Subcommittee on this matter.

Introduction (**Open**)

Messrs. J Klapproth, E. Eckert, and I. Nir discussed the GE BWR Power Uprate Program. The open session discussion centered on the key elements of the GE Uprate Program which includes: Stretch Power Uprate (SPU - 5% increase), Extended Power Uprate (EPU - up to 20% increase), Thermal Power Optimization (~1.5% increase via use of improved feed flow instrumentation pursuant to revision of Appendix K), and, Constant Pressure Power Uprate (CPPU - up to 20% increase, with no increase in RCS pressure). GE noted that the ACRS had reviewed and approved the topical reports associated with both the SPU and EPU approaches. GE anticipates a high volume of power uprate applications, beginning in the near future.

Extended Uprate Program (**Closed**)

GE representatives discussed the following topics:

- Background
- Constant Pressure Power Uprate (CPPU)
- EPU-Specific Topics
  - CPPU Performance Margin
  - Evaluation Model (EM) Codes
  - EPU Effect on Core & Fuel Design
  - EPU Startup Testing

GE adopted the CPPU approach to both minimize and focus the NRC review scope as well as limit the impact on plant operational parameters. For the latter, increases are only seen

in steam flow, feed flow & temperature, radiation source & levels, and decay heat. All aspects of the CPPU approach are dispositioned in either generic assessments, core-

dependent reload evaluations, operational assessments, or plant-specific evaluations. GE has submitted the CPPU topical for NRR review and has requested a SER on this approach by the end of this year.

Regarding the effect of EPUs on core and fuel design limits, GE maintained that no safety margins are impacted by the EPUs; rather, some operational margin is sacrificed.

Experience with the EPU startup testing at the Swiss KKL plant (17% uprate) was recounted. The power increase was instituted in step wise fashion over four years, with control and core monitoring tests performed (identical to testing in U.S. EPU plants). Additional scram-avoidance testing was also conducted (turbine-generator trips, single recirculation pump trip, etc.). GE said that any unplanned major transients will be recorded and analysis performed for EPU plants under the CPPU approach.

During discussion, GE noted that for all power uprates all plant components with a CUF > 0.5 are subject to evaluation. In response to Professor Schrock, GE said that it believes that a revised decay heat curve is generated for uprated plants. Dr. Wallis said that for future presentations, GE should provide best-estimate-plus-uncertainty values for its analyses. He said that the code predictions shown have no figures-of-merit attached to them. GE said that some plant uprates may be limited by containment temperature safety limits.

GE also noted that they are not proposing any operational testing in conjunction with CPPUs. They propose to wait for unplanned transient events.

SUBCOMMITTEE CAUCUS

Following discussion, the Subcommittee requested that the NRC staff make a presentation to the full Committee on what actions, if any, are planned relative to development of a Standard Review Plan Section on core power uprates. Dr. Wallis will lead Committee discussion of this session with a Subcommittee Report on the results of this meeting.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

Memoranda dated June 4, 2001, from P. Boehnert, ACRS, to T/H Phenomena Subcommittee Members and Consultant transmitting:

- Letter to L. Joseph Callan, Executive Director for Operations, NRC, from R. L. Seale, ACRS Chairman, Subject: Application for Power Level Increase for Edwin I. Hatch Nuclear Power Plant Units 1 and 2, dated September 15, 1998

- Letter to L. Joseph Callan, Executive Director for Operations, NRC, from R. L. Seale, ACRS Chairman, Subject: General Electric Nuclear Energy Extended Power Uprate Program and Monticello Nuclear Generating Plant Power Level Increase Request, dated July 24, 1998

- Memorandum to Commissioners from J. T. Larkins, Executive Director, ACRS, Subject: Transmittal of Foreign Trip Report of ACRS Member, dated May 16, 2001

- Handouts from Presentation by F. Eltawila, RES to ACRS at December 2000 Committee Meeting, "RES Views on Significant Power Uprate"

- Report of ACRS Fellow A. W. Cronenberg, for Presentation to ACRS at June 6-8, 2001 Meeting: "Potential Margin Reductions for Re-Licensed/Uprated Nuclear Power Plants", and Associated Slides titled: "Signature Estimates of Margin Reductions for Power Uprates/License Renewal" and "Review of Power Uprate Applications and Potential Synergistic Safety Issues".

- Memorandum, to Commissioner Diaz, from A. Vietti-Cook, Secretary, Subject COMNJD -01-0001, Power Uprate Applications, with SRM COMNJD-01-0001 attached.

- GE Nuclear Energy Topical Report, NEDC-33004P, "Licensing Topical Report - Constant Pressure Power Uprate", dated March 2001. (**proprietary**)

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NOTE: Additional details can be obtained from a transcript of the open portions of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW", and can also 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".

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING
ISSUES ASSOCIATED WITH CORE POWER UPRATES
JUNE 12, 2001
ROCKVILLE, MARYLAND

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

PRESENTATION SCHEDULE

TOPIC	SPEAKER	TIME
I. Introduction	G. Wallis, Chairman	8:30 a.m.
II. NRC Staff Presentations		8:40 a.m.
A. Introductory Remarks	J. Hopkins, NRR	
o Plant Power Uprates: Scheduled and Planned		
B. Duane Arnold Plant Audit Uprate Reviews	R. Caruso, NRR	
o RCS Depressurization: Blowdown Forces		
C. Risk-Informed Review Considerations	D. Harrison, NRR	
o Risk Impact for Shutdown Operations o Additional Questions (see Attached List)		
D. PWR Vessel Internals Embrittlement	Response to Sub. Questions	
E. RES Research Program	J. Rosenthal, RES	
F. Concluding Remarks	J. Hopkins, NRR	
	LUNCH	11:30 - 12:30 p.m.

TOPIC	SPEAKER	TIME
III. ACRS Fellow Presentation o Potential Margin Reductions Associated With Power Uprates	A. W. Cronenberg, ACRS Sr. Fellow	12:30 p.m.
	BREAK	2:00 - 2:15 p.m.
IV. GE Nuclear Energy Presentation 1. Introduction(**Open**) 2. Extended Power Uprate Program (**Closed**) o Background - Scope and Content o CPPU Topical - Details of Methodology o EM Codes - Use of Realistic Methods 3. Synergistic Effects (**Closed**) - Impact on Margin Reduction 4. Concluding Remarks	J. Klapproth/ I. Nir/E. Eckert, GE Nuclear Energy	2:15 p.m.
V. Subcommittee Caucus (**Open**) • Follow-on Items from this Meeting • Future Actions • Committee Action		4:15 p.m.
VI. Adjournment		4:30 p.m. (Est)

the examinations from June 4, 2001, to July 13, 2001. The proposed action is in accordance with the licensee's application for exemption dated March 6, 2001.

The Need for the Proposed Action

The proposed action would extend the current Oconee Nuclear Station, Units 1, 2, and 3 requalification program from June 4, 2001, to July 13, 2001. To require the licensee's operators and staff to support the comprehensive examination and operating tests scheduled during the 24-month requalification cycle could have a detrimental effect on the public interest because it would remove qualified operators from refueling operations and place them into the training program, which could interfere with the current Oconee Unit 2 refueling outage schedule. Further, this one-time exemption will provide additional operator support during plant shutdown conditions, which would provide a safety enhancement during plant shutdown operations and post-maintenance testing. The affected licensed operators will continue to demonstrate and possess the required levels of knowledge, skills, and abilities needed to safely operate the plant throughout the transitional period via continuation of the current satisfactory licensed operator requalification program. Upon completion of the examinations on July 13, 2001, the follow-on cycle will end on March 8, 2003. Future annual requalification cycles will run from March to March.

Environmental Impacts of the Proposed Action

The NRC has completed its evaluation of the proposed action and concludes, as set forth below, that there are no environmental impacts associated with the extension of the operator requalification examinations from June 4, 2001, to July 13, 2001. The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types or amounts of any effluents that may be released off site, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action does not involve any historic sites. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, there are no significant nonradiological environmental impacts associated with the proposed action.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

As an alternative to the proposed action, the staff considered denial of the proposed action (i.e., the "no-action" alternative). Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the Oconee Nuclear Station, Units 1, 2, and 3.

Agencies and Persons Consulted

In accordance with its stated policy, on May 18, 2001, the staff consulted with the South Carolina State official, Mr. Henry Porter of the Division of Waste Management, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated March 6, 2001. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov≤* (the Public Electronic Reading Room).

Dated at Rockville, Maryland, this 24th day of May 2001.

For the Nuclear Regulatory Commission.

David E. LaBarge,

Senior Project Manager, Section 1, Project Directorate II, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 01–13606 Filed 5–29–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee Meeting on Thermal-Hydraulic Phenomena; Notice of Meeting

The ACRS Subcommittee on Thermal-Hydraulic Phenomena will hold a meeting on June 12, 2001, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

Portions of the meeting will be closed to public attendance to discuss proprietary information per 5 U.S.C. 552b(c)(4) pertinent to General Electric Nuclear Energy.

The agenda for the subject meeting shall be as follows:

Tuesday, June 12, 2001–8:30 a.m. Until the Conclusion of Business

The Subcommittee will discuss potential issues for consideration by the NRC staff pertaining to its review of applications for core power uprates. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman. Written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, General Electric Nuclear Energy, the ACRS staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the cognizant ACRS staff engineer, Mr. Paul A. Boehnert (telephone 301–415–

8065) between 7:30 a.m. and 4:30 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: May 22, 2001.

James E. Lyons,

Associate Director for Technical Support.

[FR Doc. 01–13488 Filed 5–29–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATES: Weeks of May 28, June 4, 11, 18, 25, July 2, 2001.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville Maryland.

STATUS: Public and Closed.

Matters To Be Considered

Week of May 28, 2001

Wednesday, May 30, 2001
10:25 a.m.—Affirmation Session (Public Meeting) (If needed)

Week of June 4, 2001—Tentative

Tuesday, June 5, 2001
9:25 a.m.—Affirmation Session (Public Meeting) (If needed)
2:00 p.m.—Discussion of Management Issues (Closed-Ex. 2)
Wednesday, June 6, 2001
10:30 a.m.—All Employees Meeting (Public Meeting)
1:30 p.m.—All Employees Meeting (Public Meeting)

Week of June 11, 2001—Tentative

Thursday, June 14, 2001
9:55 a.m.—Affirmation Session (Public Meeting) (If needed)
10:00 a.m.—Meeting with Nuclear Waste Technical Review Board (Public Meeting)
1:30 p.m.—Briefing on License Renewal Program (Public Meeting) (Contact: David Solorio, 301–415–1973)

Week of June 18, 2001—Tentative

There are no meetings scheduled for the Week of June 18, 2001

Week of June 25, 2001—Tentative

Wednesday, June 27, 2001
9:25 a.m.—Affirmation Session (Public Meeting) (If needed)

Week of July 2, 2001—Tentative

There are no meetings scheduled for the Week of July 2, 2001

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: David Louis Gamberoni (301) 415–1651.

The NRC Commission Meeting Schedule can be found on the Internet at

http://www.nrc.gov/SECY/smji/schedule.htm

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, D.C. 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving the Commission meeting schedule electronically, please send an electronic message to dkw@nrc.gov.

Dated: May 24, 2001.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 01–13605 Filed 5–25–01; 10:16 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 7, 2001 through May 18, 2001. The last biweekly notice was published on May 16, 2001 (66 FR 27174).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

THERMAL-HYDRAULIC PHENOMENA

JUNE 12, 2001

ATTENDEES - PLEASE SIGN BELOW

PLEASE PRINT

NAME	AFFILIATION
MEHRAN GOLBABAI	WESTINGHOUSE Electric
Steve Austin	TVA
Henry L. Jones	TVA
Raymond H. Wright	TVA - BFN
Ed Hartwig	TVA - BFN
Eugene Eckert	GE
Tony Browning	NMC - Duane Arnold
Israel Nir	GE
JIM KLAPPROTH	GE
Al Hueger	Exelon
Kent Cole	GE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

THERMAL-HYDRAULIC PHENOMENA

JUNE 12, 2001

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Ralph Architzel	NRR/SPLB
BARRY ELLIOT	NRR/EMCB
M. K. TREHAN	NRR/EEIB
F. ELTAWILA	RES/DSARE
BOB PALLA	NRR/SPSB
George Thomas	NRR/SRXB
Samuel Lee	NRR/RGEB
Tony ULSES	NRR/SRXB
Jennifer Uhle	RES/DSARE
Topaz Obler	RES/DSARE
Katrina Groth	RES/DSARE
Syed Abdullahi	NRR/DSSA/SRXB
Tai Huang	NRR/DSSA/SRXB
Amar Pal	NRR/DE/EEIB
George Hubbard	NRR/~~DSSA~~ DSSA/SPLB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

THERMAL-HYDRAULIC PHENOMENA

JUNE 12, 2001

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

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Donnie Harrison	NRR/DSSA/SPSB
RALPH CARUSO	NRR/SRXB
TONY ATTARD	NRR/SRXB
Edward Kendrick	NRR/DSSA/SRXB
Jon Hopkins	NRR/DLPM
John Hannon	NRR/DSSA
K.M. CAMPE	NRR/DSSA
Stephen F LaVie	NRR/DSSA
Mark Rubin	NRR/DSSA
LARRY ROSSBACH	NRR/DLPM
HUKAM GARG	NRR/DE/EEIB
I. Ahmed	NRR/DE/EEIB
CHENG-IH WU	NRR/DE/EMEB
Michael Palko Jr.	NRR/DSSA/SPLB
Glenn Kelly	NRR/DSSA/SPSB
S. SINGH BAJWA	NRR/DLPM
Jack Rosenthal	NRC/RES/SMSAB
Joseph Staudenmeier	NRR/SRXB
Mohammed Shuaibi	NRR/DLPM
Steve Bajorek	NRC/RES/SMSAB

General Electric Company

AFFIDAVIT

I, George B. Stramback, being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the attached GE Meeting Presentation, *ACRS Thermal Hydraulic Phenomena Subcommittee Meeting - GE Power Uprate Program – Closed Session*, dated June 12, 2001. This document, marked GE Proprietary Information, constitutes a proprietary compilation of information prepared by the General Electric Company.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

Both the compilation as a whole and the marked independently proprietary elements incorporated in that compilation are considered proprietary for the reason described in items (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. That information (both the entire body of information in the form compiled in this document, and the marked individual proprietary elements) is of a sort customarily held in confidence by GE, and has, to the best of my knowledge, consistently been held in confidence by GE, has not been publicly disclosed, and is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information is classified as proprietary because it contains detailed plans, processes, results and conclusions from these evaluations, utilizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The information also is classified as proprietary because it constitutes a confidential compilation of information, including detailed results of analytical models, methods, and processes, including computer codes, and conclusions from these applications, which represent, as a whole, an integrated process or approach which GE has developed, obtained NRC approval of, and applied to perform evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of a given increase in licensed power output for a GE BWR. The development and approval of this overall approach was achieved at a significant additional cost to GE, in excess of a million dollars, over and above the very large cost of developing the underlying individual proprietary analyses.

