Attachment C (#2)

DIFFERENCES BETWEEN SANDIA AND
NUMARC ANALYSES OF DECAY HEAT REMOVAL
RELATED RISK FOR POINT BEACH

31 MARCH 1988

WHITE FLINT, MD

GLV:3867NS8

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

MAR 2 1988

MEETING NOTICE FOR: Distribution

FROM: Roy Woods, Task Manager
USI A-45

SUBJECT: DIFFERENCES BETWEEN SANDIA AND NUMARC ANALYSES OF DECAY HEAT REMOVAL RELATED RISK FOR POINT BEACH

TIME: 8:30 a.m.
March 31, 1988 (Thursday)

PLACE: White Flint 1 Building
Room 2F21

PURPOSE OF MEETING: NUMARC will present additional information requested by the NRC Staff and by Sandia personnel regarding the NUMARC analysis of decay heat removal related risk at the Point Beach Plant. Subject areas to be emphasized will be those agreed upon at the earlier 2/23/88 public meeting on the same subject with Sandia and NUMARC.

This additional information will be utilized by the NRC Staff in preparing an appendix for the USI A-45 Resolution Package. The appendix will outline the PRA methods and related assumptions that a licensee should use in a plant-specific PRA of decay heat removal related risk (such plant-specific PRAs will be the proposed resolution of USI A-45).

This meeting was requested by the NRC Staff, and is open to all interested members of the public.

The NRC contact is Roy Woods, telephone (301) 492-3568.

Roy Woods

Roy Woods, Task Manager
USI A-45

NSAC-113 MEETING AGENDA

8:30 OPENING REMARKS
(G. NEILS)

8:35 BACKGROUND TO NSAC-113
(G. VINE)

8:50 OVERVIEW OF POINT BEACH
(H. HANNEMAN)

9:20 COMPARISON OF CASE STUDY AND NSAC-113
(J. HAUGH)

9:50 BREAK

DISCUSSION OF NSAC-113 ISSUES

10:00 SMALL LOCA FREQUENCY
(W. PARKINSON)

10:15 PORV AND SRV EFFECT ON LOCA POTENTIAL
(D. PADDLEFORD)

10:30 CCW SUCCESS CRITERIA FOR HPI

(D. PADDLEFORD)

10:45 FIRE ANALYSIS

(W. PARKINSON)

- o METHODOLOGY DIFFERENCES
- o HALON SYSTEM RELIABILITY

11:15 INTERNAL FLOOD EVALUATION

(J. HAUGH)

- o THOMAS CORRELATION AND SW PUMP HOUSE EVALUATION

11:35 POINT BEACH VISUALS

(H. HANNEMAN)

12:00 LUNCH

1:00 COST ESTIMATE DIFFERENCES

(H. HANNEMAN)

- o GENERAL DIFFERENCES
- o SPRAY MODIFICATION
- o ADDITION OF DD AFW PUMP

1:20 RECOVERY ANALYSIS
(W. PARKINSON/
E. DOUGHERTY)

2:30 OTHER HRA ISSUES
(E. DOUGHERTY)
- o FEED AND BLEED
- o SUMP RECIRCULATION

3:00 OTHER RECOVERY ACTIONS
(H. HANNEMAN)
- o POINT BEACH PROCEDURES

3:30 SEISMIC HAZARD CURVE
(C. STEPP)

4:15 DISCUSSION AND WRAP-UP

5:00 ADJOURN

INDUSTRY SPONSORS/PARTICIPANTS

NUMARC	GERRY NEILS, NSP (W.G. CHAIRMAN) ROGER HUSTON, NUMARC STAFF
EPRI	JACK HAUGH GARY VINE CARL STEPP
WEP	ROGER NEWTON HARV HANNEMAN
WOG	WARREN ANDREWS
SAIC	BILL PARKINSON ED DOUGHERTY
W	DON PADDLEFORD

GLV:3867NS8

NUMARC WORKING GROUP
ON
DECAY HEAT REMOVAL
CHAIRMAN: GERRY NEILS

NAME, COMPANY	NSSS DESIGNS REPRESENTED W	GE	CE	B&W	CASE STUDY PLANTS	DHRTSG MEMBER
GERRY NEILS, NSP CHAIRMAN	X	X				
JEFF JEFFRIES, CP&L (NSAC T.F. CH.)	X	X				
ROGER NEWTON, WEP (WOG CH.)	X				POINT BEACH	
DAVE HELWIG OR GEORGE BECK, PECO (BWROG DHR CH.)		X				
ALAN LADIEU, YANKEE (WOG ANAL. CH.)	X					X
MIKE MEISNER, LP&L;			X			
DON JAMES OR MIKE SCHOPPMAN, FP&L (CEOG REP)	X		X		TURKEY POINT/ ST. LUCIE	
LARRY TAYLOR OR TED ENOS, AP&L			X	X	ANO-1	
GREGG SWINDLEHURST, DUKE	X			X		
XAVIER POLANSKI, COMMED	X	X			QUAD CITIES	
DON REEVES, NPPD		X			COOPER	
GARY VINE, EPRI (NSAC STAFF SUPPORT)						X

BRIEF HISTORY OF NSAC-113

- DEC 1985 — DHRTSG MEETING; REVIEW OF PB & QC DRAFTS
- FEB 1986 — COMMENT LETTERS TO SANDIA FROM EPRI, AIF
- MAR 1986 — DHRTSG MEETING, REVIEW OF PB/QC COMMENTS; DISCUSSION OF TP, COOPER. NRC REQUEST FOR INDUSTRY ANALYSIS
- JUN 1986 — EPRI INITIATED REANALYSIS OF POINT BEACH, SUPPORTED BY WOG AND WEP
- OCT 1986 — FIRST MEETING OF NUMARC WG; ENDORSED PB REANALYSIS EFFORT
- FEB-APR 1987 — ALL SIX CASE STUDIES DISTRIBUTED FOR FINAL REVIEW
- MAY 1987 — NUMARC (COUNCIL) INCORPORATED AND CHARTERED
- JUN 1987 — EPRI/NUMARC DHR WORKSHOP, NEW ORLEANS. NUMARC REVIEW COMMENTS ON ALL SIX CASE STUDIES. PB REANALYSIS RESULTS DISCUSSED.
- JUL 1987 — NRC (B. SHERON) LETTER TO NUMARC (G. NEILS) REQUESTING MORE INFORMATION ON PB REANALYSIS; SUGGESTED MEETING
- OCT 1987 — NSAC-113 (DRAFT) FORWARDED TO NRC BY G. NEILS
- NOV 1987 — A-45 PRESENTATION TO ACRS DHR S.C.; NSAC-113 DELIVERED TO ACRS
- JAN 1988 — A-45 PRESENTATION TO ACRS DHR S.C. ON NSAC-113

GLV:3867NS8

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

JUL 20 1987

Dr. Gerald Neils, Chairman
NUMARC Working Group on DHR
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Dr. Neils:

I received a copy of your June 22, 1987 communication to Dr. David Ericson on the subject of our USI A-45 Program on Shutdown Decay Heat Removal Requirements. Sandia National Laboratories (SNL) is in the process of studying these comments in detail; however, we are particularly interested in your indication that a separate PRA of one of the SNL Case Studies was sponsored by EPRI, the Westinghouse Owners Group, and Wisconsin Electric Power and concludes that core melt risk is about ten times lower than the SNL Case Study for the same plant. Since we perceive that this industry-sponsored study represents your quantification of the differing views outlined in your June 22nd letter, it would assist us in our deliberations to better understand your technical basis for these differences. Therefore, we ask that you identify the major items in your recent PRA study which contribute to the factor of ten difference in core melt probability, and present the technical basis for the value(s) selected (e.g., referenceable operating experience data base, human factor studies, component reliability data, external event initiating frequencies, etc).

We also wish to acknowledge the creation of the NUMARC Working Group to study the DHR issue, and we look forward to interfacing with members in the near future. Since our draft Regulatory Analysis on USI A-45 is still pre-decisional, perhaps we can consider a first meeting to focus on our review of the items you will identify as key contributors to the factor of ten difference in the core melt probability between the two PRAs. We appreciate the technical attention that NUMARC has apparently devoted to review of the six Case Studies, as indicated by Enclosure 1 of your June 22nd letter, and we intend to work with SNL to consider your comments.

Sincerely,

Brian W. Sheron

Brian W. Sheron, Director
Division of Reactor and Plant Systems
Office of Nuclear Regulatory Research

cc: D. Ericson

~~8708060056~~ 1p

NSP

Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401
Telephone (612) 330-5500

October 28, 1987

Dr. Brian W. Sheron, Director
Division of Reactor & Plant Systems
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Sheron:

In response to your letter dated July 20, 1987, I am pleased to forward a draft copy of a document entitled "EPRI/WOG Analysis of Decay Heat Removal Risk at Point Beach." This study, sponsored by EPRI and the Westinghouse Owners Group, was prepared by Science Applications International Corporation and Westinghouse Electric Corporation with the assistance of Wisconsin Electric Power Company, the owners and operators of Point Beach. The NUMARC Working Group on DHR has followed and endorsed this effort.

The primary purposes of this study were to provide a best-estimate analysis of DHR risk at a selected USI A-45 Case Study plant and to quantify the differences discussed in our June 22 comment letter on the Case Studies. The results of this Point Beach reanalysis, as they now stand, indicate an approximate factor of thirty reduction in core-melt frequency for the sequences included in the scope of the NRC study; an approximate factor of seven reduction in the offsite consequences of these sequences, over and above the core-melt frequency reduction; and an approximate 50-400% increase in the estimated cost of the various backfit proposals evaluated in the NRC study. The EPRI/WOG findings indicate that the core-melt frequency estimate for Point Beach (1.0×10^{-5} per reactor year) is a factor of ten lower than the core-melt frequency target in the NRC's Safety Goal. The EPRI/WOG study, like the NRC study, also concludes with a very high degree of confidence that an add-on, dedicated SDHR system would not be cost-beneficial for Point Beach.

We would be pleased to meet with you and members of your staff, as suggested in your letter, to discuss the methodologies, technical bases and findings contained in both studies. We have provided you with this draft report prior to publication to allow sufficient time for your staff to familiarize themselves with the EPRI/WOG reanalysis in advance the meeting. In anticipation of that meeting, EPRI and WOG are continuing to double-check the models used in their analysis against the final, as published, numerical values used in the NRC study. Although some small changes in the comparative estimates of core melt frequency could occur in some instances, we expect the

~~87~~12110067 2pp

overall results and conclusions of the EPRI/WOG study to remain essentially unchanged. In the meanwhile, we would be pleased to schedule a meeting for the first mutually convenient opportunity.

Sincerely,

G H Neils

G H Neils
Chairman
NUMARC Working Group on DHR

GHN/vf

cc: B. Lee, NUMARC
NUMARC Working Group Members
T. Speis, NRC
K. Kneil, NRC
A. Marchese, NRC
D. Ericson, Sandia National Laboratories
R. Newton, Wisconsin Electric Power Co.
W. J. Parkinson, SAIC
D. F. Paddleford, Westinghouse Electric Corp.
A. Ladieu, Chairman WOG Analysis Subcommittee
J. Taylor, EPRI
W. Layman, EPRI
T. Marston, EPRI
G. Vine, EPRI
J. Haugh, EPRI

OBJECTIVE OF NSAC-113

REANALYZE THE DHR RISK AT POINT BEACH, BY BEST ESTIMATE METHODS AND DATA. USE THE A-45 CASE STUDY OF POINT BEACH AS THE BASELINE AND POINT OF COMPARISON.

SPECIFIC OBJECTIVES

1. QUANTIFY THE CONSERVATISMS IN THE A-45 CASE STUDY, AND DEMONSTRATE THESE CONSERVATISMS AND LIMITATIONS CAN BE CORRECTED BY BEST-ESTIMATE ANALYSIS.

2. CONDUCT THE REANALYSIS WITH THE SAME SCOPE AND PLANT MODEL USED IN THE CASE STUDY, SO DIFFERENCES IN RESULTS CAN BE LIMITED TO DIFFERENCES IN INPUT DATA, SUCCESS CRITERIA, AND CONSTRAINTS ON NON-SAFETY EQUIPMENT AND HUMAN PERFORMANCE (PROVIDE FOR EASY SIDE-BY-SIDE COMPARISON).

3. DEMONSTRATE QUANTITATIVELY THE PARTICULARLY SIGNIFICANT SHORTCOMINGS IN THE CASE STUDY TREATMENT OF EXTERNAL EVENTS AND THE DEDICATED DHR SYSTEM.

4. RESPOND TO NRC REQUEST FOR QUANTITATIVE CRITIQUE OF CASE STUDIES. PROVIDE BETTER QUANTITATIVE BASIS FOR NRC REGULATORY ANALYSIS.

5. PROVIDE QUANTITATIVE BASIS FOR NRC/NUMARC DISCUSSIONS ON A-45 RESOLUTION.

GLV:3867NS8

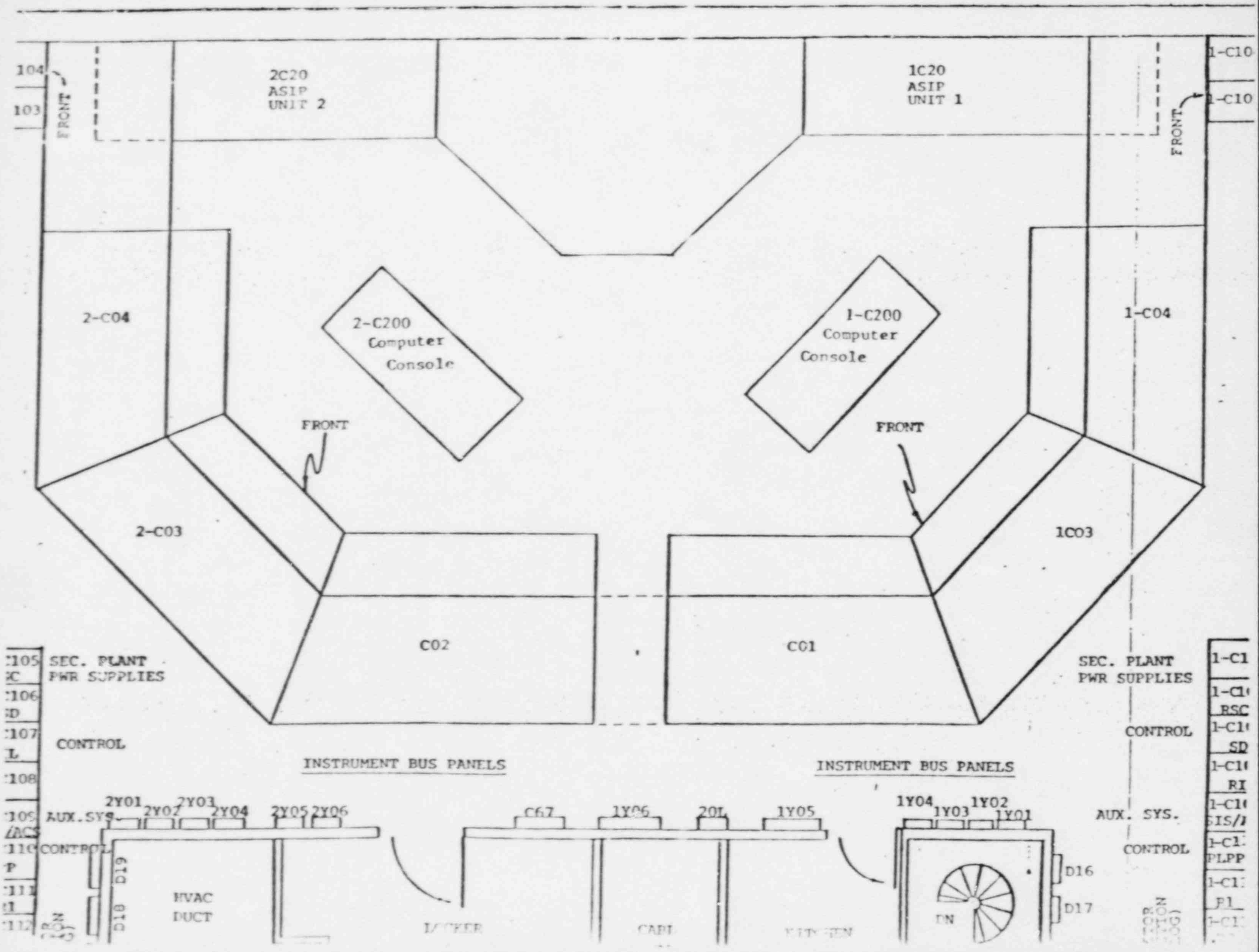
POINT BEACH DESIGN

1. Two Unit 2-Loop W PWR (497 MWe Net Each)
   - Commercial Operation: Unit 1 12/70 (17 years)
     Unit 2 10/72 (15 years)

