March 24, 2000

LICENSEES: Vermont Yankee Nuclear Power Corporation, North Atlantic Energy Service Corporation, Entergy Nuclear Generation Company, and Northeast Nuclear Energy Company

FACILITIES: Vermont Yankee Nuclear Power Station (Vermont Yankee); Seabrook Station, Unit No. 1 (Seabrook); Pilgrim Nuclear Power Station (Pilgrim); and Millstone Nuclear Power Station, Unit Nos. 2 and 3 (Millstone 2 & 3)

SUBJECT: SUMMARY OF MEETING ON MARCH 1 AND 2, 2000, REGARDING LICENSING WORKSHOP (TAC NO. MA6891)

On March 1 and 2, 2000, representatives of the licensees' staffs for Millstone 2 & 3, Pilgrim, Seabrook, and Vermont Yankee met with members of the Nuclear Regulatory Commission (NRC) staff to conduct a licensing workshop with the goals of improving the quality of licensing submittals and improving the licensing interface between licensees and the NRC.

Major topics of discussion included project manager responsibilities, licensing department responsibilities, 10 CFR 50.59, 10 CFR 50.90, qualities of a good licensing submittal, enforcement discretion, commitment management, and inservice testing/inspection. Enclosure 1 is a list of attendees and Enclosure 2 contains copies of handouts distributed at the meeting.

/RA/

Richard P. Croteau, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-271, 50-293, 50-443,
50-336, and 50-423

Enclosures: 1. List of Attendees
2. Handouts

cc w/encls: See next page

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DATE	3/21/00		3/21/00		3/21/00			

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

March 24, 2000

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Docket Nos. 50-271, 50-293, 50-443,
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Licensing Workshop Meeting

March 1-2, 2000

List of Attendees

NAME	ORGANIZATION
Len Gucwa	Vermont Yankee
Doug Pickett	NRC
Jeff Rogers	Pilgrim
Paul Russell	Northeast Utilities
Jim Emly	First Energy
Paul Willoughby	NNECO
Renee LecLerc	NAESCO
Walter Lobo	Entergy/Pilgrim
James Haley	Entergy/Pilgrim
James Connolly	North Atlantic/Seabrook
Peter Anastes	NNECO
Mike Schoppman	NEI
Steve Gatcomb	NNECO
Tracy Clark	NRC
Jim Keyes	Pilgrim
Jake Zimmerman	NRC
Bob Pulsifer	NRC
Victor Nerses	NRC
Rick Croteau	NRC
Jeff Meyer	Vermont Yankee
Clare Hsieh	Seabrook
Jim Clifford	NRC
Jeff Sobotka	Seabrook
Tim Pucko	Seabrook
Joe Malone	Seabrook
Jim DeVincentis	Vermont Yankee
Walt Anson	Vermont Yankee
Bob Leach	Millstone
Stephen Brennion	Entergy/Pilgrim
Gautam Sen	Vermont Yankee
Jack Alexander	Entergy/Pilgrim
Michael Ossing	Seabrook
Bill Sherman	Vermont Yankee
George Hess	ABB C-E Nuclear Power
Mike Sortwell	Vermont Yankee
Alan Wang	NRC

NRC/New England Licensing Workshop

March 1-2, 2000

Jointly Sponsored By:

Nuclear Regulatory Commission
and Vermont Yankee

Quality Inn and Suites
Brattleboro, Vermont

Enclosure 2

Agenda for NRC/New England Licensing Workshop
March 1, 2000

Time	Subject	Leader
8:00-8:20	Opening	Rick Croteau Jeff Meyer
8:20-8:30	Welcome	Bob Wanczyk (Vermont Yankee)
8:30-8:50	Introduction	Jim Clifford
8:50-9:00	NEI	Mike Schoppman (NEI)
9:00-10:00	PM Responsibilities	Jake Zimmerman
10:00-10:15	Break	
10:15-11:15	Licensing Department Responsibilities	Jeff Sobotka (Seabrook)
11:15-12:00	10CFR50.59/50.90	Robert Pulsifer
12:00-1:00	Lunch	
1:00-1:30	Qualities of a Good Licensing Submittal (NRC Perspective)	Vic Nerses Alan Wang
1:30-2:00	Qualities of a Good Licensing Submittal (Licensee Perspective)	Jeff Meyer (Vermont Yankee)
2:00-4:00	Breakout Session: Qualities of a Good Licensing Submittal	
4:00-4:15	Break	
4:15-4:45	Feedback from Breakout Session	
4:45	Day 1 Closing Remarks	Rick Croteau

Agenda for NRC/New England Licensing Workshop
March 2, 2000

Time	**Subject**	**Leader**
8:00-8:15	Opening	Rick Croteau Jeff Meyer
8:15-8:30	Introduction to Breakout Session	Vic Nerses Alan Wang
8:30-9:45	Breakout Session: Critique of Actual Submittal	
9:45-10:00	Break	
10:00-10:30	Feedback From Breakout Session	
10:30-11:30	NOEDs	Rick Croteau
11:30-12:30	Lunch	
12:30-1:30	Commitment Management	Paul Willoughby (Millstone)
1:30-2:30	ISI/IST/BWRVIP Topics	Pilgrim
2:30-2:45	Break	
2:45-4:30	Open Discussion on Licensing Issues (ADAMs, Electronic submittals, etc.)	Jim Clifford/ Licensees
4:30-4:40	Workshop Feedback	Rick Croteau
4:40	Closing Remarks	Jim Clifford

FEEDBACK

LICENSING WORKSHOP
Quality Inn
Brattleboro, Vermont
March 1 - 2, 2000

On a scale of 1 to 10, please provide an *overall* rating for workshop/ materials and effectiveness_____.

Excellent	**Very Good**	**Good**	**Fair**	**Unsatisfactory**
10--------9-	--8-------7-	--6-------5--	--4-------3-	----2------1--

A. PLEASE RATE THE WORKSHOP/MATERIALS, USING A SCALE OF 1 TO 10, AS TO:

1. **Accomplishment of objectives** _____
2. **Coverage of subject matter** _____
3. **Organization of subject matter** _____
4. **Suitability of instructional materials** _____

***Overall* rating for the workshop/materials** _____

B. PLEASE RATE THE PRESENTERS/FACILITATORS ON THE FOLLOWING ITEMS, USING A SCALE OF 1 TO 10.

1. **Effectiveness of presentations** _____
2. **Presenter/Facilitator's ability to answer questions** _____
3. **Presenter/Facilitator's effectiveness in keeping discussions focused on relevant topics** _____
4. **Presenter/Facilitator's courtesy and tact** _____

***Overall* rating of the presenters/facilitators** _____ _____

C. YOUR KNOWLEDGE AND SKILL LEVEL OF THE SUBJECT MATTER

1. Before taking the workshop

NONE HIGH

_____1_____2_____3_____4_____5_____6_____7_____8_____9_____10

2. After taking the workshop

NONE HIGH

_____1_____2_____3_____4_____5_____6_____7_____8_____9_____10

3. How well will you be able to use what you learned?

A great deal____ Mostly____Somewhat____Minimally_____Not at all____

D. OTHER

1. WHAT DID YOU PARTICULARLY LIKE ABOUT THE WORKSHOP?

_____

_____

_____

_____

2. WHAT WERE THE WORKSHOP'S STRENGTHS?

_____

_____

_____

_____

3. WHAT WERE THE WORKSHOP'S WEAKNESSES?

_____

_____

_____

4. DO ANY PARTS OF THE WORKSHOP NEED IMPROVEMENT?

_____

_____

_____

_____

_____

5. HOW WILL YOU USE WHAT YOU'VE LEARNED AT THE WORKSHOP?

_____

_____

_____

_____

_____

_____

_____

6. AT WHAT FREQUENCY DO YOU THINK THESE WORKSHOPS SHOULD BE HELD?

_____

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

February 4, 2000
NVY 00-11

MEMORANDUM TO: James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Richard P. Croteau, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: FORTHCOMING MEETING WITH LICENSEES FOR MILLSTONE 2 & 3, PILGRIM, SEABROOK, AND VERMONT YANKEE

DATE & TIME: March 1 and 2, 2000, 8:00 a.m. - 5:00 p.m.

LOCATION: Quality Inn and Suites
1380 Putney Road
Brattleboro, Vermont 05301
(802) 254-8701

PURPOSE: The Nuclear Regulatory Commission (NRC) and Vermont Yankee Nuclear Power Corporation will jointly sponsor a licensing workshop with the goals of improving the quality of licensing submittals and improving the licensing interface between licensees and the NRC. A proposed agenda is attached.

PARTICIPANTS:* Participants from the NRC include members of the Office of Nuclear Reactor Regulation (NRR).

NRC	UTILITIES
J. Clifford, NRR, et al.	D. Smith, Millstone Stations S. Brennion, Pilgrim Station J. Sobotka, Seabrook Station G. Sen, Vermont Yankee

Docket Nos. 50-271, 50-336, 50-423, 50-293, and 50-443

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*Meetings between NRC technical staff and applicants or licensees are open for interested members of the public, petitioners, intervenors, or other parties to attend as observers pursuant to "Commission Policy Statement on Staff Meetings Open to the Public" 59 Federal Register 48340, 9/20/94.

1. Northeast Region Licensing Workshop
2. Licensing Workshop Objectives
   a. Enhance regulatory interface
   b. Promote understanding of entire licensing process
   c. Generate proposals for change
   d. Improve licensing submittal quality
   e. Improve safety evaluation quality
   f. Exchange information on current topics of interest
3. NRC Attendees
   a. Rick Croteau - Vermont Yankee
   b. Vic Nerses - Millstone 3
   c. Jake Zimmerman - Millstone 2
   d. Bob Pulsifer - Seabrook
   e. Alan Wang - Pilgrim
   f. Tracy Clark - Licensing Assistant
4. Goals of Improved Licensing Performance
   a. Budget and resource challenges
   b. Operating plan goals
   c. Efficiency and Effectiveness
   d. Faster response to licensee needs
   e. Need for more stable regulatory environment
5. Benefits of Improved Submittals
   a. SIMPLIFY -- Reduce extent and duration of interactions between reviewer and requester (reduce RAIs, supplemental submittals)
   b. MAXIMIZE -- NRR review assets (schedule control, labor rate, use of precedents)
   c. REDUCE -- Actions rejected or withdraw -- Cost
6. Preview of Closing Session Feedback Areas
   a. Was workshop effective in meeting objectives?
   b. What parameters can be used to assess licensing submittal quality?
   c. What lessons learned can you integrate into your routine licensing practices?
   d. Suggestions for improving communications at NRC-licensee interface?
   e. Need for follow-on workshops?

New England Licensing Workshop

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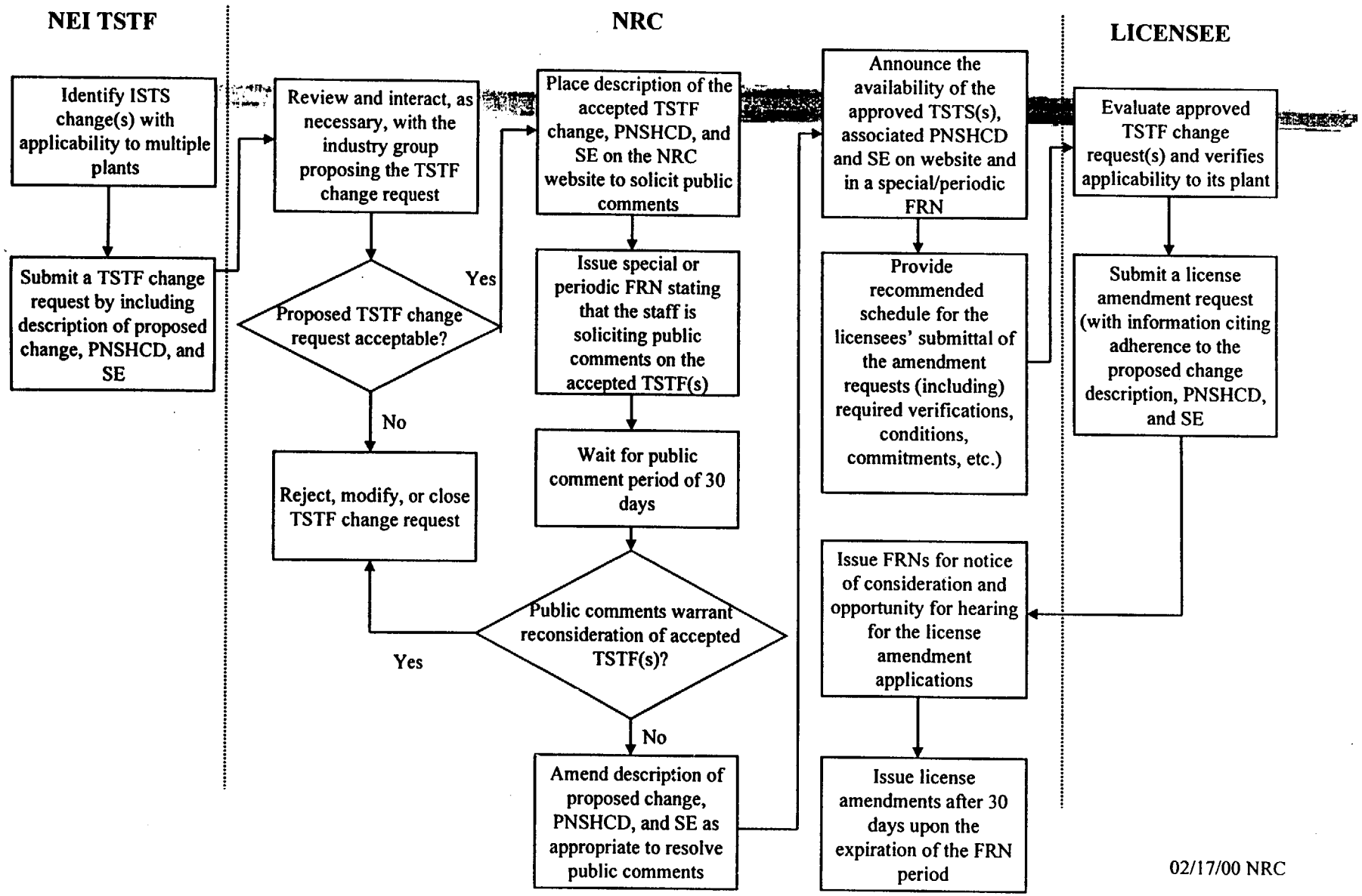
NEI LATF