To effect a change to the licensing basis of a plant requires a thorough evaluation of the impact of the change on all postulated accident and transient events, and all other regulatory requirements and commitments included in the plant's FSAR. The analytical process to perform and document these evaluations for a proposed power uprate was developed at a substantial investment in GE resources and expertise. The results from these evaluations identify those BWR systems and components, and those postulated events, which are impacted by the changes required to accommodate operation at increased power levels, and, just as importantly, those which are <u>not</u> so impacted, and the technical justification for not considering the latter in changing the licensing basis. The scope thus determined forms the basis for GE's offerings to support utilities in both performing analyses and providing licensing consulting services. Clearly, the scope and magnitude of effort of any attempt by a competitor to effect a similar licensing change can be narrowed considerably based upon these results. Having invested in the initial evaluations and developed the solution strategy and process described in the subject document GE derives an important competitive advantage in selling and performing these services. However, the mere knowledge of the impact on each system and component reveals the process, and provides a guide to the solution strategy.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the

technology base includes the value derived from providing analyses done with NRC-approved methods, including justifications for not including certain analyses in applications to change the licensing basis.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to avoid fruitless avenues, or to normalize or verify their own process, or to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. In particular, the specific areas addressed by any document and submittal to support a change in the safety or licensing bases of the plant will clearly reveal those areas where detailed evaluations must be performed and specific analyses revised, and also, by omission, reveal those areas not so affected.

While some of the underlying analyses, and some of the gross structure of the process, may at various times have been publicly revealed, enough of both the analyses and the detailed structural framework of the process have been held in confidence that this information, in this compiled form, continues to have great competitive value to GE. This value would be lost if the information as a whole, in the context and level of detail provided in the subject GE document, were to be disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources, including that required to determine the areas that are not affected by a power uprate and are therefore blind alleys, would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing its analytical process.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 8th day of June 2001.

George B. Stramback
General Electric Company

Subscribed and sworn before me this 8TH day of JUNE 2001.

Notary Public, State of California

ANNA HANLIN
Commission # 1184507
Notary Public - California
Santa Clara County
My Comm. Expires Jun 19, 2002

GE Nuclear Energy

ACRS Thermal Hydraulic Phenomena Subcommittee Meeting

GE BWR Power Uprate Program

Closed Session

Jim Klapproth
Gene Eckert
Israel Nir

June 12, 2001

Outline

- *Opening Remarks (Open Session)*
- *Introduction*
  - *Past EPU briefing to ACRS*
  - *Key elements of GE power uprate program*
  - *BWR PU Implementation status and grid MWe time line*
- **GE Extended Power Uprate Program (Closed Session)**
  - **Background**
  - **Constant Pressure Power Uprate**
  - **EPU Specific Topics**
- **Concluding Remarks**

GE Extended Power Uprate Program

- **Background**
- **Constant Pressure Power Uprate**
- **EPU Specific Topics**
  - **CPPU Performance Margin**
  - **Evaluation Model (EM) Codes**
  - **EPU effect on Core and Fuel Design**
  - **EPU Startup Testing**

Background – ELTR Approach

- **ELTR represented an innovative and aggressive initiative for nuclear plants power uprate**
  - **Introduction included extensive plant specific analyses, extensive startup testing and plant operational evaluations**
  - **Comprehensive plant specific analysis including aspects not significantly affected by power uprate**
- **ELTR addressed both increase and no change in reactor dome pressure**
  - **Significant effect of pressure increase on certain uprate aspects**
  - **Simplification for constant dome pressure uprate not addressed**
- **Plant specific submittals expanded beyond PU (MELLLA, GE14)**

EPU Background – Experience

- **Extensive experience since initial ELTR introduction**
  - **Completed EPU evaluation for 10 BWR plants**
    - 7 domestic, 3 foreign
    - Systematic and common processes applied to all recent uprates
  - **EPU evaluation underway for 5 additional BWR plants**
  - **EPU implemented at 4 BWR plants**
    - 2 domestic, 2 foreign
    - Leading uprate over 117% OLTP for BWR-6
- **Successful and routine process implementation since ELTR approval**

Dome Pressure Increase Considerations

- **Original PU approach based on dome pressure increase**
  - **Allowed increased turbine steam passing capability**
  - **Minimized turbine-generator modification**
    - Typically not applicable to EPU which requires new HPT design
- **Licensing impact of pressure increase significant**
  - **Increased evaluation impact (TS setpoints, loads, HELB, EQ)**
  - **Additional Tech Specs affected**
  - **Increased NRC review scope**

Avoiding increase in reactor pressure reduces impact on system/component design and analysis

Background – Transition to CPPU

- **Need for efficient and effective EPU evaluation and NRC review processes**
  - **Uprate aspects not significantly affected by power increase should not dilute effort and priorities**
- **Introduce approach and process simplifications to facilitate focus on aspects affected by power increase**
  - **Separate changes not related to power increase**
    - Quantify direct uprate effect
  - **Separate aspects not affected by power increase**
    - Build on SPU and EPU experience
  - **Separate operational aspects affected by power increase**
    - Align review scope with regulatory requirements

A streamlined process which maintains full assurance of safety is needed

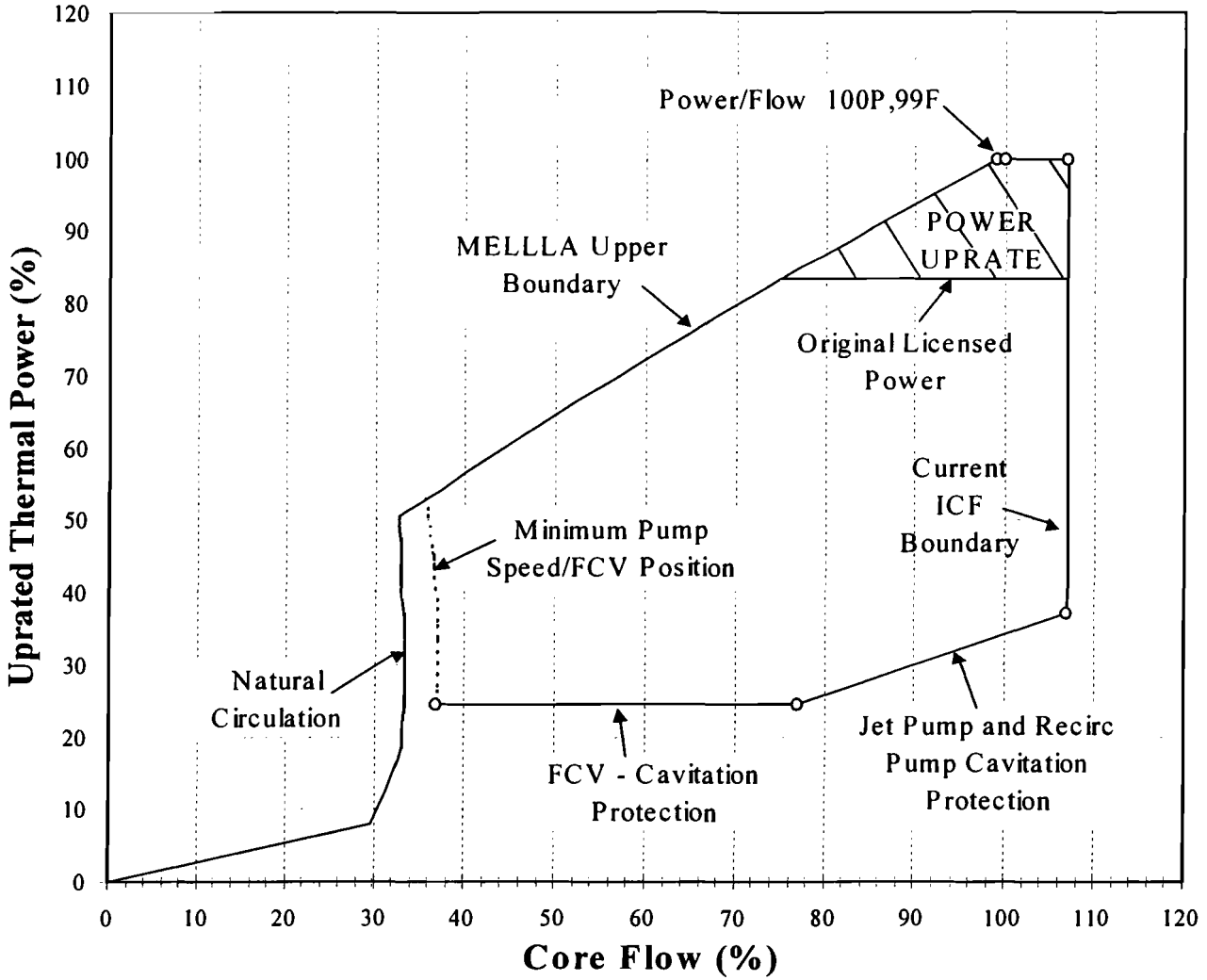
CPPU Initiative

- **Evolutionary process based on previous licensing experience**
  - **Stretch and Extended Power Uprate**
  - **Reload licensing process (GESTAR)**
  - **Improved Standard Technical Specification program**
- **Minimize changes not related to power increase**
  - **More than constant pressure…**
- **CPPU LTR provides limits, applicable methods, reference to computer codes and disposition process**
- **Plant specific supplement referencing CPPU LTR**

CPPU Changes

- **Limit power increase to 20% of Original Licensed Thermal Power (OLTP)**
  - **Increased steam flow**
  - **Increased feedwater flow**
  - **Increased feedwater temperature**
  - **Increased radiation source and levels**
  - **Increased decay heat**
- **Results in limited effect on safety related systems and performance**
- **Some modifications to non-safety related power cycle and balance of plant system required**

Power Uprate Operating Map



20% PU Heat Balance Trends

- **Sensitivity based on common 20% power increase demonstrates tightly grouped BWR performance**

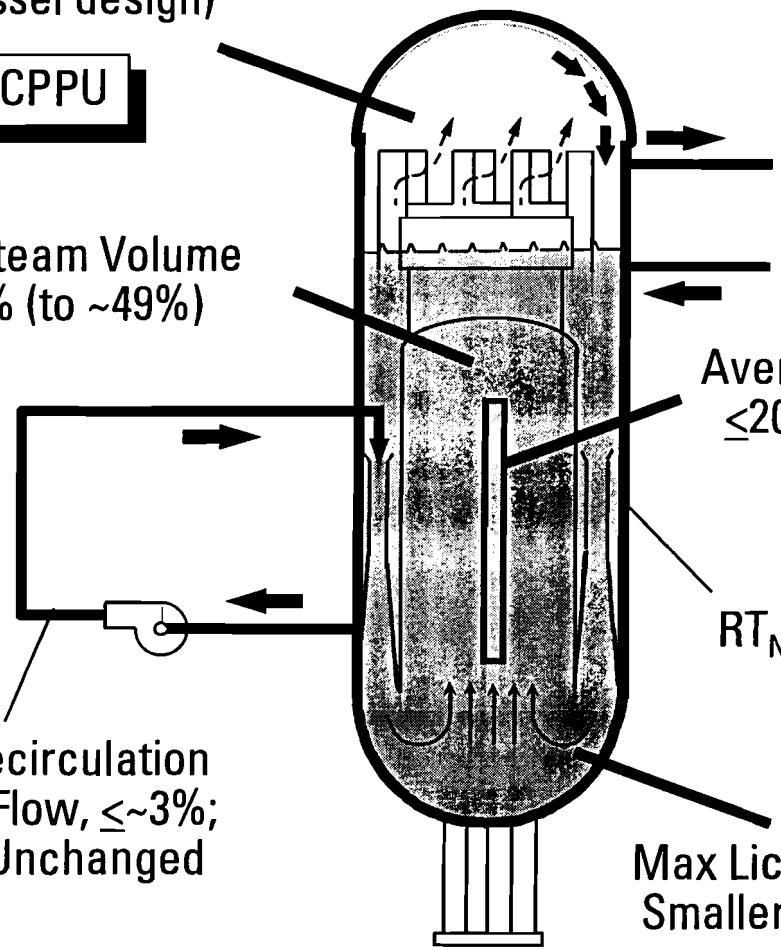
Parameter	Average Change	Range of Change
Input:		
Core Thermal Power	20 %	NA
Maximum Core Flow	0 %	NA
Max Operating Dome Press	0 %	NA
Output for GE BWR Fleet:		
Max Operating Dome Temp	No change	Constant
Steam Flow	23.2 %	22.1 to 24.1 %
Feedwater Flow	23.4 %	22.5 to 24.0 %
Feedwater Temp*	18.8°F	14.6 to 21.1°F
Core Inlet Enthalpy*	-1.9 Btu/lbm	-3.3 to –0.8 Btu/lbm

* Sensitivity study assumes no HP Turbine design changes

≤ 20% PU Effect on NSSS Conditions

ELTR: Dome Pressure, 1020 to 1095 psia (1265 psia vessel design)

No change for CPPU



Steam & FW Flows, ≤~+24%
FW Temperature, ≤~+19 °F

Average Core Steam Volume Fraction, ~+3% (to ~49%)

Average Channel Power Increase ≤20%; No Significant Change in Hot Channel Conditions (~Same operating limits)

RT_{NDT} Shift ≤~10 °F

Average Recirculation Drive Loop Flow, ≤~3%; Maximum Unchanged

Max Licensed Core Flow Unchanged
Smaller Full Power Core Flow Range

CPPU LTR Approach Key Elements

- ***Simplification:*** **Exclusion of changes not required by PU**
  - **No increase in maximum operating dome pressure**
  - **No change in plant licensing or design basis unless in LTR**
  - **Non-power uprate requests handled separately**
- ***Completeness:*** **Systematic disposition of all PU aspects**
  - **Follow previous EPU plant specific submittal outline**
- ***Focused review:*** **Plant specific results will be included in supplemental safety analysis report to the CPPU LTR**
  - **Will follow NRC pre-approved report shell**
  - **Will not repeat NRC approved CPPU LTR dispositions**
  - **Will report deviations from CPPU LTR**

