211

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

December 10, 1991

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NUMBER 211

The Committee to Review Generic Requirements (CRGR) met on Tuesday, November 26, 1991 from 8:00 a.m. to 1:30 p.m. A list of attendees at the meeting is enclosed (Enclosure 1). The following items were discussed at the meeting:

1. C. Grimes and T. Dunning of NRR presented for CRGR review a generic letter on relaxing technical specification surveillance intervals involving surveillance testing during power operation. The CRGR supported the letter; however, the CRGR suggested redrafting the letter and the associated NUREG document. The staff agreed to address the CRGR comments in a revised draft, which the CRGR planned to review quickly, by negative consent if possible. This matter is discussed in Enclosure 2.

2. T. King and J. Flack of RES provided a briefing on the status and results of its reviews of licensee submittals regarding individual plant examinations (IPE's). The CRGR requested an information copy of the final review procedures when they are completed. This matter is discussed in Enclosure 3.

3. The CRGR discussed a draft memorandum to the Executive Director for Operations describing the CRGR's experience in applying the criteria of the backfit rule. The CRGR provided comments which the CRGR staff planned to address in a revised draft. This matter is discussed in Enclosure 4.

4. The CRGR briefly discussed comments that Commissioner Curtiss had received from the licensee during a visit to the Susquehanna plant. The comments expressed concerns about new generic requirements and staff positions. It was noted that the staff was preparing a response to Commissioner Curtiss' questions related to the concerns expressed.

5. J. Richardson, B. Elliott and K. Wichman of NRR presented for CRGR review a revised generic letter on reactor vessel structural integrity. (The letter had been redrafted since it was previously reviewed at Meeting No. 210 on November 12, 1991.) The CRGR recommended in favor of the letter, subject to some revisions to be coordinated with the CRGR staff. This matter is discussed in Enclosure 5.

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6/6/94
index date

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Questions concerning these meetings minutes should be referred to Dennis Allison (492-4148).

Original Signed by:
E. L. Jordan

Edward L. Jordan, Chairman
Committee to Review Generic Requirements

Enclosures:
As stated

cc w/encl:
Commission (5)
SECY
J. Lieberman
P. Norry
D. Williams
W. Parler
Regional Administrators
CRGR Members

Distribution:
Central File w/o encl.
S. Treby
M. Taylor
J. Heltemes
T. King
G. Mizuno
K. Wickman
E. Rossi
T. Dunning
D. Ross
D. Allison
CRGR S/F

PDR (NRC/CRGR) w/o encl.
P. Kadambi
J. Sniezek
W. Minners
W. Beckner
J. Richardson
B. Elliott
C. Grimes
J. Flack
E. Jordan
J. Conran
CRGR C/F

CRGR:AEOD	DD:AEOD	C/CRGR:AEOD
DAllison:slm	DRoss	EJordan
12/6/91	12/ /91	12/9/91

ENCLOSURE 1

ATTENDANCE LIST

CRGR MEETING NO. 211

November 26, 1991

CRGR Members

E. Jordan
F. Miraglia
G. Arlotto
J. Callan
J. Moore
J. Murphy (for B. Sheron)

CRGR Staff

D. Allison
J. Conran

NRC Staff

C. Grimes
T. Dunning
R. Lobel
T. Tjader
M. Reinhardt
R. Emch
W. Beach
M. Schwartz
T. Stetla
E. Chow
W. Beckner
J. T. Chen
C. Ader
J. Flack
R. E. Depriest
T. King
F. Congel
W. Minners
S. Mays
C. Petrone
P. Kadambi
M. Taylor
J. Tsao
C. Y. Chen

Enclosure 2 to the Minutes of CRGR Meeting No. 211
Draft Generic Letter on Relaxing Technical Specification
Surveillance Intervals Related to Surveillance Testing at Power

November 26, 1991

TOPIC

C. Grimes and T. Dunning of NRR presented the subject generic letter and an associated NUREG report for CRGR review. The letter would provide guidance on requesting line item technical specification improvements to reduce certain surveillance requirements related to testing during power operation. It was based on a staff study to determine which specific requirements would warrant relaxation.

BACKGROUND

The review packaged was forwarded by a memorandum for E. Jordan from F. Miraglia dated October 31, 1991. It included:

1. Draft generic letter
2. Draft NUREG-1366, "Improvements to Technical Specification Surveillance Requirements"
3. Model Safety Evaluation Report
4. CRGR review package (answers to CRGR Charter questions).

At the meeting, the staff also provided:

1. Revised page 18 of the enclosure to the generic letter. This is provided as Attachment 1 to this enclosure.

2. "CRGR Charter Considerations" which addressed new questions in the latest version of the CRGR Charter. This is provided as Attachment 2 to this enclosure.

CONCLUSIONS/RECOMMENDATIONS

The CRGR supported the action; however, the CRGR suggested redrafting the letter and the NUREG report. The primary issue was a need to better articulate the balance struck between the benefits of frequent surveillance and the drawbacks of frequent testing during power operation. The staff agreed to address the CRGR comments in a revised draft, which the CRGR planned to review quickly, by negative consent if possible.

Additional comments discussed included the following:

1. A basis for each specific change should be included. For example, in the discussion of pressurizer heaters beginning on page 41 of the NUREG report, the two brief paragraphs preceding the conclusion did not appear sufficient.

2. It would be desirable to attach to each conclusion a finding of no (significant) increase in risk. For example, in the sample analysis on page 6 of the NUREG report it was not obvious that there has been a finding of no undue risk.

3. It would be desirable in the executive summary of the NUREG report to discuss the Commission policy statement, criteria for technical specification improvement and previous studies related to this action.

4. It would be desirable to include a paragraph or two in the generic letter and the NUREG report indicating that, from among the many requirements that involve testing at power, very few were considered to warrant relaxation.

5. It would be desirable to further discuss the PRA work done in this area (related to standard technical specifications).

6. It would be desirable to discuss the benefits of rotating or running equipment to maintain lubricant distribution.

7. It should be stated early on that sometimes increasing a surveillance interval would reduce risk.

8. The name "qualitative risk assessment" was considered confusing.

9. The CRGR review package appeared to indicate that reducing wear was a primary or sole criterion reducing surveillance requirements. This should be re-examined.

10. When licensees address the specific changes, they should review their own experience. It may not be appropriate to take maximum advantage of the proposed changes.

11. The staff indicated that it would redraft certain sections to more clearly define future NRC actions. This applied in cases where a licensee submittal, to request a relaxation, would depend upon the completion of some further NRC study or action. The CRGR noted that it may be desirable to remove items which are not yet ready for action from the enclosure to the generic letter. They could be discussed in the body of the letter.

12. The generic letter should make a clear distinction between (1) what the staff would recommend in the NUREG report and (2) what would be authorized by the generic letter.

13. The conclusions on revised page 18 appeared to go too far in that they implied that the San Onofre 1 containment spray system would have performed its intended function in total. However, the issue under discussion was restricted to clogging and the adequacy of the spray pattern.

14. The staff indicated that it would address several additional editorial comments with the CRGR staff. This included items such as a need to update certain sections of the NUREG report to reflect the current status of NRC programs (e.g., diesel generator reliability testing).

BACKFITTING

The staff noted that the action (1) was voluntary and (2) would constitute a relaxation in requirements. Thus, it was not considered neither a backfit as defined in 10 CFR 50.109 nor a request for information as defined in 10 CFR 50.54(f). The CRGR accepted these determinations without any further questions.

SAFETY GOALS

The staff noted that the action (1) would be voluntary (2) would constitute a relaxation in requirements and (3) would be expected to enhance safety. Thus, the action would be consistent within the safety goals. The CRGR accepted these determinations without any questions.

(7.6, cont.)

The following condition must be met and addressed to justify the use of this approach:

> A justification is required that the measurement of the boron concentration in the boric acid storage tank verifies the boron concentration in the BIT.

3/4.5.4 Boron Injection System - Boron Injection Tank, [W STS] TS 4.5.4.1:

The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days,

b. Verifying the boron concentration of the water in the tank <u>by measuring the boron concentration in the boric acid storage tank</u> once per 7 days, and

(Added clarification of where measurement is made.)

c. Verifying the water temperature at least once per 24 hours.

(No change for item c.)

<u>8.1 Containment Spray System (PWR)</u>

Recommendation: The surveillance interval (air or smoke flow test) should be extended to 10 years.

Recent Experience: On June 11, 1991, Southern California Edison Company reported that a containment spray system (CSS) air flow test for San Onofre Unit 1 indicated blockage of several nozzles. An investigation revealed that seven nozzles were clogged with sodium silicate, a coating material that was applied to the carbon steel CSS piping in 1977. Subsequent air flow tests conducted in 1980, 1983, and 1988 were acceptable. The licensees analysis confirmed that the CSS was capable of performing its safety function under the as found condition. The staff concludes that this event should not preclude an extension of the air flow test surveillance interval for plants with the more commonly used stainless steel piping system. Plants using carbon steel piping should justify a change in the surveillance interval in light of the San Onofre experience.

3/4.6.2 Depressurizing and Cooling Systems - Containment Spray System, [CE STS (Typ)] TS 4.6.2.1:

Each Containment Spray System shall be demonstrated OPERABLE:

d. At least once per <u>10</u> years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

(Changed the surveillance interval from "5" to "10" years.)

ATTACHMENT 1 ENCL. 2
MEETING 211

ROUTING AND TRANSMITTAL SLIP

Date 5-24-94

TO: (Name, office symbol, room number, building, Agency/Post)	Initials	Date
BCS P1 37		
2. PDR		
3.		
4.		
5.		

Action	File	Note and Return
Approval	For Clearance	Per Conversation
As Requested	For Correction	Prepare Reply
Circulate	For Your Information	See Me
Comment	Investigate	Signature
Coordination	Justify	

REMARKS

This previous Central File material can now be made publicly available.