- Licensing Action Task Force
  - Improve "licensing process"
  - Similar Task Forces at NRC & NEI
  - NEI LATF formed Nov. 1998
  - 8 utilities, B&WOG, BWROG, CEOG, WOG, ABB/CE, FTI, Westinghouse, NEI TSTF, EPRI, Winston & Strawn
  - 8 NRC/NEI meetings (Nov. 1998 - Feb. 2000)

LATF Action Items

- ⌘ Consolidated Line Item Improvement Process (CLIIP)
- ⌘ Bases/TRM Change Process
- ⌘ Unintended Tech Spec Actions (UTSA)
- ⌘ NRR Office Letter 803 (RAIs)
- ⌘ NRR Office Letter 1201 (TIAs)
- ⌘ Other process improvements (relief requests, topical report reviews, etc.)
- ⌘ Standardization of licensing submittals

Consolidated Line Item Improvement Process (CLIIP) - draft



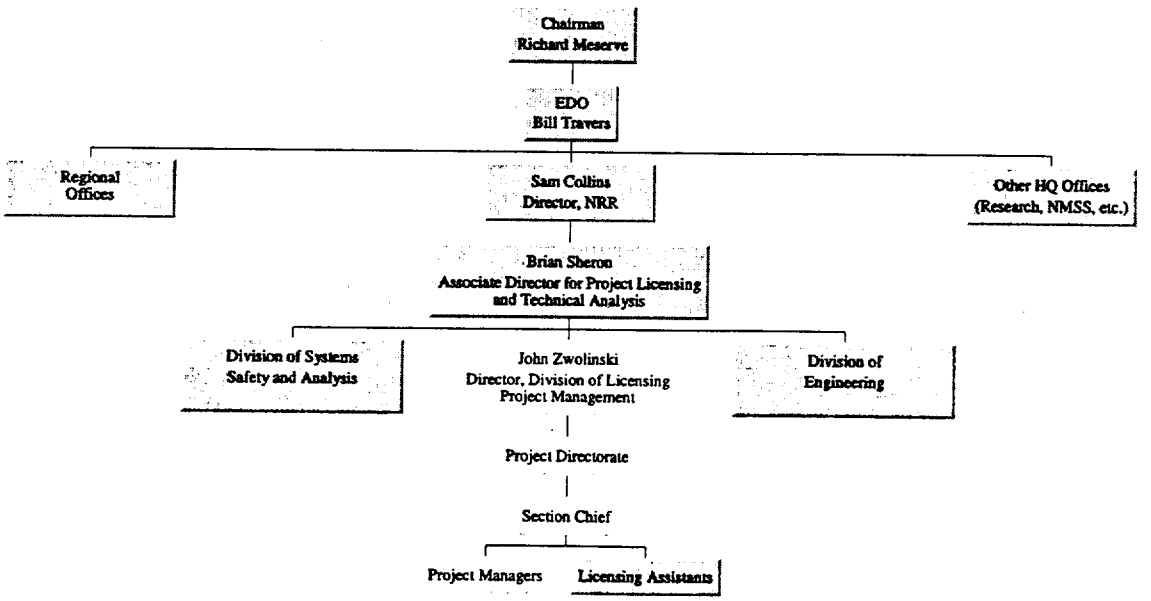
02/17/00 NRC

NRC PROJECT MANAGER RESPONSIBILITIES

NRC/New England Licensing Workshop
March 1-2, 2000

Jake Zimmerman
Division of Licensing Project Management

DLPM ORGANIZATION



BACKGROUND

- Generally, one Project Manager per site
- PM assignments are for a maximum of 5 years
- Educational background is typically engineering
- Experience is varied (nuclear industry, regional inspectors, other NRC offices)

EXPECTATIONS

- Most knowledgeable member of the staff regarding the licensing agenda for assigned facility
- Knowledgeable of plant design and operation
- Thorough understanding of NRC rules, processes and licensing requirements
- Focal Point for NRC/Licensee Correspondence
- Prioritize, Schedule, Review, Manage & Prepare all actions associated with the licensing process
- Maintain NRC information management systems

PERFORMANCE MEASURES

- Timeliness
- Effectiveness
- Efficiency
- Quality
- Quantity

STRATEGIC OUTCOME GOALS

- Maintain Safety
- Reduce Unneccessary Regulatory Burden
- Increase Public Confidence
- Increase Internal Efficiency & Effectiveness

DLPM IMPLEMENTING PLAN

- Licensing Authority
- Interfaces
- Regulatory Improvements
- Total of 75 Specific Tasks

LICENSING AUTHORITY

- Licensing Actions
- Mandated Controls
- Other Licensing Tasks

LICENSING ACTIONS

- License Amendments (TS & USQ)
- Exemptions
- Relief Requests
- License Transfers
- NOEDs
- Lead Plant Reviews

MANDATED CONTROLS

- TS Bases Changes
- UFSAR Reviews (10 CFR 50.71(e))
- Facility Changes (10 CFR 50.59)
- QA, Security, EP Program Reviews

OTHER LICENSING TASKS

- Pre-Application Reviews
- Task Interface Agreements
- 10 CFR 2.206 Petitions
- Plant-Specific Multi-Plant Actions
- Commitment Management
- Hearing Support
- Backfits
- Proprietary Information Reviews

INTERFACES

- Licensee/Owners' Groups
- NRC Headquarters
- Regional Offices
- Public

LICENSEE/OWNERS' GROUP ACTIVITIES

- Routine Communications with Licensee
- Site Visits/Drop-ins
- Lead PM on Technical Issues (MPAs, GSIs, USIs)

HEADQUARTERS

- Management Info. & Status Reports
- Incident Response
- Miscellaneous Licensee Reports
- Fee Billing Reviews
- Surveys
- General Support to other NRC Offices

REGIONS

- Morning Plant Status Calls
- Management Oversight Panels
- Routine Communications
- TS Interpretations
- Enforcement Support
- Event Followup

PUBLIC

- Controlled Correspondence
- Noticing Amendments, meetings
- Allegations
- FOIA requests
- Plant Information on NRC Web page

REGULATORY IMPROVEMENTS

- Licensing Action Task Force
- Owners' Group Interactions
- NRR Office Letters
- NRR Reinvention Effort
- Rulemaking (Risk Informing Part 50)
- Task Forces (ADAMS, Public, Y2K)
- Licensing Workshops

Responsibilities of a Utility's Licensing Organization

Responsibilities of a Licensing Organization

- Classic Licensing/Compliance
- NRC Inspection Coordination
- Generic Licensing

Classic Licensing/Compliance

Operating License/Regulation/Code Related Actions:

- License Amendment Requests/TS Changes/OL changes
- TS Bases changes
- NOEDs
- Exemption requests
- ASME Code Relief Requests
- Other Relief Requests

Classic Licensing/Compliance

Licensing Basis:

- FSAR maintenance/FSAR changes
- Technical Requirements changes
- QA Program changes per 50.54(a)
- EP Program changes per 50.54(q)
- Security Program changes per 50.54(p)
- Commitment management/Commitment changes

Classic Licensing/Compliance

Compliance/Inspection Submittals:

- Review/processing of all incoming correspondence
- Develop/Coordinate responses to Generic Letters
- Develop/Coordinate responses to Bulletins
- Develop/Coordinate routine submittals/reports
- Responses to RAIs
- Operator Licensing Submittals
- Responses to NOVs
- Responses to URIs
- Allegation responses

Classic Licensing/Compliance

Reportability:

- Prompt Report Input
- Reportability Determinations
- Develop LERs
- Develop Security Event Reports
- Part 21 Reports

Classic Licensing/Compliance

Guidance/Training/Other:

- Clarify TS and Regulatory Requirements
- Provide guidance/training on 50.59s
- Develop 50.59s
- Review of Operability Determinations
- Provide training for Regulatory issues
- Coordinate implementation of RROP
- Coordinate conference calls
- Other duties as assigned

NRC Inspection Coordination

- Interface with Resident Inspectors
- Interface with visiting inspectors
- Coordinate Regional/NRR inspections
- Coordinate NRC Executive visits and meetings
- Prepare staff for inspections
- Summarize exit meetings
- Analyze inspection reports
- Document/process commitments
- Coordinate NRC PI submittals

Generic Licensing

- Review Federal Register
  - Rule changes
  - Petitions for rulemaking
  - Requests for comment
  - Regulatory Guides
  - Licensing actions

Generic Licensing

- Document reviews
  - SECYs
  - SRMs
  - NUREGs
  - RISs
- Coordinate NEI interface

What Does Everyone Else Do?

Significant Differences from Seabrook Station's Program:

- Vermont Yankee
- Pilgrim
- Millstone

50.59/50.90

Licensing Workshop March 1 - 2, 2000

Brattleboro, Vermont

Bob Pulsifer, NRR PDI Section 2
Seabrook Project Manager

INTRODUCTION

- Goals
  - Develop Ideas for Improvement both at NRC and Utilities
  - Stimulate Discussion for Breakout Sessions
- Discuss Briefly the New 50.59 Process
  - Major Changes
  - Impacts and Benefits
- Discuss License Amendment 50.90 Change Process
  - Criteria
  - Content

10CFR 50.59

- Purpose

  - Used to Determine Whether Prior NRC Review and Approval is Necessary Before Licensee Makes:
    - Changes to facility as described in Final Safety Analysis Report (as updated) [UFSAR]
    - Changes in procedures as described in UFSAR
    - Test/experiments not described in UFSAR

  - When Prior NRC Approval is Required, Submit Application for Amendment per 10 CFR 50.90

SCHEDULE

- FINAL RULE ISSUED IN FR ON 10/4/99
- NEI SUBMITTED NEI 96-07, REV 1 IN DECEMBER 1999
- NRC REG GUIDE TO BE ISSUED IN LATE 2000
- IMPLEMENTATION IS 90 DAYS AFTER RG ISSUED

MAJOR CHANGES

- REMOVAL OF REFERENCE TO USQ
- TERM "SAFETY EVALUATION" CHANGED TO "10 CFR 50.59 EVALUATION"
- ADDED DEFINITION OF "CHANGE" AND "FACILITY AS DESCRIBED IN THE FINAL SAFETY ANALYSIS (AS UPDATED)"

MAJOR CHANGES (continued)

- WILL ALLOW FOR MINIMAL CHANGES, WITHOUT REQUIRING PRIOR NRC APPROVAL
- CHANGED "PROBABILITY" TO "INCREASE IN FREQUENCY" OR "LIKELIHOOD OF OCCURRENCE"
- MALFUNCTION OF A DIFFERENT TYPE IS BEING REPLACED WITH "MALFUNCTION WITH A DIFFERENT RESULT"

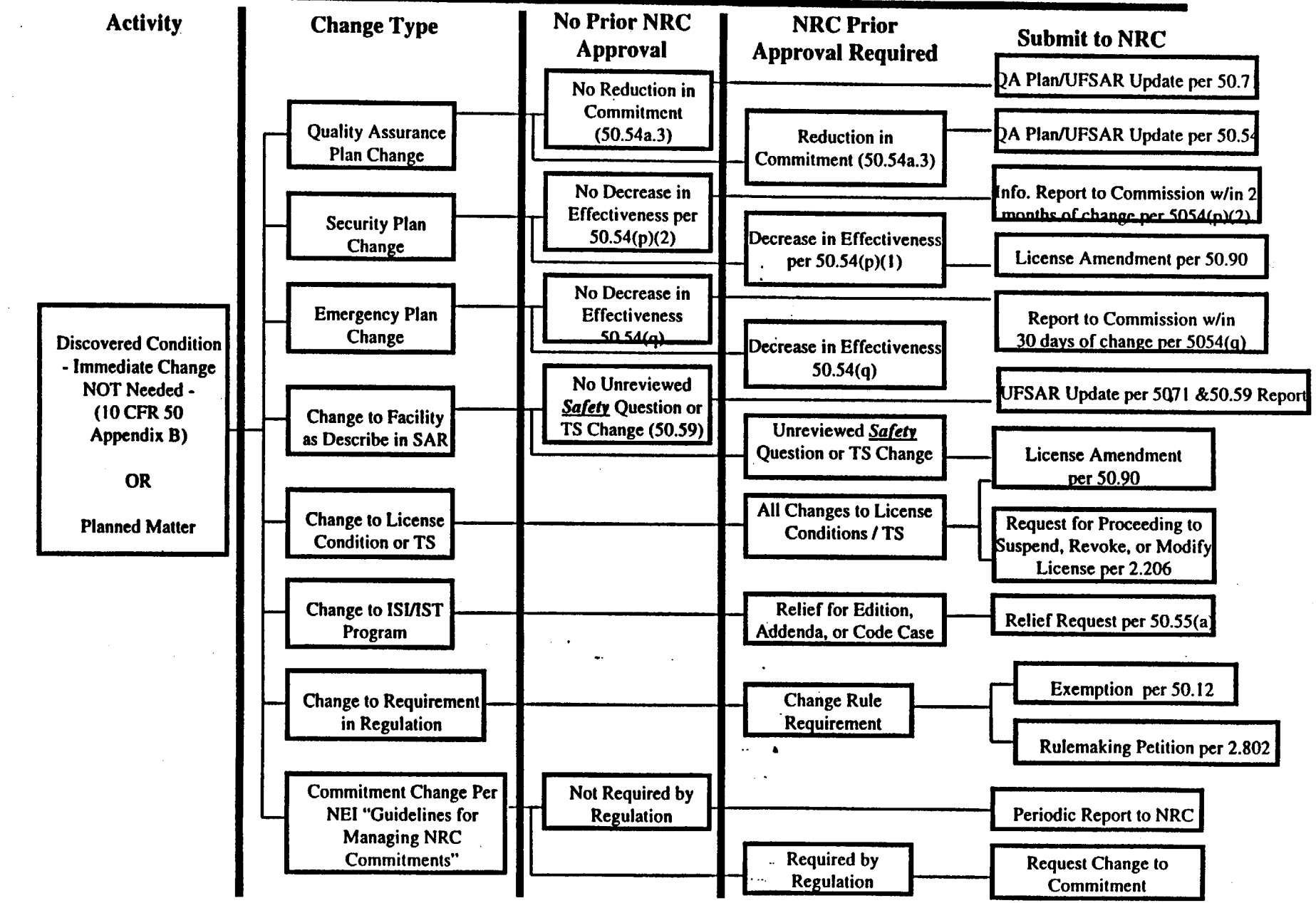
MAJOR CHANGES (continued)