Simplification: CPPU Exclusions

- **Significant reduction in LTR scope relative to original ELTR**
- **Changes other than thermal power minimized**
  - **No increase in maximum operating reactor dome pressure**
  - **No increase to maximum licensed core flow**
  - **No increase to MELLLA/MEOD upper boundary**
  - **No change to source term methodology**
  - **No new fuel mechanical design introduction**
  - **No change to fuel cycle length**
  - **No introduction of/additions to licensed operational enhancements**
- **Non-PU Changes may be submitted separately for NRC approval**

NRC review focused on power uprate effects only

Completeness: CPPU Dispositions

- **Systematic disposition of all PU aspects in CPPU LTR**
- **Four Types of Dispositions**
  - **Generic assessments**
  - **Core dependent reload evaluations**
  - **Operational assessments**
  - **Plant specific evaluations**

Generic Assessments

- **Bounding analysis**
  - **Previous ELTR bounding assessments applicable to CPPU**
  - **Additional generic studies**
- **Negligible effect due to CPPU**

Maximized generic reviews for more efficient plant specific NRC review process

Core Dependent Evaluations

- **Fuel operating cycle dependent**
- **Cycle specific core loading not established for PU submittal**
  - **Power uprate analyses on representative core loading**
  - **Cycle specific core analysis done following standard reload process**
  - **Reload analysis must be completed prior to power uprate implementation**
- **Impact for PU well known**
  - **Typical effect on safety analysis small due to no change in pressure and fuel/core loading design criteria**

Core dependent analyses performed for each reload

Operational Assessments

- Typically associated with normal plant operation or maintenance
- Not included in safety analyses or required to mitigate a design basis accident
- Separate NRC license amendment not required under 10CFR50.59 and is not performed for similar non power uprate related changes
- Will be dispositioned by utility but not included in plant specific power uprate submittal

Focused Review: Plant Specific Evaluations

- **CPPU will confirm key plant specific PU aspects**
  - **Pressure relief system**
  - **Reactor vessel fluence**
  - **Reactor vessel and internals structural performance**
  - **Containment temperature, pressure and loads response**
  - **ECCS LOCA performance**
  - **Emergency service water systems**
  - **SLCS ATWS requirement**
  - **Radiation source and accident radiological consequences**
  - **ATWS, SBO, fire protection, HELB, PSA, EQ**
- **Relative effect of PU known**
- **Allows focused, standardized plant specific analysis supplement to the NRC approved base CPPU LTR**

CPPU Initiative Summary

- **CPPU LTR facilitates BWR power uprate applications**
  - **Narrows and focuses ELTR scope**
  - **Experienced based**
  - **Utilizes established processes**
  - **Provides assurance that NRC review continues to address all power uprate safety aspects**
  - **Maximizes one time generic assessments**
  - **Eliminates repetitive reviews**

Simplified and focused process for NRC review of power uprate plant specific submittals

CPPU Implementation Target Plan

- **Schedule**
  - **Submit CPPU LTR for NRC review** **03/2001**
  - **Initial NRC feedback** **06/2001**
  - **1st CLTR based (partial) plant specific submittal** **3Q/2001**
  - **CPPU LTR SER** **12/2001**
  - **1st CLTR based (partial) plant specific SER** **1Q/2002**
  - **1st full CLTR based plant specific submittal** **1Q/2002**

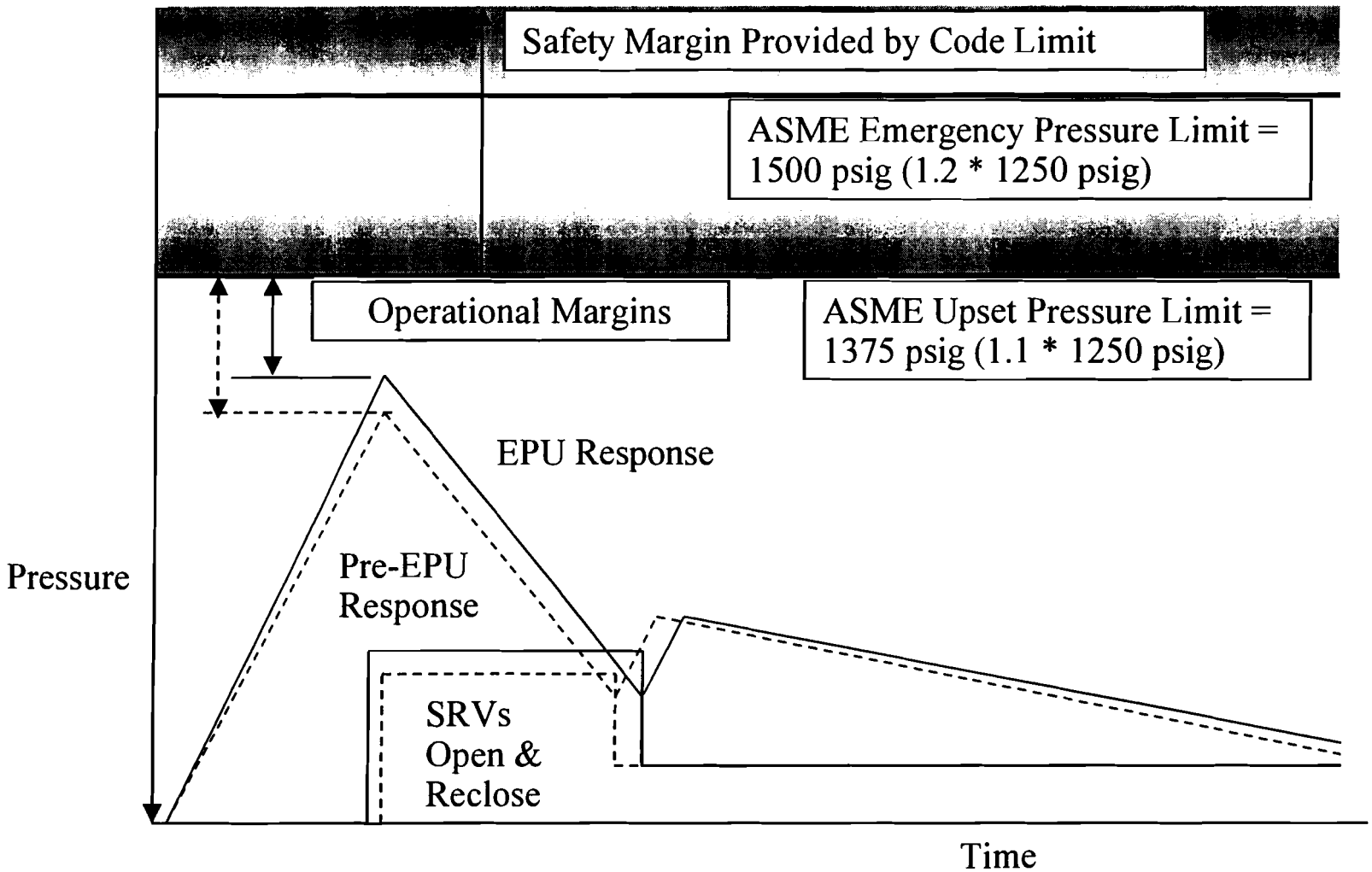
EPU Specific Topics

- **CPPU Performance Margin**
- **Evaluation Model (EM) Codes**
- **EPU effect on Core and Fuel Design**
- **EPU Startup Testing**

CPPU Effect on Performance Margin

- **Margin of safety unaffected by CPPU**
  - **Safety limits/criteria unchanged**
- **Existing NSSS design can accommodate increases in key parameters**
  - **Thermal power, steam/ FW flow, radiation, decay heat**
  - **Original NSSS design typically adequate**
- **Existing BOP design requires modifications to maintain needed operational margin**

CPPU Effect on Overpressure Protection Margin



CPPU Effect on Key Performance Parameters

CPPU Aspect	Key Parameter	CPPU Effect (Increase)	Change (Approximate)
Pressure relief system	Peak dome pressure	Steam flow	+ 10 psi
Reactor vessel RT_{NDT}	Vessel temp vs. press	Power	+ 10°F
Vessel - structural	Thermal cycles	FW flow & temp	Usage < 1.0
Internals - structural	Pressure drop	Steam flow	+ 4 psid
Containment	Peak temperature	Decay heat	+ 15°F
Containment	Peak pressure	None	< 1 psi
Containment	Peak load	Subcooling	< Current Load Definition
ECCS LOCA	PCT	None (Hot channel)	± 20°F
Radiation sources	Source term	Power	~ ΔPower
ATWS	Reactor pressure	Power	+ 75 psid
ATWS	Pool temperature	Power	+ 10°F
HELB	Peak temperature	Subcooling	+ 2°F

Evaluation Model (EM) Codes

- **All GE EPU analyses being performed with current, approved methodology (GESTAR)**
  - **Steady state and slow transients: ISCOR, PANACEA**
  - **Stability: ODYSY (Recently approved)**
  - **Fast transients: ODYN, TASC**
  - **ATWS: ODYN**
  - **LOCA-ECCS: SAFER/GESTR-LOCA**
  - **LOCA-Containment: SuperHEX**
- **Operating conditions at EPU power within range of application of these methods**
  - **Average channel void fraction increase ~3%**
  - **Hot channel power increase ~0% if on limits**
  - **Pressure within current operating and ASME ranges**

Evaluation Model (EM) Codes

- **Several methodology improvements in process**
  - **TRACG - Transients**
  - **TRACG - LOCA**
  - **Vessel Fluence Calculations**
- **Future methods not essential for EPU**
  - **Expected to provide additional operating margins when approved**

All EPU Analyses Based on Currently Approved Evaluation Model Codes

EPU effect on Core and Fuel Design

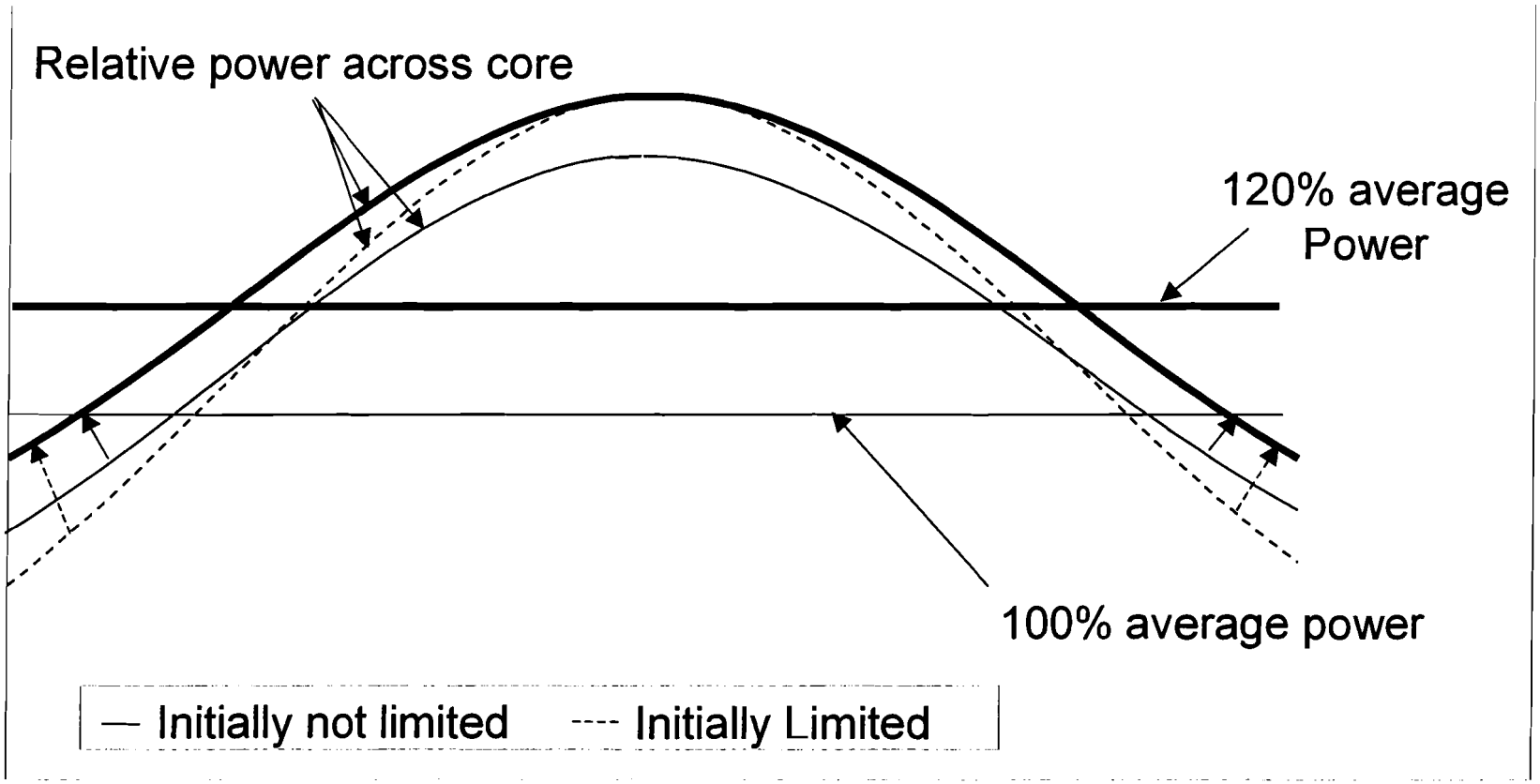
- **Changes to consider**
  - **Cycle Energy – increases**
    - Assume no change in calendar cycle length
    - Increase cycle energy by % increase in power
  - **Core average power - Increases**
  - **Operating Limit – No change**
    - Still maintain design margin MCPR, LHGR, MAPLHGR, SDM
    - Cannot operate beyond Operating Limit

EPU effect on Core and Fuel Design

- **Specific changes**
  - **Batch Size and/or enrichment Increase**
  - **Gadolinia loading may increase**
- **EPU effects**
  - **Flatter overall core design**
    - More bundles near Operating Limit
  - **Higher power bundles on periphery**
    - Low leakage philosophy maintained

