



OFFICE OF THE  
COMMISSIONER

UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555  
September 1, 1987

Stello  
Taylor  
Rehm  
Murley  
~~Beckjord~~  
Murray  
EDO R/F

MEMORANDUM FOR: Samuel Chilk, Secretary

FROM: Frederick M. Bernthal *FB*

SUBJECT: COMTR 87-3 -- ACRS COMMENTS ON THE ADVANCED NOTICE OF  
PROPOSED RULEMAKING: DEGREE REQUIREMENTS FOR SENIOR  
OPERATORS

Thoughtful people will continue to disagree on this issue. It is therefore especially important that the Commission now have the benefit of the full public airing of views that will be afforded by publication of the proposed rule. I thus do not believe that the Commission should revisit the issue at this time. ACRS comments on the Commission's contemplated rulemaking on degreed SRO's should and will be addressed during the usual notice and comment period once the NPR is published. I would note, however, that ACRS has apparently failed to appreciate fully the distinction between training and education. To my knowledge, no one has argued that a baccalaureate degree provides training essential to SRO performance.

Commissioner Carr notes that the Commission's goals should be clearly identified in the proposed rule. I agree. In particular, the arguments in support of providing a conduit for experienced operators into upper management, as well as those for enhancing professional regard for the role of senior operator should be fully articulated.

cc: Chairman Zech  
Commissioner Roberts  
Commissioner Carr  
Commissioner Rogers  
ACRS

✓ EDO

BB09160137 BB0815  
PDR REVGP NRGCRGR  
MEETING141 PNU

Cy: Ross  
Speis  
Kelber  
Burda  
Sheron/Minners  
Coffman  
Morris/Rosztoczy  
Marcus  
Telford  
DiPalo  
File (O&M-DCM)

ADVANCE NOTICE OF PROPOSED RULEMAKING

Published May 1986

- ° AFTER JANUARY 1, 1991
  - BACCALAUREATE DEGREE IN ENGINEERING OR PHYSICAL SCIENCE REQUIRED FOR SOs
  - OTHER DEGREES ACCEPTED ON A CASE-BY-CASE BASIS
  - DEGREE EQUIVALENCY UNACCEPTABLE
- ° ONE OF TWO YEARS OF REQUIRED NUCLEAR PLANT EXPERIENCE MUST BE AT GREATER THAN TWENTY PERCENT POWER
- ° SOs LICENSED PRIOR TO JANUARY 1, 1991 WOULD BE GRANDFATHERED
- ° ONE RE-EXAMINATION FOR SO APPLICANTS WHO APPLY JUST PRIOR TO JANUARY 1, 1991
- ° CONCURRENT POLICY STATEMENT TO ENCOURAGE UTILITIES TO ESTABLISH A DEGREE PROGRAM FOR REACTOR OPERATORS

Enclosure 4 to the Minutes of CRGR Meeting No. 141  
Proposed Resolution for USI A-45,  
"Shutdown Decay Heat Removal Requirements"

TOPIC

W. Minners (RES) and R. Woods (RES) presented for CRGR review the proposed resolution for USI A-45, "Shutdown Decay Heat Removal Requirements." This proposal calls for the USI to be subsumed into (i.e., to be effectively addressed by) the Individual Plant Examinations to be performed by the licensees, as specified in the IPE Generic Letter that was reviewed and endorsed by the Committee at Meeting No. 134. Copies of the briefing slides used by the staff to guide their presentation and the discussions of this matter at this meeting are enclosed (see attachment to this enclosure).

BACKGROUND

The package submitted for review by CRGR in this matter was transmitted by memorandum dated June 9, 1988, E. S. Beckjord to E. L. Jordan; that review package included the following documents:

1. Draft memorandum (undated), V. Stello, Jr. to the Commissioners, "Shutdown Decay Heat Removal Requirements (USI A-45)," and attachments as follows:
  - a. Enclosure A - Draft Report NUREG-1289, dated April 1988, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements"
  - b. Enclosure B - Draft Report NUREG-1292, dated October 1987, "Shutdown Decay Heat Removal Analysis: Summary Report"
  - c. Enclosure C - Proposed Federal Register Notice, "Shutdown Decay Heat Removal Requirements"
  - d. Enclosure D - Draft Letter from E. S. Beckjord to Congressional Oversight Committees regarding this proposed USI resolution

CONCLUSIONS/RECOMMENDATIONS

As a result of their review of this issue, including the discussions with the staff at this meeting and the background provided by their earlier review of the IPE Generic Letter at Meeting No. 134, the Committee recommended in favor of issuing the proposed resolution for USI A-45, subject to modifications to emphasize to the Commission that the external event aspect of USI A-45 is not addressed by subsuming this issue into the IPEs (the IPEs are not intended to address external events - external events will be addressed separately at a later time after development and approval of appropriate methodology).