- MARGIN OF SAFETY EVALUATION CRITERIA IS REPLACED WITH 2 NEW CRITERIA:
  - CRITERIA (vii) - EVALUATION OF INTEGRITY OF FISSION PRODUCT BARRIERS
  - CRITERIA (viii) - CHANGES TO APPROVED EVALUATION METHODS

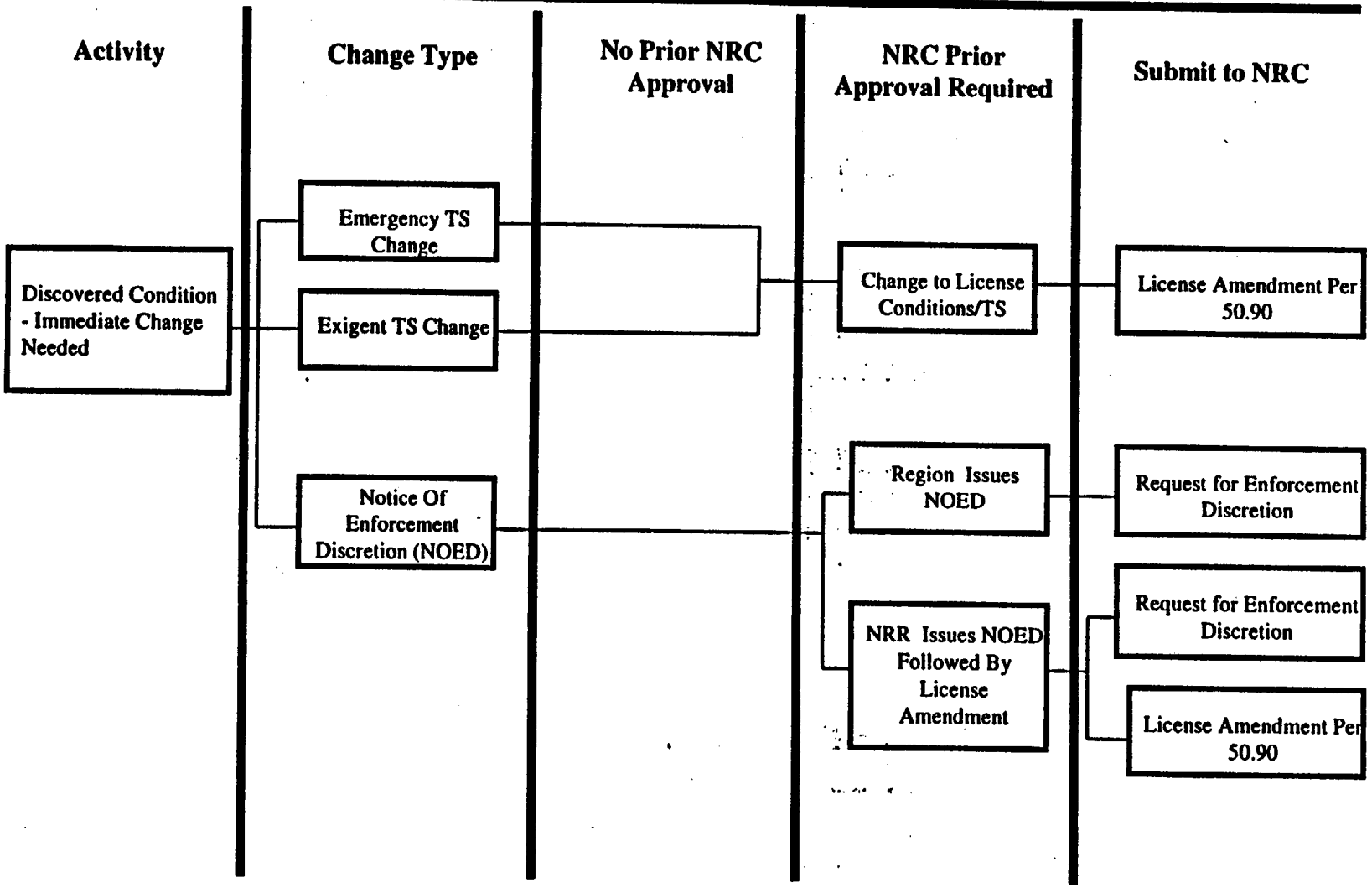
IMPACTS AND BENEFITS

- IMPACTS
  - WILL REQUIRE MAJOR REVISION TO 50.59 PROCEDURES
  - WILL REQUIRE NEW TRAINING STANDARDS TO BE DEVELOPED
- BENEFITS
  - OVERALL IMPROVEMENT OVER PREVIOUS RULE LANGUAGE
  - AGREED UPON INDUSTRY/NRC GUIDANCE

Regulatory Change Processes



Regulatory Change Processes



NRC GL 91-18: Operability

- Discovered Conditions Covered By This GL
  - Regulatory Changes Processes May Be Used As Corrective Actions
    - Subject to ALL normal Requirements and Restraints PLUS
    - Schedule Restraints of Appendix B, Section XVI
      - Promptly identify and correct
      - Timing commensurate with safety significance of issue

License Amendment - 10 CFR 50.90

- Criteria
  - Submit as specified in 10 CFR 50.4
  - Fully describe changes; form of original application
  - No significant hazards consideration [50.92(c)]
    - » No significant increase in probability or consequences of an accident previously evaluated
    - » No possibility of a new/different kind of accident from any previously evaluated
    - » No significant reduction in margin of safety

License Amendment - 10 CFR 50.90

- Content
  - Deterministic safety assessment
  - Optional - supported by risk-informed information
  - No significant hazards consideration
  - Environment input [especially for "irreversible consequences" - 10 CFR 50.92(b)]
    - » To support impact statement per 10 CFR 51.20
    - » To support environmental assessment per 10 CFR 51.21
    - » None if exclusion applies per 10 CFR 51.22(c)
  - Revised Technical Specifications or License Condition

License Amendment - 10 CFR 50.90

- Content (con't)
  - New or revised commitments identified
  - New or revised Design Basis (10 CFR 50.2) and Licensing Basis identified
  - Reference to current licensing basis
  - Reference previous pertinent SEs
  - Cost Beneficial Licensing Actions (NRC AL95-02)
    - » Total lifetime savings identified
  - Need date and basis identified
  - Implementation schedule provided

Emergency License Amendment

- Criteria
  - Must meet all License Amendment criteria from 50.91 and 50.92
  - Failure to act on request would result in
    - » Nuclear power plant shutdown
    - » Prevention of resumption of operation or increase in power up to licensed level
  - Issue without prior notice and opportunity for hearing or public comment ONLY if change would NOT involve significant hazards consideration

Emergency License Amendment: 50.91

- Content
  - License Amendment content PLUS
    - » Explanation of why emergency situation occurred
    - » Explanation of why situation could not be avoided
  - NRC publishes notice for opportunity for hearing and public comment after issuance per 2.106
- Timing
  - Amendment not issued as an emergency amendment if failure to be timely created the emergency

Exigent License Amendment

- Criteria:
  - Must meet all license amendment criteria from 50.91 and 50.92
  - Must act quickly
    - » Time does not permit 30 day public comment
    - » Does not involve significant hazards consideration
  - Issue after notice of an opportunity for hearing with at least 2 weeks for prior public comment or use local media to provide reasonable notice.
  - Provide an opportunity for hearing after issuance

Exigent License Amendment (cont.)

- Content
  - License amendment content PLUS
    - » Explanation of why exigency situation exists
    - » Explanation of why situation could not be avoided
  - Facts must match NOED request information (if NOED issued)
  - Will provide a hearing after issuance if appropriately requested

Exigent License Amendment (cont.)

- Timing
  - Amendment not issued as an exigent amendment if failure to be timely created the exigency
  - Request must be submitted within 48 hours if NOED issued
  - Amendment to be issued within 4 weeks of receipt of request

License Conditions

- License must match NRC authority file
  - Licensee should update its copy as amendments are issued
  - Several sites have noted discrepancies between authority file and their copy. Errors found in authority file and licensee copy.
  - Remove outdated license conditions through an amendment request

General Submittal Concepts

- Know the Specific Regulations Affected
- Use Flexibility Allowed by the Regulations
- Keep Staff Up-to-date With What You Need
- Submit Requests Early, Allowing Adequate Time for Staff Review
- Provide Complete, Well Written, Thorough, High Quality Submittals
- Be Prepared to Interact Promptly with the Staff
- Use and Reference Previously Approved SEs where Appropriate

Specific Guidance

- Keep PM Aware of What is Happening at Plant
- Discuss Requested Need With PM Before Submittal is Written
- Minimize Complexity of the Requests
- Provide Copies of Licensing Submittals to PM by Mail and Electronically
- Provide Future Licensing Needs to Staff Well Before Next Outage
- Plan Ahead!
  - Sholly Notice Period Expires No Earlier Than 6 to 8 Weeks after PM Receives Application

Quality of License Submittal

- Cover letter
  - Descriptive title
  - Summation of what, why and when
  - Description of proposed revision
  - Description of the regulatory processes for change
  - Risk-informed nature of the submittal, if applicable
  - Statement of proprietary info., if applicable
  - Cite applicable regulation if filed as exigent or emergency
  - Cite applicable licensee committee review and concurrence
  - Confirm State notification

Quality of License Submittal (contd.)

- Attachments
  - Safety Assessment
    - Introduction
      - Describe the change and identify affected TS sections
    - Background
      - System descriptions
      - Industry precedents/references, other lic. approvals
      - Refer. previous correspondence, mtgs., phone calls, etc
    - Safety Analysis
      - Be complete in the justification for change
      - Conformance to appl. stds.: R.Gs., SRPs, NEI docs.
      - Describe analytical methods, data and results
      - Discuss impact of change on accident analysis/risk
      - Address anticipated questions

Quality of License Submittal (contd.)

- Request for exigent/emergency approval
  - Provide detail justification for the request
    - Why the situation occurred
    - Why the situation could not be avoided
- No Significant Hazards Consideration
  - Stand alone, available for use in FR Notice
  - Answer each question fully, being understanable, concise, yet sufficiently detailed
  - Avoid excessive technical details
- Technical Specifications pages
  - Marked-up pages
  - Revised pages

Quality of License Submittal (cont'd)

- Environmental statement
  - Concise and separate discussion of environmental considerations
- 0ath and affirmation
  - Notary public statement
- Drawings
  - Consider submitting for clarification
- Commitments
  - Ident. new regulatory commitments, if any
  - Ident. any change to existing commitments, if any

- Miscellaneous
  - Supplements to original submittals need to stand alone
  - Bases only change submital

What Constitutes a Good License Submittal (Vermont Yankee Perspective)

Jeff Meyer
VY Licensing Dept.

Objectives:

- High success rate in approval of Licensing submittals necessary to support safe plant operation and company business goals.

- Clear, thorough, accurate and technically justified submittals that minimize NRC concerns, objections and Requests for Additional Information (RAIs).

- Submit to allow for timely review. Prioritize with Project Manager (PM) through periodic discussions.

VY Prioritization Scheme:

- Plant safety and/or operational impact.
- Outage improvement related.
- Cost/efficiency related (equal to above).
- Other submittals (limited benefit such as administrative, editorial, clarifications, etc).

Format/Content Issues:

Cover Letter-

- State if proprietary information is enclosed (10CFR2.790). Supply a whole version and one with deleted information.

- Environmental review statement/conclusion (10CFR51.22).

- Specify a requested approval date (as necessary).

- Typically request a 60-day implementation from date of issuance.

- Supply a contact licensing person and phone number.

- Oath or Affirmation (10CFR50.30)

- Copy to State agency (10CFR50.91). Except not proprietary enclosures.

Supporting Information & Safety Assessment-

- Introduction, background and basis for amendment request.
- Cite any applicable industry references, submittals of other licensees, applicable NuRegs, Reg. Guides, STS, etc.
- PRA input/impact (if any).