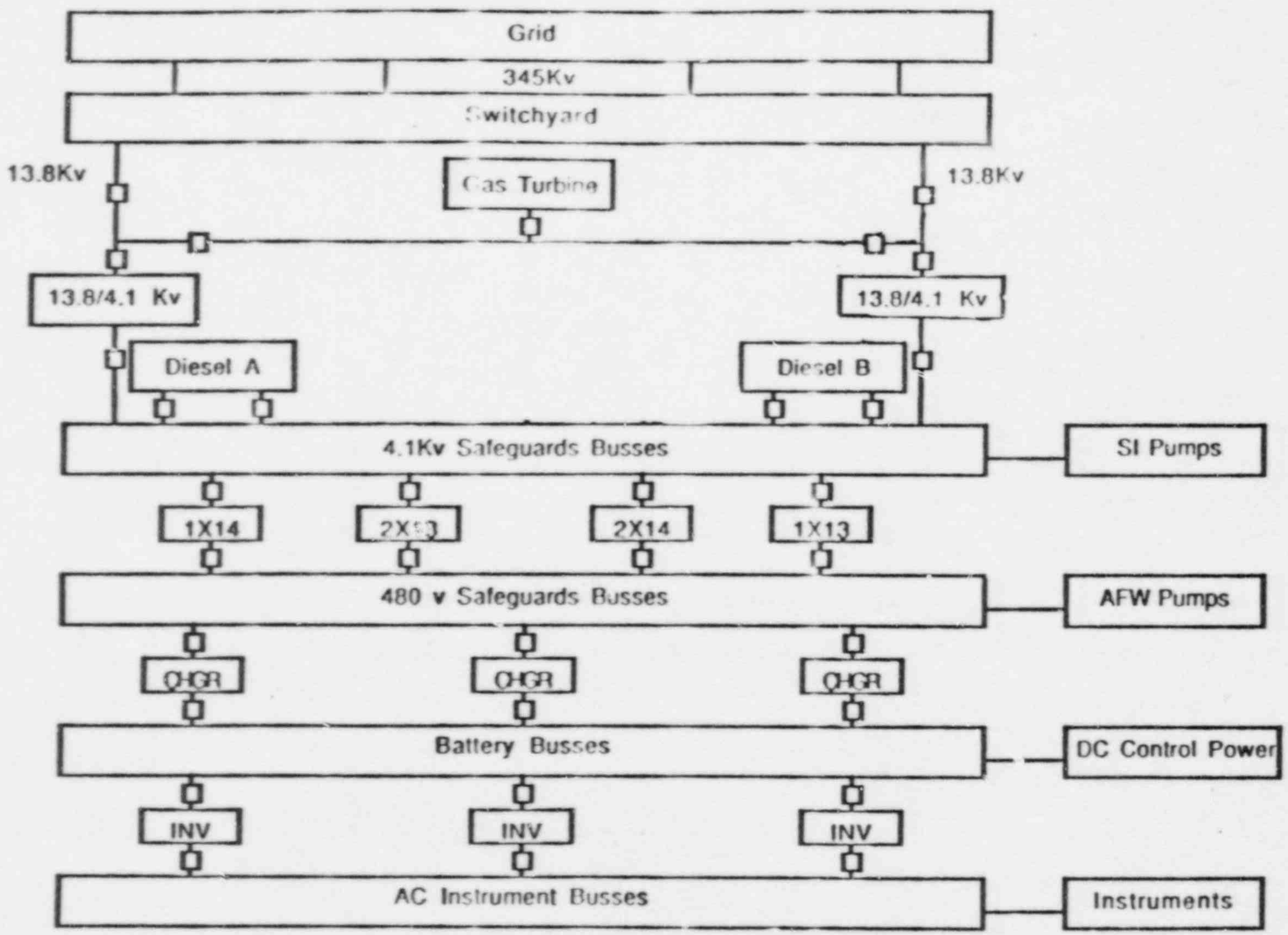
2. Common Control Room (see figure)

3. Common Safety Systems
   A. Emergency Power (AC and DC)
   B. Auxiliary Feedwater (Motor-Driven Pumps and CSTs)
   C. Service Water
   D. Spent Fuel Pool Cooling

4. Unit-Specific Safety Systems
   A. Reactor Protection
   B. Safety Injection (High and Low Head Pumps, Accumulators, RWST)
   C. Containment Isolation
   D. Containment Spray
   E. AFW (Turbine-Driven Pumps)
   F. CVCS (BAST; Charging Pumps-QA but Not Safety-Related)
   G. Component Cooling (Can Be Cross-Connected)
   H. Containment Emergency Fan Coolers.

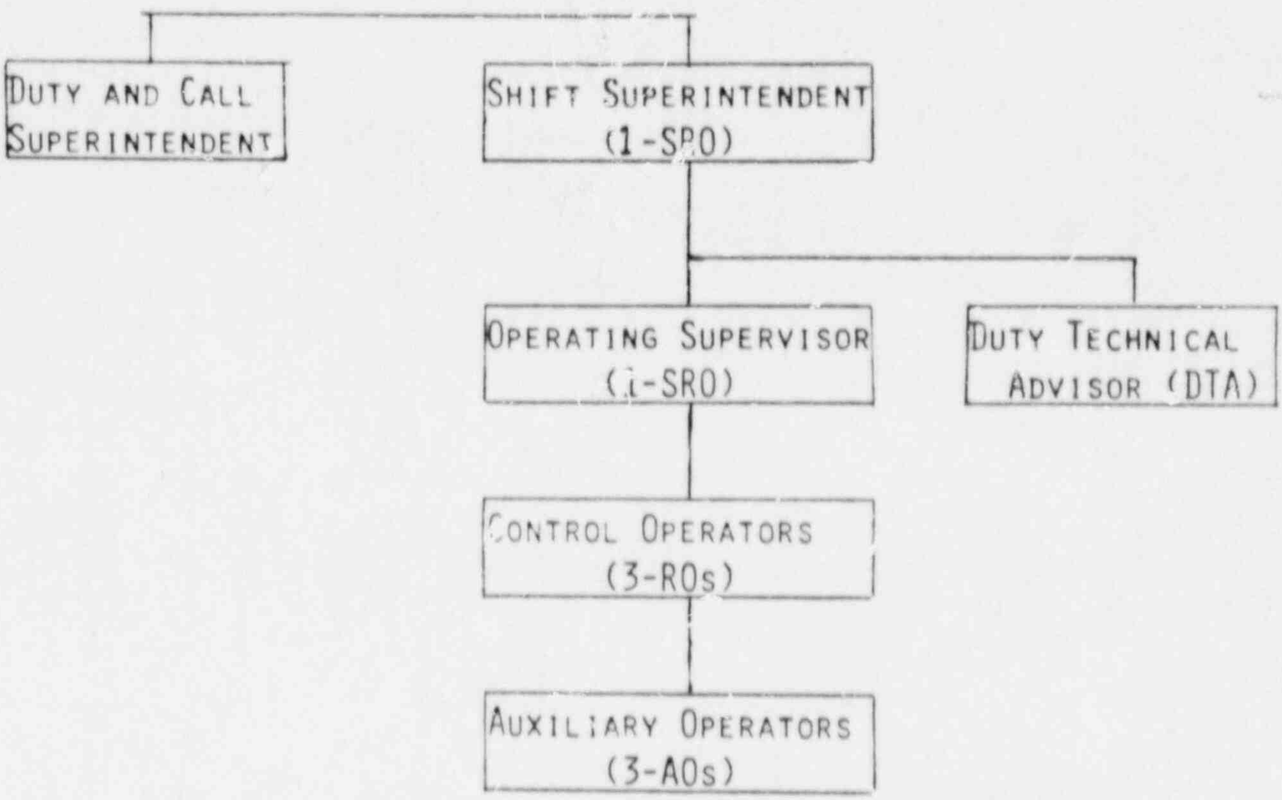


104
103
FRONT
2C20
ASIP
UNIT 2
1C20
ASIP
UNIT 1
1-C10
1-C10
FRONT
2-C04
2-C200
Computer
Console
1-C200
Computer
Console
1-C04
FRONT
FRONT
2-C03
1C03
C02
C01
SEC. PLANT
PWR SUPPLIES
CONTROL
SEC. PLANT
PWR SUPPLIES
CONTROL
INSTRUMENT BUS PANELS
INSTRUMENT BUS PANELS
AUX.SYS.
CONTROL
AUX. SYS.
CONTROL
2Y01
2Y02
2Y03
2Y04
2Y05
2Y06
C67
1Y06
20L
1Y05
1Y04
1Y03
1Y02
1Y01
D19
D18
D16
D17
HVAC
DUCT
DN
BSC
SD
RI
PLPP



Schematic of Point Beach Offsite Power Distribution.

PBNP OPERATOR STAFFING



NOTES:

1. ABOVE STAFFING LEVELS ARE FOR TWO-UNIT OPERATION.
2. ONE SRO (SS OR OS) AND ONE RO PER UNIT ARE IN THE CONTROL ROOM AT ALL TIMES.
3. DTA IS INSIDE SITE BOUNDARY AND WITHIN 10 MINUTES OF THE CONTROL ROOM.
4. DUTY AND CALL SUPERINTENDENT IS ON CALL AND WITHIN 30 MINUTES OF THE PLANT.
5. FIVE-MAN FIRE BRIGADE CONSISTS OF ONE SRO (OP SUP) AS LEADER AND FOUR OTHER OPERATORS (1RO AND 3 AOs). (TWO CREEKS FIRE DEPARTMENT IS TWO MILES FROM SITE)
6. THERE IS NORMALLY ONE AO IN AUXILIARY BLDG. AND TWO AOs IN THE TURBINE HALL (ONE FOR EACH UNIT).
7. EMERGENCY PLAN IMPLEMENTATION CAN AUGMENT ABOVE STAFF WITHIN 30 MINUTES (DCS, HP, I&C, RAD CHEM, TSC MANAGER, BACKUP OPERATORS) AND ONE HOUR (TSC AND EOF)

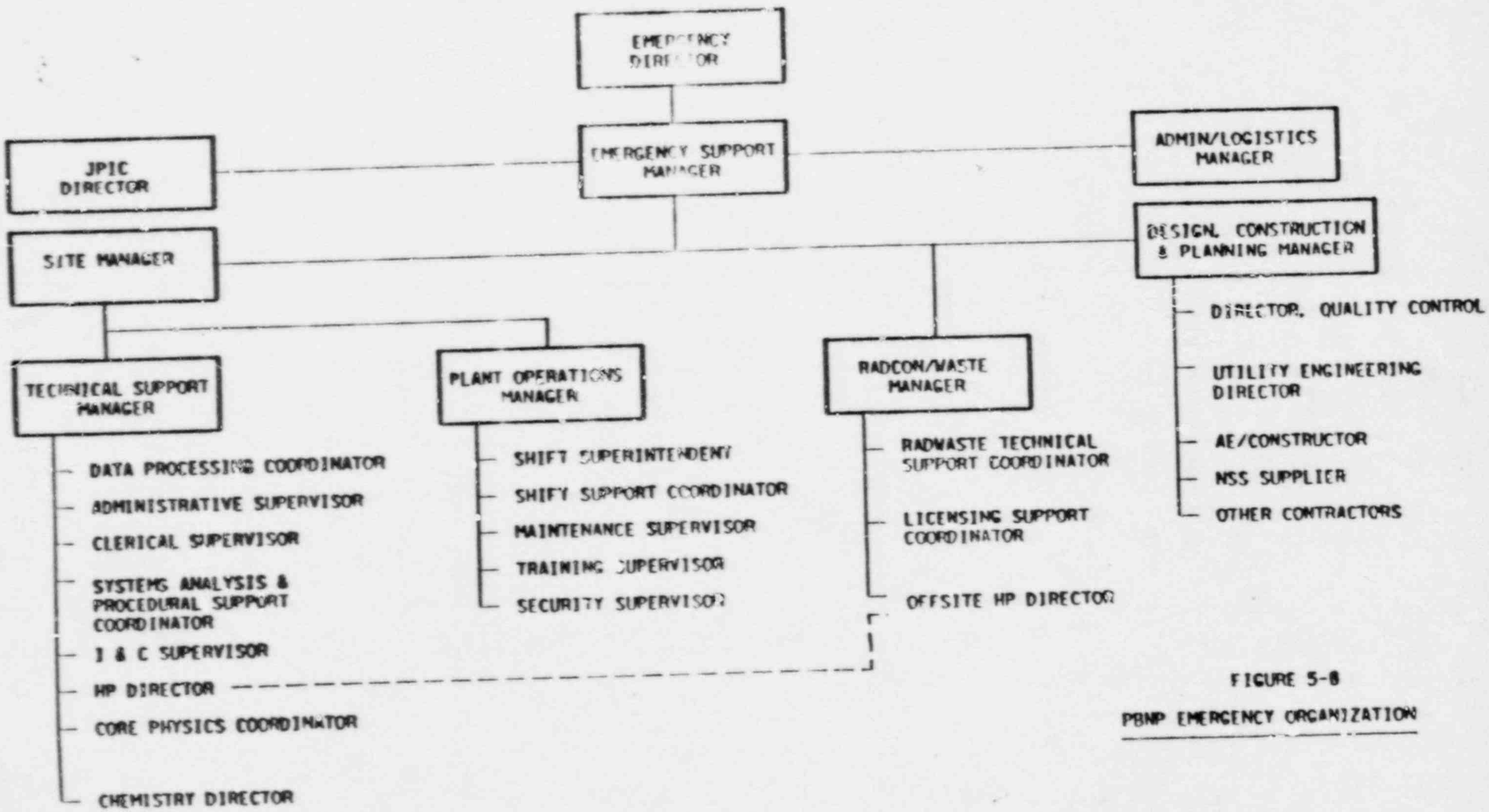


FIGURE 5-8

PBNP EMERGENCY ORGANIZATION

NOTE: THE HEALTH PHYSICS DIRECTOR (HEALTH PHYSICIST) ALSO HAS AN OPTIONAL REPORTING LINE (NOT SHOWN) DIRECTLY TO THE SITE MANGER

CCS FILE NO. 4811

POINT BEACH OPERATING PROCEDURES

1. OPERATING PROCEDURES (OPs) - NORMAL PLANT OPERATING PROCEDURES

   EXAMPLES:
   A. PLANT STARTUP, HEATUP, POWER OPERATION, SHUTDOWN, AND COOLDOWN
   B. RCP OPERATION
   C. EMERGENCY DIESEL GENERATOR
   D. GAS TURBINE OPERATION