RECOMMENDED FINAL RESOLUTION OF USI A-45  
"SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"

PRESENTED TO THE  
COMMITTEE TO REVIEW GENERIC REQUIREMENTS  
JULY 14, 1988

ROY WOODS, SENIOR TASK MANAGER  
DIVISION OF REACTOR AND PLANT SYSTEMS  
OFFICE OF NUCLEAR REGULATORY RESEARCH

## SUMMARY

- ° UNRESOLVED SAFETY ISSUE (USI A-45) APPROVED DEC. 24, 1980 (SECY-80-325)
- ° KEY QUESTIONS:
  - ° DO CURRENT REGULATIONS PROVIDE SUFFICIENT ASSURANCE THAT RISK FROM DHR FAILURES IS ACCEPTABLY LOW?
  - ° ARE IMPROVEMENTS TO DHR FUNCTION IN OPERATING PLANTS COST-BENEFICIAL?
- ° KEY CONCLUSIONS:
  - ° DHR FAILURES ARE SIGNIFICANT CONTRIBUTORS TO CORE DAMAGE FREQUENCY FROM SB-LOCAs, TRANSIENTS
  - ° VULNERABILITIES AND CORRECTIVE ACTIONS, COST/BENEFIT RATIOS ARE PLANT SPECIFIC
  - ° DEDICATED DHR SYSTEM NOT COST BENEFICIAL
  - ° DESIGNS NOT COMPARED TO "CURRENT" DBA-BASED REQUIREMENTS
- ° KEY ASSUMPTIONS:
  - ° CDF GOAL OF  $1E-05$  SELECTED BY STAFF FOR THIS APPLICATION
  - ° CONSISTENT WITH A-44 AND A-49

### USI A-45 SCOPE

- ° SIX CASE STUDIES, DHR FAILURE RELATED PRA'S (SUMMARIZED IN ENCLOSURE B)
- ° LIMITED TO SYSTEMS NEEDED TO RESPOND TO TRANSIENTS AND SMALL-BREAK LOCAs
- ° EVALUATED SUCH SYSTEMS' VULNERABILITY TO FIRE, FLOOD, SEISMIC, INSIDER SABOTAGE

## CONTENT OF REGULATORY ANALYSIS

APRIL, 1988 DRAFT REVISION DESCRIBES SIX ALTERNATIVES AND PROPOSES RESOLUTION WITH ALT. # 2 (PLANT-SPECIFIC ANALYSIS)

## ALTERNATIVE 1 - NO ACTION

COULD BE ACCEPTABLE IF NRC ANALYSIS  
RESULTS ARE OVERLY CONSERVATIVE  
(EPRI/WOG: POINT BEACH)

## ALTERNATIVE 2 - LIMITED SCOPE PRAs

SEVERE ACCIDENT PROGRAM IPEs

### ALTERNATIVE 3 - SPECIFIED SYSTEMS MODIFICATIONS

USIs AND GIS

## ALTERNATIVE 4 - DEPRESSURIZATION AND COOLING

PWR - FEED AND BLEED

## BWR - CONTAINMENT VENTING

## ALTERNATIVE 5 - DEDICATED HOT SHUTDOWN CAPABILITY

## ALTERNATIVE 6 - DEDICATED COLD SHUTDOWN CAPABILITY

## TECHNICAL FINDINGS

- FREQUENCY OF CORE DAMAGE DUE TO DHR FUNCTION FAILURE  $[P(CM)_{DHR}]$  AVERAGES 2 TO  $3 \times 10^{-4}$  PER R-YR (INCLUDES INTERNAL AND EXTERNAL CAUSES)
- SUPPORT SYSTEM FAILURES (E.G., EMERGENCY POWER, SERVICE WATER, COMPONENT COOLING) CONTRIBUTE SIGNIFICANTLY TO  $P(CM)_{DHR}$
- REDUNDANCY CONCERNS AND CONSIDERABLE SHARING OF SYSTEMS, PARTICULARLY AT SUPPORT SYSTEM LEVEL FOR SOME PLANTS
- CONCERNS WITH OVERALL GENERAL ARRANGEMENT OF EQUIPMENT FROM A SAFETY VIEWPOINT, E.G., LACK OF INDEPENDENCE, SEPARATION & PHYSICAL PROTECTION OF REDUNDANT SAFEGUARD TRAINS
- FIRE, FLOOD, SEISMIC, SABOTAGE RISK CONCERNS
- RELATIVE IMPORTANCE OF VULNERABILITIES IS PLANT-SPECIFIC
- EFFECTIVENESS OF CORRECTIVE ACTIONS IN REDUCING  $P(CM)_{DHR}$  IS PLANT SPECIFIC