No Significant Hazards Consideration-

- NSH (10CFR50.92) – narrative/description at beginning. Keep answers to the three questions simple and concise.

Revised Pages-

- Submit marked-up pages.
- Submit re-typed pages with revision bars.
- Include revised Bases pages in submittal.

Other-

- Send license amendment request electronically to Project Manager to assist with preparation of Federal Register Notice, as well as paper version per 10CFR50.4.
- Include a workbook to enhance NRC review for a complex change.

VERMONT YANKEE NUCLEAR POWER CORPORATION

185 Old Ferry Road, Brattleboro, VT 05301-7002
(802) 257-5271

May 6, 1999
BVY 99-68

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 215
Removal of Main Steam Line Isolation Valve Leakage Specifications

Pursuant to 10CFR50.90, Vermont Yankee (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating the attached proposed change into the VY Technical Specifications. This proposed change deletes the specific leak rate requirements of Technical Specifications 3.7.A.4 and 4.7.A.4 for the main steam line isolation valves.

In the transition from 10CFR50 Appendix J, Option A to Option B, as approved per Facility Operating License Amendment No. 152, the specific acceptance criteria for the main steam line isolation valves was retained. However, with the advent of the Primary Containment Leak Rate Testing Program Plan (PCLRTP) created to implement Option B, the specific valve leak rate requirements (acceptance criteria) of the Technical Specifications could be deleted and the leak rate limits controlled in the implementing program plan. VY is now proposing to exercise this option to remove the specific leak rate acceptance criteria for the main steam line isolation valves from the Technical Specifications since the criteria is redundant, is obsolete per 10CFR50 Appendix J, and constitutes a burden with no significant gain in safety. Main steam line isolation valve leakage is a component of the Technical Specification 6.15, Primary Containment Leak Rate Testing Program, combined local leak rate acceptance criteria and will continue to be governed by that specification. The specific main steam line isolation valve leak rate criteria will be relocated to the PCLRTP, which implements Technical Specification 6.15.

Attachment 1 to this letter contains supporting information and the safety assessment of the proposed change. Attachment 2 contains the determination of no significant hazards consideration. Attachment 3 provides the mark-up version of the current Technical Specification page. Attachment 4 is the retyped Technical Specification page.

VY has reviewed the proposed Technical Specification change in accordance with 10CFR50.92 and concludes that the proposed change does not involve a significant hazards consideration.

Pursuant to 10CFR50.91(b)(1), we have provided a copy of this proposed change and the associated no significant hazards consideration to the appropriate State of Vermont representative.

VY has also determined that the proposed change satisfies the criteria for a categorical exclusion in accordance with 10CFR51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared for this change.

We request that the Staff issue the subject license amendment no later than September 1999, in order to ensure implementation prior to the next scheduled refueling outage in October 1999.

If you have any questions on this transmittal, please contact Mr. Jim Devincentis at (802) 258-4236.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Robert J. Wanczyk
Director of Safety and Regulatory Affairs

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Robert J. Wanczyk, who, being duly sworn, did state that he is Director of Safety and Regulatory Affairs of Vermont Yankee Nuclear Power Corporation, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Corporation, and that the statements therein are true to the best of his knowledge and belief.

SALLY A. SANDSTRUM NOTARY PUBLIC WINDHAM COUNTY, VERMONT

Sally A. Sandstrum, Notary Public
My Commission Expires February 10, 2003

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
Vermont Department of Public Service

Docket No. 50-271
BVY 99-68

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 215

Removal of Main Steam Line Isolation Valve Leakage Specifications

Supporting Information and Safety Assessment of Proposed Change

INTRODUCTION

Option B to 10CFR50 Appendix J was implemented at Vermont Yankee (VY) per Facility Operating License Amendment No. 152[1]. With the advent of the Primary Containment Leak Rate Testing Program Plan (PCLRTP) created to implement Option B, the specific valve leak rate requirements (acceptance criteria) of the Technical Specifications could be eliminated. The individual valve leak rates are components of the combined local leak rate as governed by Technical Specification 6.15, and control of the specific valve leak rate limits is provided by the PCLRTP that implements Technical Specification 6.15.

Therefore, the following paraphrased text "the leakage from any one isolation valve shall not exceed 5 percent of the allowable leak rate" was removed from the VY Technical Specifications 3.7.A.4 and 4.7.A.4 per Amendment 152. However, the specific acceptance criteria for the main steam line isolation valves was retained in the transition from 10CFR50 Appendix J, Option A to Option B.

Technical Specifications 3.7.A.4 and 4.7.A.4 contain main steam line isolation valve leakage acceptance criteria of 15.5 scf/hr at 44 psig (P_a) and 11.5 scf/hr at 24 psig (P_t). The 15.5 scf/hr is equal to the calculated value of $0.05L_a$ (i.e., the pre-Amendment 152 individual valve leakage rate limit discussed above, where L_a equals 0.8 wt%/day [309.890 scf/hr]). The original and present Technical Specification surveillance requirement leakage acceptance criterion for the main steam line isolation valves is 11.5 scf/hr, when tested at a reduced pressure of 24 psig (P_t). During the post rule making implementation period for the original 10CFR50 Appendix J (circa 1972), VY requested relief from the requirement to test the main steam line isolation valves at 44 psig (P_a). In the Safety Evaluation dated August 19, 1983[2], VY was granted relief to continue the testing of the main steam line isolation valves by pressurizing between the inboard and outboard valves at a reduced pressure of 24 psig (P_t). VY will continue to utilize a reduced pressure test at 24 psig (P_t), pressurizing the main steam line isolation valves between the inboard and outboard valves.

VY is now proposing to exercise the option to remove from Technical Specifications the specific acceptance criteria for the main steam line isolation valves. Main steam line isolation valve leakage is a component of the Technical Specification 6.15 combined local leak rate acceptance criteria and, as such, will continue to be governed by that specification. The specific main steam line isolation valve leak rate criteria will be relocated to the PCLRTP.

Maintaining the specific main steam line isolation valve leak rate criteria in Technical Specifications is redundant, is obsolete per 10CFR50 Appendix J, and constitutes a burden with no significant gain in safety. Due to the prescriptive requirement, maintenance is required whenever a leak rate is even slightly above the specific acceptance criterion (i.e., 11.5 scf/hr). If the combined local leak rate acceptance criterion of $0.6L_a$ (maximum or minimum pathway basis depending on operating mode as prescribed by Technical Specification 6.15) is not challenged, the maintenance is unnecessary. Maintenance conducted to fulfill the redundant specification results in significant worker exposure and increased outage duration.

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[1] Letter USNRC to VYNPC, NVY 98-24, "Issuance of Amendment No. 152 to Facility Operating License No. DPR-28, VYNPS (TAC No. M99264)," dated February 26, 1998.

[2] Letter USNRC to VYNPC, NVY 83-192, "Exemption from Certain Requirements of Section 50.54(o) and Appendix J of 10 CFR 50," dated August 19, 1983.

The PCLRTP implements the requirements of 10CFR50, Appendix J Option B, and is consistent with the U.S. Nuclear Regulatory Commission Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. The technical methods and techniques for performing the Types A, B, and C tests are in accordance with the ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements." The program plan follows the Nuclear Energy Institute's "Industry Guideline for Implementing Performance-Based Option of 10CFR50, Appendix J," (NEI 94-01, Revision 0) for both the Type A and the Types B and C Test Programs. Additionally, the program plan implements the requirements of 10CFR50.55a, "Codes and Standards," Section (b)(2)(vii) *Inservice testing of containment isolation valves* to be analyzed in accordance with 4.2.2.3(e) and corrective actions performed in accordance 4.2.2.3(f) of Part 10 of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987.

The documents that provide the requirements and the approved guidelines for implementation of the Option B leak rate testing program, permit the establishment of Administrative and Corrective Action Limits for the determination of valve leak rate performance. When a Corrective Action Limit is exceeded, the containment isolation valves "...shall be declared inoperable and either repaired or replaced." Corrective Action Limits are established and controlled in the PCLRTP for all Type C local leak rate tested valves, including the main steam line isolation valves. Presently the Corrective Action Limit for a main steam line isolation valve is the Technical Specification acceptance criteria of 11.5 scf/hr. Administrative controls will be established to limit, prior to entering a mode of operation where Primary Containment Integrity is required, the combined total leak rate of the main steam line isolation valves based on the maximum pathway and consistent with the analyses of record, including the main control room habitability analysis[3,4,5]. Appropriate Corrective Action Limits will be established for the valves in accordance with the PCLRTP.

The analysis and the repair or replacement process required by the ASME/ANSI OMa-1988[6], Part 10 paragraph 4.2.2.3(f) and the corrective action and causal determination process of NEI 94-01, Revision 0, are included in the PCLRTP. The PCLRTP and other plant procedures control the repair and subsequent retest requirements for all leak rate tested valves, including the main steam line isolation valves.

SAFETY ASSESSMENT

The proposed change is not a safety concern and can be implemented without endangering the public health and safety because:

1. Regulatory Guide 1.163 provides specific guidance concerning the implementation of Option B to 10CFR50 Appendix J. The Regulatory Guide does not prescribe a specific leak rate for a main steam line isolation valve. It does provide directions for the testing intervals for these valves. The proposed Technical Specification change is consistent with the Regulatory Guide.

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[3] Letter VYNPC to USNRC, FVY 81-8, "Submittal of Information on NUREG 0737, Item III D.3.4: Control Room Habitability," dated January 12, 1981.

[4] Letter USNRC to VYNPC, NVY 82-22, "Safety Evaluation Report for III.D.3.4 Control Room Habitability Requirements (NUREG-0737)," dated February 24, 1982.

[5] Vermont Yankee Final Safety Analysis Report, Revision 15, Section 14.9.1.5.

[6] ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, "Operation and Maintenance of Nuclear Power Plants."

2. The present testing requirements of Option B of 10CFR50 Appendix J, contained in the PCLRTP, and based on the guidelines of NEI 94-01, Revision 0, continue to provide reasonable assurance that the leakage through the primary containment and components penetrating the primary containment will not exceed the allowable leakage rates in Technical Specification 6.15 and that the integrity of the containment structure is maintained during its service life. Neither Option B of 10CFR50 Appendix J nor NEI 94-01, Revision 0, requires a specific leak rate for the main steam line isolation valves.

3. Main steam line isolation valve leakage is a component of the Technical Specification 6.15 combined local leak rate acceptance criteria and, as such, will continue to be governed by that specification. The specific main steam line isolation valve leak rate criteria will be relocated to the Primary Containment Leak Rate Testing Program Plan (PCLRTP), which implements Technical Specification 6.15.

4. Prior to Facility Operating License Amendment 152, the main steam line isolation valve maximum pathway leak rate test results were added to Types B and C tests summation for evaluation compared to the $0.6L_a$ maximum pathway acceptance criteria. With the implementation of Amendment 152, the main steam line isolation valve maximum pathway leak rate test results are added to Types B and C tests summation for evaluation and for comparison to the $0.6L_a$ maximum and minimum combined leak rate acceptance criteria. The relocation of the specific leak rate acceptance criteria for the main steam line isolation valves does not change the calculated leak rate results that are compared to the $0.6L_a$ maximum and minimum combined leak rate acceptance criteria.

5. The radiological consequences of the design basis loss of coolant accident are dependent upon containment leakage rates. The leakage rate limitations assumed in the safety analyses are not impacted by this change.

Docket No. 50-271
BVY 99-68

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 215

Removal of Main Steam Line Isolation Valve Leakage Specifications

Determination of No Significant Hazards Consideration

Pursuant to 10CFR50.92, Vermont Yankee (VY) has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10CFR50.92(c).

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that contribute to the initiation of any accidents previously evaluated. Thus, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change does not affect the leak-tight integrity of the containment structure that is designed to mitigate the consequences of a loss-of-coolant accident (LOCA). The primary containment must maintain functional integrity during and following the peak transient pressures and temperatures that result from any LOCA, thereby limiting fission product leakage following the accident. Because the proposed change does not alter any of the fission product leak rate assumptions used in the design basis LOCA analysis, the analyzed consequences of the Loss of Coolant Accident are not changed.

Based on the above VY has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any parameters or conditions that could contribute to the initiation of any accident. This change eliminates redundant acceptance criteria from the Technical Specifications. The methods of performing the tests are not changed. No new accident modes are created by the removal of the acceptance criteria. No safety-related equipment or safety functions are altered as a result of this change. The removal of the acceptance criteria has no influence over nor does it contribute to, the possibility of a new or different kind of accident or malfunction from those previously evaluated.

Based on the above VY has concluded that the proposed change will not create the possibility of a new or different kind of accident from those previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The removal of the acceptance criteria does not impact the margin of safety. The $0.6L_a$ maximum and minimum pathway leak rate acceptance criteria provide the previously analyzed margin of safety. The testing method for determining the leak-tightness of the main steam line isolation valves has not changed. The leak rate test results are presently added to the Types B and C tests summation. The $0.6L_a$ maximum and minimum pathway leak rate acceptance criteria and the programmatic Corrective Action Limits provide assurance that component degradation does not impact the assumptions used to determine, nor provide a reduction in, the analyzed margin of safety.

Based on the above VY has concluded that the proposed change will not cause a significant reduction in a margin of safety.

Summary No Significant Hazards Consideration

On the basis of the above, VY has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10CFR50.92(c), in that it: (1) does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) does not involve a significant reduction in a margin of safety.

In making this determination, Vermont Yankee has also reviewed the NRC examples of license amendments considered not likely to involve significant hazards considerations as provided in the final adoption of 10CFR50.92 published in the Federal Register, Volume 51, No. 44, dated March 6, 1986.