2. REFUELING PROCEDURES (RPs) - NOT APPLICABLE TO DESIGN BASIS EVENTS

3. OPERATING INSTRUCTIONS (OIs)

   EXAMPLES:
   A. CHARGING PUMP LOCAL CONTROL
   B. HALON FIRE PROTECTION SYSTEM
   C. MOTOR-DRIVEN AFW SYSTEM
   D. TURBINE-DRIVEN AFW SYSTEM
   E. SERVICE WATER SYSTEM

4. ABNORMAL OPERATING PROCEDURES (AOPs)

   EXAMPLES:
   A. AUXILIARY FEED PUMP STEAM BINDING OR OVERHEATING
   B. EMERGENCY BORATION
   C. SERVICE WATER SYSTEM MALFUNCTION
   D. LOSS OF COMPONENT COOLING
   E. CONTROL ROOM INACCESSIBILITY
   F. SAFE TO COLD SHSUTDOWN IN LOCAL CONTROL

5. FIRE PROTECTION MANUAL

A. FIRE ATTACK PLANS (GENERAL, A, B, C FIRES, DIFFERENT EQUIPMENT)

B. FIRE EMERGENCY PLANS (DIFFERENT AREAS OF THE PLANT)

C. TRANSIENT COMBUSTIBLE CONTROL PROCEDURE

D. IGNITION CONTROL PROCEDURE

POINT BEACH EMERGENCY OPERATING PROCEDURES

- EOPs Were Developed from WOG Emergency Response Guidelines, Rev. 1

- Entry Procedures: EOP-0 Reactor Trip or Safety Injection
  ECA-0.0 Loss of All AC Power

- Supplementary and Emergency Contingency Actions (ECAs)

  Examples: a. EOP-1 Loss of Reactor of Secondary Coolant
  EOP-1.2 - Small-Break LOCA Cooldown and Depressurization
  EOP-1.3 - Transfer to Containment Sump Recirculation
  b. EOP-2 Faulted Steam Generator Isolation
  c. EOP-3 Steam Generator Tube Rupture

- Critical Safety Function Status Trees (Hardcopy and SAS Computer)

  Subcriticality
  Core Cooling
  Secondary Heat Link
  Reactor Vessel Integrity
  Containment Integrity
  Reactor Coolant Inventory

- Critical Safety Procedures

PBNP D.C POWER SYSTEM

- DC System is Shared Between Unit 1 and Unit 2

- D.C. System Consists of 4 Main Distribution Buses

- Each Main Distribution Bus Continuously Powered from Two Sources

  1. Batteries: D05, D06 — Original Station Batteries

     D105, D106 — New Station Batteries (1985)

  2. Battery Chargers-Supplied from 480V Safeguards Buses

- Battery Design - D05 & 6 - 60 Cell (Nom)

  125 Volts

  950 Amp-Hr (1 Hr. Rate)

  Seismically Analyzed (SSE)

  Designed for SBO Loads for 1 Hour

  D105 & 6 - 60 Cell (Nom)

  125 Volts

  795 Amp-Hr (1 Hr. Rate)

  Seismically Qualified-IEEE 344

  Designed for SBO Loads for 1 Hour

BATTERY MAJOR LOADS:

D05 & 6 (ORIGINAL BATTERIES)

- ESTIMATED DEMAND: 400 AMPS
  - DC CONTROL POWER (BKRS, SOLENOID VALVES, RELAYS, ANNUNCIATORS, ETC.)
  - MOVS FOR TD AFW PUMPS
  - INVERTERS FOR RED AND BLUE INSTRUMENT BUSES
  - EMERGENCY DC LIGHTING
  - EDG STARTING CIRCUITS & FIELD FLASHING
  - VARIOUS EMERGENCY OIL PUMPS (TURBINE & FEEDPUMP)

D105 & 6 (NEW BATTERIES)

- ESTIMATED DEMAND: 180 AMPS
  - BACKUP FOR EDG STARTING CIRCUITS/FIELD FLASHING
  - ALTERNATE SHUTDOWN INSTRUMENTS
  - INVERTERS FOR WHITE & YELLOW INSTRUMENT BUSES
  - SAS (SPDS) AND PPCS COMPUTERS

POINT BEACH DIESEL GENERATOR

- TWO EMERGENCY DIESEL GENERATORS - SHARED BETWEEN UNITS

- EACH EDG DESIGNED TO POWER ALL SAFEGUARDS LOADS NECESSARY FOR DESIGN-BASIS ACCIDENT MITIGATION IN ONE UNIT AND ALL NECESSARY SAFE SHUTDOWN LOADS IN OTHER UNIT

- DESIGN: GENERAL MOTORS ELECTRO-MOTIVE DIVISION

  MODEL 999-20 (20 CYLINDERS)

  RATING: 2850 KW (CONT.)

  3050 KW (30 MINUTES)

  4160 VOLTS, 3Ø, 60 HZ.

  FUEL OIL USAGE: 205 GAL/HR AT RATED LOAD

  FUEL OIL AVAILABLE: 550 GALLON BASE TANK

  550 GALLON "DAY TANK" FOR EACH DG (CAN BE CROSS-CONNECTED)

  12,000 GALLON UNDERGROUND EMERGENCY FUEL OIL TANK

  60,000 GALLON ABOVE-GROUND STORAGE TANKS (1 OF 2 NORMALLY FULL)

- EDG STARTING CIRCUITS

  REQUIREMENTS: SOLENOID VALVES FOR AIR START SUPPLY

  FIELD FLASHING

  CONTROL/ALARM CIRCUITS

  POWER: 1. DC FROM OLD STATION BATTERIES

  2. ALTERNATE FROM NEW BATTERIES (SWITCHED LOCALLY)

- EDG Reliability at Point Beach

	G01	G02	Both
No of Demands (1983 - 1987)	174	173	347
# of Demand Failures	1	0	1

Probability of Start Failure - Actual (1983-87) 2.9E-3

Hours Out of Service for Annual Maintenance (1985-1987)	529	360	889
Probability of One EDG For Maintentance	2.0E-2	1.4E-2	1.7E-2(Avg)

Total Probability of One EDG Out of Service or Failure to Start on Demand -

Actual PBNP Data (1983/85-1987)	- 2.0E-2
EPRI/WOG Study	- 2.8E-2
SNL Study	- 4.4E-2

PBNP RWSTs/CSTs

Refueling Water Storage Tanks (RWSTs)

- One Per Unit (285,000 gallons)
- 1/4"-.32" Welded Stainless Steel Shell
- 1/4" Stainless Steel Bottom
- 27' ID x 70' High (Aspects Ratio = 2.6)
- Anchorage - 27 Bolts (1 1/4" - Dia. x 4' Long)
- Designed as Seismic Class I for SSE of .12g PGA
- Mounted on Concrete Pad

Condensate Storage Tanks (CSTs)

- Two Total (Shared and Normally X-Connected-40,000 Gal. Ea)
- 1/4" Welded Carbon Steel Shell and Bottom
- 20' ID x 24' High (Aspect Ratio = 1.2)
- Anchorage - 8 Bolts (3/4" Dia. x 1 1/2' Long)
- Seismic Class III (Non-Seismic)
- Mounted on Roof of Seismic Class I Control Building inside Turbine Building

COMPONENT COOLING WATER

- Purpose: Intermediate Cooling System to Separate Radioactive Reactor Coolant from Service Water (Lake Michigan) to Minimize Potential for Radioactive Releases

- One Component Cooling Loop for each Unit

- One Pump and One Heat Exchanger Required/Unit
  (Normal Operation or Design Basis Accident)

- Two Pumps and One Unit-Specific HX/Unit with Two Shared Backup HXs
  (Each Units Pumps Can Also Be Cross-Connected)

- Pumps Powered from Safeguards 480V Buses
  (i.e., Off-Site Power, On-Site Gas Turbine Generator, or Emergency Diesel Generators)

- Important Loads from Risk Perspective
  1. RHR HXs (Cooling for ECCS Recirculation- or RHR)
  2. RHR (Low-Head SI) Pumps - Seal Cooling Only
  3. SI Pumps - Seal Cooling Only
  4. CS Pumps - Seal Cooling Only
  5. RCP Thermal Barrier (Seal Cooling-Backup to Seal Injection)

POINT BEACH PORVs, BLOCK VALVES, AND ALTERNATE VENTS

PORVs - AIR OPERATED VALVES

1. PNUEMATIC SUPPLY: INSTRUMENT AIR TO CONTAINMENT
   (N_2 BOTTLE BACKUP - LTOP ONLY)

2. SOLENOID VALVE POWER: 125 V. D.C.

BLOCK VALVES - MOTOR-OPERATED VALVES

1. POWERED FROM SAFEGUARDS 480 V MCC

2. BLOCK VALVE STATISTICS:

	1986 (# OF SHIFTS)	SEPT-DEC 1987 (# OF SHIFTS)		PERCENTAGE OF TIME
	UNIT 1	UNIT 1	UNIT 2	
BOTH CLOSED	307	0	43	22%
ONE CLOSED	465	0	5	30%
BOTH OPEN	272	294	192	48%

ALTERNATE VENT PATHS

1. REACTOR COOLANT GAS VENT SYSTEM - DC POWER
   (REACTOR VESSEL AND PRESSURIZER VENTS-7/32" ORIFICE)
2. LOOP DRAINS - NON-SAFEGUARDS AC POWER
3. LETDOWN & EXCESS LETDOWN - DC & SAFEGUARDS AC POWER
4. SAMPLING - DC POWER

EPRI/WOG Study Results

Source of Risk	Core Melt Frequency per Year NRC	EPRI/WOG	Reduction Factor
Internal	1.4E-4	2.6E-6	54
Seismic	6.1E-5	7.4E-6	8
Fire	3.2E-5	6.3E-8	500
Internal Flood	7.7E-5	<1.0E-8	>7700
External Flood	1.9E-8	<1.0E-8*	(>>2)
Wind	4.0E-6	<1.0E-8*	(>>400)
Lightning	5.8E-8	<1.0E-8*	(>>6)
Total	3.1E-4	1.0E-5	31

* Core melt frequency reduced to <1.0E-8 by explanation of errors in NRC Case Study without RMQS requantification.

INTERNAL EVENT

S_2 M H1' H2': SBLOCA + MFW FAILURE + HPRS & LPRS FAILURES W/O RHR H/X

ESTIMATES: 4.70E-5 CASE STUDY
5.80E-7 NSAC-113

PRINCIPAL DIFFERENCES:

1. SBLOCA FREQUENCY

CASE STUDY: 2.0E-2 BASED ON LEAKS <2-IN. DIA; DERIVED FROM SNL/IREP ANO-1 AND MURLEY MEMO: RCP SEAL LOCAS DOMINATE

NSAC-113: 3.0E-3 BASED ON LEAKS <2.0IN. DIA; DERIVED FROM OCONEE PRA AND INDUSTRY EXPERIENCE. CREDITS SHUTDOWN PRIOR TO RECIRCULATION ~ 20 HRS; SUMP RECIRCULATION NOT REQUIRED FOR LOCAS EXPERIENCED SO FAR

CRITIQUES OF NSAC-113:

GENERALLY ASSESS NSAC-113 ESTIMATE AS REASONABLE CONTINGENT ON SUPPORTING INFORMATION

INTERNAL EVENT (CONT'D)

2. OPERATORS FAIL TO IMPLEMENT SUMP RECIRCULATION

CASE STUDY: 1E-3

NSAC-113: 1E-4 BASED ON RULE VS DIAGNOSIS AND LONG TIME TO DEPLETE RWST

CRITIQUES OF NSAC-113:

NEED ADDITIONAL EXPLANATION

3. RECOVERY FROM RECIRCULATION FAULTS

CASE STUDY: NO CREDIT

NSAC-113: 5E-2 BASED ON REFILLING RWST OR USING CVCS; NOT APPLIED TO OPERATOR FAILURE TO IMPLEMENT RECIRCULATION

CRITIQUES OF NSAC-113:

NEED BACKUP ON PROCEDURES/TRAINING

4. CCW SUCCESS CRITERIA

APPLIED ONLY TO HPI (I.E., INJECTION FAULTS)

INCORRECT ENTRY IN TABLES 1-8, 8-3, OF NSAC-113

INTERNAL EVENT

$S_2\ M^{D_1 D_2}$: SBLOCA + MFW FAILURE + HPIS & LPIS FAILURE

ESTIMATES: 8.7E-6 CASE STUDY
9.5E-8 NSAC-113

PRINCIPLE DIFFERENCES:

1. SBLOCA FREQUENCY - SAME COMMENTS AS $S_2\ MH_1'\ H_2'$
2. CCW SUCCESS CRITERIA -

CASE STUDY: HPI DEPENDENT ON CCW AND SW AVAILABILITY

NSAC-113: CCW NOT REQUIRED FOR PT. BEACH HPI (INJECTION MODE). RECOVERY MUST OCCUR PRIOR TO SUMP RECIRCULATION

INTERNAL EVENT

T_1MLE: LOSP + MFW FAILURE + AFW FAILURE + F&B FAILURE

ESTIMATES: 6.7 E-6 CASE STUDY
7.7 E-7 NSAC-113

PRINCIPLE DIFFERENCES:

1. USE OF NEW STATION BATTERIES

CASE STUDY: ANALYSIS DID NOT INCLUDE

NSAC-113: INCLUDED OPERATOR ACTION TO USE NEW BATTERIES TO START DIESELS AFTER COMMON CAUSE FAILURE OF STATION BATTERIES

CRITIQUES OF NSAC-113:
CONCUR WITH CREDITING "NEW" BATTERIES CONTINGENT ON SUPPORTING INFORMATION

NOTE: LOSP DATA WAS OF MINOR IMPORTANCE

CASE STUDY: 8.4 E-2 BASED ON NRC GENERIC ESTIMATE (NUREG-1032)
INTERNAL EVENT (CONT'D)

NSAC-113: 6.2 E-2 BASED ON PT. BEACH SPECIFIC DATA

CRITIQUES OF NSAC-113:

- GENERALLY AGREE WITH NSAC-113
- APPLICATION OF PLANT SPECIFIC DATA

INTERNAL EVENT

T_3 Q H_1' H_2' AND T_3 Q D_1 D_2: TRANSIENT (MFW UNAVAILABLE) + SRVs FAIL TO CLOSE + EITHER (HPRS & LPRS FAILURE W/O RHR H/X) OR (HPIS & LPIS FAILURE)

ESTIMATES: 2.5 E-5 + 4.6 E-6 = 3 E-5 CASE STUDY
N/A NSAC-113

PRINCIPLE DIFFERENCES

CASE STUDY: SRVs ASSUMED TO OPEN -- FAILURE TO RECLOSE (EVENT Q) RESULTS IN TRANSIENT INDUCED LOCA

NSAC-113: EVENT Q SEQUENCES DO NOT EXIST FOR REACTOR OR TURBINE TRIPS AT PT. BEACH. NEITHER PORVs NOR SRVs WILL BE CHALLENGED BASED ON WESTINGHOUSE OPERATING EXPERIENCE.
EVENT Q SEQUENCES CONSERVATIVELY MODELED FOR LOSS OF OFFSITE POWER AND LOSS OF MAIN FEEDWATER

INTERNAL EVENT (CONT'D)

CRITIQUES OF NSAC-113:

GENERALLY AGREE WITH NSAC- 113
CONCLUSION CONTINGENT ON
THERMAL/HYDRAULICS ANALYSES

$T_2MQH'_1H_2$: LOSS OF PCS + MFW FAILURE + SRVs/PORVs FAIL TO CLOSE + HPRS & LPRS FAILURE W/O RHR H/X

PRINCIPAL DIFFERENCES:

1. SAME AS FOR S_2 INIATOR

2. STUCK OPEN PORV -

CASE STUDY: 1.4E-3 PER DEMAND. ASSUMES A PORV STICKS OPEN 7% OF THE TIME AND BLOCK VALVE FAILURE TO ISOLATE IS 1.0E-2 PER VALVE

NSAC-113: CONSERVATIVELY ASSUMES BOTH PORVs DEMAND OPEN, AND 1% OF THE TIME ONE STICKS OPEN. ALLOWS 30 MIN FOR BLOCK VALVE CLOSURE

CRITIQUES OF NSAC-113:

ACCEPTANCE OF NSAC-113 IS CONTINGENT ON VERIFICATION OF BLOCK VALVE CLOSED-CLOSED FREQUENCY

INTERNAL FLOOD SEQUENCES

ESTIMATES FOR ALL SEQUENCES:
7.7E-5 CASE STUDY
<1.0E-8 NSAC-113

PRINCIPAL DIFFERENCES:

SW PUMP HOUSE - FIRE MAIN RUPTURE

CASE STUDY: BASED ON GENERIC AUX BLDG FLOOD DATA (2.2E-2 FOR MODERATE FLOOD); 1.0E-1 ASSIGNED TO SIMULTANEOUS FAILURE OF SIX SW PUMPS; 3.48E-2 TDAFW PUMP FAILURE. TOTAL = 7.66E-5

NSAC-113: BASED ON APPLICATION OF THOMAS PIPE RUPTURE CORRELATION SIMILAR TO OCONEE PRA TOTAL ESTIMATED WAS 9.8E-5 FOR LEAKS SUFFICIENT TO DAMAGE SW PUMPS. DID NOT APPORTION FREQUENCY ACCORDING TO BREAK SIZE AS IN OCONEE PRA. DID NOT DIRECTLY APPLY 1.0E-1 FACTOR FOR SW PUMP FAILURE AS IN CASE STUDY; ASSUMED BREAK LOCATION INSTEAD.