## REG. ANALYSIS

### DIFFERENT APPROACHES TO VALUE-IMPACT ANALYSIS

- ° VALUE-IMPACT ANALYSIS PERFORMED 3 WAYS:
  - A. VALUE TERM: AVERTED DOSE TO POPULATION  
IMPACT TERM: COST OF IMPLEMENTATION
  - B. VALUE TERM: SAME AS METHOD A  
IMPACT TERM: COST OF IMPLEMENTATION LESS AVERTED  
ON-SITE COSTS
  - C. SAME AS METHODS A & B PLUS THE SAVINGS FROM  
SPECIAL CONSIDERATIONS (E.G., SABOTAGE,  
MORATORIUM, RESOLUTION OF OTHER GENERIC ISSUES,  
UNQUANTIFIABLES)
- ° RESULTS:
  - ° METHOD A - ALTERNATIVES 2, 3 & 4 MAY BE  
COST-EFFECTIVE
  - ° METHOD B - ALTERNATIVES 2, 3 & 4 MAY BE MORE  
COST-EFFECTIVE

REG. ANALYSIS (CONT'D)

DIFFERENT APPROACHES TO VALUE-IMPACT ANALYSIS

- ALTERNATIVES 2, 3, & 4 DO NOT MEET THE STAFF'S CDF GOAL OR REDUCE SABOTAGE RISK
- METHOD C - ALTERNATIVES 5 & 6 MAY BE COST-EFFECTIVE AND REACH CDF GOAL
- STAFF ENDORSES ALTERNATIVE 2, PLANT-SPECIFIC ANALYSES, ON FOLLOWING BASES:
  - A-45 CASE STUDIES SHOWED:
    - MANY RISK CONTRIBUTORS ARE PLANT-SPECIFIC
    - ONLY WAY TO IDENTIFY DHR VULNERABILITIES IS THRU PLANT-SPECIFIC EXAMINATIONS
    - EFFECTS OF CORRECTIVE ACTIONS ARE PLANT-SPECIFIC
    - CDF AT MANY PLANTS ABOVE STAFF-SELECTED GOAL
- USE OF "METHOD C" (CREDIT FOR "MORATORIUM AVOIDANCE" ETC.) GOES BEYOND VALUE/IMPACT METHODS PREVIOUSLY USED FOR USIs/GSIs

## IMPLEMENTATION DETAILS

### (ALTERNATIVE 2)

- ° NRC PROPOSES TO REQUIRE A PLANT-SPECIFIC EXAMINATION TO IDENTIFY VULNERABILITIES TO SEVERE ACCIDENTS (THE IPE PROGRAM)
- ° IPE WILL INCLUDE, BUT NOT BE LIMITED TO, DHR FAILURE RELATED CORE DAMAGE EVENTS
- ° SEPARATE, DEDICATED "A-45" EXAMINATION WOULD BE REDUNDANT
- ° WE CONCLUDE THAT A-45 SHOULD BE SUBSUMED INTO IPE
  - ° INSIGHTS GAINED FROM SIX CASE STUDIES AND EPRI-WOG ANALYSIS (PLUS NRC/SANDIA REVIEW, SEE APPENDIX D TO ENCLOSURE A) WILL BECOME EXAMPLE FOR LICENSEES.

NUMARC/EPRI/WOG POINT BEACH PRA

(APPENDIX D TO ENCL A)

(SUMMARY OF MARCH 31, 1988 MEETING WITH NUMARC)

• NRC REVIEWED FOR TWO REASONS:

1) ANSWER POSSIBLE CLAIM THAT ALTERNATIVE 1 IS JUSTIFIABLE

- IF NUMARC PRA IS CORRECT, AND
- IF PB IS A "BOUNDING" PLANT  
(THEN ALT. 1 WOULD BE ACCEPTABLE)
- WE ARE NOT CONVINCED THAT EITHER OF THE ABOVE ARE CORRECT

2) EXAMPLE OF NRC REVIEW OF AN IPE, HIGHLIGHTS DIFFERENCES BETWEEN NRC AND INDUSTRY:

- METHODS
- NUMERICAL ASSUMPTIONS

# EPRI/WOG STUDY RESULTS

(NSAC - 113)