Docket No. 50-271
BVY 99-68

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 215

Removal of Main Steam Line Isolation Valve Leakage Specifications

Marked-up Version of the Current Technical Specifications

3.7 LIMITING CONDITIONS FOR OPERATION

at normal cooldown rates if the torus water temperature exceeds 120°F.

e. Minimum Water Volume - 68,000 cubic feet

f. Maximum Water Volume - 70,000 cubic feet

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

3. If a portion of a system that is considered to be an extension of primary containment is to be opened, isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve or blind flange.

4. ~~Whenever primary containment is required, the leakage from any one main steam line isolation valve shall not exceed 15.5 scf/hr at 44 psig (P_a).~~

4.7 SURVEILLANCE REQUIREMENTS

2. The primary containment integrity shall be demonstrated as required by the Primary Containment Leak Rate Testing Program (PCLRTP).

3. (Blank)

4. ~~The leakage from any one main steam line isolation valve shall not exceed 11.5 scf/hr at 24 psig (Pt). Repair and retest shall be conducted to insure compliance.~~

Delete and replace with "(Blank)"

Docket No. 50-271
BVY 99-68

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 215

Removal of Main Steam Line Isolation Valve Leakage Specifications

Retyped Technical Specification Page

3.7 LIMITING CONDITIONS FOR OPERATION

at normal cooldown rates if the torus water temperature exceeds 120°F.

e. Minimum Water Volume - 68,000 cubic feet

f. Maximum Water Volume - 70,000 cubic feet

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

3. If a portion of a system that is considered to be an extension of primary containment is to be opened, isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve or blind flange.

4. (Blank)

4.7 SURVEILLANCE REQUIREMENTS

2. The primary containment integrity shall be demonstrated as required by the Primary Containment Leak Rate Testing Program (PCLRTP).

3. (Blank)

4. (Blank)

VERMONT YANKEE NUCLEAR POWER CORPORATION

185 Old Ferry Road, Brattleboro, VT 05301-7002
(802) 257-5271

February 11, 2000
BVY 00-020

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 231
Main Steam Isolation Valve Surveillance Requirements

Pursuant to 10CFR50.90, Vermont Yankee (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating the attached proposed change into the VY Technical Specifications. The proposed amendment is requested to be processed as an "exigent" Technical Specification change in accordance with 10CFR50.91(a)(6) due to special circumstances currently existing at the Vermont Yankee Nuclear Power Station. The exigent circumstances involve testing of one main steam isolation valve (MSIV) and are further described in VY letter to NRC (BVY 00-019) dated February 11, 2000, concerning a Notice of Enforcement Discretion.

This proposed change deletes the requirement to exercise MSIVs twice weekly by partial closure and subsequent re-opening. Testing of the MSIVs to demonstrate their safety function will continue to be performed on a quarterly basis in accordance with the Vermont Yankee Inservice Testing program and applicable provisions of Section XI of the ASME Boiler and Pressure Vessel Code. This change to the Technical Specifications is consistent with the Standard Technical Specifications[1].

Beginning with partial closure testing performed on January 17, 2000, MSIV 80-C has exhibited slower than normal re-opening time during the test. Closing times and the quarterly full stroke testing of this MSIV in accordance with the inservice testing (IST) program have been acceptable. However, the re-opening time has continued to be erratic since the January 17 test and is trending up (i.e., taking longer to re-open). This is evidenced by two other tests indicating slower than expected re-opening times. If the MSIV were to fail to re-open and continue closing, a plant transient could result. Therefore, exigent circumstances exist because continued partial-closure testing of inboard MSIV 80-C has the potential to induce an operational transient, considering the probable degraded condition of its test pilot valve. The test pilot valve is not used to test the safety function of the MSIV; its use is required to perform the twice-weekly partial closure exercise of the MSIV.

Prior to January 17, 2000, there was no indication of degradation of MSIV partial-closure testing performance. A review of inservice testing data for all MSIVs since 1996 indicates all MSIVs have met acceptance criteria relative to demonstrating isolation (full closure) times within 3-5 seconds as required by Technical Specifications and assumed in accident analyses. VY could not have anticipated the need for processing this change under 10CFR50.91(a)(6) since the circumstance described above is recently occurring and is only evident in three recent partial-closure tests. The situation was unavoidable considering the past reliable performance of the MSIVs and their pneumatic actuators. The subject test pilot valve was refurbished in 1998 as part of scheduled preventive maintenance on the MSIV pneumatic

1 NUREG 1433, Revision 1, Standard Technical Specifications General Electric Plants, BWR/4, dated April 7, 1995

actuator unit. Again, prior to January 17, 2000, VY had no indication of degradation of the suspected test pilot valve. All MSIVs continue to perform their safety function as demonstrated by acceptable full closure testing and VY believes the cause of the slower re-opening is limited to the test pilot valve, which will not prevent the MSIV from performing its safety function.

Attachment 1 to this letter contains supporting information and the safety assessment of the proposed change. Attachment 2 contains the determination of no significant hazards consideration. Attachment 3 provides the marked-up version of the current Technical Specification and Bases pages. Attachment 4 is the retyped Technical Specification and Bases pages.

VY has reviewed the proposed Technical Specification change in accordance with 10CFR50.92 and concludes that the proposed change does not involve a significant hazards consideration.

VY has also determined that the proposed change satisfies the criteria for a categorical exclusion in accordance with 10CFR51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared for this change.

Upon acceptance of this proposed change by the NRC, VY requests that a license amendment be issued for implementation within 30 days of the effective date of the amendment.

If you have any questions on this transmittal, please contact Mr. James M. DeVincentis at (802) 258-4236.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Samuel L. Newton

Samuel L. Newton
Vice President, Operations

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Samuel L. Newton, who, being duly sworn, did state that he is Vice President, Operations of Vermont Yankee Nuclear Power Corporation, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Corporation, and that the statements therein are true to the best of his knowledge and belief.

Judith A. Harris

Judith A. Harris, Notary Public
My Commission Expires February 10, 2003

JUDITH A. HARRIS NOTARY PUBLIC WINDHAM COUNTY, VERMONT

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
Vermont Department of Public Service

Docket No. 50-271
BVY 00-020

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 231

Main Steam Isolation Valve Surveillance Requirements

Supporting Information and Safety Assessment of Proposed Change

INTRODUCTION

The purpose of this proposed change is to delete Vermont Yankee (VY) Technical Specification (TS) Surveillance Requirement (SR) 4.7.D.1.d regarding exercising of main steam isolation valves (MSIVs). An associated change is made to Technical Specification Bases 4.7.D. The proposed change is consistent with industry practice and the BWR/4 Standard Technical Specifications (NUREG-1433, Rev. 1). Testing of MSIV safety function will continue to be conducted on a quarterly basis, consistent with other TS requirements, the VY Inservice Testing (IST) program, and Section XI of the ASME Boiler and Pressure Vessel Code.

SR 4.7.D.1.d requires:

> *At least twice per week, the main steam line isolation valves shall be exercised by partial closure and subsequent reopening.*

This SR is no longer necessary to assure safe reactor operation and reliability of the MSIVs. MSIV testing will continue to be performed in accordance with SRs 4.7.D.1.a, 4.7.D.1.b, and 4.7.D.1.c.

BACKGROUND

The requirement to exercise MSIVs twice weekly was originally incorporated into the TS at the time VY was first licensed to operate. The purpose of this frequent, partial stroke test was to provide indirect assurance that the valve actuator is operable and will function as intended when necessary. Because of a distinctive design, the former MSIV solenoid-operated pilot valves were susceptible to binding, which could compromise MSIV performance. To compensate for this potential, twice-weekly testing was conducted to provide assurance of valve reliability. The earlier design pneumatic control valves were replaced a number of years ago with those of a different manufacturer and different design. Long-term operating experience (VY and industry) has since demonstrated superior reliability of the replacement components.

The solenoid-operated pilot valves that were susceptible to binding were replaced, but the TS were not revised to eliminate the twice-weekly exercise of the MSIVs.

MSIV Design and Operation

The VY MSIVs are spring-closing, pneumatic, piston-operated valves designed to automatically close (fail-safe) upon loss of pneumatic pressure to the valve operator. Solenoid-operated pilot valves control valve opening and closure. Directing pneumatic pressure to the valve operator to overcome the closing force exerted by the spring opens the MSIV, while either a re-directed pneumatic pressure and/or spring force will close the MSIV.

Each of the eight (four inboard, four outboard) MSIVs is equipped with a Hiller model SA-A083 actuator unit (i.e., MSIV actuator). Each actuator unit consists of three solenoid-operated pilot valves that direct pneumatic pressure to the actuator piston. Using positioning control valves to direct nitrogen/air flow, individual MSIVs can be fast-closed or slow-closed as necessary during periodic testing.

Partial MSIV Closure Exercise

VY TS 4.7.D.1.d requires that the MSIVs be exercised at least twice weekly by partial closure and re-opening. This test entails a slow, partial closure of each MSIV. The testing arrangement is designed to give a slow closure of the MSIV to avoid rapid changes in steam flow and nuclear system pressure, which could induce a transient condition. The control room operator performs the MSIV exercise test by manually depressing a pushbutton switch, which energizes a test pilot solenoid causing a 3-way flow control valve to slowly relieve pneumatic pressure from the actuator. As the MSIV slowly closes, the operator monitors the control room panel indicating lights to verify valve movement. When the MSIV is still about 90% of full open, the operator releases the test pushbutton, reversing the flow control valve and causing the MSIV to return to the full open position.

One drawback of the partial closure test is that it does not directly test the (2-way and 4-way) pilot valves used for fast closure of the MSIVs, but rather actuates a test pilot valve of the same manufacturer. This somewhat indirect indication of MSIV reliability is not as valid a test as the quarterly, full-stroke, fast-closure of the MSIVs since it does not activate the other pilot valves, nor does it test the isolation safety function of the MSIVs.

Full Closure MSIV Testing

In addition to the twice-weekly exercise of MSIVs, they are subjected to a full, fast closure test quarterly as part of the VY IST program in accordance with TS SRs 4.7.D.1.a, 4.7.D.1.b, and 4.7.D.1.c, incorporating ASME Code Section XI requirements. Unlike the twice-weekly exercise, this quarterly surveillance tests the safety function of the valves and ensures that the closure times are within the limits of operability for the MSIVs as specified in TS Table 4.7.2 and assumed in VY accident analyses.

SAFETY ASSESSMENT

The proposed change will eliminate the requirement to exercise the MSIVs twice weekly.

Bases for the Change:

The requirement to exercise MSIVs twice weekly is unnecessary based on the design and reliability of components in the MSIV actuator. This test was originally instituted based on a concern with the reliability of MSIV actuator components. However, since the components that were susceptible to failure have been replaced and operating experience has demonstrated a high degree of reliability with the replacement components, this additional exercise testing of the MSIVs is no longer warranted. Indeed, continued twice-weekly MSIV testing may be contrary to safety and reliability goals since unnecessary testing may lead to premature equipment wear while exposing the reactor to potential operational transients because of the testing.

The supplier of the MSIV actuator unit, Ralph A. Hiller Company, was recently contacted and does not recommend actuator testing at a frequency greater than the quarterly testing specified in the VY IST program and ASME Code.

Any binding of the solenoid-operated pilot valves or other problems with the MSIVs should be identified during the quarterly surveillance. However, since the earlier model components were replaced, operating experience supports elimination of the twice-weekly testing requirement.

The BWR/4 Standard Technical Specifications (STS) (NUREG 1433, Rev. 1) notes that MSIVs should be tested in accordance with the plant's Inservice Testing Program. The STS do not contain requirements to perform MSIV exercising or testing at a greater frequency. Industry operating experience has shown that testing the valves on a quarterly basis is adequate for ensuring the MSIVs will close on demand, fulfilling their safety function. Furthermore, decreasing the MSIV test frequency reduces wear, which is caused by more frequent testing.

This reduction in testing is supported by NRC's position regarding NUREG-0737, Item II.K.3.16, which provided recommendations to reduce the frequency of challenges to safety relief valves. One of the recommendations was to reduce the frequency of testing MSIVs. Partial-stroke exercising of MSIVs can result in a transient if, for example, a test circuit malfunction causes the tested MSIV to fully close. Because the partial-stroke exercise tests are usually performed during full power operation, the resulting transient could involve reactor trips and challenge safety systems. By testing according to the approved IST program, the potential for this type of plant transient is reduced.

The requirements of TS SRs 4.7.D.1.a, 4.7.D.1.b and 4.7.D.1.c are adequate to ensure operability of the MSIVs and SR 4.7.D.1.d can be eliminated from TS.

Finally, a conforming change is being made to TS Bases 4.7.D by changing the sentence, "The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability." to "The main steam isolation valves are primary containment isolation valves and are tested in accordance with the requirements of the Inservice Testing program."

Docket No. 50-271
BVY 00-020

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 231

Main Steam Isolation Valve Surveillance Requirements

Determination of No Significant Hazards Consideration

Determination of No Significant Hazards Consideration

Description of amendment request:

This proposed change deletes the requirement to exercise main steam isolation valves (MSIVs) twice weekly.

The twice-weekly test involves the partial closure of each MSIV to about the 90% open position and re-opening to the full open position. The purpose of the twice-weekly test is not to test the safety function of the MSIVs, but to provide assurance of reliable MSIV operation because of a component that was susceptible to failure. That particular component has been replaced by one of a more reliable design, which has demonstrated reliable performance. The MSIV isolation safety function is tested during the quarterly, full-stroke, fast closure test. The MSIVs will continue to be tested quarterly in accordance with the Inservice Testing program.