EPU effect on Core and Fuel Design

- **EPU effect on core radial power shape**



EPU Startup Testing: KKL Experience

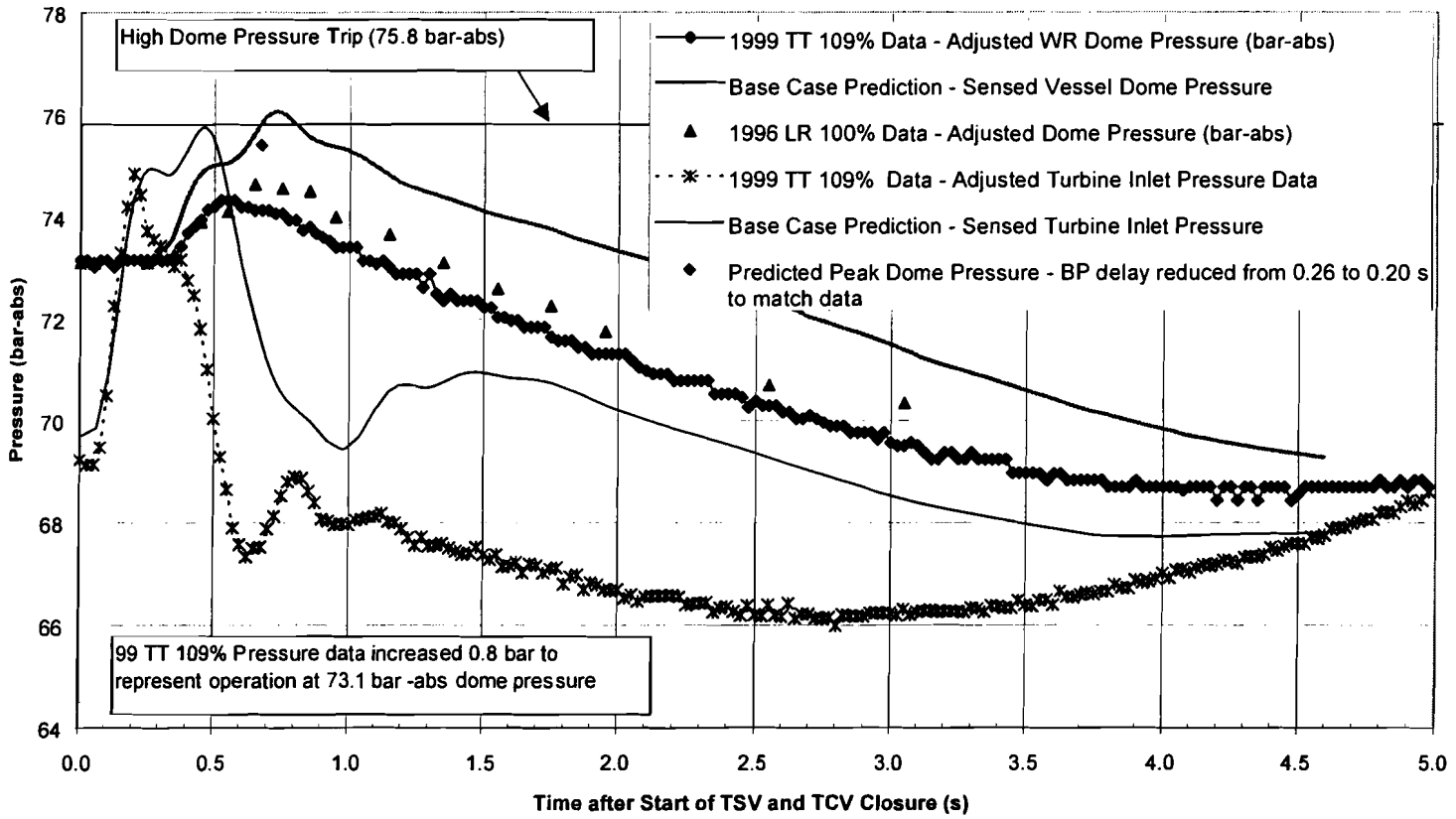
- **Step-wise increases in each cycle from 1996 to 2000 (+17%)**
- **Control system and core monitoring tests (same as US EPUs)**
- **Several successful scram avoidance tests performed in areas where equipment had been modified**
  - **Turbine-Generator trips at each power step (TCVs and BPVs had been modified) [Nominal ODYN predictions represent tests well; pressure response conservative]**
  - **Low level scram avoidance during one FW pump trip successfully performed at progressively more challenging operating conditions with modified FW System [Nominal ODYN prediction matched well]**
  - **Trip of one and two recirculation pumps performed from uprated conditions - successfully recovered without shutdown [ODYN match]**

Strong Confidence that Uprated Plant Performance is within Expected Response

EPU Startup Testing: KKL Experience

- **KKL Turbine Trip – 114% OLTP, 88% Core Flow**

1999 KKL Turbine Trip - 3420 MWt (109% of 3138MWt, 113.6% OLTP) Comparisons of Vessel Dome and Turbine Inlet Pressure Test Results to ODYN Predictions and 1996 Load Rejection Test



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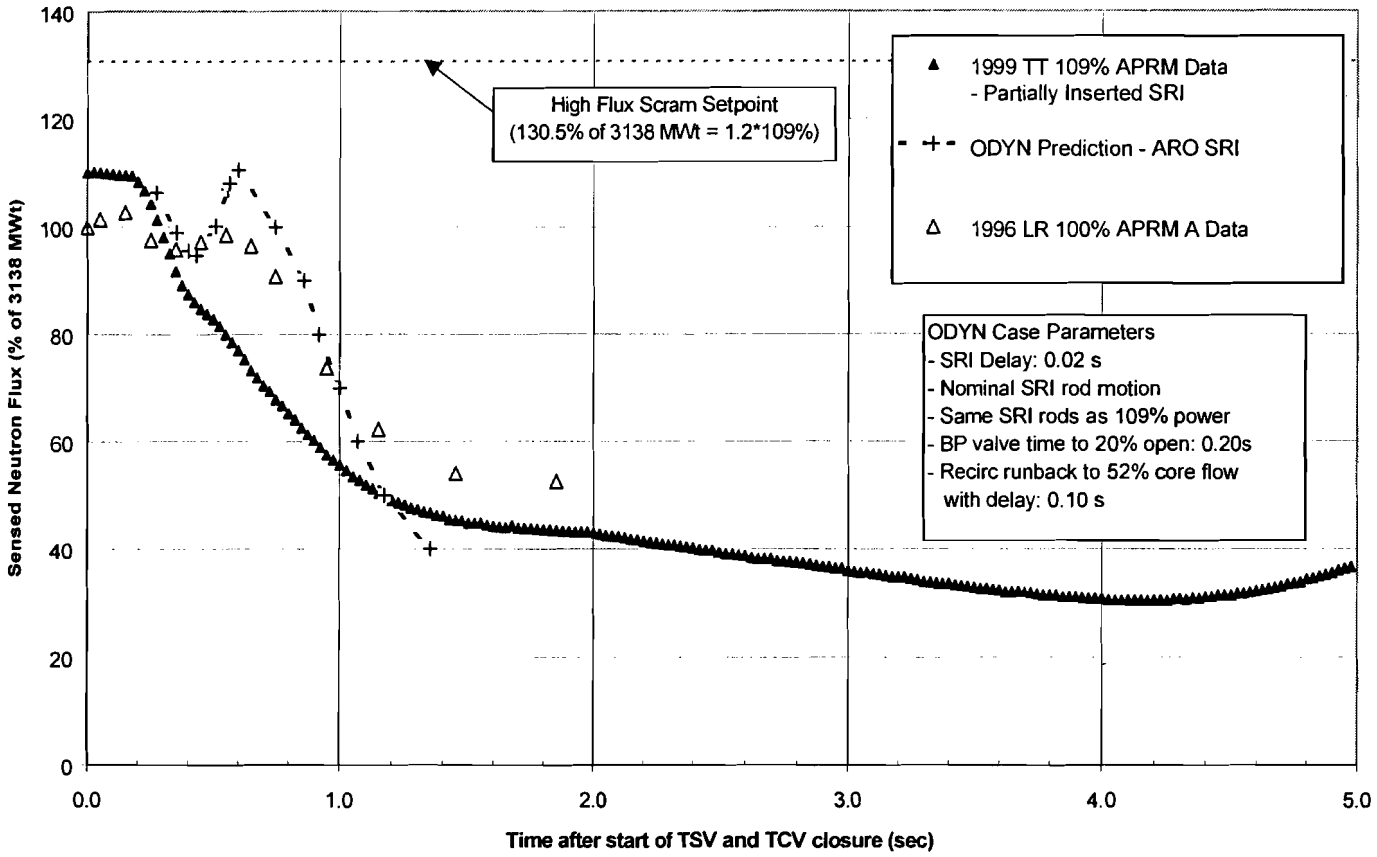
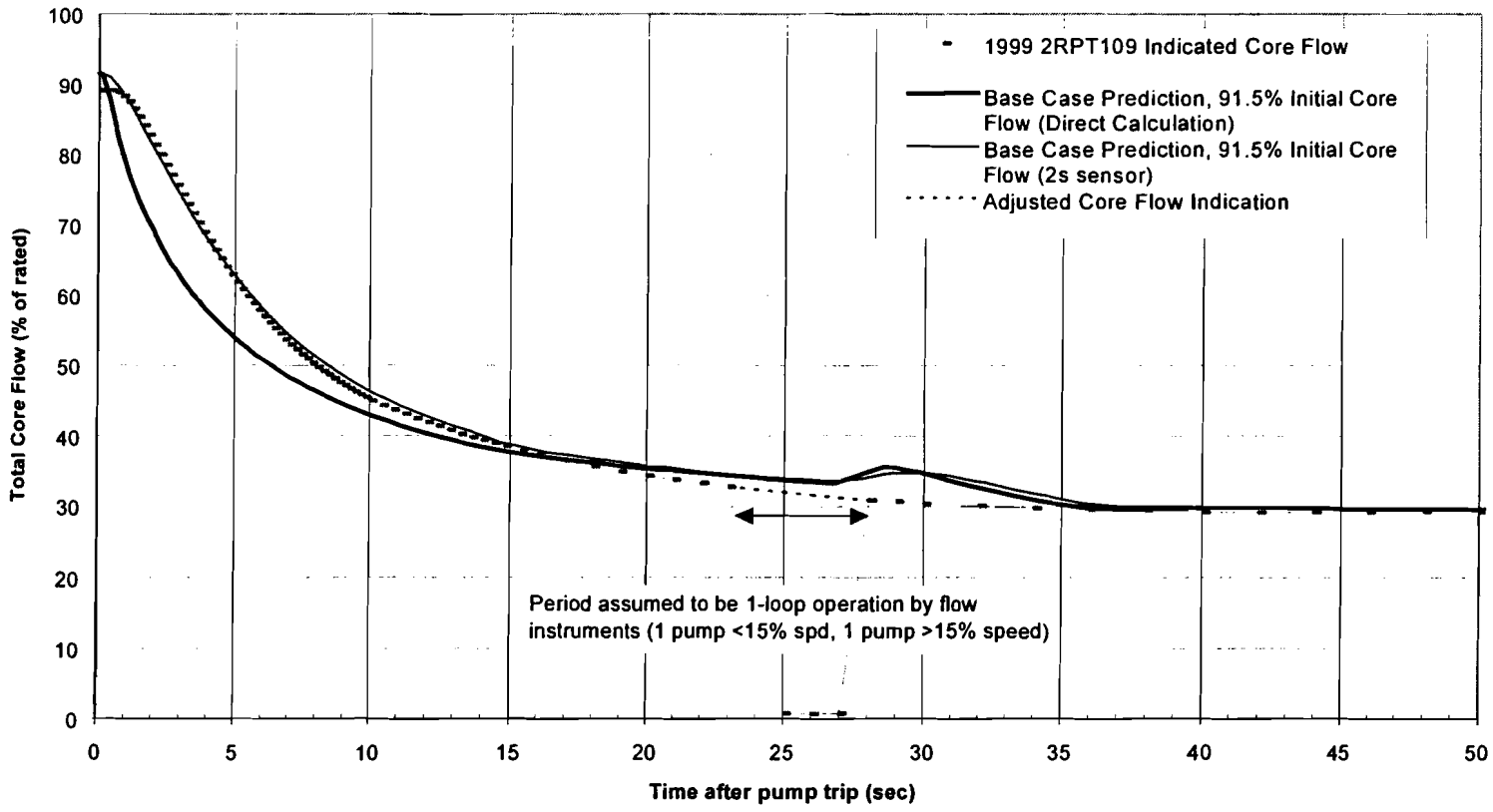


EPU Startup Testing: KKL Experience

- **KKL 2 Pump Trip – 114% OLTP, 88% Core Flow**

1999 KKL Trip of Both Recirculation Pumps from 3420 MWt (109% of 3138MWt, 113.6% OLTP)
Comparison of Core Flow Response to Prediction and 1995 Test



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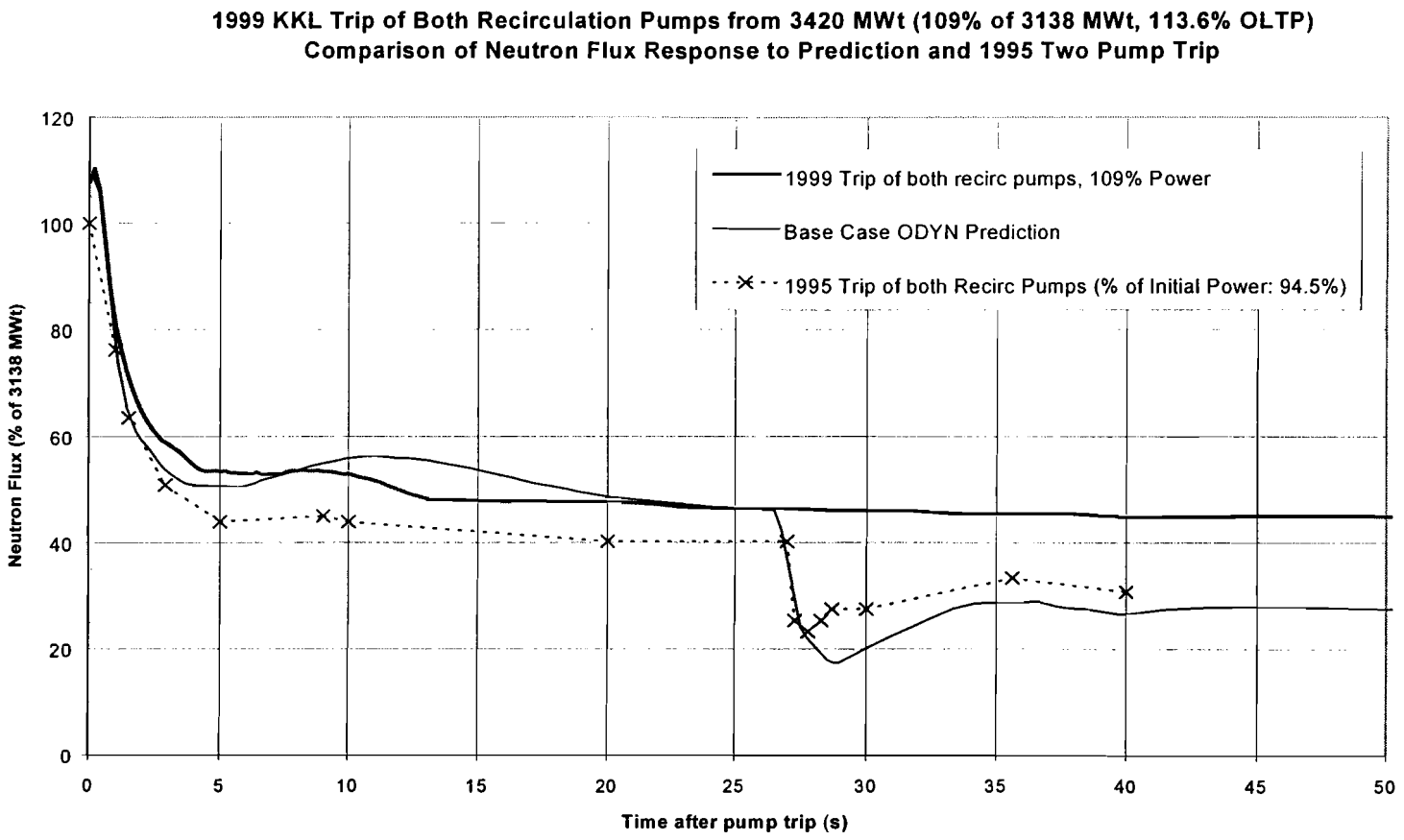