<u>SOURCE OF RISK</u>	<u>CORE DAMAGE</u>	<u>FREQUENCY PER YEAR</u>	
	<u>NRC</u>	<u>EPRI/WOG (RF)*</u>	<u>REVISED NRC (R</u>
INTERNAL	1.4E-4	2.6E-6 (54)	2.5E-5 (6)
SEISMIC	6.1E-5	7.4E-6 (8)	4.1E-5 (1.5)
FIRE	3.2E-5	6.3E-8 (500)	2.2E-5 (1.5)
INTERNAL FLOOD	7.7E-5	1.0E-8 (7700)	9.8E-7 (79)
EXTERNAL FLOOD	1.9E-8	1.0E-8 (2)	- -
WIND	4.0E-6	1.0E-8 (400)	1.7E-7 (24)
LIGHTNING	<u>5.8E-8</u>	<u>1.0E-8 (6)</u>	<u>- -</u>
TOTAL	3.1E-4	1.0E-5 (31)	9E-5 (3.5)

\*REDUCTION FACTOR COMPARED  
TO "NRC"

### CDF GOAL FOR APPLICATION TO A-45

- ° MUST ANSWER QUESTION - IS DHR FAILURE RELATED CDF HIGH ENOUGH TO JUSTIFY REQUIRING PLANT SPECIFIC ANALYSIS?
- ° STAFF CURRENTLY CONSIDERING PROPOSING MORE GENERAL USE OF OVERALL  $10^{-4}$ /RY YR CDF\*
- ° BELIEVE DHR FAILURE RELATED CDF 1/3 TO 1/2 OF OVERALL CDF
- ° THEREFORE, NEED TOTAL DHR FAILURE RELATED CDF  $3 \times 10^{-5}$
- ° TO ACHIEVE THAT, NEED QUANTIFIABLE DHR FAILURE RELATED CDF  $1 \times 10^{-5}$  (OPERATOR ERRORS, ACTS OF COMMISSION...)
- ° INTENDED ONLY FOR A-45. NOT IMPLIED THAT IPE SHOULD ADOPT.
- ° COMPATIBLE WITH STAFF'S A-44 AND A-49 GOALS

\*CONSISTENT WITH LARGE RELEASE CDF OF  $10^{-6}$ /RY FOR EXAMPLE IF 1:10 RATIO BETWEEN CMF AND CDF, AND 1:10 RATIO BETWEEN LARGE RELEASE FREQ AND CMF

# OVERVIEW OF ALTERNATIVES ON GENERIC BASIS - PWR

(COST PER P-REM - AVERAGE SITE)

<u>ALTERNATIVE</u>	<u>EXTENT OF IMPROVEMENT</u>		<u>EXTENT OF IMPROVEMENT</u>		<u>COST OF IMPROVEMENT</u>		<u>COST PER PERSON-REM</u>	
	<u>P(CM)</u>		<u>POPUL. DOSE</u>		<u>GROSS IMPACT \$</u>	<u>NET IMPACT \$</u>	<u>OFFSITE W/GROSS IMPACT (\$ PER P-REM)</u>	<u>OFF + ON SITE W/NET IMPACT</u>
	<u>INITIAL VALUE</u>	<u>% REDUCTION</u>	<u>INITIAL VALUE</u>	<u>% REDUCTION</u>				
2	2.2E-4	75%	3.7E3	62%	9.4E6	5E6	4100	2180
3	2.2E-4	10%	3.7E3	11%	0.56E6	-0.52E6	1370	NO COST
4	4.8E-4	61%	8.3E3	61%	7E6	-6.2E6	1390	NO COST
5	4.8E-4	94%	8.3E3	94%	66E6	46E6	8400	5830
6	5.7E-4	95%	9.9E3	94%	94E6	70E6	10,140	7520

## Note:

Alt. 2 & 3 Assume F&B

Alt. 4, 5 & 6 Assume No F&B

Alt. 6 Assumes + 20% for Cold Shutdown

OVERVIEW OF ALTERNATIVES ON GENERIC BASIS - BWR

(COST PER P-REM - AVERAGE SITE)

<u>ALTERNATIVE</u>	<u>EXTENT OF IMPROVEMENT</u>				<u>COST OF IMPROVEMENT</u>		<u>COST PER PERSON-REM</u>	
	<u>P(CM)</u>		<u>POPUL. DOSE</u>		<u>GROSS IMPACT \$</u>	<u>NET IMPACT \$</u>	<u>OFFSITE W/GROSS IMPACT (\$ PER P-REM)</u>	<u>OFF + ONSITE W/NET IMPACT</u>
	<u>INITIAL VALUE</u>	<u>% REDUCTION</u>	<u>INITIAL VALUE</u>	<u>% REDUCTION</u>				
2	2.2E-4	54%	2.3E4	45%	13E6	9E6	1260	870
3	2.2E-4	4%	2.3E4	4%	0.28E6	-0.12E6	300	NO COST
4	2.67E-4	30%	2.7E4	31%	1.1E6	2.7E6	120	NO COST
5	2.67E-4	84%	2.7E4	84%	80E6	69E6	3460	3020
6	3.56E-4	84%	3.6E4	84%	84E6	73E6	2690	2260