The safety function of the MSIVs is to quickly isolate the main steam lines in the event of a postulated steam line break or control rod drop accident to limit the loss of reactor coolant and/or the release of radioactive materials. The MSIVs perform a safety function by mitigating the consequences of accidents; however, an operational transient can be initiated by the inadvertent closure of MSIVs.

NRC has previously found, in other applications, the acceptability of removing the identified requirement from the Technical Specifications. The requirement for twice-weekly testing is not required to provide adequate protection of the public health and safety.

Basis for no significant hazards determination:

Pursuant to 10CFR50.92, Vermont Yankee (VY) has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10CFR50.92(c).

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The frequency of MSIV testing is not assumed to be an initiator of any analyzed event. This change will not alter the basic operation of process variables, structures, systems, or components as described in the safety analyses. The twice-weekly exercise of MSIVs is not intended to verify the safety function of the MSIVs. The safety function testing will continue to be conducted during the quarterly, full-stroke fast closure MSIV test. However, eliminating unnecessary testing of the MSIVs may reduce the probability of occurrence of an inadvertent valve closure that could lead to a plant transient condition.

Deleting the twice-weekly MSIV test is not considered to have any measurable effect on the reliability of the MSIVs to perform their safety function; therefore, the mitigating function of the MSIVs is maintained. The consequences of accidents previously evaluated will not be affected by this change because the surveillances to test MSIVs in accordance with the IST program and Section XI of the ASME Code will still be performed, assuring that MSIVs will perform their intended safety function.

Since reactor operation with the deleted surveillance specification is fundamentally unchanged, no design or analytical acceptance criteria will be exceeded. As such, this change does not impact initiators of analyzed events nor assumed mitigation of design basis accident or transient events.

These changes do not affect the initiation of any event, nor do they negatively impact the mitigation of any event. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not affect any parameters or conditions that could contribute to the initiation of an accident. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). No new accident modes are created since the manner in which the plant is operated is fundamentally unchanged. This change to surveillance requirements does not affect the design or function of safety-related equipment, nor does it eliminate testing to verify a safety function. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

Testing the MSIVs by full stroke closure on a quarterly basis is adequate to maintain reliability of the MSIVs to perform their safety function. This has been demonstrated through industry operating experience. Since frequency or method of MSIV testing is not specifically considered in any safety analysis, current safety analysis assumptions are being maintained. The reduction in testing from a twice-weekly exercise (partial closure and re-opening) while maintaining the quarterly full-stroke test is adequate to maintain the reliability of this safety function while reducing unnecessary valve wear and the potential for inducing an inadvertent transient. Consequently, margins of safety are maintained.

There is no impact on equipment design or operation, and there are no changes being made to safety limits or safety system settings that would adversely affect plant safety because of the proposed changes. Since the changes have no effect on any safety analysis assumption or initial condition, the margins of safety in the safety analyses are maintained.

Summary No Significant Hazards Consideration:

On the basis of the above, VY has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10CFR50.92(c), in that it: (1) does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) does not involve a significant reduction in a margin of safety.

In making this determination, Vermont Yankee has also reviewed the NRC examples of license amendments considered not likely to involve significant hazards considerations as provided in the final adoption of 10CFR50.92 published in the Federal Register, Volume 51, No. 44, dated March 6, 1986.

Docket No. 50-271
BVY 00-020

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 231

Main Steam Isolation Valve Surveillance Requirements

Marked-up Version of the Current Technical Specifications

3.7 LIMITING CONDITIONS FOR OPERATION

5. Core spray and LPCI pump lower compartment door openings shall be closed at all times except during passage or when reactor coolant temperature is less than 212°F.

D. Primary Containment Isolation Valves

1. During reactor power operating conditions all containment isolation valves and all instrument line flow check valves shall be operable except as specified in Specification 3.7.D.2.

4.7 SURVEILLANCE REQUIREMENTS

5. The core spray and LPCI lower compartment openings shall be checked closed daily.

D. Primary Containment Isolation Valves

1. Surveillance of the primary containment isolation valves should be performed as follows:

   a. The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and the closure times specified in Table 4.7.2 at least once per operating cycle.

   b. Operability testing of the primary containment isolation valves shall be performed in accordance with Specification 4.6.E.

   c. At least once per quarter, with the reactor power less than 75 percent of rated, trip all main steam isolation valves (one at a time) and verify closure time.

   d. ~~At least twice per week, the main steam line isolation valves shall be exercised by partial closure and subsequent reopening.~~

BASES: 4.7 (Cont'd)

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein and per Specification 4.6.E are adequate to prevent loss of more cooling from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, the isolation valve closure times are sufficient to prevent uncovering the core.

Purge and vent valve testing performed by Allis-Chalmers has demonstrated that all butterfly purge and vent valves installed at Vermont Yankee can close from full open conditions at design basis containment pressure. However, as an additional conservative measure, limit stops have been added to valves 16-19-7/7A, limiting the opening of these valves to 50° open while operating, as requested by NRC in their letter of May 22, 1984. (NVY 84-108)

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate the fuel rod cladding perforations would be avoided for the main steam valve closure times, including instrument delay, as long as 10.5 seconds. The test closure time limit of five seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided and 10CFR100 limits are not exceeded. Redundant valves in each line ensure that isolation will be effected applying the single failure criteria.

~~The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.~~

The containment is penetrated by a large number of small diameter instrument lines. The flow check valves in these lines are tested for operability in accordance with Specification 4.6.E.

The main steam isolation valves are primary containment isolation valves and are tested in accordance with the requirements of the Inservice Testing program.

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 231

Main Steam Isolation Valve Surveillance Requirements

Retyped Technical Specification Pages

Listing of Affected Technical Specifications Pages

Replace the Vermont Yankee Nuclear Power Station Technical Specifications pages listed below with the revised pages. The revised pages contain vertical lines in the margin indicating the areas of change.

Remove	Insert
156	156
171	171

3.7 LIMITING CONDITIONS FOR OPERATION

5. Core spray and LPCI pump lower compartment door openings shall be closed at all times except during passage or when reactor coolant temperature is less than 212°F.

D. Primary Containment Isolation Valves

1. During reactor power operating conditions all containment isolation valves and all instrument line flow check valves shall be operable except as specified in Specification 3.7.D.2.

4.7 SURVEILLANCE REQUIREMENTS

5. The core spray and LPCI lower compartment openings shall be checked closed daily.

D. Primary Containment Isolation Valves

1. Surveillance of the primary containment isolation valves should be performed as follows:

   a. The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and the closure times specified in Table 4.7.2 at least once per operating cycle.

   b. Operability testing of the primary containment isolation valves shall be performed in accordance with Specification 4.6.E.

   c. At least once per quarter, with the reactor power less than 75 percent of rated, trip all main steam isolation valves (one at a time) and verify closure time.

BASES: 4.7 (Cont'd)

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein and per Specification 4.6.E are adequate to prevent loss of more cooling from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, the isolation valve closure times are sufficient to prevent uncovering the core.

Purge and vent valve testing performed by Allis-Chalmers has demonstrated that all butterfly purge and vent valves installed at Vermont Yankee can close from full open conditions at design basis containment pressure. However, as an additional conservative measure, limit stops have been added to valves 16-19-7/7A, limiting the opening of these valves to 50° open while operating, as requested by NRC in their letter of May 22, 1984. (NVY 84-108)

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate the fuel rod cladding perforations would be avoided for the main steam valve closure times, including instrument delay, as long as 10.5 seconds. The test closure time limit of five seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided and 10CFR100 limits are not exceeded. Redundant valves in each line ensure that isolation will be effected applying the single failure criteria.

The main steam isolation valves are primary containment isolation valves and are tested in accordance with the requirements of the Inservice Testing program.

The containment is penetrated by a large number of small diameter instrument lines. The flow check valves in these lines are tested for operability in accordance with Specification 4.6.E.

NOTICES OF ENFORCEMENT DISCRETION

REVISED STAFF GUIDANCE - PART 9900

Rick Croteau
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

NOED POLICY

BACKGROUND AND CHRONOLOGY

- 7/85 - ENFORCEMENT GUIDANCE MEMO - TEMPORARY WAIVERS OF COMPLIANCE
- 3/93 - 10 CFR 2, APPENDIX C, SECTION VII.C (NOW NUREG-1600, SECTION VII c), NOED ADDED TO THE POLICY
- 8/ 93 - STAFF IMPLEMENTATION GUIDANCE - MANUAL CHAPTER PART 9900
- 5 -9/94 - CONGRESSIONAL, STAFF AND OIG REVIEWS
- 1/95 - MANUAL CHAPTER 9900 -REVISED GUIDANCE
- 11/95 - REVISED GUIDANCE -PART 9900
- 7/98 - OIG FOLLOW-UP REVIEW OF NOED PROCESS
- 12/98 - REVISED STAFF GUIDANCE
- 6/99 - REVISED STAFF GUIDANCE

GENERAL AND BACKGROUND INFORMATION

- PART 9900 GUIDANCE WAS REVISED ON JUNE 29, 1999.
- REVISION AS A RESULT OF:
  - Policy changes - COMEXM-98-004, dated July 28, 1998
  - OIG audit findings and recommendations -OIG/98A-06, July 30, 1998
  - Utility/NRC Interface Licensing Workshop, July 20-21, 1998
  - Staff experience with the current guidance.

SIGNIFICANT CHANGES TO THE NOED GUIDANCE

- **PROCESS IMPROVEMENTS FOR NOEDs RELATING TO SEVERE WEATHER OR OTHER NATURAL EVENTS**
  - Previously an enforcement discretion, now an NOED
  - Prior Commission approval not required

- **STAFF DOCUMENTATION CHANGES**

COMPLIANCE WITH REQUIREMENTS

- LICENSEES ARE REQUIRED TO COMPLY WITH:

  - NRC REGULATIONS
  - INDUSTRY CODES
  - UFSAR
  - LICENSE TECHNICAL SPECIFICATIONS
  - OTHER LICENSE CONDITIONS

PROCESSES FOR ADDRESSING NON-COMPLIANCE WITH REQUIREMENTS

- NOEDS ARE APPROPRIATE ONLY FOR NON-COMPLIANCE WITH TS OR OTHER LICENSE CONDITIONS

- NOEDS ARE <u>NOT APPROPRIATE</u> FOR NON-COMPLIANCE WITH:
  - REGULATIONS -PROCESS EXEMPTIONS -10 CFR 50.12
  - CODES -PROCESS RELIEFS -10 CFR 50.55a
  - UFSAR -OPERABILITY DETERMINATION GL 91-18 REV. 1 AND PROCESS LICENSE AMENDMENT -10 CFR 50.90

TWO TYPES OF NOEDs

- **(1) RADIOLOGICAL SAFETY CONSIDERATIONS (REGULAR NOED)**

FORCED COMPLIANCE WITH LICENSE WOULD INVOLVE PLANT-RELATED RISKS DUE TO UNNECESSARY TRANSIENT

- **(2) OVERALL PUBLIC HEALTH AND SAFETY CONSIDERATIONS (A SEVERE EXTERNAL CONDITION - RELATED NOED).**

FORCED COMPLIANCE WITH LICENSE MAY AFFECT GRID STABILITY, EXACERBATING IMPACTS OF SEVERE WEATHER OR OTHER NATURAL EVENTS ON OVERALL PUBLIC HEALTH AND SAFETY

DIFFERENTIATE REGULAR VS EXTERNAL CONDITION-- RELATED NOEDS

- REGULAR NOED
  - REGARDLESS OF THE CAUSE, CONTINUED OPERATION IS NEEDED FOR AVOIDING UNNECESSARY TRANSIENT
- WEATHER-RELATED NOED
  - REQUEST CONTINUED OPERATION BECAUSE THERE IS NEED FOR POWER. THIS IS A SEVERE EXTERNAL CONDITION - RELATED NOED.

OTHER PROCESS CHANGES

- ALL NOED-RELATED TELECONFERENCES ARE MADE THROUGH THE NRC HEADQUARTERS EMERGENCY OPERATIONS CENTER RECORDED TELEPHONE LINE (301) 816-5100.

- LICENSEES ARE NO LONGER REQUIRED TO STATE WHETHER:

  - prior adoption of TS enhancement initiatives (GL 87-09, Line Item Improvements or the Improved Standard TS) would have obviated the need for the NOED

  - the noncompliance involves a USQ

- FOR ALL NOEDs (REGIONAL OR NRR) REGION TO OPEN AN UNRESOLVED ITEM (URI).

- This will facilitate:
  - tracking
  - verification of resolution activities
  - documentation and closure of inspection
  - enforcement action determination

CONSIDERATION FOR GRANTING FOR SEVERE WEATHER-RELATED NOEDs

- NATURE OF THE EMERGENCY, POTENTIAL CONSEQUENCES TO THE PLANT , AND CHALLENGES TO OFF-SITE AND ON-SITE POWER SOURCES.

- THE STAFF WILL BALANCE THE OVERALL PUBLIC HEALTH AND SAFETY BENEFIT WITH THE PLANT RISK OF CONTINUED OPERATION

Road Map

- An NOED checklist is provided in Attachment D
- This is only an aid to assure adherence to this guidance.
- It's use is discretionary
- It is a companion, not a substitute, for the detailed guidance.

1. The TS or other license conditions that will be violated.

2. The circumstances surrounding the situation, including root causes, the need for prompt action and identification of any relevant historical events.

3. The safety basis for the request, including an evaluation of the safety significance and potential consequences of the proposed course of action.