INTERNAL FLOOD SEQUENCES (CONT'D)

CRITIQUES OF NSAC-113:

FURTHER JUSTIFICATION OF NSAC-113 METHODOLOGY IS REQUIRED.

SEISMIC EVALUATIONS

ESTIMATES FOR ALL SEQUENCES:

6.1E-5 CASE STUDY

7.4E-6 NSAC-113

PRINCIPAL DIFFERENCES:

CASE STUDY:

- o GENERATED A SEISMIC HAZARD CURVE BASED ON ZION (SSMRP).
- o CALCULATED RWST FAILURE DUE TO BUCKLING AND ANCHOR PULLOUT.
- o NO CREDIT GIVEN FOR SEISMIC RECOVERY ACTIONS.
- o DID NOT INCLUDE NEW STATION BATTERIES.

NSAC-113:

- o SEISMIC HAZARD CURVE WAS REDUCED FROM CASE STUDY VALUES BY FACTOR OF TWO FOR 1-3XSSE AND FIVE FOR >3XSSE.
- o RWST NOT EXPECTED TO FAIL CATASTROPHICALLY (INSTANTANEOUSLY) AT LOW ACCELERATIONS; ALLOWS ADDITIONAL TIME FOR RECOVERY.

SEISMIC EVALUATIONS (CONT'D)

- ALLOWED CREDIT FOR RECOVERY ACTIONS
  - E.G., USE OF ALTERNATE WATER SUPPLIES IN PLACE OF CST AND RWST
- INCLUDED NEW SEISMIC I BATTERIES.

CRITIQUES OF NSAC-113

- UNABLE TO EVALUATE ADEQUACY OF RWST AND NSAC-113 MODIFIED HAZARD CURVE WITHOUT MORE DETAILS.
- GENERALLY CONCUR WITH ALLOWING CREDIT FOR RECOVERY PROVIDED THOSE ACTIONS CAN BE SUBSTANTIATED.
- CONCUR WITH CREDITING NEW BATTERIES. CONTINGENT ON REVIEW OF BATTERY DESIGN DETAILS.

FIRE EVALUATION

ESTIMATES FOR ALL SEQUENCES:

3.2E-5 CASE STUDY

6.3E-8 NSAC-113

PRINCIPAL DIFFERENCES:

CASE STUDY:

- o CREDITS TWO TRAIN HALON SYSTEM IN SWITCHGEAR ROOM FIRE
- o CREDITS 1 TRAIN HALON SYSTEM IN A FW PUMP ROOM FIRE
- o HALON SYSTEM FAILURE ESTIMATED AT 0.2 PER DEMAND BASED ON DATA REPORTED IN MILLSTONE PRA ET AL.
- o TDAFW PUMP FAILURE ESTIMATED AT 0.1 DURING AFW PUMP ROOM FIRE. NO CREDIT GIVEN FOR 4160V SWITCHGEAR ROOM FIRE.
- o AUX BLDG GENERIC FIRE FREQUENCY DATA RATIONED BY THE AMOUNT OF COMBUSTIBLES IN THE AFW PUMP ROOM TO THAT IN ENTIRE AUX BLDG. TRANSIENT COMBUSTIBLE FIRE EVALUATED USING UCLA COMPBURN (1983) CODE.
- o DID NOT INCLUDE FIRES <u>STARTING</u> IN CABLE TRAYS AND ELECTRICAL PANELS

FIRE EVALUATION (CONT'D)

NSAC-113:

- o CREDITS 2 TRAIN HALON SYSTEM FOR ALL SEQUENCES.
- o HALON SYSTEM FAILURE ESTIMATED AT 0.06 DERIVED FROM DOE HALON SYSTEM RELIABILITY DATA BASED ON ACTUAL FIRES.
- o FREQUENCY OF TRANSIENT COMBUSTIBLE FIRE ESTIMATED TO BE OF LESSER IMPORTANCE. DID NOT USE COMPBURN.
- o CONSIDERED FIRES <u>STARTING</u> IN CABLE TRAYS AND ELECTRICAL PANELS BASED ON GENERIC PLANT DATA.

CRITIQUES OF NSAC-113:

ENDORSED GIVING CREDIT FOR SECOND HALON TRAIN. GENERALLY PREFERRED THE CASE STUDY APPROACH FOR INITIATING EVENT DATA; QUESTIONS ASKED ABOUT USE OF DOE HALON RELIABILITY DATA.

SMALL LOCA FREQUENCY

NRC Case Study Assumptions

EPRI/WOG Assumptions

Need for Recirculation

Summary of Other PRAs

Small LOCA Classes

Conclusion

NRC CASE STUDY ASSUMPTIONS

All Small LOCAs Require Recirculation

Murley to Eisenhut Memo on RCP Seal Failure

Charging System Not Available

EPRI/WOG ASSUMPTIONS

Experience Justifies a Rare Event Probability for Small LOCAs Requiring Recirculation

Data Is Adequate to Model Likelihood Of Reaching RHR or Isolating the Break Before Recirculation is Required

Small LOCA Frequency is Valid for Injection Faults

NEED FOR RECIRCULATION

Plant Experience Indicates that Shutdown Can Occur Prior to Emptying the RWST

Generic Thermal Hydraulics Analysis Shows RHR Entry Can Be Reached Prior to Sump Recirculation

NRC Case Study Makes Similar Assumption Regarding More Stressful Case, I.E., No Injection

SUMMARY OF OTHER PRAS

Most Reference Murley Memo Directly or Indirectly

Pipe Breaks Generally Assessed With Lower Frequency Than RCP Seal LOCA

Categorization Scheme Might Be Best

COMPARISON OF EPRI/WOG AND NRC CASE STUDY SMALL LOCA FREQUENCY TO OTHER PRAS

PRA Study	Value	Comments/References
NRC Case Study	2.0E-2	IREP Procedures Guide which references ANO IREP which references Murley memo
EPRI/WOG Study	3.0E-3	Oconee PRA which references isolable small LOCA at Zion as one event in total PWR experience as of early 1980s
WASH-1400	1.0E-3	Based on nuclear and non-nuclear experience
ANO-1 IREP Calvert Cliffs IREP	2.0E-2	Both studies reference memo, identify the value as an RCP seal LOCA value and identify a pipe break value of 1.0E-3
Oconee PRA	3.0E-3	Based on one event in approximately 160 years of PWR experience and updated with no events at Oconee
Indian Point	2.0E-2	Based on three events, the isolable Zion LOCA in 1975, an RTD blowout at Surry in 1972, and an RCP seal failure at Indian Point in 1977
Seabrook PRA	2.8E-2	Basis, i.e., data, is proprietary. Event broken down further to yield: LOCAs requiring recirculation - 1.0E-3 non-isolable LOCAs not requiring recirculation - 5.8E-3, and isolable LOCAs not requiring recirculation - 2.3E-2
ASEP (NUREG-1150)	2.0E-2	Basis is an average of many of above PRAs. Dismisses Murley memo, but includes indirectly by referencing other PRAs which in turn reference the memo

SMALL LOCA CATEGORIES

Small LOCAs Requiring Recirculation

Isolable Small LOCAs Not Requiring Recirculation

Non-Isolable Small LOCAs Not Requiring Recirculation

CONCLUSION

EPRI/WOG May Overestimate Small LOCAs Requiring Recirculation (Dominant Accident)

EPRI/WOG May Underestimate Small LOCAs Requiring Injection (20% of Total Small LOCA Contribution)

VERY SMALL LOCA RECOVERY W/O SUMP RECIRCULATION

O NRC CASE STUDY INCLUDES VERY SMALL LOCA CATEGORY - FREQ OF 0.02/YR WITH SUBSEQUENT FAILURE OF SUMP RECIRCULATION AS SIGNIFICANT

O BUT LOCA WITH FREQ DOWN TO AT LEAST 0.002/YR ARE RECOVERED WITHOUT SUMP RECIRCULATION

O CONFIRMED BY EXPERIENCE SINCE NO SUMP RECIRCULATION EVENTS IN OVER 500 PLANT YEARS

O WOG DEVELOPED EMERGENCY RESPONSE GUIDELINE ES-1.2, "POST LOCA COOLDOWN AND DEPRESSURIZATION" TO REDUCE RCS TEMP AND PRESS BELOW 200F AND 400 PSIG RHR ENTRY CONDITIONS

O ACCOMPLISHED BY COOLDOWN WITH A SG AND REDUCING SI FLOW AND ESTABLISHING NORMAL CHARGING WHEN MINIMUM SUBCOOLING AND PRESSURIZER LEVEL CONDITIONS ARE MET

O BACKGROUND DOCUMENT FOR ES-1.2 INCLUDES GENERIC APPLICATION (TREAT T/H CODE) TO A ONE INCH COLD LEG BREAK CASE AND TO A STUCK OPEN PORV CASE. IN BOTH, RHR ENTRY CONDITIONS REACHED WELL BEFORE RWST DRAINED

O WEP HAS DEVELOPED PT. BEACH SPECIFIC PROCEDURE CONSISTENT WITH ES-1.2

FREQUENCY OF SAFETY VALVE OPENING IN W PWR'S

- O NRC CASE STUDY PRESUMES 7 TRANSIENT/RY WITH 7% CHANCE OF OPENING SRV - OR ABOUT 0.5/RY

- O W SURVEY AND ANALYSIS INDICATES PORV OPENINGS FROM OPERATIONAL TRANSIENTS AT FREQ OF ABOUT 0.23/RY PRE TMI AND 0.12/RY POST TMI

- O W SAFETY VALVE OPENINGS EXTREMELY RARE - ONE THE ORDER 0.01/RY OR LESS

- O TRANSIENTS OPENING PORV RELATIVELY INFREQUENT

  - TRANSIENTS WITH DIRECT OR ANTICIPATORY REACTOR AND TURBINE TRIP DO NOT OPEN PORV'S

  - TRANSIENTS WITH LARGE PRIMARY SECONDARY MISMATCH AND NO IMMEDIATE TRIP CAN CAUSE PORV'S TO OPEN (LARGE LOSS OF STEAM LOAD, MSIV CLOSURE), REDUCED COOLING (LOCKED ROTOR), REACTIVITY ADDITION (ROD WITHDRAWAL), AND MASS ADDITION (SI OR STEAM BREAK WITH SI) - ALL OF WHICH INFREQUENT

- O BLOCKED PORV DOES NOT CAUSE SRV OPENING BECAUSE: (1) SET POINT 150 PSI HIGHER, (2) HIGH PRESSURE TRIP AT 2400 PSIA

- O EVENTS OPENING SRV ARE COMPOUNDED TRANSIENTS SUCH AS LOAD REJECTION WITHOUT STEAM DUMP OR DIRECT REACTOR TRIP UNDER ADVERSE CONDITIONS, OR POSTULATED ACCIDENTS SUCH AS LOSS OF MAIN AND AUXILIARY FW, OR ATWS, OR RCCA EJECTION ..WHICH ARE MUCH LESS FREQUENT

DIVERSE MECHANISMS FOR PRESSURE CONTROL

MECHANISM	SET PRESSURE	DESIGN TRANSIENT
HEATERS	2250 PSIA	NORMAL OPERATION
SPRAY	2275-2325	10% SET LOAD CHANGE
PORV	2350	50% STEP LOAD REDUCTION
REACTOR TRIP	2400	TOTAL LOAD REJECTION
SAFETY VALVES	2500	LOAD REJECTION W/O IMMEDIATE TRIP

- O EACH MECHANISM DESIGNED TO PREVENT OPERATION OF NEXT MECHANISM FOR ITS DESIGN RANGE OF TRANSIENTS
- O THE MECHANISMS ARE DESIGNED WITH REDUNDANCY AND HAVE SUBSTANTIAL OVERCAPACITY (IN STARTUP TEST SPRAY WAS ADEQUATE TO PREVENT PORV OPENING FOR 25% LOAD REDUCTION, ONE PORV ADEQUATE TO PREVENT HIGH PRESSURE TRIP, ETC.)