Note:

Alt. 2 & 3 Assume Cont. Vent

Alt. 4, 5 & 6 Assume No Cont. Vent

Alt. 6 Assumes + 20% for Cold Shutdown



PWR Core Melt Probability by Vulnerability -  
Base Case with Recovery

Generalized Comparable Vulnerabilities	Plant A Prob. Cont.		Plant B Prob. Cont.		Plant C Prob. Cont.		Plant D Prob. Cont.	
AFWS Turbine Pump	<u>7E-6</u>	2%	5E-6	2%	8E-7	1%	<u>7E-6</u>	4%
Station Batteries	<u>4E-6</u>	1%	5E-7		3E-6	4%	<u>2E-5</u>	9%
Diesel Generators	4E-7				1E-6	2%	<u>5E-6</u>	3%
LT Station Blackout	<u>4E-5</u>	12%	8E-7					
Pump Common Mode	2E-5	5%	1E-5	4%	2E-6	2%	3E-6	1%
Valve Common Mode	1E-5	3%					2E-5	11%
NPIS & RWST Valves			4E-6	2%			<u>6E-6</u>	3%
Recirculation Switchover	<u>2E-5</u>	7%			9E-7	1%		
LPI/R System	<u>2E-5</u>	5%	8E-6	3%			<u>1E-5</u>	6%
LP Pump Cooling	<u>1E-5</u>	4%						
CCW System	2E-6	1%						
Service Water System			1E-6	1%				
SIS & Manual Actuation			4E-6	2%	2E-7		5E-6	3%
Seismic Cab. & Racks	<u>5E-5</u>	16%					<u>6E-5</u>	33%
Seismic RWST & CST	<u>1E-5</u>	3%	<u>1E-5</u>	6%	<u>1E-5</u>	17%	<u>1E-5</u>	8%
Spray Fire Header Rupt.	<u>8E-5</u>	25%						
Fire AFW Pump Room	<u>1E-5</u>	4%						
Fire 4160 Sw Gear Rm	<u>2E-5</u>	6%						
Fire Cable Spreading Room			<u>8E-5</u>	32%	<u>4E-5</u>	59%	<u>6E-6</u>	3%
Wind Chimney Collapse			2E-5	10%				
Wind DG Exhaust Stack	<u>4E-6</u>	1%					<u>5E-6</u>	3%
Flood Safety Systems			<u>5E-5</u>	19%	3E-6	4%	7E-6	4%
Lightning DC Power	1E-7				2E-7		2E-7	
Unspecified Vulnerabilities	1E-5	4%	5E-5	19%	6E-6	8%	2E-5	10%
Total Core Melt Prob.	3.1E-4		2.4E-4		7.4E-5		1.8E-4	

**PWR Internal and Special Emergency Core Melt Probabilities**

Initiating Event	Plant A		Plant B		Plant C		Plant D	
	Prob.	% of Total	Prob.	% of Total	Prob.	% of Total	Prob.	% of Total
Internal	1.4E-4	45	7.1E-5	32	1.4E-5	19	8.8E-5	49
Seismic	6.1E-5	19	7.3E-6	3	1.3E-5	18	7.3E-5	41
Fire	3.3E-5	10	7.5E-5	33	4.4E-5	59	5.8E-6	3
Internal Flood	7.7E-5	25	NA		NA		NA	
External Flood	1.9E-8		4.6E-5	20	3.2E-6	4	7.2E-6	4
Extreme Wind	4.0E-6	1	2.4E-5	11	1.6E-8		5.3E-6	3
Lightning	5.8E-8		2.6E-6	1	2.0E-7		1.8E-7	
TOTAL	3.13E-4		2.36E-4		7.48E-5		1.79E-4	
Internal	1.4E-4	45	7.1E-5	32	1.4E-5	19	8.8E-5	49
Special Emergency	1.7E-4	55	1.6E-4	68	6.0E-5	81	9.1E-5	51