EPU Startup Testing: Hatch Experience

- **Hatch Initial EPU Operation**
  - **Successful approach to EPU on both units**
  - **Fine tuning of pressure controls on unit that did not enlarge turbine**
- **Hatch Turbine Trip**
  - **Unplanned Unit 2 trip on May 5, 1999**
  - **All plant parameters within all safety criteria and safety analyses**
  - **All plant parameters equal to or less severe than post-trip nominal calculation (e.g., no power increase, pressure rise and SRV opening as expected)**

Summary: CPPU Testing

- **Safety parameters within nominal predictions using current models**
- **Equipment operating within existing pressure and temperature conditions**
- **Unplanned large transients to be recorded and nominal post-event analysis performed for EPU plants with no pressure increase**

Strong confidence that uprated plant safety performance is within expected response

Concluding Remarks

- **Extensive experience with EPUs**
  - **EPUs implemented at 4 BWRs**
  - **NRC review on-going for 5 additional plants**
- **Plant safety margins maintained**
- **Constant pressure power uprate approach reduces EPU impacts**
- **High volume of EPU review requests anticipated**
  - **Two additional submittals in 2001**
  - **Expect 4 BWR submittals per year**
- **Streamlined approach proposed by GE**

POWER UPRATE RISK-INFORMED REVIEW QUESTIONS

1. Power uprates apparently lead to increased frequency of release of small amounts of radioactivity (short of causing fatalities). These increases are not reflected in increases in CDF or LERF. Do we need to have additional risk-informed acceptance criteria on the frequency of fission product releases of all magnitudes?

2. Power uprates can be accommodated by reducing the margins between calculated values and the limits established for the figures-of-merit related to design basis accidents. Can the licensee use up all of these margins? Do we need to include uncertainty determination in the analyses to determine the probabilities associated with exceeding the limits?

3. If power uprates lead to significant increases in burnup, do we reflect this in our PRA evaluations related to core melt behavior (e.g., ECCS success criterion) and fission product release?

4. Will power uprates change the fission product source term associated with "gap release"? Will power uprates affect the "iodine spike" magnitude and the associated releases? Are these changes accounted for in the DBA analyses?

5. Comparison of times required and times available before and after power uprates for operator actions in response to DBAs.

6. Risk-Informed Uprate Review Considerations
   - Need for Assessment of Operational Data for Uprated Plants
   - Evaluation of Leibstadt Risk Study for Uprated Plants

R

ACRS BRIEFING

ON

THE POWER UPRATE PROGRAM

June 12, 2001

J. Hopkins, Senior Project Manager, NRR

AGENDA

- Introduction ..... J. Hopkins, NRR
- Duane Arnold Audit ..... R. Caruso, NRR
- Risk Informed Review Considerations ..... D. Harrison, NRR
- RES Research Program ..... J. Rosenthal, RES
- Concluding Remarks ..... J. Hopkins, NRR

EXTENDED POWER UPRATES (15 - 20%)

Plant	Uprate	Submittal Date	Review Completion
Duane Arnold	15%	11/20/2000	10/15/2001 T
Dresden, Units 2&3	17%	12/29/2000	11/30/2001 T
Quad Cities, Units 1 &2	17.8%	12/29/2000	11/30/2001 T
Clinton	20%	06/18/2001 T	03/30/2002 T

ACRS BRIEFING

ON

THE POWER UPRATE PROGRAM

June 12, 2001

J. Hopkins, Senior Project Manager, NRR

AGENDA

- Introduction ..... J. Hopkins, NRR
- Duane Arnold Audit ..... R. Caruso, NRR
- Risk Informed Review Considerations ..... D. Harrison, NRR
- RES Research Program ..... J. Rosenthal, RES
- Concluding Remarks ..... J. Hopkins, NRR

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(3)

R

Duane Arnold Power Uprate Audit Results

Ralph Caruso
Chief, BWR Systems and
Nuclear Performance Section
Reactor Systems Branch
US Nuclear Regulatory Commission

June 12, 2001

Background

- DAEC Power Uprate
- Staff Performing Review for Compliance with ELTR2 Topical Report
- Staff Review Includes Audit of Various Licensing Calculations in Support of Uprate Conditions
- Audit Performed on March 26-29 by Team of 4 Staff Members

Audit Scope

- SAFER/GESTR LOCA Methodology
- Long-Term Stability Option I-D
- GEXL14 Correlation
- Reactor Core Design
- Methodology and Uncertainties for Safety Limit MCPR

Findings

- SAFER/GESTR
  - Analyses for Rated Conditions Complied with SER Conditions and the Codes Were Appropriately Applied.
  - Analyses of Single-loop Conditions Used Uncertainties Derived from TRAC and Full Power Operation, Which May Not Be Directly Applicable to Off-rated Conditions.
  - However, Conservative Penalties Are Applied to Single-loop Operation, and Plants Rarely Operate in this Condition.

- Long-Term Stability Solution
  - Option I-D still Applicable to DAEC
  - Power Uprate and Change of Licensing Methodology Will Increase Operator Reliance on SOLOMON On-Line Stability Monitoring System

- Recommendation that DAEC and Fuel Vendor Work Together to Increase Operator Confidence in Ability of SOLOMON to Predict Instability

- GEXL14
  - Staff Review Identified Use of COBRAG-generated Data in GEXL14 Database
  - Staff and GE Discussing Appropriateness of COBRAG to Generate Data

- Reactor Core Design
  - Methods Continue to be Used Appropriately

- Safety Limit MCPR
  - Methods Continue to be Used Appropriately

Conclusions

- Approved Methods Continue to Be Used Appropriately at Uprated Power Levels

- GEXL14 Correlation Database Evaluation Continues to be Discussed with GE

- Staff Will Continue to Perform Audits of Analysis Methods and Results for Upcoming BWR Extended Power Uprates
  - Dresden 2,3, Quad Cities 1,2
  - Clinton

POWER UPRATES

RISK-INFORMED REVIEW CONSIDERATIONS

A DISCUSSION BETWEEN
THE NRC STAFF AND THE ACRS

JUNE 12, 2001

Power Uprates
Risk Informed Review Considerations

Current Review Considerations - Based on Duane Arnold

- Duane Arnold Review Still On-going
- Not Submitted as a Risk Informed Licensing Action
- PRA Quality
  - Reflects Plant Configuration and Operating Procedures
  - Peer Reviewed
  - Staff Considering Review of the Peer Review

Power Uprates
Risk Informed Review Considerations

Current Review Considerations - Based on Duane Arnold

- Initiating Event Frequencies
  - No Change Anticipated
- Success Criteria
  - Thermal Hydraulic Evaluations Performed to Establish/Confirm Success Criteira
  - No Significant Impact Identified

Power Uprates
Risk Informed Review Considerations

Current Review Considerations - Based on Duane Arnold

- Operator Response
  - Thermal Hydraulic Evaluations Performed to Establish Event Timing Sequences
  - Decreased Timing Increases HEPs
    - Responses to ATWS for SLC Injection, Inhibiting ADS, Reactor Water Level Control
    - Response to Non-ATWS With Reactor at High Pressure for Depressurization on HPI Failure
  - Increased Internal Events CDF by 1.1E-6/year
  - Increased LERF by 1.39E-7/year
  - Increased External Events CDF by 2E-8/year

Power Uprates
Risk Informed Review Considerations

Current Review Considerations - Based on Duane Arnold

- Component Reliability
  - No Change Anticipated
  - Functionality and Reliability Maintained by Component Monitoring Programs
    - Maintenance Rule, Erosion/Corrosion, Environmental Qualification, Instrument Trending
- Shutdown Risk
  - No Significant Impact Anticipated

Power Uprates
Risk Informed Review Considerations

ACRS Topics/Questions

- PRA Quality
- Risk Impact on Shutdown Operations
- Power Uprate Risk-Informed Review Questions

Power Uprates
Risk Informed Review Considerations

PRA Quality

- Not Risk-Informed Licensing Actions
- Licensees are Required to Meet Existing Deterministic Requirements
- Staff Ensure No Significant Risk Changes or New Vulnerabilities are Introduced
  - Following RG 1.174 Framework
    - ΔCDF and ΔLERF
    - Industry Peer Reviews
    - Staff May Review PRAs/Peer Reviews

Power Uprates
Risk Informed Review Considerations

Risk Impact on Shutdown Operations

- Increased Decay Heat During Shutdown
  - Extended Time DHR Systems Needed to Remain in Service and Available
  - Reduced Time to RCS Boiling
  - Reduced Time for Operator Responses to Loss of DHR Before Boiling
  - Reduced Time for Containment Closure Given a Postulated Loss of DHR if RCS is Open

Power Uprates
Risk Informed Review Considerations

ACRS Power Uprate Risk-Informed Review Questions

1. Power uprates apparently lead to increased frequency of release of small amounts of radioactivity (short of causing fatalities). These increases are not reflected in increases in CDF or LERF. Do we need to have additional risk-informed acceptance criteria on the frequency of fission product releases of all magnitudes?
2. Power uprates can be accommodated by reducing the margins between calculated values and the limits established for the figures-of-merit related to design basis accidents. Can the licensee use up all of these margins? Do we need to include uncertainty determination in the analyses to determine the probabilities associated with exceeding the limits?
3. If power uprates lead to significant increases in burnup, do we reflect this in our PRA evaluations related to core melt behavior (e.g., ECCS success criterion) and fission product release?
4. Will power uprates change the fission product source term associated with "gap release"? Will power uprates affect the "iodine spike" magnitude and the associated releases? Are these changes accounted for in the DBA analyses?
5. Comparison of times required and times available before and after power uprates for operator actions in response to DBAs.
6. Risk-Informed Uprate Review Considerations
   a. Need for Assessment of Operational Data for Uprated Plants
   b. Evaluation of Leibstadt Risk Study for Uprated Plants

Power Uprates
Risk Informed Review Considerations

Question 1: Need for Additional Acceptance Criteria for Frequency of Release of All Magnitudes

- RG 1.174 Philosophy: Increases in CDF and Risk are Small and Consistent with Commmission's Safety Goal Policy Statement
  - Use ΔCDF and ΔLERF as Quantitative Acceptance Guidelines
  - Do Not Address Impacts on Frequency of Routine or Small Releases
- No Additional Risk Metrics are Necessary to Assure Risk Impact from Uprates is Consistent with Commission's Safety Goal Policy Statement and to Allow Assessment of RG 1.174 Guidelines

Power Uprates
Risk Informed Review Considerations

Question 2: Margins

- Licensing Limits Established by Regulation (e.g., Appendix K -2200°F), by Approved Topical Report, or by Internal Design Constraints
- Analyses Have Not Included Uncertainty Determinations in the Past, But More Analyses are Based on Statistical Uncertainties
  - MCPR Limits
  - Realistic LOCA
  - Criticality Calculations
- Licensees and Vendors Control Use of Margins Up To Approved Limits

Power Uprates
Risk Informed Review Considerations

Question 3: Need to Reflect Increased Burnup in PRA

- Thermal-Hydraulic Codes Used to Establish PRA Success Criteria Should Reflect Core Burnup
- In Using ΔLERF, Increased Fission Product Releases From Increased Core Power Will Not Be Reflected
  - Quantity of Radionuclides Released Will Increase, But Release Fractions Will Be Essentially Unchanged
  - Time to Release May Be Shortened, But Not Expected to Shift Releases From Late to Early Release Categories

Power Uprates
Risk Informed Review Considerations

Question 4: Impact on DBA Source Terms

- Fission Product Inventory Will Increase
  - Gap Fraction Considered a Function of Burnup
  - Power Uprate Has No Direct Impact on Gap Fraction
  - Iodine Appearance Rate and Spiking Factor Based on Tech. Spec. Equillibrium RCS Activity
  - 500x Spike Multiplier Will Continue to Compensate for Uncertainty in the Iodine Spike Rate

Power Uprates
Risk Informed Review Considerations

Question 5: Operator Time Required vs. Time Available for Pre- and Post-Uprate Conditions

- Operator Response Timing is One of the Main Areas Observed to be Impacted by Power Uprates
  - Shorter Response Time Available Results in Larger HEPs
  - Typical Impacts are to ATWS Response (e.g., SLC Initiation and Inhibiting ADS) and High-Pressure System Failure Response

Power Uprates
Risk Informed Review Considerations

Question 6a: Need to Assess Operational Data

- Licensees Currently Track and Trend Operational Data
  - Maintenance Rule
  - Corrective Action Program
- Significant Impacts Should be Self-Revealing
  - Staff Considering Ways Monitoring Programs, Performance Indicators, etc. Could be Used to Provide Early Indication of Power Uprate Impacts

Power Uprates
Risk Informed Review Considerations

Question 6b: Evaluation of Leibstadt Study Power Uprates

- Addressed 14.7% Power Increase
  - Studied Same Factors as NRC for CDF Impacts
  - No Significant Effect on CDF
    - 2% Increase Mainly From Increased HEPs
  - No Effect on Frequency of Release Categories
  - Release Increase of 25-30%
    - 15% Increase in Activity Inventory in Core
    - 15% Increase Due to Faster Accident Sequences
  - Overall Risk Remains Small
    - Within Uncertainty Band of Phenomena
    - No Cliff Edge Effects Identified

R

RES Plans to Assess Synergistic Effects of Power Uprates

Presented to ACRS Subcommittee on Power Uprates

June 12, 2001

Jack Rosenthal

Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

Identification of Issues

Quantitative Analyses at Extended Power for Risk Dominant Sequences

- Anticipated Transient Without Scram
- Station Blackout
- Loss of Heat Removal
- Loss of Coolant

Review of success criteria at extended power for range of sequences in PRA

Review of Generic Issues

- Power/Flow Stability
- Torus Hydro-Dynamic Loads

Identification of Issues (Continued)

Primary System

- Reactor pressure vessel- higher fluence induces embrittlement
- Pipe loading

Containment Systems

- Higher suppression pool temperatures correspond to reduced NPSH

Control Systems

- Pressure and flow control, turbine and steam bypass

Human Actions

- Response time

Signature Estimates of Margin Reductions for Power Uprates/License Renewal

August W. Cronenberg
ACRS Fellow

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Outline

- Margins in the Regulatory Process
- Margin Estimates for Power Uprates
- Margin Estimates for Renewal Plants
- Findings To Date

2

Margins in Regulatory Process

- Webster (Margin):
  - Spare amount allowed for contingencies.
  - Bare minimum below which something is no longer desirable.
- General Design Criteria (10CFR50-App. A)

  Criterion 10: Reactor core and associated coolant, control, and protection systems shall be designed with sufficient margin to assure acceptable design limits shall not be exceeded.