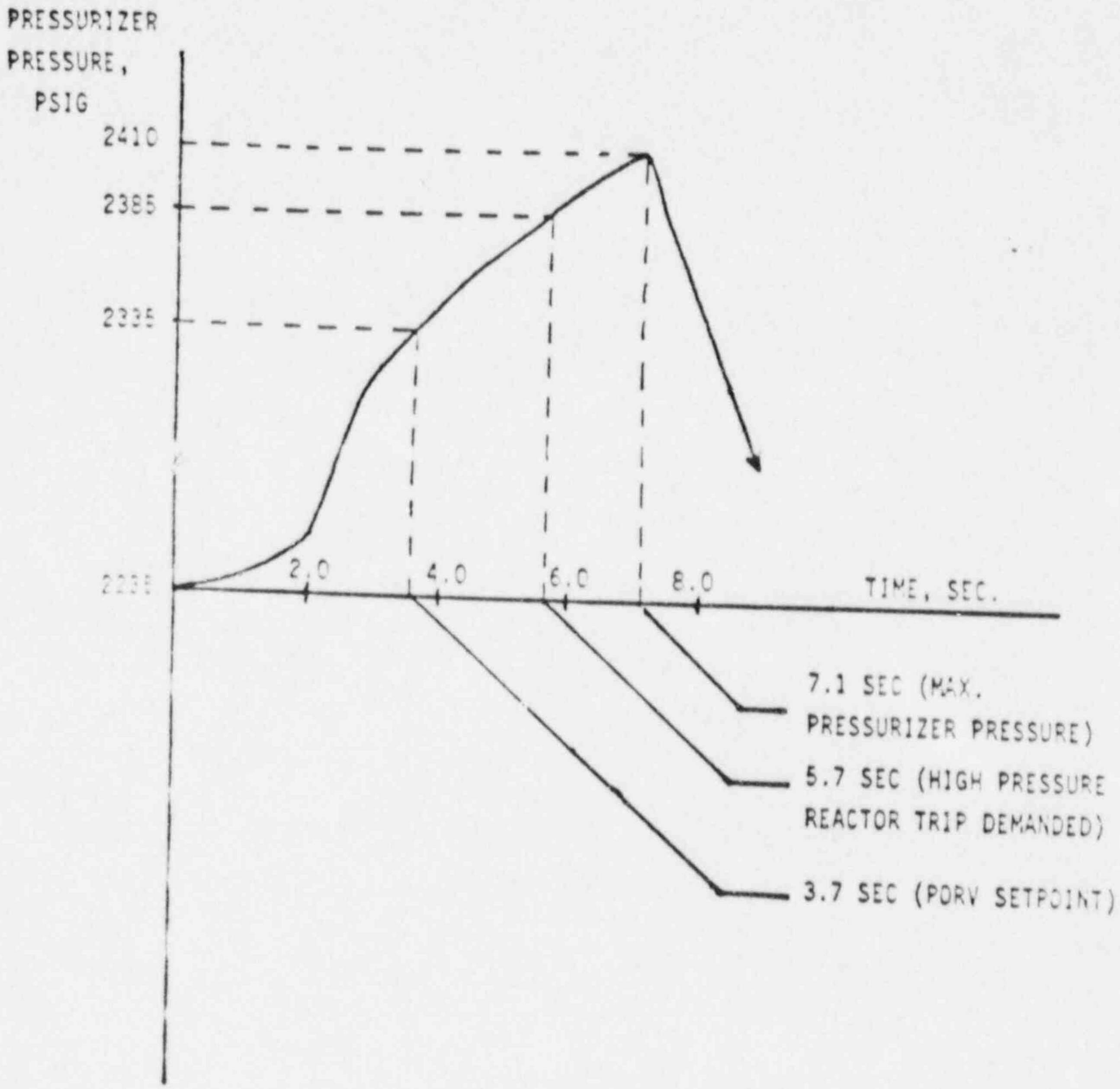


Figure 1

PRESSURIZER PRESSURE RESPONSE FOR NET LOAD TRIP (GRID DISCONNECT) FROM FULL POWER WITH NO PRESSURIZER PORVS OPERATIONAL

PT. BEACH - SAFETY INJECTION PUMP

PUMP TYPE: BYRON JACKSON MODEL 4X6X9C, 8-STAGE, DVMX

PUMP BEARINGS: ANTI-FRICTION RING OILED RADIAL AND THRUST BEARINGS. COOLING VIA AIR FANS LOCATED ON EACH SHAFT END TO PROVIDE BEARING HOUSING COOLING.

MECHANICAL SEALS: STANDARD END FACE RUBBING SEALS
ONE SEAL ON EACH SHAFT END
MECHANICAL SEAL COOLERS UTILIZE CCW

SEAL TYPE: JOHN CRANE SEAL TYPE 1 3 INCH SEAL

AFFECT OF LOSS OF CCW ON SAFETY INJECTION PUMP

- COULD IMPACT ONLY THE MECHANICAL SEALS SINCE ONLY SEAL COOLERS UTILIZE CCW

- SEAL COOLERS BASICALLY EMPLOYED ON PUMP APPLICATIONS WHERE FLUID TEMP EXCEEDS 160F FOR A SUSTAINED PERIOD OF TIME

- JOHN CRANE COMPANY TESTS - CRANE REPORT "SEAL PERFORMANCE TESTING FOR NUCLEAR PLANT SAFETY INJECTION SYSTEMS", BULLETIN NO. 3472

  O JOHN CRANE TYPE 1 SEALS TESTED FROM 0-400 PSIG AND 140-300F WITH NO SEAL COOLING BEING EMPLOYED

  O FOR 160F THE PROJECTED SEAL LIFE WOULD BE GREATER THAN 3 YEARS CONTINUOUS OPERATION (W/O EXTERNAL SEAL COOLING)

- FOR INJECTION CONDITIONS OF SHORT PERIOD OF INJECTION FROM BORIC ACID STORAGE TANK (~170F) FOLLOWED BY INJECTION FROM RWST, OPERATION OF SI PUMPS FOR 24 HOURS WILL NOT RESULT IN MORE THAN EXPECTED NORMAL WEAR

- CONSEQUENCES OF SEAL FAILURE - LEAKAGE, BUT NOT CATASTROPHIC FAILURE OF PUMP (FLOOR DRAINS DESIGNED FOR EXPECTED LEAK RATES; BACKUP PACKING RING ALSO AVAILABLE)

FIRE CASE ANALYSIS

COMPARISON OF NRC CASE STUDY AND EPRI/WOG

INITIATING EVENT FREQUENCY DISCUSSION

BASIS FOR HALON SYSTEM RELIABILITY

COMPARISON OF NRC CASE STUDY AND EPRI/WOG

Plant Design Differences

New Batteries

Redundant Halon System in AFW Pump Room

New Source of Halon System Reliability Data

Differing Application of Fire PRAs for Initiating Event Frequency

Human Reliability Analysis Of AFW Operation

INITIATING EVENT FREQUENCY

Divide Frequency Into Component Parts

Development Similar to Limerick Analysis

Similar Totals For Each Room

DIVIDE FREQUENCY INTO COMPONENT PARTS

Frequency for a Room

Location Within the Room

Intensity

DEVELOPMENT SIMILAR TO LIMERICK ANALYSIS

Cable Tray Fires

Electrical Panel Fires

Large Transient Combustible Fires

SIMILAR TOTALS FOR EACH ROOM

30% of NRC Case Study for AFW Pump Room

70% of NRC Case Study for Switchgear Room

HALON SYSTEM RELIABILITY

Source - Summary of Fire Protection Programs of USDOE, Calendar Year 1986

Uses Actual Fire Experience

Recent Fire Experience Versus EPRI/WOG Assessment

Comparison to Millstone 3 PRA Data Source

NSAC-113

SW PUMP HOUSE FLOODING EVALUATION

- THOMAS PIPE RUPTURE CORRELATION IS BASED ON LEAK BEFORE BREAK CONCEPT

$$P_C = \frac{P_C}{P_L} \cdot P_L = \frac{P_C}{P_L} \cdot P \cdot Q_E$$

$$= \frac{P_C}{P_L} \cdot P \cdot [Q_P + AQ_W] \cdot B \cdot F \cdot S \cdot \Sigma$$

WHERE

P_C = PROBABILITY OF RUPTURE

P_C/P_L = FRACTION OF LEAKS THAT RESULT IN RUPTURES

P = PROBABILITY OF LEAKAGE PER Q_E-YR

Q_E = SIZE AND SHAPE FACTOR FOR PARENT MATERIAL AND WELDS

Q_P = SIZE AND SHAPE FACTOR FOR PARENT MATERIAL

Q_W = SIZE AND SHAPE FACTOR FOR WELD MATERIAL

A = WELD PENALTY FACTOR (=50)

B = SYSTEM DESIGN AGE FACTOR

F = PLANT AGE FACTOR

S = MODIFIER FOR QUALITY OF MATERIALS

Σ = MODIFIER FOR SPECIFICS OF STRESS AND FATIGUE

RANGES AND RECOMMENDED VALUES GIVEN BY THOMAS:

$0.05 < P_c/P_L < 0.10$, $P_c/P_L \sim 0.06$

$1E{-}9 < P < 1E{-}7$/QE-YR

$P \sim 1E{-}8$/QE-YR

$A=50$, $\Sigma \sim 1$

$B \sim 1$ BEYOND 10 YEARS

$F \sim 1.7$ EXTRAPOLATED TO 40 YEARS

SIZE AND SHAPE FACTORS

$$Q_P = \frac{L_P D_P}{T_P^2}$$

$$Q_W = \frac{N\ L_W D_W}{T_W^2}$$

$L_W = 1.75\ T_P$ (DEFINED BY THOMAS)

$D_W \sim D_P$

$T_W \sim T_P$

HENCE $Q_W = \frac{1.75\ N\ D_W}{T_W}$

WHERE

L_P = PIPE LENGTH

L_W = WELD LENGTH

D_P = DIAMETER OF PIPE

D_W = DIAMETER OF WELD

T_P = THICKNESS OF PIPE

T_W = THICKNESS OF WELD

N = NUMBER OF WELDS

FORMULATION IN NSAC-113

$$P_C = P\ \left(\frac{P_C}{P_f}\right)\ [Q_P + A*S*Q_W]\ B*F$$

NOTE THAT Q_P SHOULD BE MULTIPLIED BY S BUT FOR S~1, AS IS THE CASE FOR "AVERAGE" COMMERCIAL GRADE PIPE, THERE IS NO LOSS OF GENERALITY AS SHOWN.

POINT BEACH 10-IN. DIAMETER, CARBON STEEL FIRE MAIN AT 125 PSIG. A BREAK IN THE 3-FT. CENTER OF THE PIPE SPAN ABOUT THE T-JUNCTION WITH THOMAS NORMAL VALUES AND F~2

$$Pc = 1E\text{-}8\ (0.06)\ [1440 + 50\ (1)(70)]\ (2)(1)$$

$$Pc = 5.93E\text{-}6$$

IF LEAK IS SUFFICIENT TO DAMAGE SW PUMPS, LET $Pc/PL \sim 1.0$ THEN $Pc = 9.88E\text{-}5$

WEPCo Cost Estimate for Intake Structure Shield Wall Extension (Spray) Modification

Architect/Engineer Cost:

Design - 500 Man-Hours at $110/Hr	$ 55,000
Installation /Removal-630 Man-Hours at $110/Hr	69,000
	$124,000

Includes:
1. QA Design Calculations for Seismic Class I Structure
2. Independent Review of Calculations
3. Onsite Inspection after Initial Design
4. Floor Loading Analysis
5. Installation Procedures
6. Construction Drawings/Specifications
7. Work Planning Including Sequence
8. Revision of Design During Construction
9. Final As-Built Drawings

Material:

Structural Steel/Other Construction Materials	$ 32,000

(Note: Non-Standard Steel Required-20Ft Standard)

WEPCo Cost: 400 Man-Hours at $50/Hour	$ 20,000

Writing Specifications, Purchase Orders, Engineering and Supervisory Review

CONTRACTOR COST:

REMOVAL - 1000 MAN-HOURS AT $22.50/HOUR	$ 22,500
INSTALLATION - 4000 MAN-HOURS AT $22.50/HOUR	90,000
	$112,500

INCLUDES:
1. 2 DAYS OF PLANT ACCESS TRAINING FOR 8 PEOPLE
2. 1 DAY OF WELDER QUALIFICATION FOR 2 PEOPLE
3. AWS D1.1 WELDING COMPLIANCE (REQUIRED BY WISCONSIN)
4. INTERFERENCE BY ERECTING NEW WALL PRIOR TO REMOVING EXISTING WALL (OTHERWISE 2-UNIT SHUTDOWN REQUIRED)
5. REWORK FOR FIELD INTERFERENCE WITH DEAD TIME FOR ENGINEERING DESIGN AND REVIEW
6. SETUP AND TEARDOWN TIME

TOTAL COST ESTIMATE $288,000*

* A 2-UNIT SHUTDOWN MAY BE REQUIRED FOR THIS MODIFICATION AT ABOUT $450,000/DAY FOR REPLACEMENT POWER

WEPCo Cost Estimate for Installation of Diesel-Driven Auxiliary Feedwater Pump (Internal 9)

WEPCo Estimate: $ 18,000,000

Includes:
1. Two Diesel Driven AFW Pumps (One/Unit)
2. Associated Instrumentation and Controls, Starting System, Cooling System, Fuel Oil System
3. Tie-In to Existing AFW Suction and Discharge Lines with Seismic Class I Piping
4. Construction of Seismic Class I and Tornado-Missile Resistant Design Building

Basis:
1. A Two-Loop W Plant Installed Two Additional Motor-Driven AFW Pumps in Existing Seismic Building in 1979. Cost was $16,000,000
2. WEPCo Recently Estimated Cost of Installation of 3rd Emergency Diesel Generator in a Seismic Class I, Tornado-Missile Resistant Building for ~$7,000,000
3. Another Two-Loop W Plant is Installing Two New Emergency Diesel Generators in a New Building - Estimated Cost is about $20,000,000

RECOVERY ANALYSIS

Methodology

Basis for Quantification

Example - Manual Operation of TD AFW Pump

NRC CASE STUDY METHODOLOGY

DETERMINE TIME AVAILABLE

IDENTIFY RECOVERABLE FAILURES

QUANTIFY NON-RECOVERY EVENT

REQUANTIFY DOMINANT ACCIDENT SEQUENCES

METHODOLOGY

Identify Potential Recovery Actions

- Operator interviews

- Past PRAs

Consider Feasibility for Accident Scenarios

Quantification

Cut Set Review and Implementation

Verification by Walkdown

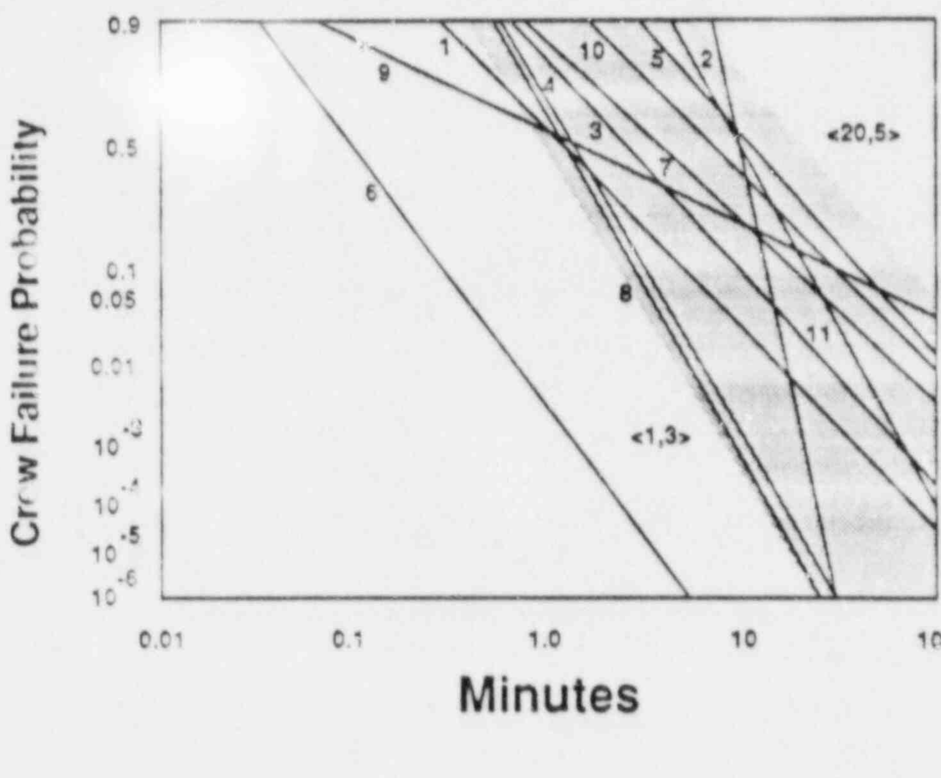
BASIS FOR QUANTIFICATION

Simulator Data

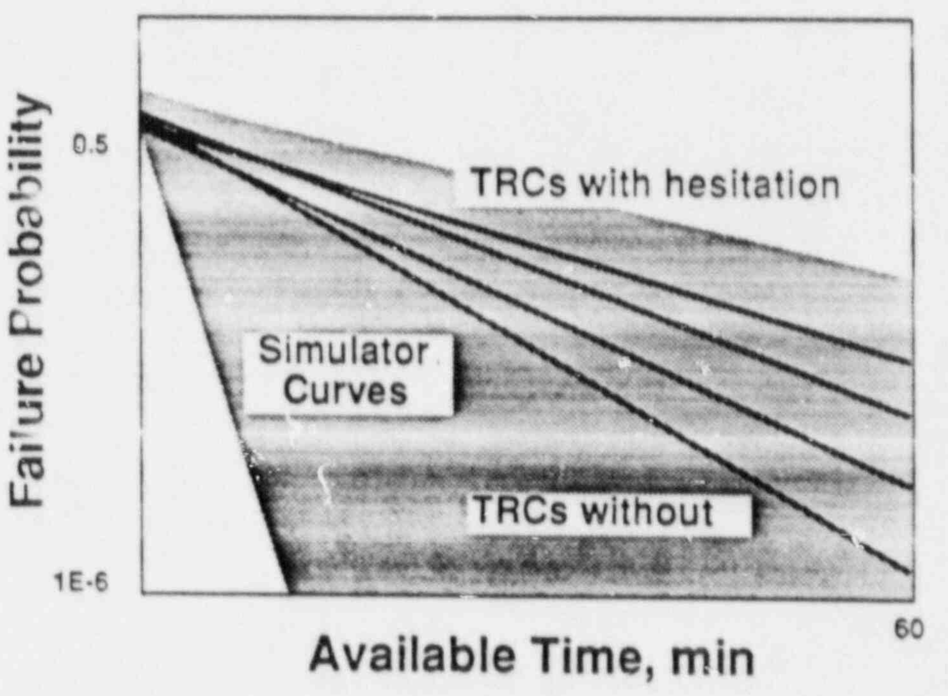
Comparison of EPRI/WOG and Other Studies

DATA FOR CREW RESPONSES

Taken from the LaSalle Nuclear Power Plant Simulator



TRCS IN COMPARISON WITH SIMULATOR DATA



NUMBER COMPARISON

Time	EPRI/WOG[1]				ASEP[2]	
	SLI=0.5		SLI=0.7			
min	no H[3]	H	no H	H	lower	upper
10					0.01	0.1
15	0.006	0.05	0.002	0.02	(0.003)	(0.03)
20					0.001	0.01
30	0.0003	0.01	0.00006	0.006	0.0001	0.001

[1] Page 4-9, NSAC/113.
[2] NUREG/CR-4772, A. D. Swain, *ASEP HRA Procedure*, Sandia National Laboratories, February 1987.
[3] H stands for hesitation, or some source decisional burden.