# BWR Vulnerabilities and Proposed Modifications

Plant E		Plant F	
<u>Vulnerability</u>	<u>Modification</u>	<u>Vulnerability</u>	<u>Modification</u>
Failure of 2 of 3 Diesel Generators	Add a 4th Diesel Generator	Failure of both Diesel Generators	Add a 3rd Diesel Generator
Failure of both Station Batteries and subsequent failure to flash the DG fields	Add Dedicated Battery to at least 1 DG or add 3rd Station Battery	Failure of both Station Batteries and subsequent failure to flash the DG fields	Add Dedicated Battery to at least 1 DG or add 3rd Station Battery
Failure of Diesel Generator Jacket Cooling	Additional DG Cooling Water Pump or cross tie the existing Cooling Water Lines	Failure of RB Closed Cooling Water System due to PTO of 2 NC MOVs	Add Bypass Line with NC Manual Valve
Failure of 125 VDC Circuit Breaker Control Power	Automatic transfer of Loads or Battery Chargers	Flow Diversion failure of RB Closed Cooling Water System	Add a second Isolation MOV that automatically closes in an accident
Fire in Control Room or Cable Spreading Room	Enhance procedures for operating the Safe Shutdown Pump	Flow Diversion failure of RB Service Water System due to failure of an Isolation MOV	Add Automatic Actuator to another Blocking MOV to prevent loss of 2 Cooling Loops
Seismic Failure of Station Batteries	Seismically qualify and upgrade Battery Racks and Supports	Fires in Cable Expansion Room	Add a one-hour Fire Barrier around HPCI & RBSW Power Cables
Seismic Failure of 4160 VAC Buses	Add Seismic Restraints at top of Bus Cabinets	External Flood	Develop Procedures for Safe Shutdown for very high flood crests
		Seismic Event	Add or strengthen supports & braces to several important components

**BWR Internal and Special Emergency  
Core Melt Probabilities**

<u>Initiating Event</u>	<u>Plant E</u>		<u>Plant F</u>	
	<u>Probability</u>	<u>% of Total</u>	<u>Probability</u>	<u>% of Total</u>
Internal	9.9E-5	50	2.9E-4	66
Seismic	8.3E-5	42	8.1E-5	18
Fire	1.3E-5	7	1.1E-5	3
Internal Flood	NA		NA	
External Flood	9.8E-8		5.0E-5	12
Extreme Wind	1.4E-7		3.8E-6	1
Lightning	1.7E-6	1	1.8E-6	
TOTAL	2.0E-4		4.4E-4	
Internal	9.9E-5	50	2.9E-4	66
Special Emergency	1.0E-4	50	1.5E-4	34

Enclosure 5 to the Minutes of CRGR Meeting No. 141  
Proposed NRC Bulletin on Thimble Tube Thinning

TOPIC

C. E. Rossi (NRR) presented for CRGR review a proposed NRC Bulletin requesting that licensees/permittees for Westinghouse-designed PWRs (1) establish an inspection program to monitor incore neutron monitoring system thimble tube integrity, and (2) implement this program by inspecting the thimble tubes at the next (or first) refueling outage. Copies of the briefing slides used by the staff to guide their presentation and the discussions of this matter at this meeting are enclosed (see attachment to this enclosure).

BACKGROUND

The package submitted for CRGR review in this matter was transmitted by memorandum dated July 8, 1988, J. H. Sniezek to E. L. Jordan; that package included the following documents:

1. Proposed NRC Bulletin, "Thimble Tube Thinning in Westinghouse Reactors"
2. Summary of Proposed Bulletin (in accordance with Section IV.B of the CRGR Charter)

CONCLUSIONS/RECOMMENDATIONS

As a result of their review of this matter at this meeting, the Committee recommended in favor of issuing the proposed bulletin, subject to minor modifications and clarifications (to be coordinated with the CRGR staff), as follows:

1. Modify the wording of the proposed bulletin under "Purpose:" to specify that licensees establish and implement a program to monitor and confirm thimble tube integrity.
2. Reconsider the appropriateness of the completion schedules specified in the draft bulletin for licensees who have not yet done any inspections, in particular for such licensees who are now in a cold shutdown condition that is expected to last long enough to allow such inspections to be completed before restart. If for any reason (e.g., an increase in the rate or number of thimble tube degradations reported) the staff determines that the specified completion schedules must be accelerated such as to result in mandated shutdowns or delayed restarts, the bulletin should come back to CRGR for further review.
3. Include explicitly in the bulletin recognition of licensees who already have thimble tube monitoring programs, and include a provision that allows such licensees to continue inspections on the schedules already established in their programs.

4. Under "Reporting Requirements," include an explicit reminder of the reporting requirements in Part 50.72 and 50.73 of the regulations, as they apply to instances of thimble tube degradation that may be (or already have been) noted by licensees in the prescribed inspections.
5. Expand the sentence encouraging licensees to work collectively to address this issue, e.g., encourage workshops for sharing information and experience, and for jointly developing appropriate acceptance criteria and inspection frequencies).
6. In the last sentence of the first paragraph under "Description of Circumstances," delete the words following "...facility's design basis..." and substitute instead the phrase "...during flux mapping operations or a transient event."