MATERIAL RELATED TO CA6R MEETING NO. 211

CC (LIST ONLY) JEAN RATAJE, PDR L STREET

DO NOT use this form as a RECORD of approvals, concurrences, disposals, clearances, and similar actions

FROM: (Name, org. symbol, Agency/Post)	Room No.—Bldg.
DENNIS ALLISON	Phone No. 24148

5041-102

OPTIONAL FORM 41 (Rev. 7-76)
Prescribed by GSA
FPMR (41 CFR) 101-11.206

910308

DF03 1/1

SENT TO PDR ON

MATERIAL RELATED TO CRGR MEETING NO. 211
TO BE MADE PUBLICLY AVAILABLE

1. MEMO FOR J. TAYLOR FROM E. JORDAN DATED 12-10-91
SUBJECT: MINUTES OF CRGR MEETING NUMBER 211
INCLUDING THE FOLLOWING ENCLOSURES WHICH WERE NOT PREVIOUSLY RELEASED:

a. ENCLOSURE 2
A SUMMARY OF DISCUSSIONS OF A PROPOSED GL on Relaxing TS Surveillance Intervals Related to Surveillance Testing @ Power

b. ENCLOSURE 3
A SUMMARY OF DISCUSSIONS OF A PROPOSED Status and Results of the NRC Staff's Review of Licensee Submittals Regarding Individual Plant Examinations

c. ENCLOSURE 4
A SUMMARY OF DISCUSSIONS OF A PROPOSED Memorandum to the Executive Director for Operations Regarding CRGR Experience in Applying the Backfit Rule

2. MEMO FOR E. JORDAN FROM F. Miraglia DATED 10-31-91
FORWARDING REVIEW MATERIALS ON A PROPOSED Improvements to Technical Specifications Surveillance Requirements

3. MEMO FOR E JORDAN FROM W. Minners DATED 10-31-91
FORWARDING REVIEW MATERIALS ON A PROPOSED Review of Seabrook IPE Submittal - Internal Events

4. MEMO FOR E JORDAN FROM ______ DATED ______
FORWARDING REVIEW MATERIALS ON A PROPOSED

______

d. Enclosure 5
A Draft GL on Reactor Vessel Structural Integrity

CRGR CHARTER CONSIDERATIONS

Item 1 asks the question if the objective can be achieved by setting a readily quantifiable standard that has an unambiguous relationship to a readily measureable quantitiy and is inforceable.

The changes in the surveillance intervals were proposed primarily based upon operating experience and engineering judgement. These would be difficult to quantify in terms of a readily measurable standard.

Risk Based TS would be responsive to this approach, but are a future consideration.

The guidance provided is for the most part straight forward and indicates the changes to the wording of the Standard TS to effect a change in the specified surveillance intervals. The wording may be different for plants that have TS that follow a custom TS format.

Item 4 address curcurrences by Program Offices:

TS improvements are being carried out by NRR which is the applicable program office. All technical divisions within NRR have concurred with this proposal. Comments on an earlier draft of the NUREG were solicited from other program offices. Specifically, RES provided extensive on an earlier draft of the NUREG. Also, data were requested from AEOD staff.

ATTACHMENT 2, ENC 2
MEETING 211

Items 7 & 9 address clarification of Backfits.

Backfit considerations are not applicable since action by licensees is voluntary for proposing TS changes.

Subpart J under item 7 addresses prioritization and scheduling in light of of other ongoing activities.

Generic letters on TS improvements are provided to project managers in a memorandum with guidance on processing license amendment requests. This guidance addresses the prioritization of the licensing action as Priority 3 items in the NRR priority ranking system for reviews. This priority applies to TS changes that are not needed to correct a safety problem or support continued plant operation or prevent a derate.

Item 11 clarifies information requested under 50.54(f)

The only information requested (and not under 50.54(f)) is related to the licensees time and cost to prepare an amendment request to implement proposed TS changes and an estimate of the long term costs that would be saved by the proposed action. There is an affirmative statement that any action taken is voluntary and not a backfit under 50.109. However, the CRGR staff suggested that the generic letter should state that the action is not a request for information under 50.54(f), and we have no objection to this clarification since some information is being requested from licensees that propose TS changes.

Item 12 address an assessment of how the porposed action relates to the Commission Safety Goal.

We judge the changes in surveillance intervals to be a net benefit to safety and consistent with the objectives of the safety goal. However, they are voluntary actions which we do not deem are necessary to satisfy the safety goal.

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Enclosure 3 to the Minutes of CRGR Meeting No. 211
Briefing on the Status and Results of the
NRC Staff's Review of Licensee Submittals
Regarding Individual Plant Examinations (IPE'S)

November 26, 1991

TOPIC

T. King and J. Flack of RES provided the subject briefing. Several licensee submittals had been received and, for some, the review had been completed. The staff discussed the schedules for additional submittals and reviews, the status of ongoing reviews and the results, to date, of the reviews.

Handouts used by the staff in its presentation are provided in Attachments 1 and 2 to this enclosure.

BACKGROUND

The CRGR was provided with the following background information prior to the briefing:

1. Part of a Commission paper, SECY-91-359, dated November 7, 1991, subject: Status of Implementation Plan for Closure of Severe Accident Issues and Status of Individual Plant Examination.

2. Memorandum dated June 3, 1991 for R. Wessman from C. Ader and W. Beckner, subject: Questions on Seabrook Individual Plant Examination (IPE) Submittal.

3. Memorandum dated October 31, 1991 for F. Congel from W. Minners, subject: Review of Seabrook Individual Plant Examination (IPE) Submittal - Internal Events.

CONCLUSIONS/RECOMMENDATIONS

There were no recommendations to the EDO as a result of this briefing.

The staff noted that the first few reviews were conducted using a draft review procedure and the procedure would be finalized in the near future. The CRGR requested an information copy of the final procedure when it is available and indicated that the CRGR may request a presentation that time. However, the CRGR did not wish to review the procedure before it is approved.

INDIVIDUAL PLANT EXAMINATION PROGRAM

11/26/91

ATTACHMENT 1 TO ENC. 3
MEETING # 211

Individual Plant Examination (IPE) Program

Background

- o IPE consists of two parts:

  - Internal events - initiated in 9/89 via GL-88-20, supplement 1
  - External events (IPEEE) - initiated in 7/91 via GL-88-20, supplement 4

- o Staff review of internal and external event IPE submittals consists of two steps:

  - Step 1 - screening review - all submittals
    - 4 person staff review teams look at each submittal

  - Step 2 - more detailed review - selected submittals
    - staff review team plus contractor assistance

- o Reviews to be documented via SER supplements

Status

- o All generic letters and guidance documents necessary for licensees to implement the program have been issued

IPE Program (Con't)

- Eight submittals received:

			Internal Events	External Events
-	Yankee Rowe	- 12/89	X	X
-	Millstone 3	- 9/90	X	X
-	Oconee 1, 2, and 3	- 11/90	X	X
-	Seabrook	- 3/91	X	X
-	Turkey Point 3 and 4	- 6/91	X	X
-	Surry 1 and 2	- 8/91	X	
-	Fitzpatrick	- 9/91	X	
-	McGuire 1 and 2	- 11/91	X	X

- Most of remaining internal event submittals due in FY 1992
- Licensee schedules for external event submittals due - 12/91

IPE PROGRAM (CON'T)

o EXPECTED COMPLETION DATES FOR STAFF REVIEW OF INTERNAL EVENT SUBMITTALS (I.E. DRAFT SERs PREPARED). ALL REVIEWS ARE STEP 1 EXCEPT TURKEY POINT WHICH IS STEP 2

- YANKEE ROWE	-	COMPLETE
- MILLSTONE 3	-	2/92
- OCONEE 1, 2, AND 3	-	4/92
- SEABROOK	-	COMPLETE
- TURKEY POINT 3 AND 4	-	2/92
- SURRY 1 AND 2	-	5/92
- FITZPATRICK	-	7/92
- MCGUIRE 1 AND 2	-	TBD

o EXTERNAL EVENT REVIEWS TO BE SCHEDULED AFTER STAFF REVIEW PLAN IS DRAFTED.

IPE PROGRAM CON'T

- o PER AGREEMENT WITH NRR, COMPLETE RESPONSIBILITY FOR IPE REVIEWS PICKED UP BY RES. NRR MANAGEMENT TO BE PROVIDED WITH PERIODIC STATUS REPORTS

- o LICENSEE REPORTED INFORMATION FROM SUBMITTALS RECEIVED TO DATE SUMMARIZED IN ATTACHED TABLE.

- o STAFF VIEWS TO DATE ON SUBMITTALS RECEIVED (INTERNAL EVENTS ONLY):
  - YANKEE ROWE - GENERALLY ACCEPTABLE. WEAK IN HUMAN FACTORS AREA. OPEN ITEM ON H^2 DETONATION
  - MILLSTONE 3 - GENERALLY ACCEPTABLE. WEAK IN HUMAN FACTORS AREA.
  - OCONEE 1, 2, AND 3 - GENERALLY ACCEPTABLE. WEAK IN HUMAN FACTORS AREA. PROPOSED GSI RESOLUTIONS NOT WELL DOCUMENTED.
  - SEABROOK - ACCEPTABLE
  - TURKEY POINT 3 AND 4 - TBD
  - SURRY 1 AND 2 - TBD (VULNERABILITY REPORTED TO NRR)
  - FITZPATRICK - TBD
  - MCGUIRE 1 & 2 - TBD

- o TECHNICAL ASSISTANCE COMMERCIAL CONTRACTS ARE IN PLACE TO SUPPORT INTERNAL EVENT STEP 2 REVIEWS.

- o TECHNICAL ASSISTANCE COMMERCIAL CONTRACTS TO SUPPORT STEP 2 EXTERNAL EVENT REVIEWS TO BE PLACED IN FY 92.

- o REVIEW OF NUMARC/EPRI ALTERNATE FIRE EVALUATION METHODOLOGY COMPLETE.