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General Design Criteria (continued)

- Criterion-31: The reactor coolant pressure boundary shall be designed with sufficient margin to assure…it behaves in non-brittle manner and probability of rapidly propagating fracture is minimized.
- Criterion-50: The containment, including access openings, penetrations….shall be designed…without exceeding design leakage rate and with sufficient margin…to reflect…metal-water and other chemical reactions…

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Margin Requirements More Explicitly Spelled Out

- Regulatory Guidance/Standard Review Plan (e.g. acceptance criteria for design Press., P-T limits, stress limits, allowable materials, ductility limits, etc.)
- ASME Boiler & Pressure Vessel Code
- Am. Nat. Standards Inst. (ANSI)
- Other

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Impact of Power Uprates on Plant Operating Conditions & Margins

- **Primary System Conditions**
  - Core Power/Coolant Enthalpy
  - Core Flow Rate/Coolant Temperature
  - Fuel Temperature
  - BWR-Steam Dome Temp./Press.
- **Secondary System Conditions**
  - Steam Generator Flow Rate/Temp.
  - Feedwater Flow Rate/Temperature
  - Feedwater Pumping Requirements

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Current Power Uprate Applications

Plant	Power Uprate	Year Start
Duane Arnold/BWR	15-%	1975
Dresden-2/BWR Dresden-3/BWR	17-% 17-%	1970 1971
Quad Cities-1/BWR Quad Cities-2/BWR	17-% 17-%	1973 1973
Brunswick-1/BWR Brunswick-2/BWR	15-% 15-%	1977 1975
Clinton/BWR	20-%	1987
Arkansas Nuclear One-2/PWR (C-E)	7.5-%	1978

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Margin Estimates for Power Uprates (Hatch Case Study)

- Hatch Plant Characteristics
  - GE-BWR/4 (direct-cycle)
  - Mark-I Containment (inverted light-bulb/ torus suppression pool)
- Power Level
  - 1974 Unit-1/1979 Unit-2 = 2436 MWt
  - 1995 Units 1 & 2 = 2558 MWt (5-% uprate)
  - 1997 Units 1 & 2 = 2763 MWt (8-% uprate)
- License Renewal Application (under review)

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Summary of Hatch Uprate Conditions

Parameter	Hatch Unit-1			Hatch Unit-2		
Thermal Power, MWt (Year at power)	2436 (1975)	2558 (1995)	2763 (1997)	2436 (1979)	2558 (1995)	2763 (1997)
% Power Uprate (from prior value)	–	5-%	8-%	–	5-%	8-%
Core Coolant Flow Rate, 10^6 lb_m/hr (at %-Power)	68.3-82.4 (87-105)	68.3-82.4 (87-105)	68.3-82.4 (87-105)	67-80.9 (87-105)	67-80.9 (87-105)	67-80.9 (87-105)
Vessel Steam Flow, 10^6 lb_m/hr	10.0	10.6	11.5	10.5	11.1	12.0
Steam Dome Pressure, psig	1015	1050	1050	1015	1050	1050
Steam Dome Temp., F	547	551	551	547	551	551
Full-Power Feedwater Flow, 10^6 lb_m/hr	10.1	10.7	11.6	10.5	11.2	12.1
Full-Power Feedwater Temp., F	388	393	398	420	424	425

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Summary Hatch-I Operational Margins

Residual Margin = (Design Limit – Value) / (Design Limit)

Power Level, MWt	Parameter Value	Residual Margin, %
Main Steam-line Pressure (Design Limit = 1250 psig)		
Original = 2436	1015 psig	18.8
1st Uprate = 2558	1050 psig	16
2nd Uprate = 2763	1050 psig	16
Main Steam-line Temperature (Design Limit = 575 F)		
Original = 2436	546 F	5.04
1st Uprate = 2558	---	---
2nd Uprate = 2763	551 F	4.17

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Summary Hatch-I Operational Margins

Power Level, MWt	Parameter Value	Residual Margin, %
[illegible]		
Original = 2436	1130 psig	31.5
1st Uprate = 2558	—	—
2nd Uprate = 2763	1088 psig	34.1
[illegible]		
Original = 2436	392 F	30.2
1st Uprate = 2558	—	—
2nd Uprate = 2763	400 F	28.8

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Hatch-I Vessel DBA-LOCA Margins

Power Level, MWt	Predicted Stress, ksi	Residual Margin, %
[illegible]		
Original = 2436	8.95 ksi	41.4
1st Uprate = 2558	9.05 ksi	40.8
2nd Uprate = 2763	---	---
[illegible]		
Original = 2436	52.7 ksi	24.6
1st Uprate = 2558	53.0 ksi	24.2
2nd Uprate = 2763	---	---
[illegible]		
Original = 2436	---	---
1st Uprate = 2558	64.5 ksi	40.1
2nd Uprate = 2763	90.0 ksi	16.4
[illegible]		
Original = 2436	31.5 ksi	37.9
1st Uprate = 2558	34.8 ksi	31.4
2nd Uprate = 2763	34.9 ksi	31.2

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Hatch-I Containment DBA-LOCA Margins

Power Level, MWt	Parameter Value	Residual Margin, %
Peak Drywell Pressure (Design Limit = 56 psig/Max=62 psig)		
Original = 2436	47.9 psig	14.5
1st Uprate = 2558	49.6 psig	11.4
2nd Uprate = 2763	50.5 psig	9.8
Peak Drywell Gas Temperature (Design Limit = 281 F)		
Original = 2436	290 F (for short time only)	Exceeds design limit for short time
1st Uprate = 2558	292 F (for short time only)	Exceeds design limit for short time
2nd Uprate = 2763	293 F (for short time only)	Exceeds design limit for short time
Peak Suppression Pool Temperature (Design Limit = 281 F)		
Original = 2436	198 F	29.5
1st Uprate = 2558	202 F	28.1
2nd Uprate = 2763	208 F	26.0

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Hatch-I License Renewal/Margin Summary

$CUF = N_{OBE}/f_1 + N_{Scram}/f_2 + N_{Startup}/f_3 + \ldots\ldots N_{other}/f_n$

Residual Margin (RM) = 1 - [CUF]

Component	Hatch Count	CUF at 40 yr	RM at 40 yr	CUF at 60 yr	RM at 60 yr
Residual Heat Removal Suction Piping	2	0.57	43-%	0.77	23-%
Reactor Vessel Equalizer Piping	1	0.52	48-%	0.64	36-%
Core Spray Replacement Piping	1	0.16	84-%	0.19	81-%
Feedwater Piping	2	0.61	39-%	0.83	17-%
Standby Liquid Control Piping	1	0.24	76-%	0.25	75-%
Steam Condensate Drainage Piping	2	0.66	34-%	0.89	11-%

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Hatch-I License Renewal/Margin Summary

Relative Residual Margin (RR-Margin) = 1 - $\frac{T_{limit}(@\ EFPY) - 157}{157}$

[illegible]	[illegible]	[illegible]
Min. Temperature @ 54 EFPY	291.4 F	14.4
Min. Temperature @ 48 EFPY	283.6 F	19.4
Min. Temperature @ 44 EFPY	277.6 F	23.2
Min. Temperature @ 40 EFPY	271.7 F	26.9
Min. Temperature @ 36 EFPY	265.2 F	31.1
Nil-irradiation damage T-limit at bottom head = 157 F		

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Margin Estimate/Data Sources

- Power Uprate Operational Conditions (e.g. Main-Steamline Press.)
  - Design Limit: FSAR & ULAR
  - Uprate Press: ULAR or ULAR-SAR
- Power Uprate DBA-LOCA Predictions (e.g. bolt stress)
  - Component Design Limit: FSAR & ULAR-SAR
  - Component DBA-LOCA Stress: ULAR-SAR-Appendix (GE sub-contract report; summary tables of LOCA stress for a limited number of components, which can be different than assessed in original FSAR or prior uprate-SAR)
- License Renewal/Component Aging Effects-TLAA
  - LRA/TLAA-Chap. 4
  - CUF (Contractor Refs. To Chap.-4, *Structural Integrity Assoc.)*

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Summary & Observations to Date

- *Safety Margins* used in very broad sense in regulatory process.
- Difficulty in self-consistent data for margin impact.
- Some success for Hatch. Generally reduced margins indicted for various components for both power uprates and plant life extension.
- License Amendment-SARs and NRC-SERs do not appear to be of sufficient detail or consistency for in-depth/quantitative assessment of margin impact for multiple licensing actions (glean bits & pieces).

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Observations/Suggestions for Power Uprates

- NRC Uprate Review: Centers on assessment that current regulatory requirements are satisfied (minimal or no requirements for risk impact, margin reductions, impact of multiple licensing actions/synergies)
- Endorse Prior Recommendations for Uprate Standard Review Plan
  - 1996 Main Yankee Lesson Learned Report (Cota, Cubbage, Dorman—NRR)
  - 1996 Power Uprate Review (Scientech)
  - 1999 Review of Operational Events for Power Uprates (ACRS Fellow Report)
- Uprate-SRP Should Include Following:
  - Standardized listing of all systems, structures, & components (SSC) subject to uprate review
  - Assessment of impact of uprate on SSC margins for both operational and DBA conditions (clear definition of methods/criteria)

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Power Uprate/Suggestions (cont'd)

- Recommend Legacy Tables (Original FSAR→ Latest Uprate)
  - Standardized Table of plant conditions impacted by power uprates, and other license amendments (LA) impacting same plant conditions
  - Standardized Table of DBA predicted loads on SSC impacted by power uprate, and other LA impacting same loads
  - Standardized Table of margins impacted by power uprates, and other LA impacting same margins
- Risk assessment requirements for significant power uprates
- Bottom Line: A clear understanding of safety & margin implications of <u>significant power uprates</u> to an <u>aging fleet</u> of plants

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Potential Margin Reductions Associated with Power Uprates

ACRS Fellow: August W. Cronenberg

483rd ACRS Meeting (June 6-8, 2001)

Recent headline news of electrical power shortages in California and other western states point to an ever increasing demand for electrical power, which is exacerbated by the fact that little new generating capacity has been added to the national grid over the past decade. Electricity shortfalls have been compounded by uncertainties/dislocations stemming from deregulation. These developments appear to have affected a more positive public attitude toward continued use of nuclear power. All this has lead to industry initiatives at plant life extension, power uprates, and requests for a longer fuel cycle at ever higher burnups. Indeed industry is aggressively moving toward the goal of maximizing power generation and investment recovery from its aged population of nuclear power plants. Of the approximately 100 nuclear units currently in operation, it is estimated that upwards of 80 may apply for life extensions beyond their current 40-year license. It is also anticipated that many plants will apply for significant increases in power level; recent examples being the 15-% uprate request for the Duane Arnold plant and the 17-% increase for the Dresden and Quad City units. The agency is under considerable pressure to respond to these initiatives and approve such license modifications in a timely manner. The ACRS is concerned about the overall safety implications of such licensing actions; in particular the potential for reductions in design basis *margins* [a] owing to multiple licensing actions; which is the subject of this presentation.

A prior ACRS-fellow study [1] of operational events noted for power uprated plants points to potential *synergistic*[b] safety implications of power uprates when combined with fuel life extensions to higher burnup and plant aging phenomena. Examples cited in that study include control rod insertion problems noted in high-burnup/high-power fuel assemblies for the uprated Wolf Creek plant (uprate/high-burnup synergism), as well as several pipe failure events via corrosion/erosion effects, notably at the uprated Callaway-PWR and Susquehanna-BWR plants (uprate/aging synergism). Additionally risk implications have been assessed for the 15-% power uprate for the Swiss Leibstadt-BWR plant [2]. In that study the risk impact of the power increase was based on an estimation of reduced operator response time to a core uncovery event due to the elevated decay-heat at the higher power. The Leibstadt risk study also points to a similar power-increase/risk-increase dependence owing to a greater severe accident containment failure probability at the higher power level (due to increased containment over-pressure with increased power level). The Leibstadt study thus points to increased risk for plants operated at ever higher power levels.

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a) Margin (from Webster): a spare amount or measure allowed for contingencies; a bare minimum below which something becomes no longer desirable

b) Synergistic (from Webster): the cooperative action of discrete agencies such that the total effect is greater than the sum of the effects taken independently.

Although the agency is moving toward a more risk informed approach to regulation, requirements for risk information to support even singular license actions are minimal. For multiple license actions, say for plant life extension and power increase, or for higher fuel burnup with reduced inspection requirements, there is no regulatory requirement for an integrated risk assessment of the compounding/synergistic impact of multiple licensing actions. A brief examination of the License Renewal (LR) Rule further illustrates the point.