SLIDE 1  
7/14/88

## NRC BULLETIN

### THIMBLE TUBE THINNING IN WESTINGHOUSE REACTORS

#### COMPLIANCE PROBLEM

- WESTINGHOUSE DESIGNED NUCLEAR POWER REACTORS ARE NOT ENTIRELY IN COMPLIANCE WITH GDC 14 WHICH STATES THAT THE RCS PRESSURE BOUNDARY SHALL BE DESIGNED TO HAVE AN EXTREMELY LOW PROBABILITY OF:

ABNORMAL LEAKAGE

RAPIDLY PROPAGATING FAILURE

GROSS RUPTURE

#### SYSTEM PROBLEM

- THINNING OF INCORE NEUTRON MONITORING SYSTEM THIMBLE TUBES (PART OF RCS PRESSURE BOUNDARY)

#### CAUSE

- FLOW INDUCED VIBRATION

#### SIGNIFICANCE

- POTENTIALLY NON-ISOLABLE LEAK OF REACTOR COOLANT THAT MAY NOT DRAIN TO CONTAINMENT SUMP
- AREA EQUIVALENT TO 3 THIMBLE TUBES RESULTS IN A LOCA
- AREA EQUIVALENT TO 8 THIMBLE TUBES MAY RESULT IN AN EVENT THAT DOES NOT SATISFY LOCA ACCEPTANCE CRITERIA

#### GENERIC APPLICATIONS

- ALL WESTINGHOUSE DESIGNED NUCLEAR POWER REACTORS THAT UTILIZE BOTTOM MOUNTED INSTRUMENTATION ARE BELIEVED TO BE EXPERIENCING THIMBLE TUBE THINNING

ACTIONS REQUESTED

- o ESTABLISH AND IMPLEMENT AN INSPECTION PROGRAM TO MONITOR THIMBLE TUBE INTEGRITY

ACTION SCHEDULE

- o IMPLEMENT INSPECTION PROGRAM AT NEXT (OR FIRST) REFUELING OUTAGE THAT BEGINS 90 DAYS AFTER THE RECEIPT OF THE BULLETIN

REPORTING REQUIREMENTS

- o LETTER CONFIRMING THAT THE INSPECTION PROGRAM HAS BEEN ESTABLISHED AND IMPLEMENTED

REPORTING SCHEDULE

- o LETTER MUST BE SUBMITTED WITHIN 30 DAYS OF THE COMPLETION OF THE NEXT (OR FIRST) REFUELING OUTAGE THAT BEGINS 90 DAYS AFTER THE RECEIPT OF THE BULLETIN

COST TO ADDRESSEES

- o COST TO INDUSTRY OVER THE NEXT 40 YEARS IS CONSERVATIVELY ESTIMATED TO BE \$15,900,000

RADIOLOGICAL EXPOSURE

- o ESTIMATED TO BE MINIMAL (ON THE ORDER OF 100 PERSON-MILLIREM PER INSPECTION)

BURDEN TO NRC

- o ESTIMATED TO BE 50 PERSON-HOURS



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## SUMMARY OF OPERATIONAL EXPERIENCE

### DOMESTIC FACILITIES

- 23 FACILITIES (INCLUDING ONE 14-FOOT CORE) ARE KNOWN TO HAVE DETECTED THIMBLE TUBE THINNING
- 4 KNOWN CASES OF THIMBLE TUBE LEAKS (LEAKS HAVE TYPICALLY OCCURRED WHILE EITHER INSERTING OR RETRACTING THE PROBE DURING FLUX MAPPING)
- THESE LEAKS HAVE BEEN SMALL - CLOSURE OF MANUAL ISOLATION VALVE HALTED LEAK AND FACILITY CONTINUED TO OPERATE
- PROBLEMS WITH THIMBLE TUBES TYPICALLY HAVE NOT BEEN REPORTED TO THE NRC

### FOREIGN FACILITIES

- APPROXIMATELY 53 FACILITIES (INCLUDING FOURTEEN 14-FOOT CORES) ARE KNOWN TO HAVE DETECTED THIMBLE TUBE THINNING
- APPROXIMATELY 12 KNOWN CASES OF THIMBLE TUBE LEAKS