INFORMATION FROM IPEs SUBMITTED TO DATE*

	YANKEE ROWE	MILLSTONE 3	OCONEE 1,2,3	SEABROOK
o OVERALL CORE DAMAGE FREQUENCY	2×10^{-5}**	7×10^{-5}	1×10^{-4}	1.1×10^{-4}
o MAJOR CONTRIBUTORS TO CDF	o INTERNAL EVENTS - LOCAs (74%) - ATWS (8%) - SGTR (5%) - ISLOCA (1%) - LOOP (4%) - OTHER (8%)	o INTERNAL EVENTS (80%) - LOCAs (30%) - ATWS (5%) - STGR (2%) - LOOP (8%) - OTHER (35%) o - EXTERNAL EVENTS (20%) - SEISMIC (13%) - FIRE (7%)	o INTERNAL EVENTS (21%) - LOCAs (9%) - LOOP (4%) - OTHER (8%) o - EXTERNAL EVENTS (79%) - SEISMIC (45%) - FIRE (20%) - TORNADO (9%) - FLOODS (5%)	o INTERNAL EVENTS (55%) - LOCAs (4%) - ATWS (5%) - STATION BLACKOUT (19%) - LOSS OF COMPONENT COOLING (19%) - OTHER (7%) o EXTERNAL EVENTS (45%) - SEISMIC (13%) - FIRE (24%) - FLOOD (5%) - OTHER (3%)
o CONDITIONAL CONTAINMENT FAILURE	NOT PROVIDED	o NO CONTAINMENT FAILURE (69%) o CONTAINMENT FAILURE (31%) - LATE FAILURE (25%) - BASEMAT MELTTHRU (6%) - OTHER < 1%)	o NO CONTAINMENT FAILURE (60%) o CONTAINMENT FAILURE (40%) - LATE FAILURE (3%) - BASEMAT MELTTHRU (31%) - ISOLATION FAILURE (6%) - OTHERS (< 1%)	o NO CONTAINMENT FAILURE (20%) o CONTAINMENT FAILURE (80%) - LATE FAILURE (80%) - SMALL EARLY FAILURE/ BYPASS (14%) - LARGE EARLY FAILURE/ BYPASS (1%)
o UNRESOLVED GSIs PROPOSED FOR RESOLUTION	NONE	NONE	3***	NONE
o PLANT/PROCEDURE MODIFICATIONS PROPOSED AS A RESULT OF IPE	REDUCE SUSCEPTABILITY OF SWITCHGEAR ROOM TO FLOODING	NONE****	NONE****	NONE****

* All values are Licensee reported values.
** Internal events only.
*** - GI-23 Reactor coolant pump seal failures
- GI-105 - ISLOCA
- GI-130 - Essential service water pump failure at multiplant sites
**** Since the submittal is based upon a previously performed PRA, licensees have indicated that appropriate changes were made prior to the IPE.

INFORMATION FROM IPEs SUBMITTED TO DATE*

	TURKEY POINT 3	SURRY 1&2	FITZPATRICK
o **OVERALL** CORE DAMAGE FREQUENCY	$1.0E^{-4}$/YR	$7.4E^{-5}$/YR** $9.9E^{-5}$/YR-Flood	$1.9E^{-6}$/YR**
o **MAJOR** CONTRIBUTORS TO CDF	o INTERNAL EVENTS(95%) - LOCAs (29%) - ISLOCAs (3%) - TI LOCAs(56%) - ATWS (2%) - SGTR (4%) - TRANSIENT (4%) o EXTERNAL(5%) INTERNAL FLOOD (1%) SURGE (< 1%) WIND (4%)	o INTERNAL EVENTS (w/o-Flood) - LOSS OF OFFSITE POWER (21%) - LOCA (29%) - INTERFACING SYS LOCA (2%) - TRANSIENTS (34%) - ATWS --- - SGTR (14%) EXTERNAL INTERNAL FLOOD ONLY - $1.1E^{-3}$ W/ MOD - $9.9E^{-5}$	o INTERNAL EVENTS - STATION BLACKOUT (91%) - TRANSIENTS (8%) - ATWS (<1.0%) - LOCAs W/LOSS OF ALL ECCS INJECTION (<1.0%)
o CONDITIONAL CONTAINMENT FAILURE	- CONT OK (32%) - CFL NO CCI (56%) - CFL CCI (12%)	W/FLOODING / W/OUT FLD. NO FAILURE (20%) (53.5%) CONTAINMENT BYPASS (4.5%) (16.7%) ALL EARLY FAILURES (1.2%) (.6%) LATE FAILURE (72.8%) (25.3%) MELT THROUGH (1.2%) (3.9%)	o NO FAILURE (39%) o FAILURE (61%)
o UNRESOLVED GSIs PROPOSED FOR RESOLUTION	NONE	o GI-23 "Pump Seal LOCA"	o USI A-47 "Safety Impl of Control Systems"
o PLANT/PROCEDURE MODIFICATIONS PROPOSED AS A RESULT OF IPE	o TO BETTER HANDLE A SURGE DURING A HURRICANE, AN IMPROVED PROCEDURE IS TO BE IMPLEMENTED BEFORE JUNE 1992. o CHARGING PUMP MODIFICATION MADE	IMPLEMENT STONE & WEBER MODIFICATIONS AND PROPOSED MODIFICATIONS TO MAKE CDF ~$9.9E^{-5}$/YR: -sump pump improvements -procedure improvements -inspection improvement	o NONE

INDIVIDUAL PLANT EXAMINATION (IPE) OVERVIEW

ATTACHMENT 2 TO ENC. 3 MEETING # 211

PRESENTATION TO THE
COMMITTEE TO REVIEW GENERIC REQUIREMENTS

November 26, 1991

John H. Flack
Severe Accident Issue Branch
Division of Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

INTRODUCTION

- o BACKGROUND
- o IPE OBJECTIVES AND SCOPE
- o LICENSEES SUBMITTAL PLANS
- o NRC REVIEW PLAN
- o PRELIMINARY FINDINGS
- o INSIGHTS DATA BASE
- o STATUS

BACKGROUND

- SEVERE ACCIDENT POLICY STATEMENT
  August 8, 1985

- INTEGRATION PLAN FOR CLOSURE OF
  SEVERE ACCIDENT ISSUES
  (SECY 88-147) May 25, 1988

- GENERIC LETTER 88-20,
  - Issued November 23, 1988
  - Supplement 1 August 29, 1989
    (Internal Events IPE)
  - Supplement 2 April 4, 1990
    (AM Strategies)
  - Supplement 3 July 6, 1990
    (CPI Recommendations)
  - Supplement 4 June 27, 1991
    (External Events IPEEE)

OBJECTIVES OF IPE PROCESS

- TO ACHIEVE AN OVERALL APPRECIATION OF SEVERE ACCIDENT BEHAVIOR
- TO IDENTIFY DOMINANT SEQUENCES
- TO OBTAIN A QUANTITATIVE UNDERSTANDING OF CORE DAMAGE AND RELEASE OF RADIOACTIVITY
- IF NECESSARY, REDUCE THE OVERALL PROBABILITIES OF CORE DAMAGE AND RELEASE OF RADIOACTIVITY

SCOPE (INTERNAL EVENTS)

REF. GENERIC LETTER 88-20
NUREG-1335

- o FRONT-END ANALYSIS
  - LEVEL I PRA
  - IDCOR IPEM WITH ENHANCEMENTS
  - OTHER
- o BACK-END ANALYSIS
  - LEVEL II PRA

SCOPE (INTERNAL EVENTS) - (CONTINUED)

- REFERENCE PLANT ANALYSIS CONSISTENT WITH APP.1 TO G.L. 88-20

o TREATMENT OF INITIATORS FROM FULL POWER OPERATION

o INTERNAL FLOOD

o EXTERNAL EVENTS (OPTIONAL)

o USIs AND GSIs (OPTIONAL)

o HEALTH EFFECTS (LEVEL III) NOT REQUIRED

ADDITIONAL EFFORTS FOR LICENSEES WITH FULL SCOPE PRAs

- LICENSEE'S INVOLVEMENT
- TREATMENT OF VULNERABILITIES
- EVALUATION OF POST PRA MODIFICATIONS
- TREATMENT OF MULTIPLE UNITS
- RESOLUTION USI A-45
- TREATMENT OF INTERNAL FLOOD
- CONTAINMENT PERFORMANCE IMPROVEMENTS
- PEER REVIEW

TREATMENT OF USIs AND GSIs

- METHODOLOGY SHOULD BE ABLE TO IDENTIFY VULNERABILITIES
- CONTRIBUTION TO CORE DAMAGE OR UNUSUALLY POOR CONTAINMENT PERFORMANCE
- TECHNICAL BASIS FOR RESOLVING THE ISSUE

TREATMENT OF USIs AND GSIs (CONTINUED)

ISSUES BEING ADDRESSED:

GI-23 REACTOR COOLANT PUMP SEAL FAILURE

GI-47 SAFETY IMPLICATIONS OF CONTROL SYSTEMS

GI-105 ISLOCA

GI-130 ESSENTIAL SERVICE WATER PUMP FAILURE AT MULTI-UNIT SITES

LICENSEE SUBMITTAL PLANS

- 45 SUBMITTALS (64 UNITS)

  - LEVEL I PRA
  - CONTAINMENT ANALYSIS CONSISTENT WITH G.L. 88-20 APP. 1

- 21 SUBMITTALS (32 UNITS)