4. The basis for the licensee's conclusion that the noncompliance will not be of potential detriment to the public health and safety and that a significant hazard consideration is involved.

5. The basis for the licensee's conclusion that the noncompliance will not involve adverse consequences to the environment

6. Any proposed compensatory measure(s).

7. The justification for the duration of the noncompliance.

11. For NOEDs involving severe weather or other natural events, the licensee must provide:

a. details of the basis and nature of the emergency; potential consequences of forced compliance with the license conditions to the plant and to exacerbation of the emergency situation. The licensee must provide the name, organization and telephone number of the official that made the emergency assessment

b. status, and potential challenges to offsite and onsite power sources, and the impact of the emergency on plant safety.

c. demonstrated actions taken to avert and/or alleviate the emergency situation, including steps taken to avoid being in the noncompliance, as well as efforts to minimize grid instabilities (e.g., coordinating with other utilities and the load dispatcher organization for buying additional power or for cycling load, shedding interruptible industrial or non-emergency loads).

8. A statement that the request has been approved by the facility organization that normally reviews safety issues (Plant Onsite Review Committee, or its equivalent).

9. The request must specifically address which of the criteria specified in Section B is satisfied and how.

10. If a follow-up license amendment is required, the request must include marked-up TS pages showing the proposed TS changes. and a commitment to submit the actual license amendment request within 48 hours.

Ultimate Heat Sink TS requires:

If Service Water temperature exceeds 95F - Restore temperature to ≤ 95F within 8 hours or be in MODE 3 within 6 hours,
and MODE 5 within 36 hours.

Condition:

Due to hot weather conditions, service water temperatures are cycling above 95F for times approaching 8 hours and have, in the past, stayed above 95F for more than 8 hours.

A severe and sustained period of high temperatures is causing record energy demand and system reliability level alerts have been declared.

95F is the input value for containment analysis. Analysis previously performed indicates that components cooled by SW can perform their intended functions with SW temperatures up to 99F (does not include containment analysis).

CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

Serial: RNP-RA/99-0149

July 31, 1999

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REQUEST FOR NOTICE OF ENFORCEMENT DISCRETION
TECHNICAL SPECIFICATION SECTION 3.7.8 - ULTIMATE HEAT SINK

Dear Sir or Madam:

Carolina Power & Light (CP&L) Company requests a Notice of Enforcement Discretion (NOED) regarding compliance with the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Facility Operating License paragraph c, which requires that the facility be operated in conformity with the application, as amended. This NOED and supporting information were discussed during a teleconference with the Nuclear Regulatory Commission (NRC) Staff on July 31, 1999.

A severe and sustained period of high temperatures is causing record energy demand in the Carolinas. As a result, System Reliability Alert Levels 1 and 2, have been invoked on a number of occasions during the period from July 21 through July 30, 1999. Reliability Alert Level 1 is implemented when projected load and reserve requirements will utilize available capacity. Reliability Alert Level 2 is implemented when projected load and reserve requirements are marginally greater than the available capacity. In addition, administrative controls have been implemented during this same period to restrict maintenance and operational activities that have a risk of adversely affecting plant reliability.

The severe and sustained period of hot weather described above, combined with the thermal and hydrological characteristics of the UHS, have resulted in a situation where the existing 8 hour Completion Time may not be of sufficient duration to allow UHS temperature to return below 95°F. Additionally, an extended period of this severely hot weather may further result in several long temperature excursions above 95°F and could result in unwarranted plant power

reductions and shutdowns during a time of record energy demand. This NOED requests that the current 8 hour Completion Time be extended to 72 hours until an exigent TS amendment, submitted to NRC on July 30, 1999, can be approved. The proposed Completion Time will better accommodate the severe current weather patterns.

The 95°F temperature limit prescribed by Technical Specification (TS) Limiting Condition for Operation (LCO) 3.7.8 Condition A for operability of the UHS is the design basis for the Service Water (SW) system. This limit is also the acceptance criterion for a Surveillance Requirement (SR) 3.7.8.2. The current Completion Time for restoring SW temperature to within 95°F is 8 hours. Upon exceeding this Completion Time, TSs require the plant to be in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The requested NOED would allow 72 hours to restore the UHS temperature to within the limits of Condition 3.7.8.A (i.e., 95°F). If restoration does not occur within 72 hours, the plant would be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours.

Long term resolution of this situation has been addressed by previous submittals to include UHS Required Actions and Completion Times in the event that SW temperature exceeds the design limit, and to increase the UHS LCO temperature value from 95°F to 97°F. Specifically, on June 26, 1998, a Technical Specification change request was submitted that would establish permanent Required Actions and Completion Times in the event that SW temperature exceeds 95°F. The June 26, 1998 submittal is being reviewed by the NRC Staff in conjunction with an industry Technical Specification Task Force item. Also, on May 27, 1999, a Technical Specification change request was submitted to increase the maximum allowable UHS temperature from 95°F to 97°F. Due to the nature and complexity of this May 27, 1999 submittal, NRC approval of this proposed Amendment was requested by June 30, 2000.

HBRSEP, Unit No. 2, implemented Improved Technical Specifications (ITS) on November 13, 1997. There are no TS line item improvements associated with this NOED, and NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," and approved generic changes do not currently include a Required Action or Completion Time of this type. Therefore, this NOED is not a result of failure to adopt ITS or approved line-item improvements to the TS. Note, however, that prior to the implementation of the ITS there were no UHS temperature limits in the superseded custom TS. This NOED is requested based upon plant specific considerations and is not generic in nature.

Attachment I provides information in support of the request for NOED in accordance with NRC Administrative Letter (AL) 95-05, Revision 2, "Revisions to Staff Guidance for Implementing NRC Policy on Notices of Enforcement Discretion."

This request for NOED has been reviewed and approved by the Plant Nuclear Safety Committee.

The basis for CP&L's conclusion that the noncompliance will not be a detriment to the public health and safety and a Determination of No Significant Hazards Considerations are provided within the Attachment to this letter. The basis for CP&L's conclusion that the noncompliance will not involve adverse consequences to the environment is provided in the Environmental Impact Consideration provided in the Attachment.

If you have any questions concerning this matter, please contact me or Mr. H. K. Chernoff.

Very truly yours,

R. L. Warden
Manager - Regulatory Affairs

CTB/ctb

Attachment
c: Mr. Max K. Batavia, Chief, Bureau of Radiological Health (SC)
Mr. L. A. Reyes, Regional Administrator, USNRC, Region II
Mr. R. Subbaratnam, USNRC Project Manager, HBRSEP
USNRC Resident Inspector, HBRSEP
Attorney General (SC) (w/out Attachments)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR ENFORCEMENT DISCRETION
TECHNICAL SPECIFICATION SECTION 3.7.8, ULTIMATE HEAT SINK

Technical Specification Section That Will Be Violated

The requested Notice of Enforcement Discretion (NOED) will result in the violation of Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.7.8, Required Action A.1, which requires Service Water (SW) temperature be restored to less than or equal to 95°F within 8 hours of exceeding 95°F. This NOED and an associated exigent TS amendment request submitted on July 30, 1999, propose to allow a 72 hour Completion Time to restore SW temperature to the limit specified by TS LCO 3.7.8 Condition A Required Action A.1. The remaining limits and surveillances associated with the Ultimate Heat Sink (UHS) are unchanged.

Circumstances Requiring the Request

The current UHS temperature limit provided in TS LCO 3.7.8 Condition A is 95°F, with an associated 8 hour Completion Time for restoring temperature to less than or equal to 95°F. South Carolina has been in a period of sustained severe hot temperature. As a result of these temperatures and record system load conditions, the potential for grid instabilities resulting from the forced shutdown of a 700 MW power plant would be very high. It is of great importance to the health and safety of the public that every practicable effort be made to maintain this unit on line. Regional power demand remains extremely high and therefore, regional reserves are not available to replace this unit. System loads are critical and the loss of generation from HBRSEP Unit No. 2 would put CP&L in a potential situation of rotating blackouts until weather conditions moderate. Continuation of this weather pattern is creating a situation where thermal cycling of the lake may require longer than the existing 8 hour Completion Time. This situation is further exacerbated by extreme demand on the CP&L system grid which places a high importance on the continued operation of H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2.

The UHS provides the heat sink for operating and decay heat produced by various plant components during normal operation, transients, and accidents. The SW system and Component Cooling Water (CCW) system are used to transfer heat from plant components to the UHS. The SW system draws water directly from the UHS to provide cooling water to several plant components. Also, the SW system cools the CCW system, which in turn, cools other plant components. The CCW system serves as an intermediate barrier to prevent leakage of potentially radioactive fluid directly to the SW system and environment from plant components containing reactor coolant.

The two principal safety functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident. The basic performance requirements for the UHS are that a 22 day supply of water be available, and that the design basis temperatures of safety-related equipment not be exceeded. These performance requirements are verified through periodic surveillances which assure that lake water level is ≥ 218 feet mean sea level and SW inlet temperature is ≤ 95°F while the plant is operating in MODES 1, 2, 3, and 4. If either of these surveillances are not satisfied while the plant is operating in MODES 1, 2, 3, and 4, the plant is required to be in MODE 3 within 6 hours and in MODE 5 within 36 hours. An 8 hour Completion Time is currently provided to restore SW temperature 95°F or less, after which a plant shutdown would be initiated. The current requirement for SW inlet temperature to be ≤ 95°F was incorporated into the TS upon implementation of the Improved TS in November 1997.

In anticipation of the UHS temperature exceeding 95°F, a change was requested to TS LCO 3.7.8 by a letter dated June 26, 1998, that would allow plant operation above 95°F for up to 8 hours. The purpose of the change was to reduce the risk associated with plant shutdown transients. The TS change was supported by an engineering evaluation, which concluded that the components that rely on the SW system for cooling are able to operate at a SW temperature of up through 99°F. The current containment analyses use a SW temperature of 95°F as a limiting input parameter. Since there is a low probability that a DBA would occur during the proposed 72 hour Completion Time, and the magnitude of any temperature increase above 95°F is expected to be small, this proposed change is of low risk significance.

Prior to the June 1998 TS change being approved, unusually hot and dry weather conditions prompted CP&L to request a Notice of Enforcement Discretion (NOED) by letter dated June 27, 1998, until the TS change could be approved. The request proposed a similar change to TS LCO 3.7.8 with an upper temperature limit of 99°F, and as a long-term resolution for this condition, committed to perform an engineering analysis to justify an increase in the allowed SW temperature. The request for a NOED was accepted by the NRC on July 1, 1998.

Based on a request from the NRC Staff, CP&L subsequently submitted a supplement to the requested change to TS LCO 3.7.8 by a letter dated July 22, 1998, that limited the effective period of the change until September 30, 1998. The provisions of License Amendment No. 179, which were effective through September 30, 1998, were issued by letter dated July 29, 1998.

In March 1999, CP&L requested a TS change that would allow UHS temperature to be above 95°F (and less than 99°F) for 8 hours before a plant shutdown is required. Further evaluation and calculations performed since the summer of 1998 showed that the change did not increase the core damage frequency, had a negligible effect on the large early release frequency, and reduced the potential for plant shutdown transients. The proposed change did not request a long-term increase in UHS temperature because supporting engineering analyses were still in progress.

In April 1999, after discussion with the NRC Staff, CP&L requested a one-time TS change that would allow UHS temperature to be above 95°F and less than 99°F for 8 hours before a plant shutdown is required. The one-time change was requested because approval of the TS change submitted in March 1999 was not feasible by the requested date of June 30, 1999.

Safety Assessment

The UHS temperature is a function of insolation, operation of HBRSEP, Unit Nos. 1 (fossil) and 2 (nuclear), hydrology of Lake Robinson watershed, and meteorological conditions, including shifts in wind direction and velocity, which affect the efficiency of evaporative cooling, natural convection, and diurnal radiant heat losses. Average heat input to Lake Robinson due to insolation is comparable to the heat input from HBRSEP, Unit Nos. 1 and 2, during summer operation. Condenser cooling water and SW discharged from the plant is returned to Lake Robinson via a 4.2 mile discharge canal which originates just east of the plant, parallels the west shore of the lake, and terminates in the lake near its upper end. During full power operation, the nominal transit time of water through the discharge canal is approximately 3.5 hours. Hence, the effect of a plant shutdown in the event that the SW temperature limit is exceeded will not immediately be effective on the temperature of SW entering the plant. However, in the summer months during periods of hot weather, a diurnal effect of alternating insolation of the lake during the day and increased radiant and evaporative heat loss during the night results in a cyclic variation of lake temperature.

In support of HBRSEP's May 27, 1999, request for a UHS TS change, an evaluation of the effects of SW temperature in excess of 95°F was performed. SW system temperature is an input to the containment analysis contained in UFSAR Section 6.2. It is also a design assumption for the Spent Fuel Pool Cooling System, Auxiliary Feedwater System, CCW System and its loads, the Emergency Diesel Generators, Containment Air Recirculation Cooling System, room coolers for certain safety-related areas, and non-safety-related systems. Where SW temperature is relied upon to maintain these components within operating limits, this evaluation found that these components could perform their safety-related functions with SW temperatures above the 95°F limit and up through 99°F. The current containment analyses use a SW temperature of 95°F as a limiting input parameter. Since there is a low probability that a DBA would occur during the proposed NOED duration of 72 hours, and the magnitude of any temperature increase above 95°F is expected to be small, this request involves a low risk significance.