### 12-FOOT VERSUS 14-FOOT CORES

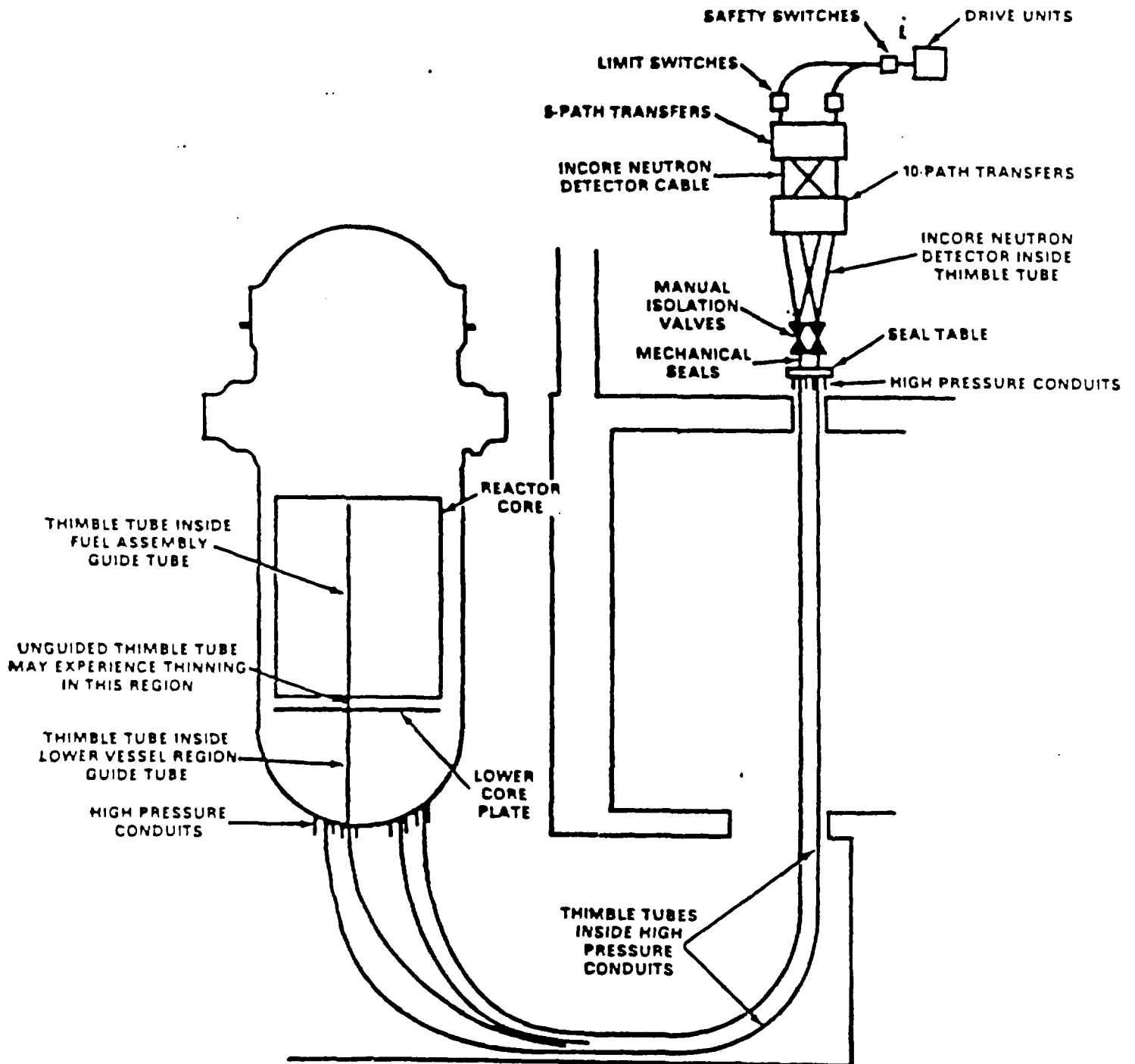
- THINNING OF THIMBLE TUBES IN 14-FOOT CORES APPEARS TO BE FASTER THAN IN 12-FOOT CORES
- SOUTH TEXAS IS THE ONLY 14-FOOT CORE IN THIS COUNTRY
  - 1) LICENSEE HAS ESTABLISHED PROGRAM TO MONITOR THIMBLE TUBE INTEGRITY
  - 2) LICENSEE HAS COMMITTED TO INSTALL AUTOMATIC ISOLATION VALVES AT FIRST REFUELING

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SUMMARY OF ECT INSPECTIONS AT D.C. COOK UNIT 2

- 12 THIMBLE TUBES HAD INDICATIONS OF < 10% THRU WALL WEAR
- 15 THIMBLE TUBES HAD INDICATIONS OF 10% TO 40% THRU WALL WEAR
- 4 THIMBLE TUBES HAD INDICATIONS OF 40% TO 50% THRU WALL WEAR
- 8 THIMBLE TUBES HAD INDICATIONS OF 50% TO 60% THRU WALL WEAR
- 19 THIMBLE TUBES HAD INDICATIONS OF > 60% THRU WALL WEAR  
(OF THESE, 8 HAD INDICATIONS OF > 90% THRU WALL WEAR)

## TYPICAL WESTINGHOUSE INCORE NEUTRON MONITORING SYSTEM



# TYPICAL WESTINGHOUSE INCORE NEUTRON MONITORING SYSTEM