  - LEVEL II PRAs

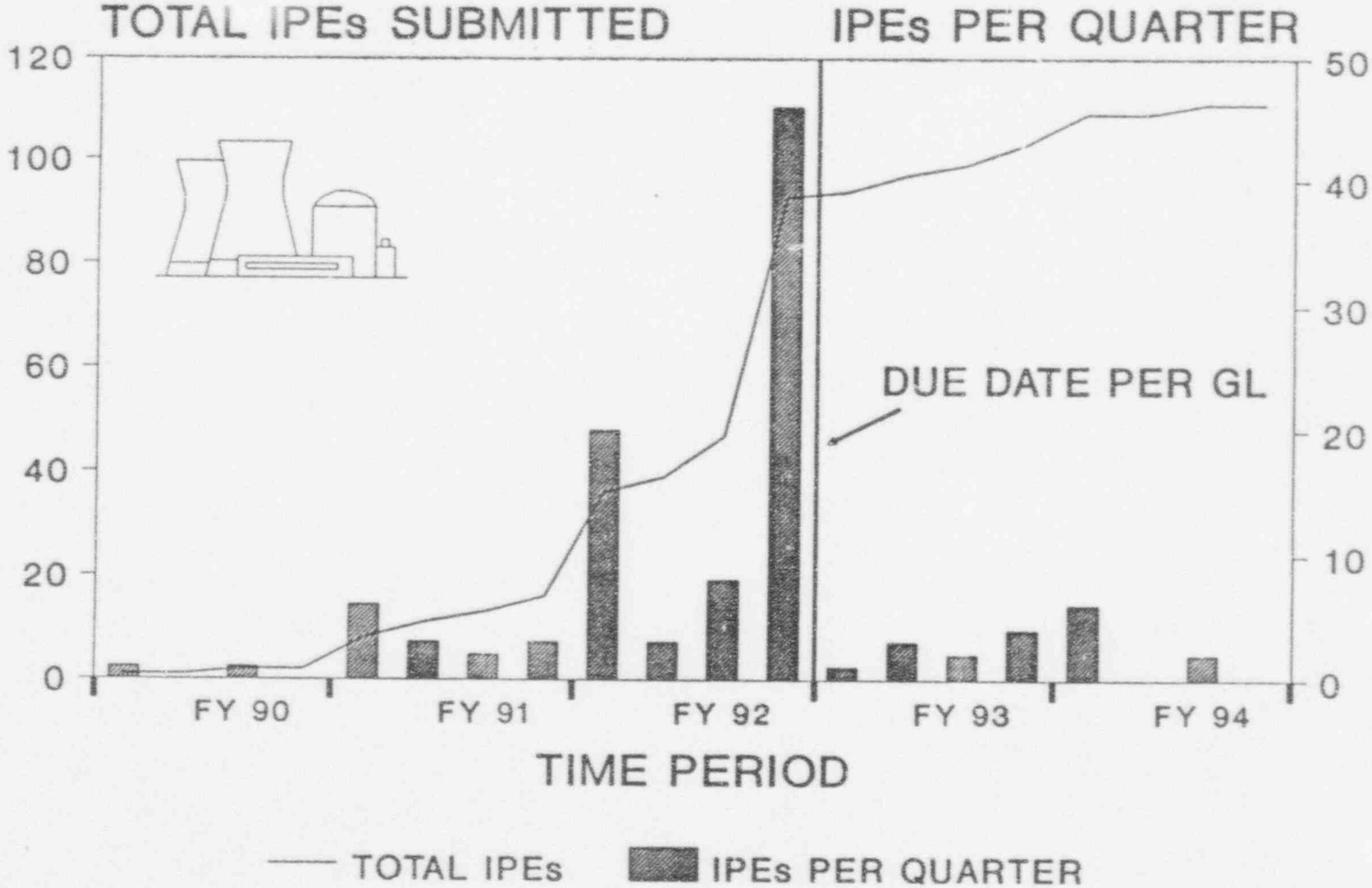
- 11 SUBMITTALS (16 UNITS)

  - LEVEL III PRAs

- 1 SUBMITTAL (2 UNITS)

  - INDEPENDENT METHODOLOGY

IPE SUBMITTAL SCHEDULE



SUBMITTALS RECEIVED

YANKEE ROWE	12/89
MILLSTONE 3	9/90
OCONEE 1,2,3	11/90
SEABROOK	3/91
TURKEY PT.	6/91
SURRY	8/91
FITZPATRICK	9/91
MCGUIRE	11/91

SUBMITTALS EXPECTED IN 1991

BEAVER VALLEY	11/91
MILLSTONE 1	11/91
BROWNS FERRY 1,2,3,	12/91
SUSQUEHANNA 1,2	12/91
CATAWBA	12/91
COOK 1,2	12/91
WATTS BAR	12/91

IPE REVIEW OBJECTIVES

(1) To determine whether the licensees' IPE process met the intent of Generic Letter 88-20
- Completeness
- Consistency with PRA experiences
- Consideration of CPI recommendations
- Utility participation
- Licensee in-house peer review
- Appropriate response to identified vulnerabilities

(2) To identify important insights and findings for data base storage
- gain generic insights

IPE REVIEW PLAN

- STEP 1 NRC TEAM REVIEWS
  - TEAM LEADER
  - FRONT-END SYSTEMS ANALYST
  - BACK-END CONTAINMENT/SOURCE TERM ANALYSTS
  - HUMAN RELIABILITY ANALYST

- NRC TEAM REPORT WHICH WILL IDENTIFY ANY:
  - INCONSISTENCIES
  - UNIQUE CHARACTERISTICS

- STEP 2 CONTRACTOR REVIEW
  - ENHANCE THE NRC TEAM'S EVALUATION OF THE LICENSEE'S IPE PROCESS

STEP 1 REVIEW PROCESS

- o SUBMITTAL REVIEW
- o GATHERING OF TECHNICAL INFORMATION
- o FORMULATION OF LICENSEE QUESTIONS
- o REVIEW OF LICENSEE RESPONSE
- o INTERACTION WITH LICENSEE
- o DETERMINATION OF STEP 2
- o STEP 2 (OPTIONAL)
- o REVIEW STEP 2 REPORT (OPTIONAL)
- o DEVELOP DRAFT SER
- o ISSUE SER

STEP 2 REVIEW PROCESS

THE CONTRACTOR TEAM WILL:

- INVESTIGATE FURTHER STEP 1 CONCERNS
- PERFORM A SITE VISIT TO:
  - (1) AUDIT TIER 2 INFO
  - (2) PERFORM WALKTHROUGH
  - (3) INTERVIEW KEY PERSONNEL
- SUBMIT TECHNICAL EVALUATION REPORT

PRELIMINARY FINDINGS

- INTERNAL FLOOD IMPORTANT CONTRIBUTOR
- IPEs ARE BECOMING LIVING DOCUMENTS
- LIMITED TREATMENT OF HRA
- SIGNIFICANT EFFORT
- CDF RANGES FROM 10[-4]/YR TO 10[-6]/YR

PRELIMINARY FINDINGS (CONTINUED)

- CONTRIBUTORS:

LOOP	(91% -> 4%)
LOCA	(74% -> 4%)
TRANSIENTS	(35% -> 4%)
SGTR	(14% -> 2%)
ATWS	(8% -> 1%)
ISLOCA	(4% -> 1%)

LICENSEE DEFINITION OF VULNERABILITIES

- o Risk significant contributor that might be missed absent a systematic search

  - cost benefit test
  - peer reviews
  - engineering appraisals

- o contribute more than 50% of total frequency for a given risk measure

  > 2E-4/yr CDF
  > 2E-6/yr Early Release

VULNERABILITIES (Continued)

- A failure (component, human error, maintenance action, procedure) that is significant greater than any other, i.e., a factor of three or more.

- Identify cost-effective improvements for functional sequences:

  > 10E-6/yr (core-melt frequency)

  > 10E-8/yr (containment by-pass)

YANKEE ROWE IPE

- o BASED ON A PSS COMPLETED IN 1983

- o BNL REVIEW (NUREG/CR-4589)

- o LICENSEE INVOLVED

- o PSS EMPLOYED TO IDENTIFY ENHANCEMENT OPPORTUNITIES

- o POTENTIAL ENHANCEMENTS UNDER CONSIDERATION:

  - primary depressurization capability

  - cavity injection

YANKEE ROWE (CONTINUED)

- PTS ISSUE BEING PURSUED ELSEWHERE
- LIMITED TREATMENT OF HUMAN FACTORS
- SEVERAL BACK-END ASSUMPTIONS NEED TO BE SUBSTANTIATED
- IN CONSIDERATION OF CPI RECOMMENDATIONS
  - limited modeling

YANKEE ROWE (CONTINUED)

- potential for local detonation not evaluated explicitly.

- o EXTENSIVE DHR CAPABILITY
- o IN GENERAL, MET THE INTENT OF G.L. 88-20
- o CPI ISSUE OPEN
- o LIVING DOCUMENT

SEABROOK IPE

- BASED ON A PSS COMPLETED IN 1983
- BNL REVIEW NUREG/CR-4552, LLNL REPORT
- LICENSEE INVOLVED
- PSS EMPLOYED TO IDENTIFY ENHANCEMENT OPPORTUNITIES
- POTENTIAL ENHANCEMENTS UNDER CONSIDERATION:
  - Automatic RCP Seal Injection Pump
  - Manual Charging Pump

SEABROOK (CONTINUED)

- Alternate AC Power Source
- Accident Management Strategies

- o VARIOUS TECHNIQUES USED TO TREAT HUMAN ERROR
- o CPI RECOMMENDATIONS ADDRESSED
- o NO DHR VULNERABILITIES
- o MET THE INTENT OF G.L. 88-20
- o LIVING DOCUMENT

IPE DATA BASE

OBJECTIVE: TO PROVIDE GENERIC INSIGHTS INTO:

- PLANT SPECIFIC ACCIDENT CONTRIBUTORS
- PLANT RESPONSE DURING OFF NORMAL CONDITIONS

DATA:

- DEPENDENCY MATRIX
- ACCIDENT SEQUENCES
- SUCCESS CRITERIA
- DOMINANT CONTRIBUTORS
- UNIQUE FEATURES

IPE REVIEW STATUS

78 IPE SUBMITTALS

2 IPE REVIEWS COMPLETED

5 IPE SUBMITTALS UNDER REVIEW

- 4 UNDER STEP 1

- 1 UNDER STEP 2

20 IPE SCHEDULES SLIPPED

cord#	PLANT	DATE	ORGDATE
1	YANKEE ROWE	12/20/89	/ /
2	MILLSTONE 3	09/10/90	/ /
3	OCONEE 1,2,3	11/30/90	/ /
4	SEABROOK 1	03/01/91	/ /
5	TURKEY POINT 3,4	06/30/91	/ /
6	FITZPATRICK	08/30/91	12/30/90
7	SURRY 1,2	08/31/91	/ /
8	BEAVER VALLEY 2	11/01/91	09/29/91
9	MILLSTONE 1	11/01/91	/ /
10	MCGUIRE 1,2	11/04/91	12/31/91
11	WATTS BAR	12/01/91	/ /
12	BROWNS FERRY 1,2,3	12/01/91	/ /
13	SUSQUEHANNA 1,2	12/15/91	06/30/91
14	CATAWBA 1,2	12/31/91	/ /
15	COOK 1,2	12/31/91	/ /
16	MONTICELLO	03/01/92	11/26/91
17	SAN ONOFRE 1	04/01/92	01/02/92
18	ZION 1,2	04/01/92	11/30/91
19	DIABLO CANYON 1,2	04/15/92	09/01/92
20	PILGRIM 1	04/30/92	/ /
21	DRESDEN 2,3	04/30/92	/ /
22	FERMI	04/30/92	/ /
23	PALO VERDE 1,2,3	04/30/92	/ /
24	NORTH ANNA 1,2	06/01/92	12/31/91
25	INDIAN POINT 2	06/30/92	/ /
26	GRAND GULF 1	06/30/92	/ /
27	HADDAM NECK	06/30/92	/ /
28	PERRY I	07/15/92	/ /
29	NINE MILE POINT 2	07/31/92	/ /
30	LIMERICK 1	07/31/92	/ /
31	LIMERICK 2	07/31/92	/ /
32	WATERFORD	08/28/92	03/31/92
33	SOUTH TEXAS 1,2	08/29/92	/ /
34	MAINE YANKEE	08/29/92	12/31/91
35	DUANE ARNOLD	08/31/92	/ /
36	COOPER	08/31/92	10/31/91
37	ANO 2	08/31/92	12/31/91
38	ANO I	08/31/92	12/31/91
39	ROBINSON 2	08/31/92	/ /
40	BRUNSWICK 1,2	08/31/92	/ /
41	DAVIS-BESSE	09/01/92	/ /
42	WNP-2	09/01/92	/ /
43	BEAVER VALLEY 1	09/01/92	09/29/91
44	FARLEY 1,2	09/01/92	/ /
45	PEACH BOTTOM 2,3	09/01/92	/ /
46	SAN ONOFRE 2,3	09/01/92	/ /
47	COMANCHE PEAK 1	09/01/92	/ /
48	INDIAN POINT 3	09/01/92	/ /
49	GINNA	09/01/92	/ /
50	SEQUOYAH 1,2	09/01/92	/ /

51	TROJAN	09/01/92	/ /
52	PALISADES	09/30/92	/ /
53	HATCH 1,2	09/30/92	/ /
54	OYSTER CREEK	09/30/92	/ /
55	WOLF CREEK	09/30/92	/ /
56	CALLAWAY	09/30/92	/ /
57	BIG ROCK POINT	09/30/92	/ /
58	BYRON 1,2	09/30/92	/ /
59	VOGTLE 1,2	09/30/92	/ /
60	CLINTON	09/30/92	/ /
61	CALVERT CLIFFS 1,2	09/30/92	/ /
62	KEWAUNEE	12/01/92	08/01/91
63	POINT BEACH 1,2	12/31/92	03/31/92
64	MILLSTONE 2	01/01/93	/ /
65	RIVER BEND 1	01/31/93	10/01/92
66	CRYSTAL RIVER 3	03/01/93	09/01/92
67	THREE MILE ISLAND 1	06/01/93	/ /
68	SUMMER	06/30/93	08/31/92
69	QUAD CITIES 1,2	06/30/93	/ /
70	SALEM 1,2	07/31/93	/ /
71	HOPE CREEK	07/31/93	/ /
72	NINE MILE POINT 1	07/31/93	/ /
73	HARRIS 1	08/31/93	/ /
74	BRAIDWOOD 1,2	10/31/93	/ /
75	FORT CALHOUN	12/01/93	/ /
76	VERMONT YANKEE	12/31/93	/ /
77	ST. LUCIE 1,2	12/31/93	/ /
78	PRAIRIE ISLAND 1,2	03/01/94	02/10/93
79	LA SALLE 1,2	06/30/94	/ /

Enclosure 4 to the Minutes of CRGR Meting No. 211
Discussion of draft Memorandum to the Executive Director for Operations (EDO) Regarding CRGR Experience in Applying the Backfit Rule

November 26, 1991

TOPIC

The CRGR discussed the subject memorandum , which had been drafted by the CRGR staff. The draft memorandum indicated that the backfit rule criteria had worked well enough and served their intended purpose; however, there had been difficulties and it was suggested, for the sake of discussion, that the backfit rule be revised.

The CRGR discussed the areas where difficulties had been experienced. Because the difficulties had been resolved within the framework of the current rule, the CRGR did not generally favor a revision to the backfit rule. However, it appeared that modification of the staff's guidance on backfitting would be appropriate in several instances.

The CRGR staff planned to redraft the memorandum to address the CRGR's comments.

CONCLUSIONS/RECOMMENDATIONS

No recommendations to the EDO resulted from this discussion.

<u>Enclosure 5 to the Minutes of CRGR Meeting No. 211</u>
<u>Draft Generic Letter on Reactor Vessel</u>
<u>Structural Integrity</u>

November 26, 1991

TOPIC

J. Richardson, B. Elliott and K. Wichman of NRR presented the subject letter for CRGR review. During the staff's review of neutron embrittlement in the Yankee Rowe reactor vessel, a concern was raised about compliance with the requirements of 10 CFR 50.60 and 10 CFR 50.61 and fulfilling commitments made in response to Generic Letter 88-11. The draft generic letter would request information under 10 CFR 50.54(f) to determine the extent to which all licensees of nuclear power plants were complying with such requirements and commitments.

The letter had been previously considered by the CRGR staff at Meeting No. 210. Subsequently, the staff redrafted the letter to address CRGR comments. Primarily, this involved narrowing the scope of the information requested and clarifying the reasons for requesting the information.

A redrafted letter provided prior to the meeting is included as Attachment 1 to this enclosure. Another redrafted letter, provided at the meeting, is included as Attachment 2 to this enclosure.

BACKGROUND

The original review package and other background information were documented in the minutes of Meeting No. 210.

CONCLUSIONS/RECOMMENDATIONS

The CRGR recommended in favor of the letter, subject to some revisions to be coordinated with the CRGR staff.

Specific comments discussed included the following:

1. The staff assured the CRGR that it was asking only for information to verify compliance with existing rules and commitments. It was not seeking to impose any new requirements or positions via this generic letter.

2. The staff would continue the editing process in consultation with the CRGR staff and a technical editor in order to prepare the final letter.

3. The language should generally say

   a. "licensees are requested to..." rather than "licensee shall...". This reflects the language of 10 CFR 50.54(f).

b. "With respect to Appendix H ... provide ..." rather than "Provide ... to demonstrate compliance with Appendix H".

c. "Report predicted charpy upper shelf ... as of December 16, 1991" rather than "... on December 16, 1991."

d. In the first paragraph on page 6, the term "screening criteria" should be changed to something else.

BACKFITTING AND SAFETY GOALS

As discussed in the Minutes of Meeting No. 210, the CRGR agreed with the staff's determinations that the generic letter would not be a backfit and the safety goals would not be material, subject to narrowing the information requested, which had been done.

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS (EXCEPT YANKEE ATOMIC ELECTRIC COMPANY, LICENSEE FOR THE YANKEE-ROWE NUCLEAR POWER STATION)

SUBJECT: REACTOR VESSEL STRUCTURAL INTEGRITY
(GENERIC LETTER 91-)

Purpose

In Section 50.60(a) of Title 10 of the Code of Federal Regulations (10 CFR 50.60(a)), the U.S. Nuclear Regulatory Commission (NRC) requires that licensees for all light water nuclear power reactors meet fracture toughness requirements and have a material surveillance program for the reactor coolant pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. Pursuant to 10 CFR 50.60(b), where the requirements of Appendices G and H to 10 CFR Part 50 cannot be met, an exemption is necessary pursuant to 10 CFR 50.12. Further, there are fracture toughness requirements in 10 CFR 50.61 for protection against pressurized thermal shock events for pressurized water reactors. In addition, licensees and permittees have made commitments in response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," to use the methodology in Regulatory Guide 1.99, Revision 2, to predict the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The 10 CFR 50.60 and 10 CFR 50.61 requirements and Generic Letter 88-11 are in the overall regulatory program to maintain the structural integrity of the reactor vessel. This generic letter is part of a program to evaluate reactor vessel integrity and take regulatory actions, if needed, to ensure that licensees and permittees are complying with 10 CFR 50.60 and 10 CFR 50.61, and are fulfilling commitments made in response to Generic Letter 88-11. Additional discussion of the applicable regulatory requirements is contained in Enclosure 1 to this letter. The NRC is requiring information on compliance under the provisions of 10 CFR 50.54(f).

ATTACHMENT 1
TO ENCLOSURE 5
MEETING # 211

Assessment of Embrittlement for the Yankee-Rowe Nuclear Power Station Reactor Vessel

Recent NRC concerns regarding the neutron embrittlement of the Yankee Rowe reactor vessel prompted the staff to perform a safety assessment of the Yankee Rowe reactor vessel. The staff found that the licensee for Yankee Rowe might not be in compliance with 10 CFR 50.60 and had not properly completed the assessment required in 10 CFR 50.61. Further, the licensee for Yankee Rowe had incorrectly applied the methodology in Regulatory Guide 1.99, Revision 2.

The staff found that the Charpy upper shelf energy of the Yankee Rowe reactor vessel material could be as low as 35.5 ft-lb, which is less than the 50 ft-lb value required in Appendix G to 10 CFR Part 50. However, the licensee for Yankee Rowe had not performed the required actions in Paragraph IV.A.1 or V.C of Appendix G to 10 CFR Part 50. Since then, the licensee for Yankee Rowe has performed an analysis in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 using criteria being developed by ASME to demonstrate margins of safety equivalent to those in the ASME Code.