NOTES

1. ASEP includes another TRC; SAIC TRC system another two.
2. match between two systems is not qualitatively accurate.
3. the distinction between ASEP curves is not well-specified.
4. ASEP does not consider decision making significant.
5. ASEP does not, thus, consider hesitation or burden.

EXAMPLE - MANUAL OPERATION OF TD AFW PUMP

Two Actions

- Manual Start
- Alternate Water Supply

Manual Start

- Fire in AFW Pump Room
- Fire in 4160 Volt Switchgear Room

Alternate Water Supply

- Long-Term Station Blackout
- Seismically-Induced CST Failure

FEED AND BLEED ISSUES

Factors That

Enhance Reliability	Detract From Reliability
cue is simple	goal conflict
action set is simple	competing recovery
action in control room	uncertainty in phenomena
easily simulable	unanticipatable system status
	incredulity in losing all feedwater

Assessed Failure Probabilities

Sandia	0.003
SAIC TRC System	0.05-0.000006
NUREG/CR-1278 (20-60 min)	0.01-0.0001
Other PRAs	0.05-0.0001
Westinghouse Owners' Group	0.01

Note

Both hesitation and possible over use of feed and bleed have been noted in different PWRs.

ECCS RECIRCULATION ISSUES

Factors That

Enhance Reliability	Detract From Reliability
cue is simple	step order is significant
action set is simple	multiple mission (flow & cooling)
action in control room	never performed
prior success in injection	DHR can avoid recirc in some cases
procedure has contingencies	
easily simulable	

Assessed Failure Probabilities

Sandia	0.003/0.001
SAIC TRC System	0.05-0.000006
NUREG/CR-1278 (20-60 min)	0.01-0.0001
Other PRAs	0.003-0.0001

Note

No evidence of conflict or unusual burden associated with recirculation—should be little decisional stress

OTHER KEY RECOVERY ACTIONS

STARTING EDGS ON LOSS OF STATION BATTERIES (SEISMIC AND SBO)

DC CROSS CONNECT

ALTERNATE SOURCE TO RWST (SEISMIC AND RECIRCULATION RECOVERY)

COMPONENT COOLING CROSS CONNECT

COMMENTS FOR CONSIDERATION IN NRC'S A-45 REGULATORY ANALYSIS BASED ON NUMARC REVIEW LETTER/NSAC-113

1. DEDICATED SDHR FAILS <u>ALL</u> COST BENEFIT MEASURES BY A WIDE MARGIN

2. MANY OF THE REASONS FOR MUCH LOWER RISK AT PB GENERALLY APPLY TO OTHER CASE STUDY PLANTS.

3. WITH POSSIBLE EXCEPTION OF SEISMIC, ALL OTHER "EXTERNAL RISK" FACTORS AS ANALYZED BY A-45 CASE STUDIES SHOWN TO BE INSIGNIFICANT.

4. A-45 HAS NOT YET ACCOUNTED FOR SAFETY IMPROVEMENTS FROM OTHER NRC AND INDUSTRY PROGRAMS (IN PARTICULAR, SBO AND SEISMIC MARGINS).

5. <u>BEST-ESTIMATE ANALYSIS</u> IS ESSENTIAL FOR CREDIBLE USEFUL RESULTS. ANY ADDITIONAL MARGIN (IF NEEDED) SHOULD BE ADDED AT <u>END</u> OF ANALYSIS. IMPORTANT LESSON FOR IPE PROCESS.

6. U.S. NUCLEAR POWER PLANT OPERATING EXPERIENCE IS THE BEST SOURCE OF CREDIBLE DATA FOR BEST-ESTIMATE ANALYSIS, AND BEST FOUNDATION FOR "DEFINING THE PROBLEM." (E.G., A-44)

A-45 RESOLUTION VIA OTHER PROGRAMS

MAJOR SOURCES OF RISK IN SANDIA STUDIES	PROGRAMS ADDRESSING THESE RISK SOURCES
• STATION BLACKOUT	A-44, NUMARC
• CONTAINMENT RELIABILITY, FAILURE MODES, CONSEQUENCES	NUMARC, IDCOR, NUREG-1150
• PWR AFW RELIABILITY, BWR H.P. CORE COOLING	INPO, NSSS OGs
• SBLOCA	EPRI, INPO, NSSS OGs
• PCS LOSSES (SCRAMS)	INPO, NSSS OGs, NSAC, AEOD
• DC POWER RELIABILITY	NSAC, INPO
• VALVE PERFORMANCE	EPRI, INPO, AEOD
• FIRE PROTECTION	APP. R, INPO
• INTERNAL FLOODING, LIGHTNING	NSAC, INPO, A-17
• SEISMIC RISK	EPRI, SQUG, A-46, A-17
• SABOTAGE	UTILITY SECURITY PROGRAMS, FITNESS FOR DUTY RULE

3895NS7

ATTACHMENT C (#1)

"EPRI/WOG ANALYSIS OF DECAY HEAT REMOVAL RISK AT POINT BEACH", NSAC/113, March, 1988.

This report was prepared by Science Applications International Corporation and Westinghouse Electric Corporation.

The report is copyrighted and so cannot be processed into the NRC document control center without written permission from the copyright holder. This was not known to the author of this meeting summary in time to request and obtain the necessary permission. Therefore a copy cannot be included here. The author is sorry for the inconvenience.

However, the report can be ordered from:

Research Reports Center (RRC)
Box 50490
Palo Alto, CA 94303

Phone (415) 965-4081

Attachment C (#3)

EPRI

Electric Power
Research Institute

March 30, 1988

Ms. Elaine Gorham-Bergeron
Division 6413
Sandia National Laboratories
P.O. Box 5800
Albuquerque, NM 87185

Reference: Letter from E. Gorham-Bergeron, SNLA to G. Vine, NUMARC dated March 8, 1988

Dear Ms. Bergeron:

Your recent request for additional information regarding certain details of the EPRI/WOG analysis of DHR at Point Beach was provided to me by Gary Vine. The enclosed material is provided in response to that request. Although the time available to us did not permit the preparation of a full response to all of your questions, each of the topics identified in your letter will be addressed at the NUMARC-NRC meeting on March 31, 1988. Our presentation material at that meeting will constitute our responses to the several open items identified in the enclosure.

With regard to the application of SAIC's Risk Management Query System (RMQS) in the EPRI/WOG study, we previously extended an invitation through Ken Adams (SNLA) that you, Ken and Dave Ericson (ERCI) visit Bill Parkinson at SAIC's Los Altos, CA office. Although a meeting could not be scheduled prior to the March 31, 1988 review meeting, due to constraints on Dave Ericson's availability, we would be pleased to reschedule a meeting at your convenience.

Sincerely yours,

John J. Haugh

John J. Haugh
Project Manager
Nuclear Safety Analysis Center
Nuclear Power Division

JJH/jph 3970NS8

Enclosure: "Responses to SNL Questions on EPRI/WOG Analysis of DHR at Point Beach"

3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303 Telephone (415) 855-2000
Washington Office: 1019 Nineteenth Street, NW, Suite 1000, Washington, DC 20036 (202) 872-9222

cc: R. Woods, NRC/RES
K. Adams, SNLA/6413
D. Ericson, Jr., ERCI
G. Neils, NSP (Chairman, NUMARC Working Group on DHR)
R. Newton, WEP (Chairman, Westinghouse Owners Group)
A. Ladieu, YAEC (Chairman, WOG Analysis Subcommittee)
W. Andrews, Southern Co. Services (Vice Chairman, WOG Analysis Subcommittee)
W. Parkinson, SAIC
W. Layman, EPRI
T. Marston, EPRI
C. Stepp, EPRI
G. Vine, NUMARC

ENCLOSURE

Responses to SNL Questions on EPRI/WOG Analysis of DHR at Point Beach

March 30, 1988

I. Plant Systems

I.1 The contention the HPI does not require CCW for cooling of bearings. Is there engineering data to support this contention? How long can HPI operate without cooling.

Response:

1. Pump Description

   a. SI Pump Type:

      Byron Jackson Model 4x6x9C, 8-stage, DVMX
      B-J Drawing No. 2E-2002, Rev. B

   b. Pump bearings: anti-friction ring oiled radial and thrust bearings. Cooling is via air fans located on each shaft end to provide bearing housing cooling.

   c. Mechanical seals: standard end face rubbing seals. One seal on each shaft end. Mechanical seal coolers are employed which utilize component cooling water.

      John Crane Seal Type 1-3 in. seal
      Crane Drawing No. F-SP-13257

2. Loss of CCW Event Affect

   The loss of CCW event is assumed to be applicable to the injection phase of the accident and not the recirculation phase such that the pumped water from the RWST typically is about 100°F. Note that the SI pumps initially take suction from the boric acid storage tanks which are maintained at approximately 170°F. This water volume is small (<5000 gals) so that the time the SI pumps are subjected to this temperature is short. The loss of CCW event will only impact on the pump mechanical seals since only the seal coolers utilize CCW. However, seal coolers are basically employed on pump applications where the fluid temperature exceeds 160°F for a sustained period of time. When temperatures are below 160°F, seal cooling has little effect on increasing seal life.

John Crane Seal Co. performed a series of tests in the late 1960's to test the performance and estimate the life of mechanical seals used in nuclear applications. These tests were documented in a Crane Report, entitled, "Seal Performance Testing for Nuclear Power Plant Safety Injection Systems," Bulletin No. 3472. John Crane Type 1 seal tests were performed on seals at pressures from 0-400 psig, and temperatures of 140-300°F, with no external seal cooling being employed. Seal life and performance is a function of temperature and pressure. The test results show that for a defined "normal condition" of 160 degrees F and 400 psig the projected seal life would be greater than 3 years of continuous operation (without external seal cooling).

Operation of the SI pumps without seal cooling for 24 hours will not result in more than expected normal wear for this seal under the maximum temperature and pressure conditions stated. Lastly, although no cooling water is circulating through the seal coolers, the pumped water circulating through these coolers still obtains some cooling by virtue of the fact that the pumped water in the seal chamber is now circulating through an air to water heat exchanger.

I.2 Capabilities of new batteries and details of installation. Where are these new batteries located? What systems are served? Are these batteries "on-line" like the station batteries or are they isolated?

Response: The information requested will be presented at the March 31, 1988 meeting.

I.3 Venting capability option as alternative to one PORV. How are these vents actuated?

Response: Venting capability was not credited in NSAC-113. Although the DC vent valves appear sufficiently large to compensate for a failed PORV or block valve, the lines are orificed downstream. The small size of the orifice precluded the crediting of the vent valves as an alternative for feed and bleed without more detailed thermal hydraulics analysis. The vent valves are actuated by DC power. The configuration of the vent paths is such that either train of DC power can be used.

It should be noted that significant conservatism exists in the thermal hydraulics analysis which indicates that both PORVs are required for feed and bleed. It is believed that the additional DC vent capacity could be beneficial when combined with these conservatisms. The most significant impact would be to eliminate feed and bleed failures caused by loss of DC bus initiating events and those events where a single PORV or block valve fails to open.

II. Failure Data

II.1 Halon system reliability. Is there a more complete reference than the personnel communication cited? Where were the data collected? What is the justification for applying it to commercial nuclear reactors?

Response: The reference for Halon system reliability is Summary of Fire Protection Programs of USDOE Calendar year 1986, pp. 29-32. This report is updated annually.

The reference indicated that for 17 fires in which automatic Halon suppression systems were involved, all fires were extinguished. The EPRI/WOG study conservatively assumed one failure thereby yielding a failure probability of 1/17.

In a follow-up phone call with Mr. Walter Maybee of DOE (301-353-5609), we have learned that 2 failures have occurred recently. Complete documentation is not yet available to DOE, but will be included in the 1987 update of the above reference. (Publication will be some months since annual fire reports from individual DOE facilities are not due until April 15.) Consequently, information on recent successes is not documented and no new failure probability could be generated.

Of the two reported failures, one failure occurred at Richland and one at Brookhaven. The Richland failure did include failure of the system to automatically initiate. The Brookhaven failure included failure of the system to manually initiate. The automatic system was determined to be operable. (The fire did not reach sufficient intensity to initiate the automatic system.)

Because of the nature of the Brookhaven failure and the lack of credit given in the EPRI/WOG study to manual initiation of Halon, the Halon system reliability estimate presented in NSAC-113, i.e., 1/17, remains reasonable. Including the second fault in the data base, would double Halon unreliability and increase fire risk by a factor of 4 in the EPRI/WOG estimate (since redundant systems are credited).

Through discussions with Mr. Maybee, we were able also to explain the apparent difference between these Halon reliability estimates and those quoted in the Millstone 3 PRA. The American Nuclear Insurers, the source for the Halon reliability estimates in the Millstone 3 PRA, generally quote Halon reliability estimates using acceptance test data rather than actual experience in extinguishing fires.

Acceptance test data is not an adequate basis for predicting Halon system reliability in the event of a fire at a nuclear facility for two reasons. First, an acceptance test is part of the design checkout phase of system design and implementation. If the Halon system fails to meet its acceptance test, the system is modified and retested until the required concentrations are delivered and maintained for the required time interval. Usually an acceptance test failure is a small variation from the criteria and only minor modifications are required.

Second, the acceptance test criteria are conservative. Mr. Maybee noted that whereas most fires are extinguished (according to research data) by a 3% concentration, the acceptance tests generally require a 5% concentration to be held for 10 minutes. Further, the experience quoted in the above reference indicates that Halon systems are more capable than their design bases suggest. In one case a Halon system put out a so-called "deep seated" fire, e.g., a fire starting at the bottom of a trash container. According to Mr. Maybee the research data suggests that Halon would not have put out such a fire.