The SW system success criteria as credited in the Probabilistic Safety Analysis (PSA) have been evaluated. The number of SW pumps required under various accident scenarios is dependent upon SW flow rates and is not impacted by the increased SW temperature. Although the increased temperature may decrease the efficiency of heat exchangers cooled by SW, these systems remain capable of performing their intended functions as credited in the PSA. In addition, each additional shutdown during a cycle results in an additional 3.7E-7 to

the annual core damage frequency, and an additional 6.3E-10 to the annual large early release frequency. It has also been determined that a small increase in peak containment pressure would have a negligible affect on the probability of containment failure.

The existing TS introduces the possibility of unwarranted plant shutdown transients. The risk associated with these transients could be reduced by an extension to the existing 8 hour Completion Time. The requested NOED would not allow continuous operation above the maximum design temperature of the SW system. If SW temperature exceeds the 95°F limit, 72 hours would be allowed to restore temperature to less than or equal to 95°F. Additionally, the requested NOED provides a reasonable likelihood for restoration of the LCO before requiring the plant to enter into a shutdown transient. An increase in SW temperature in excess of the design limit is expected to be small due to the limited time allowed by the proposed NOED in conjunction with the slow rate of temperature increase experienced from thermal changes in Lake Robinson. If the LCO is not restored within the revised Completion Time, Condition B of LCO 3.7.8 would be entered and a plant shutdown would be required.

The UHS for HBRSEP, Unit No. 2, is Lake Robinson, as noted in Updated Final Safety Analysis Report (UFSAR), Section 9.2.4, "Ultimate Heat Sink." Lake Robinson was developed for use initially for condenser cooling of HBRSEP, Unit No. 1, a fossil plant. When HBRSEP, Unit No. 2, a nuclear plant, was licensed on July 31, 1970, the unit was designed to use Lake Robinson both for condenser cooling and as the UHS. HBRSEP, Unit No. 2, was licensed in accordance with the proposed draft General Design Criteria, prior to the promulgation of 10 CFR 50, Appendix A. Therefore, the UHS was not designed to satisfy the requirements of the final General Design Criteria. Additionally, the UHS was not designed to satisfy Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Position C.1, which stipulates a 30 day cooling supply. The UHS for HBRSEP, Unit No. 2, is capable of providing cooling water for at least 22 days following a Design Basis Accident (DBA), as stated in the Bases to LCO 3.7.8, "Ultimate Heat Sink."

As shown above, the proposed NOED will not be of potential detriment to the public health and safety.

Proposed Compensatory Actions

For the duration of the NOED, SW system temperatures will be monitored hourly when temperature is in excess of 95°F. Should temperature exceed 99°F, the plant will perform the Required Actions specified within existing LCO 3.7.8, Condition B.

During periods when SW system temperatures exceed 95°F, administrative controls will be implemented to restrict maintenance and operational activities that have a risk of adversely affecting plant reliability.

HBRSEP has previously submitted a proposed TS change, to be processed in an exigent manner, which will implement the proposed Completion Time on a longer-term basis.

Justification of Duration of the Noncompliance

This NOED is requested to remain in place until the proposed exigent change to TS, submitted by letter dated July 30, 1999, is approved by the NRC. Current engineering analyses demonstrate that supported systems are not adversely affected by a SW temperature of 99°F. The revised Completion Time is consistent with those provided for other systems of similar or greater safety significance. Also, based upon prior operational experience, the magnitude of temperatures exceeding 95°F are not expected to be significant.

Justification for the Proposed NOED

In accordance with AL 95-05, Section B, 2.0, "Situations Affecting Radiological Safety," Item 1, the requested NOED is necessary avoid the transient effects of a plant shutdown due to transitory temperature effects upon the UHS, and Section B, 3.0, "Situations Arising from Severe Weather or Other Natural Events."

The existing TS introduces the possibility of additional plant shutdown transients. The risk associated with these transients could be reduced by the proposed 72 hour Completion Time to restore the LCO. The requested NOED would not allow continuous operation above the maximum design temperature of the SW system. If SW temperature exceeds the 95° F limit, 72 hours would be allowed to restore the temperature to less than or equal to 95°F before a plant shutdown would be required. Additionally, the requested NOED provides a reasonable likelihood for restoration of the LCO before requiring the plant to enter into a shutdown transient.

No Significant Hazards Consideration Determination

CP&L has evaluated the plant conditions that result from this requested NOED and have concluded that the requested NOED does not involve a significant hazards consideration. The conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration is discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The requested NOED does not involve any physical alteration of plant systems, structures or components. The requested NOED increases the allowed time for the plant condition where SW temperature exceeds 95°F. SW system temperature is not assumed to be an initiating condition for any accident analysis evaluated in the safety analysis report (SAR).

Therefore, the proposed time increase for restoration of SW temperature to less than or equal to 95°F does not involve an increase the probability of an accident previously evaluated. The SW system supports operability of safety-related systems used to mitigate the consequences of an accident. The magnitude of any increase in SW temperature in excess of the design limit is expected to be small based on historical data and experience for the UHS. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the SAR.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   This requested NOED does not involve any physical alteration of plant systems, structures or components. The increased time for SW temperature to exceed 95°F does not introduce new failure mechanisms for systems, structures or components not already considered in the SAR. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The requested NOED will allow an increase in the allowed time that SW temperature may exceed 95°F, and thereby delays the requirement to initiate a plant shutdown. There are design margins associated with systems, structures and components that are cooled by the SW system that are affected. The SW system temperature is an input assumption for mitigating the effects of design basis accidents. By maintaining an appropriate time limit on the allowed time for SW temperature to exceed 95°F, any potential impact on equipment operating margins is minimized. Therefore, there is no significant reduction in margin of safety associated with this change.

Environmental Impact Consideration

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. CP&L has reviewed this request against these criteria and determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9).

Requested Change

The requested NOED involves an allowance to continue operation for a period of 72 hours with the UHS at a temperature greater than the temperature limits provided in Technical Specifications Section 3.7.8. This period of time may allow diurnal effects to return the UHS to within its temperature limits prior to initiating a plant shutdown.

Basis

The requested NOED meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

1. As demonstrated in the No Significant Hazards Consideration Determination, the requested NOED does not involve a significant hazards consideration.

2. The requested NOED is limited to the provision of increasing the Completion Time for the UHS to exceed of its 95°F temperature limit. This request does not allow for an increase in plant power level, does not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. There will be a slight increase in the temperature of the plant cooling water effluent, but the effect is very small and the effluent will remain within the limits provided by the National Pollutant Discharge Elimination System (NPDES) permit. Therefore, the request does not result in a significant change in the types, or significant increase in the amounts, of any effluent that may be released offsite.

3. The requested NOED does not involve a physical change to the facility design, configuration, maintenance, or testing. The request is limited to allowing operation to continue for a period of 72 hours. Therefore, the requested NOED does not affect individual or cumulative occupational radiation exposure.

Managing Regulatory Commitments

Paul R. Willoughby
Supervisor - Regulatory Affairs
Millstone Nuclear Power Station
Northeast Nuclear Energy Company

Overview

- What is a commitment?

- Communicating with the NRC
  - Who can make a commitment?
  - How are commitments made?

- Maintenance of commitments
  - Where do I keep commitments?
  - Review... How often?

- Now that I have a commitment, what do I do?
  - Changing commitments
    - Why?
    - How?

- Discussion question: What about commitments embedded within SERs, responses to Generic Letters, 50.54(f) information requests, NOV responses?

What is a Commitment?

- An explicit statement to take a specific action agreed to or volunteered by a licensee that has been submitted in writing to the Commission on the docket.

What is it not?

- It is not an obligation: a legally binding requirement imposed on licensees through applicable rules, regulations, orders, and licenses (including technical specifications and license conditions).
- It is not oral statements: representations of an intent to make a commitment.
- It is not voluntary enhancements or descriptive information: ongoing practices not directly related to the cause of an event.

Communicating with the NRC

- **Who can make a commitment?**
  - **Though anyone communicating with the NRC can appear to make a commitment, licensees should have guidelines in place delineating who can make them, typically…**
  - **an officer of the company, someone who has the authority to commit the resources to carry out the commitment.**

- **How are commitments made?**
  - **In writing…**
  - **On the docket…**
  - **Explicit statements…**
  - **Distinguished from voluntary enhancements or statements of fact.**

Maintenance of Commitments

- **Where do I keep commitments?**
  - **Retrievable files**
  - **Database**
  - **Corrective Action Program**

- **Review… How often?**
  - **NRC uses a two-year time period as an indication that an adverse condition has been corrected.**
  - **Corrective action program should capture recurrences, provide indication that commitment change should be made.**

Now that I have a commitment, what do I do?

ANSWER: Change it, of course.

- **Changing commitments**
  - **Why?**
    - **More effective way to minimize recurrence of condition.**
    - **Unsuccessful corrective action.**
    - **No longer necessary due to changing plant conditions.**
    - **Commitment may never have been necessary.**
    - **Commitment captured within process or program.**
  - **How?**
    - **Reasoned management decision-making process.**
    - **5 Questions**
      1. **Codified change process… 50.59, 50.54(a)?**
      2. **Significant to safety…50.92 criteria?**
      3. **Necessary to achieve compliance with obligation?**
      4. **Reliance by NRC on commitment?**

5. Commitment made to minimize recurrence of adverse condition?

DISCUSSION QUESTION

- What about commitments embedded within SERs, responses to Generic Letters, 50.54(f) information requests, NOV responses... is it appropriate to change these commitments... and if so, why?

NRC/LICENSEE

LICENSING WORK SHOP

March 2, 2000

ISI/IST/BWRVIP TOPICS

Contact: Pilgrim -Steve Brennion
-Walter Lobo

I. RELIEF REQUESTS

1. Relief Requests

   - IST/ISI Requirements - 10 CFR 50.55a(a)(3)(f) or (g)

2. Elements of a "Good" Relief Request:

   - Describe the Existing Requirement

   - Describe the Basis for Relief
     ---Topic for Discussion---

   - Propose Alternatives to the Requirements

   - Demonstrate the Proposed Alternatives would provide an Acceptable Level of Quality and Safety [(a)(3)(i)]

OR

- Demonstrate Compliance with the Specific Requirements would result in hardship or unusual difficulty without compensating increase in the Level of Quality and Safety [(a)(3)(ii)]

  e.g. Radiological (ALARA)
  Accessibility
  (Propose Alternatives)

- If the ISI/IST requirement conflicts with the TS, propose a TS change [(f)(5)(ii) and (g)(5)(ii)]

  - For TS Change Follow License Amendment Process

- Provide a Schedule for the effective date of Relief

3. Scope of Submittal Requirement: [(f)(5)(iii) and (g)(5)(iii)]

-Notify the NRC PM in advance ASAP

-Submit Relief Request (or TS Change) in accordance with 50.4 to support NRC determination under Para. (f) or (g)(6)(i)&(ii)

4. Types of ISI Relief Requests

- Relief from ASME Code under 10 CFR 50.55a
- Relief from Augmented Examinations under Bulletins, GLs, SEs, RG.
- Relief from Vessel Internals under BWRVIP /BWROG Guidelines

II. DISCUSSION: BASIS FOR RELIEFS

1. NRC Requirement for Approval

   - Determination of Acceptable Level of Quality and Safety

2. ASME Code Relief

   - Basis Exits or can be Provided

   - Seek NRC Approval of Code Cases

Issue: Can we use NRC Approved Other Licensee Docketed Info?
What level of info. Needed in the Basis?
Do we need a Relief Request at all?
Can we provide info on the Docket instead of a Relief?

3. Augmented Examination Relief

- Basis can be Developed using BWROG/ BWRVIP) or GE documents

- Use NRC Generic SERs on BWROG/ BWRVIP Documents

- Use Precedents (Previously Approved on Other Dockets)

- Resolve with the NRC on a case by case basis

Issue: Can we use applicable partial BWROG or industry Documents to develop Basis for Relief?

Can we use previously approved Relief Requests on other Dockets for the Basis?

4. Relief from Vessel Internal Examinations

- ASME Code is silent on Examinations
- Examination are derived through BWRVIP documents
- BWRVIP documents are in NRC Review/Approval Process
- NRC Issues Generic SERs requiring examinations

Issue: Do we need relief from Generic NRC SER requirements?
Are the Generic SER requirements binding on licensees?
Can we describe the extent of examinations in the Refueling Outage Plan submittals?
Do we need NRC approval prior to the outage?

HAT WE HAVE DONE

- We have successfully transmitted digitally signed documents to participants in the low level waste hearing pilot.
- We have corrected the Internet Explorer problem.
- We have designed the ADAMS processing module for electronic submittals.

HAT WE HAVE DONE

- We have developed and implemented a system for the issuance of digital certificates.
- We have defined system requirements.
- We are completing the contracting process for implementing the production system.

WHAT WE NEED TO DO

- Do some programming to allow for multiple signatures, pull down bars for form information, access control, upgrade the EIE server and other minor programming tasks.
- Complete the Instruction and Procedures Guide.
- Install the Electronic Submittal production system.

WHAT WE NEED TO DO

- *Begin the three plant pilot.*
- *Issue a blanket exception citing 50.4(c) for each licensee to begin submitting electronically for a plant if they so wish.*
- *Validate the process.*
- *Build the encryption process.*

THE RULES

- *Participation is voluntary.*
- *Three acceptable formats:*
  - *PDF*
  - *Word Perfect*
  - *Word*
- *Initially a 5 Mb limit unless we are notified in advance.*

THE RULES

- *You may be asked to resend documents for a variety of reasons.*
- *No proprietary , privacy or safeguards until we get the encryption module in place.*

SCHEDULE

Although it sounds like a hedge, we <u>ARE</u> going to implement <u>ASAP.</u>

■ Three plant pilot	Mid February
■ Part 50 submittals	Mid April
■ Rule Change	Late summer

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