While reviewing the surveillance program for the Yankee Rowe reactor vessel, a concern was identified regarding compliance with the requirements of Appendix H to 10 CFR Part 50. ASTM E 185 requires that the licensee take sample specimens from actual material used in fabricating the beltline of the reactor vessel. These surveillance materials shall include one heat of base metal, one butt weld, and one weld "heat affected zone." The licensee for Yankee Rowe terminated the material surveillance program in 1965. Therefore, Yankee Rowe had no material surveillance program on July 26, 1983, when Appendix H to 10 CFR Part 50 became effective. Further, the samples irradiated at Yankee Rowe before 1965 were comprised only of base metal.

The licensee for Yankee Rowe had used the methodology in Regulatory Guide 1.99, Revision 2 to predict the effects of neutron embrittlement. However, the staff found that the methodology in Regulatory Guide 1.99, Revision 2 was incorrectly applied by the licensee. The specific issues were (1) heat treatment of reactor vessel material, (2) the irradiation temperature, (3) chemistry composition of reactor vessel material, and (4) results of material surveillance program.

The licensee reported that the heat treatment of the Yankee Rowe vessel resulted in a coarse grain structure that may affect the sensitivity of the material to neutron irradiation. The irradiation temperature at Yankee Rowe is between 454 ^{o}F and 520 ^{o}F, which is below the nominal irradiation temperature of 550 ^{o}F used in developing Regulatory Guide 1.99, Revision 2. A lower irradiation temperature increases the effect of neutron embrittlement. The chemical composition of the reactor vessel welds is unknown. The material sensitivity to neutron embrittlement depends on its chemical content. The limited results of the surveillance program from Yankee Rowe indicated that the shift in the reference temperature exceeds the mean-plus-two standard deviations as predicted by the procedures in Regulatory Guide 1.99, Revision 2.

The staff also found that the licensee for Yankee Rowe had not considered plant-specific information in assessing compliance with 10 CFR 50.61. Although the surveillance data indicate neutron embrittlement exceeding that being projected and the Yankee Rowe plant operates at a low temperature, the licensee for Yankee Rowe had not taken these into account. When plant-specific information is considered, the Yankee Rowe reactor vessel may have exceeded the screening criteria in 10 CFR 50.61. Since then, the licensee for Yankee Rowe has performed a probabilistic fracture mechanics analysis in accordance with 10 CFR 50.61(b)(4) and the staff is continuing its review.

Based on the staff experience with the Yankee Rowe review, the staff has a concern that this may not be an isolated case regarding compliance with 10 CFR 50.60 and 10 CFR 50.61 and regarding fulfillment of commitments made in response to Generic Letter 88-11. Thus, the staff is issuing this generic letter to obtain information to assess compliance with these regulations and fulfillment of commitments. The staff is continuing to pursue this concern with the Yankee Atomic Electric Company. Therefore, the Yankee Atomic Electric Company need not respond to this generic letter.

Actions Covered by this Generic Letter

The NRC regulations require that all addressees have reactor vessel material surveillance programs in accordance with Appendices G and H to 10 CFR Part 50 or have obtained an exemption pursuant to 10 CFR 50.12. Addressees are also required to meet 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events."

Addressees shall perform the following actions to demonstrate compliance with Appendix H to 10 CFR Part 50:

Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 or do not have an integrated surveillance program approved by the NRC, shall describe actions to be taken to assure compliance with Appendix H to 10 CFR Part 50. For addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50, they shall indicate when the revised program will be submitted to the NRC Staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees shall indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b) and to provide their technical justification.

Addressees shall perform the following actions to demonstrate compliance with Appendix G to 10 CFR Part 50:

Addressees whose Charpy upper shelf energy is predicted to be less than 50 ft-lb at the end of its license using the guidance in Paragraph C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2 shall provide to the NRC the predicted Charpy upper shelf energy for the limiting beltline weld and the plate or forging at the end of the current license term and on December 16,1991 and describe the actions taken pursuant to Paragraph IV.A.1 or V.C or Appendix G to 10 CFR Part 50.

Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition shall describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

(a) the results from all Charpy and drop weight tests for all unirradiated beltline materials

(b) the heat treatment received by all beltline and surveillance materials

(c) the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricated each beltline weld

(d) the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld

(e) the chemical composition, in particular the weight percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material

(f) the identity of the heat of wire used for determining the weld metal chemical composition if different than Item (c) above

In committing to Generic Letter 88-11, licensees have committed to calculate radiation embrittlement in accordance with the procedures documented in Regulatory Guide 1.99, Revision 2 and to account for radiation embrittlement for core critical operation with irradiation temperatures less than 525°F. This guide indicates that the procedures are applicable for a nominal irradiation temperature of 550°F and operation below 525°F is considered to produce greater embrittlement. To determine whether plants have operated for significant time with the core critical and with an irradiation temperature less than 525°F, licensees whose reactor vessels are predicted at the end of the current license term to accumulate a neutron fluence greater than 10^{16} n/cm^2 (E>1MeV) during the portion of operation with irradiation temperatures less than 525°F, shall report the amount of time, the neutron fluence and the irradiation temperature above and below 525°F. The applicable neutron fluence for BWR plants are the values at the ¼T location and for PWR plants are the values at the inside of surface. Fluence of 10^{16} does not indicate that a revision to the pressurized thermal shock analysis or pressure-temperature curves are needed. It is intended as a screening criteria to gather information for further review. These licensees shall describe the consideration given to determining the effect of lower irradiation temperature on the reference temperature and Charpy upper shelf energy.

In committing to Generic Letter 88-11, licensees have committed to compare the results of the surveillance capsule with the values predicted by Regulatory Guide 1.99, Revision 2. If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee shall report the information and describe the effect of the surveillance results on the adjusted reference temperature and predicted Charpy upper shelf energy for each beltline material at the end of the current license and on December 16, 1991.

Reporting Requirements

Pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), each addressee shall submit a letter within 90 days of the date of this generic letter providing the information described under "Actions Covered by this Generic Letter." The letter shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation. In addition, a copy shall be submitted to the appropriate Regional Administrator. This generic letter requests information that will enable the NRC to verify licensee compliance with its current licensing basis with respect to reactor vessel fracture toughness and material surveillance for the reactor coolant pressure boundary. Accordingly, an evaluation justifying this information request is not necessary under 10 CFR 50.54(f).

Backfit Discussion

This generic letter requests information that will enable the NRC Staff to determine whether licensees are complying with their prior commitments and/or license conditions with respect to 10 CFR 50.60, 10 CFR 50.61, and Generic Letter 88-11. The staff is not establishing a new position with respect to such compliance in this generic letter. Because the staff is requesting information to verify licensee compliance with their previously established commitments and is not establishing any new staff position, this generic letter does not constitute a backfit. Thus, 10 CFR 50.109 does not apply and no backfit analysis need be prepared.

Request for Voluntary Submittal of Impact Data

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires May 31, 1994. The estimated average number of burden hours is 100 person hours for each addressee's response, including the time required to assess the requirements, search data sources, gather and analyze the data, and prepare the required letters. These estimated average burden hours pertain only to the identified response-related matters and do not include the time to implement the actions required by the regulations. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to Ronald Minsk, Office of Information and Regulatory Affairs (3150-0011), NEOB-3019, Office of Management and Budget, Washington, D.C. 20503, and to the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, Division of Information Support Services, Office of Information and Resources Management, Washington, D.C. 20555.

Although no specific request or requirement is intended, the following information would be helpful to the NRC in evaluating the cost of complying with this generic letter:

(1) the licensee staff's time and costs to perform requested inspections, corrective actions, and associated testing

(2) the licensee staff's time and costs to prepare the requested reports and documentation

(3) the additional short-term costs incurred as a result of the inspection findings such as the costs of the corrective actions or the costs of down time

(4) an estimate of the additional long-term costs which will be incurred in the future as a result of implementing commitments such as the estimated costs of conducting future inspections or increased maintenance

If you have any questions about this matter, please contact one of the NRC technical contacts or the lead project manager listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosure:
Listing of Recently Issued Generic Letters

Technical Contacts:
Barry J. Elliot, NRR
(301) 492-0709

Keith R. Wichman, NRR
(301) 492-0757

Lead Project Manager:

Enclosure 1

Regulatory Requirements Applicable to

Reactor Vessel Structural Integrity

10 CFR 50.60

Pursuant to 10 CFR 50.60, all light water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to 10 CFR Part 50.

The fracture toughness of the reactor coolant pressure boundary required by 10 CFR 50.60 is necessary to provide adequate margins of safety during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. The material surveillance program required by 10 CFR 50.60 monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel.

Appendix G to 10 CFR Part 50 requires that the reactor vessel beltline materials must have Charpy upper shelf energy of no less than 50 ft-lb throughout the life of the vessel. Otherwise, licensees are required to provide demonstration of equivalent margins of safety in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 or perform actions in accordance with Paragraph V.C of Appendix G to 10 CFR Part 50.

Appendix H to 10 CFR Part 50 requires the surveillance program to meet the American Society for Testing and Materials (ASTM) Standard E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear

Power Reactor Vessels." Further, Appendix H to 10 CFR Part 50 specifies the applicable edition of ASTM E 185. Appendix H to 10 CFR Part 50, as amended on July 26, 1983, requires that the part of the surveillance program conducted before the first capsule is withdrawn must meet the requirements of the 1973, the 1979, or the 1982 edition of ASTM E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code under which the reactor vessel was purchased. The licensee may also use later editions of ASTM E 185 which have been endorsed by the NRC. The test procedures and reporting requirements for each capsule withdrawal after July 26, 1983 must meet the requirements of the 1982 edition of ASTM E 185 to the extent practical for the configuration of the specimens in the capsule. The licensee may use either the 1973, the 1979, or the 1982 edition of ASTM E 185 for each capsule withdrawal before July 26, 1983.

The licensees, especially those with reactor vessels purchased before ASTM issued the 1973 edition of ASTM E 185, may have surveillance programs that do not meet the requirements of Appendix H to 10 CFR Part 50 but may have alternative surveillance programs. The licensee may use these alternative surveillance programs in accordance with 10 CFR 50.60(b) if the licensee has been granted an exemption by the Commission under 10 CFR 50.12.