As discussed by Bonaca [3], the License Renewal (LR) Rule rests on the regulatory principle that a nuclear plant can continue to operate for as long as it complies with its current licensing basis (CLB) and satisfies the current body of regulations. This approach stems from the regulatory framework that compliance with current regulations provides assurance of adequate protection. Although the license renewal process allows for risk considerations to enter into the review, the use of risk information is an option largely left to the applicant. There is no explicit regulatory requirement to assess potential reductions in "margins" from those specified in the original FSAR, only that design margins not be exceeded. Nevertheless, extending plant life beyond 40 years involves the implicit recognition that excess margin is used (reduced) to compensate for age degradation during the allowable extension to 60 years. For example, at the end of 60 years mechanical components can be expected to be closer to their fatigue limits than at the end of the original 40 year license. The reactor vessel will be more brittle and closer to the pressurized thermal shock limit at 60 years than at 40 years, and so on.

To examine the concern of potential margin reductions for power uprates and plant life extension, a case study was made for the Hatch-BWR plant, owing to the fact that this plant has received approval for two power uprates (5-%, 8-%) and is currently under review for license renewal. An assessment was made of the impact of changes in plant operating conditions, component fatigue estimates, estimates of component loads (stress) under design basis accident (DBA) conditions, etc, for increased power and plant life extension.

For the primary system pressure boundary an increase in power generally leads to some increase in primary system pressure or temperature, and thus some reduction in "margin" to the prescribed design temperature/pressure limits specified in the ASME Boiler and Pressure Vessel Code. The same trend is noted for the feedwater piping, due to increased temperature at uprated conditions, as is evident from inspection of Table A.

The most notable changes in "margin" for power uprates appears to be associated with predictions for design basis accidents (DBA). Table B shows DBA-LOCA stress predictions for various Hatch-1 reactor vessel components. For bolting to the Hatch-1 vessel access cover plate, Table B indicates stress loads of 64.5 ksi (ksi = kilo-pound force per square inch) at the first power uprate, with an associated residual margin of 40.1-% compared to the design limit of 107.7 ksi. The residual margin is reduced to 16.4-% at the second uprate, owing to higher predicted blowdown stresses (90 ksi) at the second uprate. This reduction in stress margin is quite dramatic, when compared to the 8-% power increase between these two uprates. In other cases, the change in margin with a power is less dramatic, which largely stems from the fact that the parameter in question (e.g. stress, temperature, pressure,) remains well below the design limit.

With regards to plant life extension, margin trends were estimated for several passive components for which time-limited aging analysis (TLAA) was performed for the Hatch license renewal application. TLAA estimates for primary and feedwater piping largely center on estimates of the cumulative usage factor (CUF) for cyclic loadings on such piping during the period of extended operation. For Hatch-1 the CHU equals 0.56 for feedwater piping at 40 years, which increases to CHU=0.72 at 60 years (allowable limit, CHU=1.0). Similar CHU trends are indicated for primary system components and piping. For some passive components the residual margin at the end of the 60-year extension period is quite minimal. For example, CHU is estimated to be 0.95 for the torus suppression-pool at the end of 60-years, indicating only 5-% residual CHU margin at the end of the license renewal period. Such estimates indicate evidence of margin reduction associated with plant life extension.

The most frustrating aspect of this examination was the difficulty in obtaining self-consistent information with regards design parameters and associated margins. For example, although a re-examination of DBA-LOCA stress loads is stipulated for power uprates under 10CFR50.92 requirements, often the component stress predictions summarized in the licensee's Uprate Safety Analysis Report (SAR) are for different components then analyzed in the original FSAR; or if for the same component not at the same location. Likewise, thermal/mechanical models to predict the DBA-LOCA stress loads at the different power levels may involve changes in analysis methods, or alterations of thermal-hydraulic characteristics due to fuel designs changes that may accompany the power uprate For these reasons it was difficult to make a one-to-one comparison of margin, nevertheless some success at a quatitative evaluation of "margin reduction" was achieved.

References:

1. A. W. Cronenberg, *Potential Synergistic Safety Issues Related to Reactor Power Uprates,* Proc. Am. Nucl. Soc., San Diego, Ca (June 2000).

2. M. Khatib-Rahbar, E. G. Cazzoli, and A. Kuritzky, *An assessment of the Risk-Impact of Reactor Power Upgrade for a BWR-6 MMARK-III Plant,* Proc. of PSAM-3 Meeting, Crete, Greece, (1997).

3. M. Bonaca, *Potential Synergistic Effects of Industry Initiatives to Extend Plant Life, Increase Production, and Reduce Regulatory Burden*, ACRS Internal Memo, (April, 2000).

Table A. Hatch-1 Operational Margins

Residual Margin = Design Limit - Value / Design Limit		
Power Level, MWt	Parameter Value	Residual Margin, %
Main Steam-line Pressure (Design Limit = 1250 psig)		
Original = 2436	1015 psig	18.8
1st Uprate = 2558	1050 psig	16
2nd Uprate = 2763	1050 psig	16
Main Steam-line Temperature (Design Limit = 575 F)		
Original = 2436	546 F	5.04
1st Uprate = 2558	---	---
2nd Uprate = 2763	551 F	4.17

Table B. Hatch-1 Reactor Vessel Margins for DBA-LOCA Conditions

Residual Margin = Design Limit - Value / Design Limit		
Power Level, MWt	**Predicted Stress, ksi**	**Residual Margin, %**
Vessel Shroud at Support Weld (Design Limit = 15.28 ksi)		
Original = 2436	8.95 ksi	41.4
1st Uprate = 2558	9.05 ksi	40.8
2nd Uprate = 2763	---	---
Vessel Shroud at Head Bolts (Design limit = 69.9 ksi)		
Original = 2436	52.7 ksi	24.6
1st Uprate = 2558	53.0 ksi	24.2
2nd Uprate = 2763	---	---
Vessel Access Hole Cover Plate at Bolts (Design Limit = 107.7 ksi)		
Original = 2436	---	---
1st Uprate = 2558	64.5 ksi	40.1
2nd Uprate = 2763	90.0 ksi	16.4
Jet pump at Diffuser Base (Design limit = 50.7 ksi)		
Original = 2436	31.5 ksi	37.9
1st Uprate = 2558	34.8 ksi	31.4
2nd Uprate = 2763	34.9 ksi	31.2

Review of Power Uprate Applications and Potential Synergistic Safety Issues

August W. Cronenberg
ACRS Fellow

Outline

- Power Uprate Overview
  Applications, NRC Review Procedures,
  Altered Plant Conditions
- Events Noted for Uprated Plants
- Potential Synergistic Safety Issues
- Observations & Recommendations

Summary of Power Uprate Applications

Plant	Reactor Type	Year Startup	Original Power(MWt)	Year Power Uprate	Uprated Power(MWt)	% Power Increase
Oyster Creek	BWR	1969	1690	1971	1930	14.20
Calvert Cliffs-1	PWR	1977	2560	1977	2700	5.46
Main Yankee	PWR	1972	2440	1978	2630	7.79
Millstone-2	PWR	1975	2560	1979	2700	5.47
Fort Calhoun	PWR	1973	1420	1980	1500	5.63
St. Lucie-1	PWR	1976	2560	1981	2700	5.46
Cook-2	PWR	1976	3391	1983	3411	0.58
Duane Arnold	BWR	1975	1593	1985	1658	4.08
St. Lucie-2	PWR	1983	2560	1985	2700	5.47
Salem-1	PWR	1977	3338	1986	3411	2.19
North Anna	PWR	1976	2775	1986	2893	4.25
Callaway	PWR	1985	3411	1988	3565	4.51
Main Yankee*	PWR	1972	2440	1989	2700	10.65
Indian Point-2	PWR	1974	2758	1990	3071	11.35
Fermi-2	BWR	1987	3293	1992	3458	5.01
Wolf Creek	PWR	1985	3411	1993	3565	4.51
Vogel 1&2	PWR	1987	3411	1993	3565	4.51
Peach Bottom-2	BWR	1974	3293	1994	3458	5.01
Susquehanna 1&2	BWR	1983	3293	1994	3441	4.49
Surry 1&2	PWR	1972	2441	1995	2546	4.30
Nine Mile-2	BWR	1988	3323	1995	3467	4.33
Hatch 1&2	BWR	1975	2436	1995	2558	5.00
Limerick-2	BWR	1988	3293	1995	3458	5.01
Limerick-1	BWR	1985	3293	1996	3458	5.01

* Denotes second power uprate, percent power uprate based on original power level.

Slide 3

Licensee Uprate Responsibilities

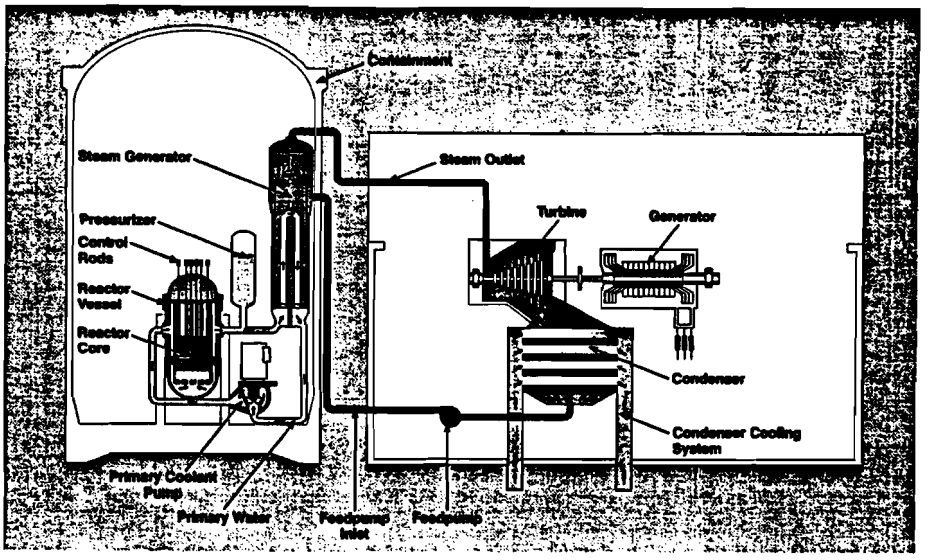
- Uprate Application takes form of License Amendment Request
- Application Must Include Power Uprate Safety Analysis Report (SAR)
- SAR Generally Follows GE-BWR Generic Guidance (Series of Licensing Topical Reports through-1996) or W-PWR Generic Review Plan (WCAP-10263, 1983)

  AR Centers on Re-evaluation of Design Basis cidents, No Significant Hazards Assessment, nges to Plant Conditions and Technical ifications

Slide 2

Slide 4

Plant Characteristics Impacted by Power Uprates



Primary Coolant System (PCS)
Core Power
Core Inlet/Outlet Enthalpy
Vessel Outlet/Inlet Coolant Temperatures
Fuel Temperature
Primary Coolant Flow Rate

Secondary Coolant System (SCS)
Steam Generator Steam Flow Rate
Feedwater Flow Rate
Feedwater Temperature
Feedwater Pumping Requirements

NRC Uprate Responsibilies

- NRC staff reviews licensee's uprate SAR
- NRC reports findings in Safety Evaluation Report (SER)
- NRC review centers on an evaluation of the impact of power increase on plant operations and safety
- No agency uprate Standard Review Plan or standardized acceptance criteria
- No agency requirements for independent code analysis of plant uprate conditions

Power Uprate Events

- Maine Yankee: Deliberate/Faulty LOCA Analysis by Licensee
- Wolf Creek/North Anna: Control Rod Insertion Problems
- Callaway/Susquehanna: Guillotine Pipe Ruptures
- Brunswick: Faulty use of DBA Criteria
- Limerick: Predicted Δ-K/K Less Than Measured

Maine Yankee

- Allegations of deliberate/ faulty LOCA submittal, DBA clad limit of 2200°F exceeded for uprate conditions.
- LOCA analysis performed with altered decay heat & critical flow models.
- NRC-SER did not question licensee analysis. No independent analysis performed by NRC staff.
- Subsequent investigations concurred with allegations.

LESSON: Need for independent NRC staff analysis (code calculations) for uprate reviews.

Wolf Creek / North Anna

Wolf Creek: 4.5-% Uprate 1996
Control rod insertion problems for 5 high burnup rods (47 GWD).

North Anna: 4.3-% Uprate 1986
2 control rod assemblies fail to fully insert in high burnup (47-49 GWD) assemblies in spent fuel pool.

NRC-SER: Neither Uprate SER addressed potential changes in fuel or control rod performance for high-burnup/high-power conditions.

LESSON: Need for NRC staff review of potential synergistic high-burnup/high-power effects.

Pipe Ruptures

- Susquehanna, Callaway, Tsuruga, etc
  (53 rupture events pipe Diam. > 2")
- Erosion (flow)/Corrosion (age) mechanism is major cause of pipe ruptures
- Empirical evidence for synergistic flow/aging effects
- LESSON: Need for NRC staff review of potential synergistic flow/aging effects

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Incorporating NUCLEONICS

NUCLEONICS WEEK

Vol. 40 No. 51 December 23, 1999

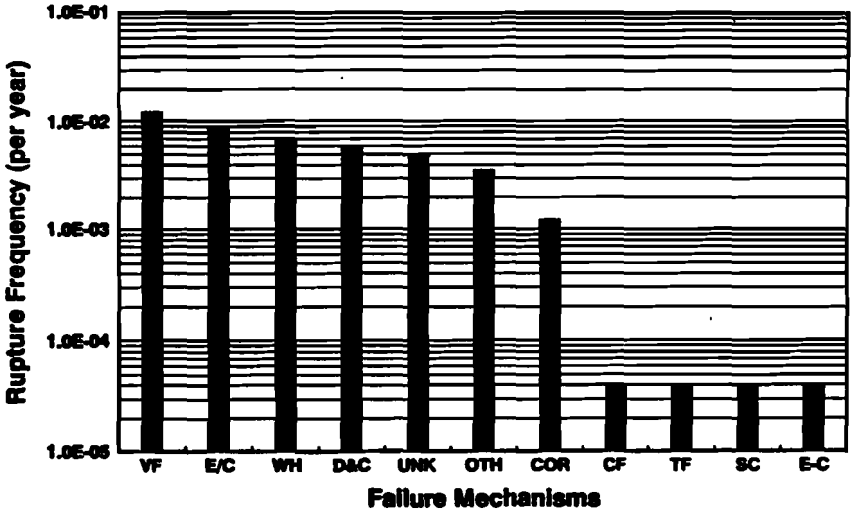
SUSQUEHANNA SHUTDOWN MAY SIGNAL NEW GENERIC PROBLEM FOR BWRs

" A recent Susquehanna-2 forced outage could be the result of weld fatigue from increased vibrations from a power uprate in 1995, and NRC is looking at potential generic implications for other uprated BWRs..."

" **BWR uprates have increased the speed** of recirculation pumps and caused increased vibrations in the recirculation systems, said Sam Hansell, NRC resident inspector at Susquehanna."