In conclusion, the Halon system reliability estimate quoted in NSAC-113 is reasonable for automatic system operation. Furthermore, that estimate appears to be more appropriate than the estimate provided in the Millstone 3 PRA.

II.2 Diesel generator failure rates. What run times are assumed/used in the data base?

Response: The diesel generator failure probabilities used in the EPRI/WOG study were taken directly from industry-wide experience collected for NSAC-108. These failure probabilities are determined on a mission basis where the mission of the diesel is to respond to an unplanned event or to meet the requirements of a monthly or annual test. No specific failure rate, i.e., failures per hour, is calculated in NSAC-108; consequently, no specific run time is used in the EPRI/WOG analysis.

Before applying the NSAC-108 data, a judgment was made as to its applicability to a PRA analysis. It was judged that the EPRI/WOG mission was similar enough to the mission defined in NSAC-108 to justify its application in NSAC-113.

NSAC-108 describes certain criteria for the "load run" portion of the test and for unplanned event data. If in a test or unplanned event, the diesel generator ran for significantly less than one hour, the event was not counted as a "load run" success. In the typical monthly test, the largest portion of the data base, the diesel is run for at least one hour. In the annual test, about 1% of the data base, the diesel is run typically for 24 hours. It is likely that the average mission of nuclear plant diesel generators is greater than one hour. Point Beach specific information submitted to EPRI for NSAC-108 indicates an average run time of 3.3 hours per start based on 1158.7 hours accumulated in 347 starts.

The NRC Case Study assumed a mission time for the diesels of 8 hours. This assumption is in conflict with the assumption inherent in the EPRI/WOG data base. However, as mentioned in NSAC-113, a Level 1 PRA would generally consider a time dependent analysis for diesel generator run fault sequences. A time dependent analysis would indicate the average necessary mission time for the diesels.

Such an analysis in the Florida Power Corporation Crystal River Unit 3 PRA found that after about 2 hours, continued successful diesel operation was risk insignificant. This finding occurs because after about two hours the chance of recovering offsite power increases substantially and the importance of continued diesel operation decreases proportionately.

The NRC Case Study did not give credit for offsite power recovery after 30 minutes, even for the so-called "long term station blackout" sequences. This conservatism, together with the 8 hour run time, significantly influences the NRC Case Study results. This level of conservatism is not consistent with the state of the art in Level 1 PRAs. The EPRI/WOG analysis provides a more realistic basis for diesel generator mission time. Based on the Point Beach plant specific mission times, the NSAC-108 data base appears to be consistent with that basis.

II.3 PORV block valve positioning data. What is the source of this data? How as it substantiated? Are records kept?

Response: The NRC Case Study assumed the following probabilities for block valve position:

both closed	1.0
one closed, one open	0.0
both open	0.0

No basis was provided for this assessment.

In contrast, the EPRI/WOG study used the following estimates of block valve position:

both closed	0.01
one closed, one open	0.50
both open	0.49

This assignment was based on initial estimates offered by WEP. As discussed below, the safety impact of these numbers was judged to be relatively insignificant in the EPRI/WOG model.

In an attempt to verify the accuracy of those initial estimates, WEP recently conducted an in-depth survey of Point Beach Unit 1 PORV block valve positioning data for the entire year 1986. In addition, a more limited review covering only the last quarter of 1987 was conducted for both Unit 1 and Unit 2. This involved tallying the changes in PORV block valve positions as recorded in the shift log books. (The positions of the block valves are recorded once per shift.) The percentage of time that the block valves were in each position during the periods investigated was as follows:

September - December 1987, Unit 1:

both closed	0.0
one closed, one open	0.0

both open	1.0

September - December 1987, Unit 2:

both closed	0.18
one closed, one open	0.02
both open	0.80

Full year 1986, Unit 1:

both closed	0.29
one closed, one open	0.45
both open	0.26

Total for above periods, both units:

both closed	0.22
one closed, one open	0.30
both open	0.48

The Unit 1 valve position data for the last quarter of 1987 clearly is in conformance with the initial estimate for both block valves being closed. The Unit 2 data for the same time frame and the Unit 1 data for 1986, however, suggest a much higher incidence of both valves being closed than was previously estimated. Taken in the aggregate, these data would suggest a frequency of 22% for both units during the time periods investigated. Although this value is considerably greater than the 1% estimate used in NSAC-113, it remains substantially less than the 100% estimate used in the NRC Case Study.

Given that the plant data suggests that the frequency of both block valves being closed may be higher than previously thought, it is appropriate to re-examine the relative risk significance of this parameter. As applied to Point Beach, the block valve positioning data can enter the DHR risk evaluation in three instances. These include situations where:

- a closed block valve prevents its PORV from opening and sticking open,
- both block valves being closed leaves the SRVs as the only pressure relief devices, and
- any closed block valves must be opened for feed and bleed.

Each of these situations is examined in turn.

If either PORV block valve is closed, that PORV cannot open spuriously to cause a LOCA, nor can it stick open after a loss of offsite power transient, or a loss of main feedwater transient. The EPRI/WOG study did not credit this beneficial effect of either or both PORV block valves being closed.

If both PORV block valves are closed, it appears that the NRC Case Study considered that the SRVs might be susceptible to opening during a primary system pressure transient. Based on experience with previous PRAs and thermal hydraulics analyses, the EPRI/WOG study assumed that the SRVs would not open on loss of offsite power or loss of main feedwater. Hence, the change in block valve positioning data has no impact on the transient induced LOCA frequency calculated in NSAC-113.

If any block valve is closed at event initiation and feed and bleed is required, the block valve or valves must be opened for feed and bleed to succeed. The EPRI/WOG analysis included the effect of PORV block valve position in the failure probability for the valve. The effect of the new PORV positioning data is being evaluated and will be presented in the meeting.

III. Analytical Methodology

III.1 SAIC's Risk Management Query System (RMQS). What are the constraints and limitations of this system? Is this software generally available? What documentation is available?

Response: RMQS contains a series of linked data bases. The data bases include:

- initiating events
- accident sequence cut sets
- system or super component cut sets
- components
- component types
- risk measures

Each data base contains a numerical estimate and descriptors. RMQS allows the user to change any entry and determine its propagating effect. For example, the reliability of a diesel generator can be changed and the impact on core melt frequency can be determined.

The constraints and limitations of RMQS are predominantly affected by its application by the analyst. For example, the user may also add events to the data base. By adding events to the accident sequence cut set, recovery can be credited. In this activity, the user must be careful to avoid adding a recovery which would be inapplicable due to other failures in the cut set. This limitation is true regardless of how recovery is applied, i.e., whether done for the NRC Case Study or for NSAC-113.

Another constraint is the ability to load the model correctly even before recovery is assessed. Often PRAs are not as traceable as even the authors would prefer. Sometimes inconsistencies are identified. As mentioned in NSAC-113, the Point Beach model in RMQS was benchmarked against the NRC Case Study results. These results differed by only a few percent from the total core melt frequency.

RMQS can be purchased from SAIC subject to license agreement. The code has been subject to a Quality Assurance program and has passed two audits/surveys by utilities. Like the code, documentation such as the RMQS Users Manual is proprietary.

III.2. PORV performance and recirculation for small LOCAs. What analysis/experiments support assumptions about PORVs performance? Is there documented analysis to support the claim that recirculation will not be required for small LOCA?

Response (a): Opening of pressurizer safety valves during power operation is a very rare event on Westinghouse PWRs. The reasons are (1) Westinghouse used a very conservative approach for primary pressure relief; and (2) transients requiring relief are much less frequent than commonly assumed. Based on engineering judgment and estimates of knowledgeable Westinghouse engineers, we estimate the rate of pressurizer safety valve opening (during power operation) to be roughly 0.01 safety valve openings/reactor year. This is in sharp contrast to the NRC Case Study assumption of approximately 0.5 safety valve openings per reactor year due to an assumed 7 pressure transients per reactor year with a 7% chance of opening the safety valves in each. This difference of a factor of 50 can be very important in arriving at an unisolatable small LOCA frequency (should a safety valve fail to reclose there are no back-up block valves as in the case of the PORVs) and is due to several causes as follows:

- The NRC Case Study assumes 0.5 transients per year (7 transients per year times 7% probability) reaching the PORV setpoint. Westinghouse survey and analysis results indicate about 0.23 pre TMI and 0.12 post TMI PORV openings per reactor pear from an operational transient.

- The NRC Case Study assumes Point Beach PORVs are both blocked 100% of the time. Point Beach plant records indicate both are blocked approximately 20 percent of the time on average.

- The NRC Case Study assumes that a transient reaching the PORV setpoint would reach the safety valve setpoint if the PORV block valves are closed. The vast majority of such transients would still not reach the safety valve setpoint because the safety valve setpoint is much higher (2500 psia compared to the PORV 2350 psi setpoint). Additionally a high pressure reactor trip would occur at 2400 psia to relieve the mismatch if there had not been a direct or earlier reactor trip. This is supported by the infrequency of safety valve openings in practice.

The Westinghouse conservative approach to pressure relief consists of diversity, redundancy, and overcapacity.

- Diversity: Four mechanisms are provided to control rapid increases in pressure: Spray; PORVs, Reactor Trip; and Safety Valves. (Normal pressure control is provided by regulating pressurizer heaters. Only fairly rapid changes in pressure require spray.) Each of these four mechanisms is designed to prevent operation of the next mechanism for its design range of transients. The following set points illustrate this point:

Mechanism	Set Pressure	Design Transient
Heaters	2250 psia	Normal operation
Spray	2275-2325	10% step load change
PORV	2350	50% step load reduction
Reactor Trip	2400	Total load rejection
Safety Valves	2500	Load rejection without immediate trip

- Redundancy: Two (or three) spray valves, PORVs, reactor trip circuits, and safety valves are provided.

- Overcapacity: Plant startup tests at Mihama unit 1 (1971) demonstrated the very conservative sizing basis. One test performed a 25% step load reduction (versus a 10% step used in the design basis): Spray was adequate to prevent PORV opening. Another test performed a 100% step load reduction (versus 50% in the design basis): One PORV (of the two installed) proved more than adequate to prevent high pressure reactor trip (it cycled open and shut several times, requiring about half of its full relieving capacity.)

In addition, large load rejections or other transients that would require pressure relief occur infrequently. Normal load changes are 1% per minute or less. Large load changes requiring pressure relief occur only as a test or a fault (on the power grid).

Following the TMI event, Westinghouse made a survey of operating plants. The results showed, that as a result of these reasons, PORV operation due to at-power transients occurred at a frequency of 0.12 per reactor-year. (Another approximate 0.1 PORV openings/R-Y was the result of testing or other I&C error or because spray had been blocked, and a third approximate 0.1 PORV opening/R-Y occurred at cold shutdown. (See Reference 1.)

Even if PORVs are not available (blocked out), safety valves would seldom open. The reactor trip (set 100 psi below the safety valves) typically begins reducing core power within one second of pressure reaching the set point. Thus, only the most extreme transient can reach the safety valve set pressure. Indeed, one cost-saving idea was once submitted to eliminate the safety valves as unnecessary for overpressure protection. (The idea was supported by analyses showing no overpressurization for any PWR Condition 2 design basis event. The idea was rejected since safety valves are desirable for hypothetical events such as control rod ejection and ATWT, and also Westinghouse's desire to preserve its conservative design basis.)

An informal survey was made of experienced Westinghouse engineers knowledgeable of PWR operating experience. They could recall only one instance of a transient opening a pressurizer safety valve. (A total turbine load rejection without steam dump or direct reactor trip under adverse conditions.)

Numerous PRAs, including those by Westinghouse, have assumed higher frequencies than the values estimated above. This is largely because safety valve openings are generally not part of the dominant contributors to risk. Therefore, over-estimation is conservative, but does not cause a significant increase in the total risk. Other PRAs have not considered it worth the effort to prove safety valve openings are less frequent than assumed from a simplistic model.

One such simplistic model, for instance, is NUREG/CR-1363 (Reference 2) Statistics derived from Licensee Event Reports (LERs). This reference tabulates both safety valve "demands" and "failures". However, essentially all the "demands" are safety valve test (to check the opening pressure), and the "failures" are failures to open within a prescribed tolerance of the intended set pressure (e.g., 2485 psig ± 25 psi).

Another example of such cost-effective conservatism is WCAP-9804 (Reference 3). This reference lists all design transients and assigns to each a frequency of occurrence and a probability of safety valve opening. (Both the frequency and the probability were conservatively over-estimated.) The sum of the probabilities for safety valve opening for all individual transients suggests a total frequency of safety valve opening of about 0.1/R-Y, even though the same reference reviews PWR operational experience and "concludes that no operational openings or failures of pressurizer safety valves have occurred domestically during approximately 181 reactor years of operation and specifically, 2,9493,324 hours of safety valve operation."

In conclusion, the estimated frequency in which operational transients cause PORVs and pressurizer safety valves to open are of the order of 0.1 and 0.01 per reactor year respectively. Most transients could not cause PORV opening, e.g., as transients with direct or anticipatory reactor and turbine trip. Transients which could open the PORVs are those which exhibit a strong mismatch between primary power generation and secondary heat removal (loss of steam load, MSIV closure), reduced cooling transients locked RCP rotor), reactivity addition transients causing core power increase (rod withdrawal) and RCS mass addition transients (SI or steam break with SI if the plant has a high head design SI system) all of which are relatively infrequent. Transients which could open safety valves are compounded transients such as load rejection without steam dump or direct reactor trip under adverse conditions, or loss of main and auxiliary feedwater, or postulated accidents such as RCCA ejection, which are much less frequent.

In NSAC-113 we have conservatively assumed that NRC Case Study transient categories T1 (loss of offsite power) and T2 (complete PCS interruptions) can lead to PORV opening but that the more frequent T3 (reactor trip/turbine trip transients) do not. However, we do not assume these transients cause safety valve opening because it is contrary to experience. Rather, safety valve opening has been considered for transients such as a total loss of main and auxiliary feedwater, or unspecified transients with frequency 0.01 per year.

REFERENCES

1. Letter, T. M. Anderson to NRC, NS-TMA-2078, May 1, 1979.
2. NUREG/CR-1363, "Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants," 1980; and Rev. 1, 1982.
3. WCAP-9804, "Probabilistic Analysis and Operational Data in Response to NUREG-0737, Item II.K.3.2 for Westinghouse NSSS Plants," D. C. Wood, 1981.

Response (b): The EPRI/WOG study concludes that small LOCAs requiring sump recirculation are unlikely. This conclusion is supported by the fact that no small LOCA requiring recirculation has occurred in 500 reactor years of Westinghouse PWR experience (greater than 500 reactor years when all PWR experience is considered).

Besides this experience which identifies that small LOCAs will be terminated prior to recirculation, thermal hydraulics analysis and plant procedures support this assessment. The WOG-developed Emergency Response Guidelines include guideline ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, providing actions to reduce the RCS temperature and pressure to 200°F and 400 psig for small LOCAs where SI can keep up with break flow at pressures above the shutoff head of the low-head SI pumps. The supporting BACKGROUND document includes a generic application of guideline ES-1.2 to a one inch cold leg break case and to a stuck open PORV case. In both cases RHR conditions are shown to be reached well before the RWST is drained for both low pressure and high pressure ECCS designs.

The background analyses were performed with the Westinghouse TREAT T/H network code. The ERGs and BACKGROUND document have been submitted to and reviewed by the NRC. Wisconsin Electric Power has developed a plant specific procedure consistent with this guideline.

Finally, it should be noted that the NRC Case Study credits similar response to a small LOCA followed by HPSI failure. The NRC Case Study credits the operators with depressurizing the primary system to below the shutoff head of the low-head SI pumps (i.e., below RHR entry conditions) so that LPSI can be used for injection. The EPRI/WOG analysis credits this as well, but also credits that the operators will perform the similar actions under the less stressful case where HPSI works and significantly more time is available.