The licensee must monitor the test results from the material surveillance program. According to Paragraph III.C of Appendix H to 10 CFR Part 50, the results of the surveillance program may indicate that a technical specifications change is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits.

10 CFR 50.61

Pursuant to 10 CFR 50.61, there are fracture toughness requirements for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature. If the projected reference temperature exceeds the screening criteria established in 10 CFR 50.61, licensees are required to submit an analysis and schedule for such flux reduction programs as are reasonably practicable to avoid exceeding the screening criteria. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, the licensee shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed. 10 CFR 50.61(b)(1), as amended effective June 14, 1991 (56 Fed Reg 22300 et. seq., May 15, 1991), requires that licensees submit their assessment by December 16, 1991, if the projected reference temperature will exceed the screening criteria before the expiration of the operating license.

Plant-specific information is required to be considered in assessing the level of neutron embrittlement as specified in 10 CFR 50.61(b)(3). This information includes but is not limited to the reactor vessel operating temperature and surveillance results.

Prediction of Irradiation Embrittlement

Paragraph V.A of Appendix G to 10 CFR Part 50 requires the prediction of the effects of neutron irradiation on reactor vessel materials. The extent of neutron embrittlement depends on the material properties, thermal environment, and results of the material surveillance program. In Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," the staff stated that it will use the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" in estimating the embrittlement of the materials in the reactor vessel beltline. All licensees and permittees have responded to Generic Letter 88-11 committing to use the methodology in Regulatory Guide 1.99, Revision 2 in predicting the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The methodology in Regulatory Guide 1.99, Revision 2 is also the basis in 10 CFR 50.61 in projecting the reference temperature.

Call Larry.

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS (EXCEPT YANKEE ATOMIC ELECTRIC COMPANY, LICENSEE FOR THE YANKEE ROWE NUCLEAR POWER STATION)

SUBJECT: REACTOR VESSEL STRUCTURAL INTEGRITY (GENERIC LETTER 91-)

Purpose

INSERT A

(9) In Section 50.60(a) of Title 10 of the Code of Federal Regulations (10 CFR 50.60(a)), the U.S. Nuclear Regulatory Commission (NRC) requires that licensees for all light water nuclear power reactors meet fracture toughness requirements and have a material surveillance program for the reactor coolant pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. Pursuant to 10 CFR 50.60(b), where the requirements of Appendices G and H to 10 CFR Part 50 cannot be met, an exemption is necessary pursuant to 10 CFR 50.12. Further, there are fracture toughness requirements in 10 CFR 50.61 for protection against pressurized thermal shock events for pressurized water reactors. In addition, licensees and permittees have made commitments in response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," to use the methodology in Regulatory Guide 1.99, Revision 2, to predict the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The 10 CFR 50.60 and 10 CFR 50.61 requirements and Generic Letter 88-11 are in the overall regulatory program to maintain the structural integrity of the reactor vessel. This generic letter is part of a program to evaluate reactor vessel integrity and take regulatory actions, if needed, to ensure that licensees and permittees are complying with 10 CFR 50.60 and 10 CFR 50.61, and are fulfilling commitments made in response to Generic Letter 88-11. Additional discussion of the applicable regulatory requirements is contained in Enclosure 1 to this letter. The NRC is requiring information on compliance under the provisions of 10 CFR 50.54(f).

INSERT B

ATTACHMENT 2 TO
ENCLOSURE 5
MEETING NO 211

Assessment of Embrittlement for the Yankee-Rowe Nuclear Power Station Reactor Vessel

Recent NRC concerns regarding the neutron embrittlement of the Yankee Rowe reactor vessel prompted the staff to perform a safety assessment of the Yankee Rowe reactor vessel. The staff found that the licensee for Yankee Rowe might not be in compliance with 10 CFR 50.60 and had not properly completed the assessment required in 10 CFR 50.61. Further, the licensee for Yankee Rowe had incorrectly applied the methodology in Regulatory Guide 1.99, Revision 2.

The staff found that the Charpy upper shelf energy of the Yankee Rowe reactor vessel material could be as low as 35.5 ft-lb, which is less than the 50 ft-lb value required in Appendix G to 10 CFR Part 50. However, the licensee for Yankee Rowe had not performed the required actions in Paragraph IV.A.1 or V.C of Appendix G to 10 CFR Part 50. Since then, the licensee for Yankee Rowe has performed an analysis in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 using criteria being developed by ASME to demonstrate margins of safety equivalent to those in the ASME Code.

~~While reviewing the surveillance program for the Yankee Rowe reactor vessel,~~ a concern was identified regarding compliance with the requirements of Appendix H to 10 CFR Part 50. ASTM E 185 requires that the licensee take sample specimens from actual material used in fabricating the beltline of the reactor vessel. These surveillance materials shall include one heat of base metal, one butt weld, and one weld "heat affected zone." The licensee for Yankee Rowe terminated the material surveillance program in 1965. Therefore, Yankee Rowe had no material surveillance program on July 26, 1983, when Appendix H to 10 CFR Part 50 became effective. Further, the samples irradiated at Yankee Rowe before 1965 were comprised only of base metal.

The licensee for Yankee Rowe had used the methodology in Regulatory Guide 1.99, Revision 2 to predict the effects of neutron embrittlement. However, the staff found that the methodology in Regulatory Guide 1.99, Revision 2 was incorrectly applied by the licensee. The specific issues were ~~(1) heat treatment of reactor vessel material~~, (1) ~~(2)~~ the irradiation temperature, (2) ~~(3)~~ chemistry composition of reactor vessel material, and (3) ~~(4)~~ results of material surveillance program.

INSERT C

~~The licensee reported that the heat treatment of the Yankee Rowe vessel resulted in a coarse grain structure that may affect the sensitivity of the material to neutron irradiation.~~ ¶ The irradiation temperature at Yankee Rowe is between 454 ^{o}F and 520 ^{o}F, which is below the nominal irradiation temperature of 550 ^{o}F used in developing Regulatory Guide 1.99, Revision 2. A lower irradiation temperature increases the effect of neutron embrittlement.

¶ ~~The chemical composition of the reactor vessel welds is unknown. The material sensitivity to neutron embrittlement depends on its chemical content.~~ INSERT D ¶ The limited results of the surveillance program from Yankee Rowe indicated that the ~~shift~~ increase in the reference temperature exceeds the mean-plus-two standard deviations as predicted by the procedures in Regulatory Guide 1.99, Revision 2. INSERT E

Based on the factors discussed above the staff ~~also~~ found that the licensee for Yankee Rowe had not considered plant-specific information in assessing compliance with 10 CFR 50.61. ¶ ~~Although the surveillance data indicate neutron embrittlement exceeding that being projected and the Yankee Rowe plant operates at a low temperature, the licensee for Yankee Rowe had not taken these into account. When~~ plant-specific information is considered, the Yankee Rowe reactor vessel may have exceeded the screening criteria in 10 CFR 50.61. Since then, the licensee for Yankee Rowe has performed a probabilistic fracture mechanics analysis in accordance with 10 CFR 50.61(b)(4) and the staff is continuing its review.

Based on the staff experience with the Yankee Rowe review, the staff has a concern that this may not be an isolated case regarding compliance with 10 CFR 50.60 and 10 CFR 50.61 and regarding fulfillment of commitments made in response to Generic Letter 88-11. Thus, the staff is issuing this generic letter to obtain information to assess compliance with these regulations and fulfillment of commitments. The staff is continuing to pursue this concern with the Yankee Atomic Electric Company. Therefore, the Yankee Atomic Electric Company need not respond to this generic letter.

Requested Information

~~Actions [redacted] this Generic Letter~~

~~The NRC regulations require that all addressees have reactor vessel material surveillance programs in accordance with Appendices G and H to 10 CFR Part 50 or have obtained an exemption pursuant to 10 CFR 50.12. Addressees are also required to meet 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events."~~

1 A *are requested to provide the following information* ~~Addressees shall perform the following actions to demonstrate compliance with Appendix H to 10 CFR Part 50:~~

a. - - -

ii Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 *and* ~~or~~ do not have an integrated surveillance program approved by the NRC, *should* ~~shall~~ describe actions *taken or* to be taken to assure compliance with Appendix H to 10 CFR Part 50. ~~For~~ addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50, ~~they shall~~ *should* indicate when the revised program will be submitted to the NRC Staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees *should* ~~shall~~ indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b). ~~and to provide their technical justification.~~ ? *was this to come later or now?*

2. Addressees *are requested to provide* ~~shall perform~~ the following *information* ~~actions~~, to demonstrate compliance with Appendix G to 10 CFR Part 50:

A.

i Addressees whose Charpy upper shelf energy is predicted to be less than 50 ft-lb at the end of its license using the guidance in Paragraph C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2 shall provide to the NRC the predicted Charpy upper shelf energy for the limiting beltline weld and the plate or forging at the end of the current license term and on December 16,1991 and describe the actions taken *or to be taken* pursuant to Paragraph IV.A.1 or V.C or Appendix G to 10 CFR Part 50.

ii Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition shall describe the consideration given to the following material properties in their evaluat...

ANY LEAD IN?

performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

(a) the results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material and the method of determining the unirradiated reference temperature from the Charpy and drop weight test,

(b) the heat treatment received by all beltline and surveillance materials

(c) the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricated each beltline weld.