Slide 11

Frequencies of Pipe Ruptures Mechanisms Through 1995



VF	Vibrational Fatigue	COR	Corrosion Attack
E/C	Erosion/Corrosion	CF	Corrosion Fatigue
WH	Water Hammer	TF	Thermal Fatigue
D&C	Design & Construction	SC	Stress Corrosion Cracking
UNR	Unreported Cause	E-C	Erosion -Cavitation
OTH	Others (human error, frozen pipes, etc.)		

Slide 12

Other Events

Brunswick 1 & 2 (BWR)

- Licensee DBA Analysis based on wet-well design limit of 220°F.
- NRC did not challenge 220°F analysis.
- Correct value of 200°F (Licensee).

Limerick-1 (BWR)

- Code predicted $\Delta K/K$ for restart less than measured for high-power/high-burnup core.
- Code inadequacies for high-power/high-burnups.

LESSON: Need for agency uprate SRP and assessment of adequacy of core physics codes for high-power/high-burnup/high enrichments.

Potential Synergistic Effects

High-power/High-Burnup Synergisms

Rod Fretting: Flow induced rod vibration leading to contact wear with adjacent structures. Increased core flow at uprated power and Zry-irradiation growth for high burnup may lead to fretting.

Axial Power Offset: Boron added to compensate for excess reactivity of high-enrichment/high-burnup rods. Crud buildup for long fuel duty times. Boron gettered by crud, leads to axial power offset. Effect is compounded at high-power core locations.

Potential Synergistic Effects

High-Power/Ageing Synergisms

Electrical Components: Insulation breakdown due to irradiation effects, exacerbated by elevated temperatures.

Fluid -Mechanical Components (pumps, piping, valves): Subject to flow assisted corrosion (FAC). Elastomer seal degradation at high temperatures.

Instrumentation & Control Systems: Ageing of I&C exacerbated at elevated temperatures and fluid velocities (vibration).

Observations & Recommendations

- Current uprate application/ review process largely centers on re-evaluation of DBA conditions at uprated power. Nil consideration of potential synergistic effects.
- Events show indirect evidence for synergistic power-uprate/ageing and power-uprate/burnup effects.
- NRC Standard Review Plan (SRP) for power uprates should be issued (in progress). SRP should include acceptance criteria for synergistic effects.
- NRC uprate reviews should include staff T-H and core physics code calculations to verify licensee analysis.
- NRC uprate reviews should include comparison of safety measures (CDF, QHO, LERF) for prior & uprate power.

R

GE Nuclear Energy

ACRS Thermal Hydraulic Phenomena Subcommittee Meeting

GE BWR Power Uprate Program

Open Session

Jim Klapproth
Gene Eckert
Israel Nir

June 12, 2001

Outline

- **Opening Remarks (Open Session)**
- **Introduction**
  - **Past EPU briefing to ACRS**
  - **Key elements of GE power uprate program**
  - **BWR PU Implementation status and grid MWe time line**
- *GE Extended Power Uprate Program (Closed Session)*
  - *Background*
  - *Constant Pressure Power Uprate*
  - *EPU Specific Topics*
- *Concluding Remarks*

Opening Remarks

- **Extensive experience with EPUs**
  - **EPUs implemented at 4 BWRs**
  - **NRC review on-going for 5 additional plants**
- **Plant safety margins maintained**
- **Constant pressure power uprate approach reduces EPU impacts**
- **High volume of EPU review requests anticipated**
  - **Two additional submittals in 2001**
  - **Expect 4 BWR submittals per year**
- **Streamlined approach proposed by GE**

Past EPU briefing to ACRS

- June 2-3, 1998
- Described EPU approach developed from successful 5% Stretch Power Uprate (SPU) program
- No change to Licensing Basis of plants
- NSSS expected to be capable of EPU, even with potential operating pressure increase
- Feasibility phase established BOP changes
- Operation constrained to MELLLA boundary
- Combination of Generic + Plant-specific submittals

Past EPU briefing to ACRS

- Discussed contents of ELTR1 (Guidelines) and ELTR2 (Generic Evaluations)
- Discussion was coordinated with Monticello EPU submittal (106.3% OLTP)
- Same approach used shortly afterward by Hatch 1, 2
- Responded to ACRS Questions

Concluded that EPU Approach Effective and Acceptable

GE PU Program Key Elements

- **Power Uprate licensing**
  - **Stretch Power Uprate (SPU)**
    - Up to 105% of original licensed thermal power (OLTP)
  - **Extended Power Uprate (EPU)**
    - Up to 120% OLTP
  - **Thermal Power Optimization (TPO)**
    - Improved feedwater flow measurement uncertainty
  - **Constant Pressure Power Uprate (CPPU)**
    - Up to 120% OLTP, no dome pressure increase
- **Power Uprate Implementation**
  - **On-line PU Implementation**

Stretch Power Uprate

- **Initial programmatic effort focused on design basis stretch**
- **Licensing Topical Report No. 1 (1991) - Guidelines**
  - **Establishes approach and basis for BWR power uprate**
  - **Defines scope and format of plant-specific submittals**
- **Licensing Topical Report No. 2 (1991) - Evaluations**
  - **Documents generic Safety Analysis for some aspects**
  - **Avoids redoing complete FSAR**
  - **Licensing based on plant-specific evaluation referencing LTR 1/2**

Extended Power Uprate

- **Follow-on program focused on significant uprates**
  - **Extended power uprate: 105-120% for GE BWR products**
- **Licensing Approach emulates SPU program**
  - **Generic licensing methodology with some generic dispositions**
  - **Address more detailed issues (e.g., expansion of operating domain)**
  - **Maintain existing plant design basis**
- **Extended LTR (ELTR-1) submitted** **1Q 1995**
- **ELTR-2 submitted** **1Q 1996**
- **NRC approved ELTR-1/2 in 1998**
- **Hatch and Monticello applications approved in 1998**

Thermal Power Optimization

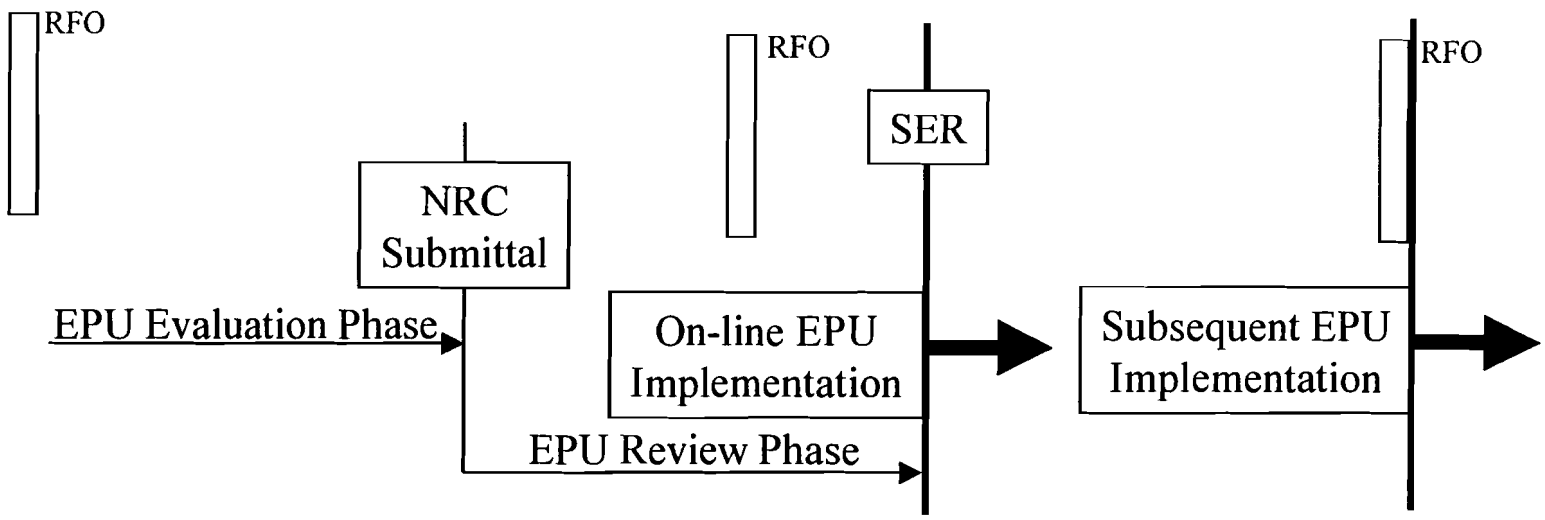
- NRC rulemaking changed requirement in 10CFR50 App K
  - Based on reduction of reactor thermal power uncertainty
- GE approach based on NRC approved uprate process
  - LTR-1, LTR-2 (SPU), ELTR-1, ELTR-2 (EPU)
- Significant simplifications for < 2% thermal power increase
- Maintain LTR/ELTR process for rigor and completeness
- Generic dispositions and evaluations facilitate NRC review and plant-specific implementation
  - Captured in TPO LTR (TLTR)
  - TLTR submitted for NRC review in August 2000
- Expect 3 plant specific submittals in 1Q02

Constant Pressure Power Uprate

- **Facilitates BWR power uprate applications**
  - **Introduces simplifications and focus process**
  - **Experienced based**
  - **Utilizes established proven processes**
  - **Provides assurance that NRC review continues to address all power uprate safety aspects**
  - **Separates effects of other, non-power uprate licensing changes**
- **CPPU LTR submitted for NRC review in 3/01**
- **Efficient power uprate process**
  - **Consistent with NRC Staff Requirements Memorandum regarding Power Uprate Applications**

On-Line PU Implementation

- License and implement power uprate without increase in reactor dome pressure
- Permits initial power increase to existing BOP capacity immediately following receipt of OL amendment
- Subsequent BOP mods to achieve full uprate

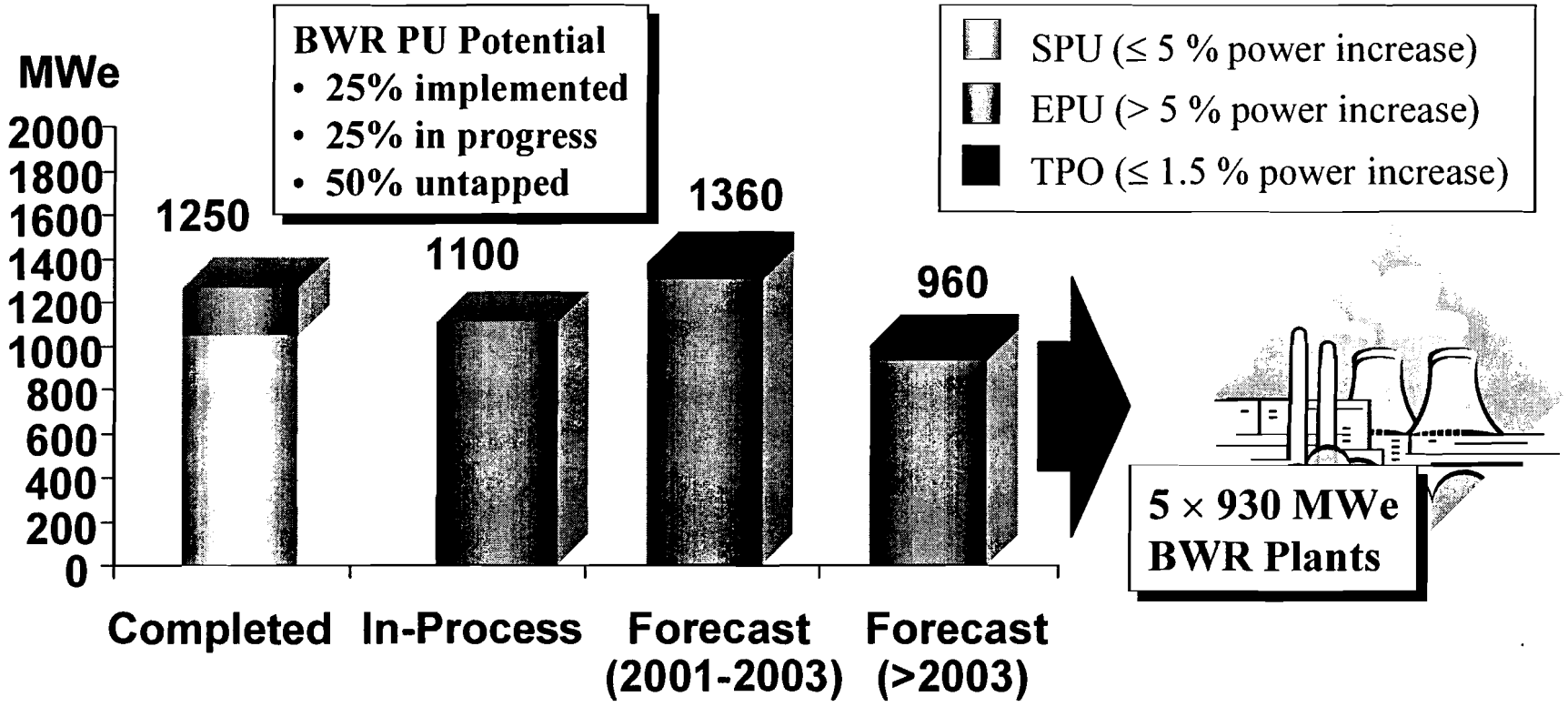


GE PU Experience/Status

Plant	PU (~% OLTP)	Dome Pr Increased	Plant	PU (~% OLTP)	Dome Pr Increased
Duane Arnold	105	Yes	Laguna Verde - 1, 2	105	No
Cofrentes	105	Yes	LaSalle - 1, 2	105	No
Hatch - 1, 2	105	Yes	Perry	105	No
Susquehanna - 1, 2	105	Yes	Hatch - 1, 2	113	No
WNP-2	105	Yes	Monticello	106	No
Limerick - 1, 2	105	Yes	Cofrentes *	110	No
Peach Bottom - 2, 3	105	Yes	Duane Arnold *	120	No
Fermi 2	105	Yes	Dresden - 2, 3 *	117	No
FitzPatrick	105	Yes	Quad Cities - 1, 2 *	117	No
Brunswick - 1, 2	105	Yes	Clinton *	120	No
NMP-2	105	Yes	Brunswick - 1, 2 *	120	No
Browns Ferry - 2, 3	105	Yes	Browns Ferry - 2, 3*	120	No
KKM	114	Yes			
KKL	120	Yes			
River Bend	105	Yes			

* In progress

GE BWR PU MWe Time Line



Supports NRC Staff Requirements memo on additional PU

High power uprate volume for NRC review is anticipated