(d) the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld

(e) the chemical composition, in particular the weight percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material

(f) the identity of the heat of wire used for determining the weld metal chemical composition if different than Item (c) above

Insert F

~~In committing to Generic Letter 88-11, licensees have committed to calculate radiation embrittlement in accordance with the procedures documented in Regulatory Guide 1.99, Revision 2 and to account for radiation embrittlement for core critical operation with irradiation temperatures less than 525°F. This guide indicates that the procedures are applicable for a nominal irradiation temperature of 550°F and operation below 525°F is considered to produce greater embrittlement.~~ To determine whether plants have operated for significant time with the core critical and with an irradiation temperature less than 525°F, licensees whose reactor vessels are predicted at the end of the current license term to accumulate a neutron fluence greater than 10^{16} n/cm^2 (E>1MeV) during the portion of operation with irradiation temperature less than 525°F, shall report ~~the amount of time,~~ the neutron fluence and the irradiation temperature above and below 525°F. The applicable neutron fluence for BWR plants are the values at the ¼T location and for PWR plants are the values at the inside of surface. Fluence of 10^{16} does not indicate that a revision to the pressurized thermal shock analysis or pressure-temperature curves are needed. It is intended as a screening criteria to gather information for further review. These licensees shall describe the consideration given to determining the effect of lower irradiation temperature on the reference temperature and Charpy upper shelf energy.

iii Addressees are to report how they considered their surveillance results on the predicted amount of embrittlement ~~In committing to Generic Letter 88-11, licensees have committed to compare the results of the surveillance capsule with the values predicted by Regulatory Guide 1.99, Revision~~ 2. If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper-shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee shall report the information and describe the effect of the surveillance results on the adjusted reference temperature and predicted Charpy upper shelf energy for each beltline material at the end of the current license and on December 15, 1991.

Same good description [illegible] format

Reporting Requirements

Pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), each addressee shall submit a letter within ~~90~~ 120 days of the date of this generic letter providing the information described under "~~Actions Covered by this Generic Letter~~ Requested Information." The letter shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation. In addition, a copy shall be submitted to the appropriate Regional Administrator. This generic letter requests information that will enable the NRC to verify licensee compliance with its current licensing basis with respect to reactor vessel fracture toughness and material surveillance for the reactor coolant pressure boundary. Accordingly an evaluation justifying this information request is not necessary under 10 CFR 50.54(f).

Backfit Discussion

This generic letter requests information that will enable the NRC Staff to determine whether licensees are complying with their prior commitments and/or license conditions with respect to 10 CFR 50.60, 10 CFR 50.61, and Generic Letter 88-11. The staff is not establishing a new position with respect to such compliance in this generic letter. Because the staff is requesting information to verify licensee compliance with their previously established commitments and is not establishing any new staff position, this generic letter does not constitute a backfit. ~~Thus, 10 CFR 50.109 does not apply and no backfit analysis need be prepared.~~ and no documented evaluation or backfit analysis need be prepared.

Request for Voluntary Submittal of Impact Data

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires May 31, 1994. The estimated average number of burden hours is ~~100~~ 200 person hours for each addressee's response, including the time required to assess the requirements, search data sources, gather and analyze the data, and prepare the required letters. These estimated average burden hours pertain only to the identified response-related matters and do not include the time to implement the actions required by the regulations. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to Ronald Minsk, Office of Information and Regulatory Affairs (3150-0011), NEOB-3019, Office of Management and Budget, Washington, D.C. 20503, and to the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, Division of Information Support Services, Office of Information and Resources Management, Washington, D.C. 20555.

Although no specific request or requirement is intended, the following information would be helpful to the NRC in evaluating the cost of complying with this generic letter

(1) the licensee staff's time and costs to perform requested inspections, corrective actions, and associated testing

(2) th[illegible] staff's time and costs to prepare the requested reports and docum[illegible]

(3) the additional short-term costs incurred as a result of the inspection findings such as the costs of the corrective actions or the costs of down time

(4) an estimate of the additional long-term costs which will be incurred in the future as a result of implementing commitments such as the estimated costs of conducting future inspections or increased maintenance.

If you have any questions about this matter, please contact one of the NRC technical contacts or the lead project manager listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosure:
Listing of Recently Issued Generic Letters

Technical Contacts:
Barry J. Elliot, NRR
(301) 492-0709

Keith R. Wichman, NRR
(301) 492-0757

Lead Project Manager:

Enclosure 1

Regulatory Requirements Applicable to Reactor Vessel Structural Integrity

10 CFR 50.60

Pursuant to 10 CFR 50.60, all light water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to 10 CFR Part 50.

The fracture toughness of the reactor coolant pressure boundary required by 10 CFR 50.60 is necessary to provide adequate margins of safety during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. The material surveillance program required by 10 CFR 50.60 monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel.

Appendix G to 10 CFR Part 50 requires that the reactor vessel beltline materials must have Charpy upper shelf energy of no less than 50 ft-lb throughout the life of the vessel. Otherwise, licensees are required to provide demonstration of equivalent margins of safety in accordance with Paragraph IV.A.1 of Appe[illegible]x G to 10 CFR Part 50 or perform actions in accordance with Paragra[illegible] C of Appendix G to 10 CFR Part 50.

Appendix H to 10 CFR Par[illegible] requires the surveillance program to meet the American Society for Te[illegible]g and Materia[illegible] (ASTM) Standard E 185, "Standard Practice for Conducting [illegible]eillance Tests for Light-Water Cooled Nuclear

Power Reactor Vessels." Further, Appendix H to 10 CFR Part 50 specifies the applicable edition of ASTM E 185. Appendix H to 10 CFR Part 50, as amended on July 26, 1983, requires that the part of the surveillance program conducted before the first capsule is withdrawn must meet the requirements of the 1973, the 1979, or the 1982 edition of ASTM E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code under which the reactor vessel was purchased. The licensee may also use later editions of ASTM E 185 which have been endorsed by the NRC. The test procedures and reporting requirements for each capsule withdrawal after July 26, 1983 must meet the requirements of the 1982 edition of ASTM E 185 to the extent practical for the configuration of the specimens in the capsule. The licensee may use either the 1973, the 1979, or the 1982 edition of ASTM E 185 for each capsule withdrawal before July 26, 1983.

The licensees, especially those with reactor vessels purchased before ASTM issued the 1973 edition of ASTM E 185, may have surveillance programs that do not meet the requirements of Appendix H to 10 CFR Part 50 but may have alternative surveillance programs. The licensee may use these alternative surveillance programs in accordance with 10 CFR 50.60(b) if the licensee has been granted an exemption by the Commission under 10 CFR 50.12.

The licensee must monitor the test results from the material surveillance program. According to Paragraph III.C of Appendix H to 10 CFR Part 50, the results of the surveillance program may indicate that a technical specifications change is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits.

10 CFR 50.61

Pursuant to 10 CFR 50.61, there are fracture toughness requirements for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature. If the projected reference temperature exceeds the screening criteria established in 10 CFR 50.61, licensees are required to submit an analysis and schedule for such flux reduction programs as are reasonably practicable to avoid exceeding the screening criteria. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, the licensee shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed. 10 CFR 50.61(b)(1), as amended effective June 14, 1991 (56 Fed Reg 22300 et. seq., May 15, 1991), requires that licensees submit their assessment by December 16, 1991, if the projected reference temperature will exceed the screening criteria before the expiration of the operating license.

Plant-specific information is required to be considered in assessing the level of neutron embrittlement as specified in 10 CFR 50.61(b)(3). This information includes but is not limited to the reactor vessel operating temperature and surveillance results.

Prediction of Irradiation Embrittlement

Paragraph V.A of Appendix G to 10 CFR Part 50 requires the prediction of the effects of neutron irradiation on reactor vessel materials. The extent of neutron embrittlement depends on the material properties, thermal environment, and results of the material surveillance program. In Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," the staff stated that it will use the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" in estimating the embrittlement of the materials in the reactor vessel beltline. All licensees and permittees have responded to Generic Letter 88-11 committing to use the methodology in Regulatory Guide 1.99, Revision 2 in predicting the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The methodology in Regulatory Guide 1.99, Revision 2 is also the basis in 10 CFR 50.61 in projecting the reference temperature.

INSERT A

PURPOSE

The purpose of this generic letter is to obtain information needed to assess compliance with requirements and commitments related to reactor vessel integrity in view of certain concerns raised in the staff's review of reactor vessel integrity for the Yankee Rowe Nuclear Power Station.

INSERT B

During the NRC staff's review of reactor vessel integrity for the Yankee Rowe Nuclear Power Station, concerns were raised regarding licensee compliance with certain requirements and commitment.

INSERT C

The Regulatory Guide indicates that for irradiation temperatures less than 525°F, embrittlement effects should be considered to be greater than predicted by the methods of the guide. However, insufficient adjustment had been made to account for this effect.

INSERT E

The regulatory guide indicates that credible surveillance data should be used to predict the increase in reference temperature resulting from neutron irradiation.

The staff implemented Regulatory Guide 1.99, Revision 2 through issuance of Generic Letter 88-11. In commiting to Generic Letter 88-11, licensee have committed to calculate radiation embrittlement in accordance with the procedures documented in Regulatory Guide 1.99, Revision 2. Hence, in accordance with the limitations in Section 1.3 of the regulatory guide, licensees should consider the effects on irradiation embrittlement of core critical operation with irradiation temperatures less than 525°F and in accordance with section 2 of the regulatory guide, licensees are to consider the effects of the results from their surveillance capsules.

The Summer 1972 Addenda of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code are the earliest code requirements for testing materials to determine their unirradiated reference temperature. Since the Yankee Rowe reactor vessel was constructed to an ASME Code earlier than the Summer 1972, it had not been sufficiently tested to determine its unirradiated reference temperature. The licensee extrapolated the available test results to determine an unirradiated reference temperature. The staff determined that the licensee's extrapolation was non-conservative.

In addition, the chemical composition of the Yankee Rowe reactor vessel welds is unknown. The material's sensitivity to neutron embrittlement depends on its chemical content. The licensee assumed that the chemistry of their welds was equivalent to that of the BR-3 reactor vessel in Mol, Belgium. However, the licensee could not identify the heat number of the wire used to fabricate the Yankee Rowe welds. Since the chemical composition, in particular the amount of copper, depends upon the heat number of the weld wire, the licensee was assuming a chemical composition that was not based on its plant specific information.

INSERT F

3. Addressees are requested to provide the following information to demonstrate compliance with their commitments in Generic Letter 88-11.

   i. Addressees are to report how they considered the effect on embrittlement of operation at irradiation temperature less than 525°F.