III.3 Revision of seismic hazard curve. What is the rationale/analysis, other than that cited, for EPRI/WOG's revision of seismic hazard curve.

Response: The EPRI/WOG comment on the Point Beach seismic hazard curve accepts the method used to prescribe the curve as reasonable. It was suggested that the prescribed curve is conservative by a factor of two to five based on recent computations at the Braidwood site by LLNL and EPRI. The Braidwood site is in the same tectonic region with Zion and Point Beach and also is a reasonable basis for comparison. The comparison shows a factor of about 10 at low acceleration level (0.1g) increasing to more than a factor of 40 at very high acceleration level (1.0 g). A basic assumption made in prescribing the Point Beach seismic hazard curve is that its slope is the same as the Zion hazard curve. The Braidwood comparison shows divergence between the LLNL and EPRI results with increasing acceleration level. Moreover, the probability of exceeding the Braidwood SSE is about a factor of ten lower than the assumed 2.5E-4 per year assumed for Point Beach. Thus considering local site amplification, the prescribed Point Beach hazard curve should be considered conservative.

III.4 RWST failure mode. What data supports the assertion that RWST will not fail catastrophically? Was a fragili'y analysis performed?

Response: The supporting evidence for the assertion that the RWST will not fail catastrophically is based on the attached letter. That basis did not include a fragility analysis for the Point Beach RWST. While the letter does argue that experience indicates that tanks have stronger capacity than the theoretical calculations performed in the NRC Case Study, additional capacity was not credited in NSAC-113.

January 27, 1987

Mr. R. K. Hanneman
WISCONSIN ELECTRIC POWER COMPANY
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

Dear Harv:

At your request, we have reviewed briefly the performance of anchored larger storage tanks in recent earthquakes. We have, in particular, reviewed the performance of tanks that are in facilities that are included in the SQUG data base. Further, we reviewed other available data on tanks from earthquakes and sites not covered by our investigations for SQUG. The goal of this review was to (1) determine whether we have experience data with tanks similar to the Point Beach RWST and CST; and (2) to briefly review the performance of such tanks in past earthquakes and study their damage and failure modes. These data would then be used to evaulate the fragilities and failure modes of tanks such as the RWST and CST.

The RWST is 70' high and 26' in diameter. It is anchored with 27 bolts using welded saddles to the bottom tank rung. The bottom rung steel plate thickness is 0.33 in. The CST is 30' high and 25' in diameter. It is also anchored with bolts. The thickest plate is 0.33 in.

In general, large vertical tanks have performed poorly in strong earthquakes compared to other equipment such as pumps and motors. In particular, numerous gross failures are known; we have documented many such cases.

Typically, the failed tanks are large unanchored tanks. They were usually subjected to accelerations well above 0.25g; most often they were located in the most intensely shaken area of the earthquakes or were subjected to long duration shaking; i.e., in excess of 40 seconds. We doubt that gross failures of heavy industrial tanks can be found in areas with accelerations less than about 0.20g PGA in the free field.

Because the tanks of interest are anchored, we then narrowed our search to anchored tanks. We are currently unaware of a gross failure of an anchored industrial tank which has equal to or better anchorage than the Point Beach RWST and CST. Our data base contains more than 20 anchored tanks for which we have detailed data. The aspect ratios of the tanks envelope those of the Point Beach tanks. The free field PGAs exceed 0.20g. All of the tanks were subjected to ground shaking with PGAs greater than to several times greater than the static lateral coefficients for which they were designed. Further, we reviewed the reported damage to several anchored tanks in recent strong events (California and Japan). We found several instances of damage to anchorages or steel plates. Gross failure did not occur. None had tears in the steel plate that led to evacuation of contents. One poorly anchored tank in Chile had a slow leak that was attributed more to corrosion than the earthquake. This type of damage is not relevant for the R' ST since it is constructed of stainless steel. The seismic literature may contain evidence of gross failures of reasonably well anchored industrial grade tanks. Currently we are unaware of any such failures. I personally have not witnessed any in the more than 20 earthquakes that I have investigated.

Based on our brief review, as outlined in the above abbreviated summary, we believe that significant margins against rapid loss of contents of the RWST and CST exist. Specifically, even if damage occurs, such as plate bulging or anchorage pullout, the tanks will not lose their contents quickly. Higher margin would be provided by flexible inflow piping. At the Point Beach SSE level, leakage is highly unlikely; if some develops, it is likely to be very slow and the tanks should be available for at least 30 minutes at a withdrawal rate of, say 200 GPM.

The above arguments are qualitative and based on experience and many analyses. They can be quantified. Experience data exist and our analytical methods can be tested against such data.

Very truly yours,

K. Bell

for Peter I. Yanev
EQE Incorporated
San Francisco

PY/maw

xc: Mr. Bill Parkinson, SAIC

mk17/hanne.ltr

III.5 Thomas pipe break correlation. What is the basis for applying this correlation in this study? What is the rationale for changes/revisions to Thomas' original work? What is the basis of data used in numerical calculation?

Response: The Thomas pipe break correlation (1) was proposed in 1981. It represents an approximation strategy to estimate the probability of catastrophic pipe rupture (Pc) which is related to the probability of pipe failure resulting in leakage (Pl). The generalized approach was based on "analysis of actual service failure statistics (1)." The approach considers the pipe geometric factors and the number of weldments as the most important factors in determining the failure probability; it also makes allowance for aging effects.

Basis for Application in NSAC-113: The Thomas correlation has been utilized in developing pipe failure probabilities in the Oconee PRA (2). That application was reviewed by Brookhaven National Laboratories and was judged to be an acceptable methodology (3). The Thomas correlation, as utilized in the Oconee PRA, was applied in the EPRI/WOG analysis (NSAC-113). In this case it was used to estimate the failure probability for a low pressure (125 psig), carbon steel fire main. That 10-in. diameter pipe runs overhead the six service water (SW) and two fire pumps in the Point Beach SW pump room.

Unlike the Oconee PRA, however, NSAC-113 did not credit the apportioning the computed probability of occurrence according to break size. In addition, NSAC-113 differed from the Case Study analysis by not crediting a multiplier (≈0.1) to accommodate the ability of the SW pumps to withstand the spray impingement from the ruptured fire main.

Use of the Thomas Correlation in NSAC-113): NSAC-113 uses a formulation of the Thomas correlation that directly results from the derivation in the original work presented in Reference 1. The derivation of the Thomas correlation follows.

Thomas relates the probability of catastrophic pipe rupture (Pc) to the probability of pipe failure resulting in leakage (Pl) as

$$Pc \propto Pl = Pl \times \frac{Pc}{Pl}$$

Where Pc/Pl is estimated from actual pipe rupture statistics, i.e., the fraction of pipe leaks resulting in pipe rupture.

Thomas suggests that, in general, Pl is related to:

- Qe, the quantification of pipe geometry factors (e.g., size and shape) and the number of weldments
- F, an age factor
- B, a design experience-related, learning-curve factor
- S, a measure for quality differences
- Σ, a summation of factors for failure causes.

Thus

$$Pl \propto Qe \cdot F \cdot B \cdot S \cdot \Sigma$$

In this formulation, the term groupings can be considered as:

Qe · F represents a global estimate for Pl

B · S represents plant specific modifiers to the global estimate

Σ represents a modifier accounting for fatigue and other factors.

In estimating the individual terms, the following relationship is used:

$$Qe = Qp + A\ Qw$$

where Q_p is the size and shape factor for the parent material

Q_w is the size and shape factor for weld material

A is a penalty factor applied to weldments.

In general

$$Q = \frac{DL}{t^2}$$

where D is the pipe diameter

L is the pipe length

t is the pipe wall thickness

Thus $Q_p = \frac{D_p L_p}{t_p^2}$ for the pipe parent material

$Q_W = \frac{D_w L_w}{t_w^2}$ for a single weldment

and hence $Q_w = N \frac{D_w L_w}{t_w^2}$ for N weldments.

Thomas notes that because the length of the weld is defined arbitrarily as

$$L_W = 1.75\ t_w,$$

a penalty factor (A=50) should be applied to Q_w.

For full penetration weld, $t_w = t_p$.

Thomas also notes that leakage failure rates are typically in the range of 10^{-7} to 10^{-9}/Qe-yr, i.e.,

$$P \simeq 10^{-8}/\text{Qe-yr}$$

which would imply a constant failure rate with age.

To account for age related effects, the factor F is applied according to Figure 1 which is reproduced from Reference 1.

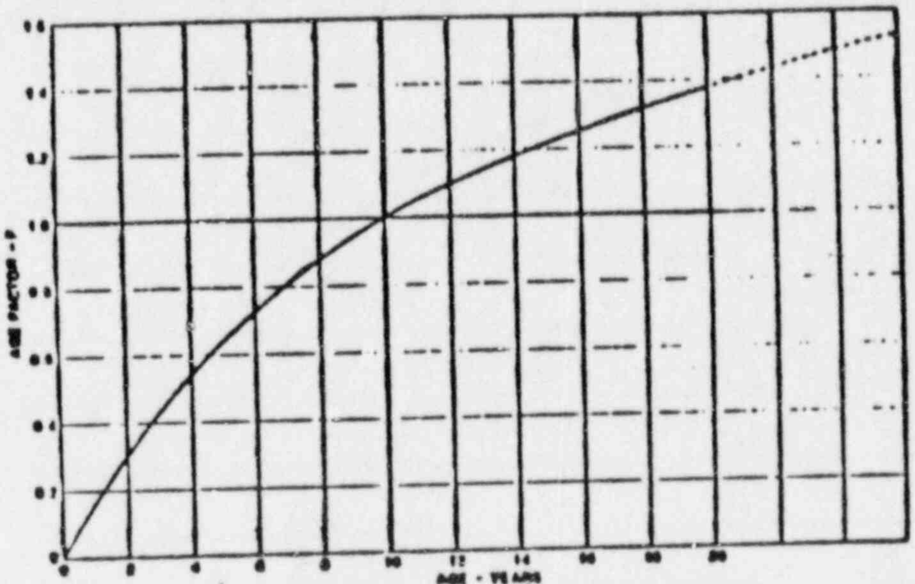


Fig 1 A simple arithmetic average of all the various age factors

Extrapolation of Figure 1 to 40 years, yields F ~2.

To account for design experience and other learning-curve effects, Thomas suggests that factors related to the technology as a whole may be neglected entirely, or they may be assumed to be incorporated within the global statistics for P1 (i.e., within the product QeF). To account for the age of the design of the specific plant, the factor B is applied according to Figure 2 which is reproduced from Reference 1.

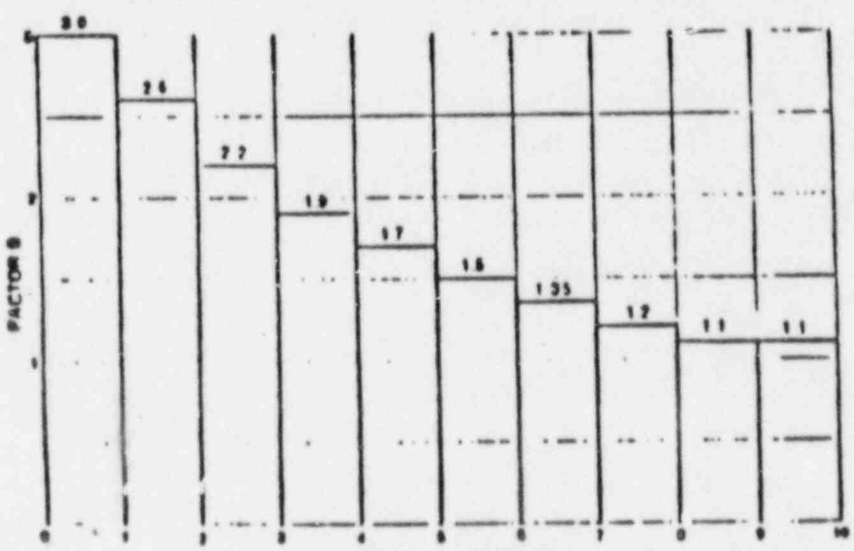


Fig 2 Hypothetical design learning curve

Note that the curve asymptotically approaches unity, where age is measured from the start of service. Because the curve is hypothetical in nature, Thomas notes that it must be used with caution. Nevertheless, it appears fair to say that older, established designs should be expected to have failure rates that are average as compared to newer untried designs. After 10 years service, $F \sim 1$.

Since the primary intent of the Thomas paper was the prediction of failure in nuclear systems, an allowance was made for the higher quality of nuclear versus commercial grade piping and components. The quality factor (designated herein as S) is intended to account for better design, manufacturing, operation, and in-service inspection practices for nuclear grade piping and components. Hence, for nonnuclear grade carbon steel piping, the logical inference is to assume a quality equal to that of average commercial grade installations, i.e., $S \sim 1$.

The last factor in the Thomas correlation (designated herein as Σ) is intended to account for plant specific details of stress, fatigue, environment, etc. Thomas notes, however, that "there is no need to factor for any detailed causes of failure when the component is being subjected to average conditions" (1). Hence, for the carbon steel fire main seeing ambient conditions over essentially its entire service life, it appears warranted to assume $\Sigma \sim 1$.

As an overall statistic, Thomas suggests that 5-10% of all leaks are ruptures, i.e.,

$$0.05 < \frac{Pc}{Pl} < 0.10$$

Based on detailed review of the four data sources cited in Reference 1, Thomas suggests a nominal value

$$\frac{Pc}{Pl} \simeq 0.06$$

but cautions that this value may be slightly optimistic especially for pressure vessels, as opposed to small pipes. He also suggests that this value should be augmented by more detailed fracture mechanics modeling. For the low pressure fire main, however, the nominal value appears reasonable for a first approximation.

Thus the full formulation of the Thomas correlation is

$$Pc = P\,[Qp + AQw]\;\; F \cdot B \cdot S \cdot \Sigma \cdot \frac{Pc}{Pl}$$

where $10^{-7} < Pl < 10^{-9}$

$$0.05 < \frac{Pc}{Pl} < 0.10$$

and

$$A = 50.$$

Note that the statement of the correlation in NSAC-113 is

$$Pc = P\,[Qp + A *S* Qw]\;(Pc/Pl)$$

where Σ is taken as unity. To be precise, the term Qp also should be multiplied by S. For S=1, however, there is no loss of generality in the formulation used.

Application of Data in NSAC-113: To calculate a best estimate value for Pc using the Thomas correlation, the following numerical values were used for a 3-ft length of carbon steel fire main:

$$P = 10^{-8}\;(10^{-7} < P < 10^{-9})$$

$$\frac{Pc}{Pl} = 0.06\;(0.05 < \frac{Pc}{A} < 0.10$$

$$A = 50$$

$$Dp = 10 \text{ in}$$

$tp = 0.5$ in

$Lp = 36$ in

$Dw = 10$ in

$tw = 0.5$ in

$N \simeq 2$

$F \simeq 2$ (=1.735 @ 40 yrs)

$B \simeq 1$

$S \simeq 1$

$\Sigma \simeq 1$

Thus

$$Pc = P \left(\frac{Pc}{Pi}\right) (Qp + AQw)\ F \cdot B \cdot S \cdot \Sigma$$

$$= 10^{-8}\ (0.06)\ (Qp + 50\ QW)\ (2)\ (1)\ (1)\ (1)/\text{year}$$

For a three foot pipe with two welds:

$$Qp = \frac{DpLp}{tp^2} = \frac{10\ (36)}{(0.5)^2} = 1440$$

and $$Qw = \frac{DwLw}{tw^2} = \frac{1.75\ N\ Dw\ tw}{tw^2}$$

$$= \frac{1.75\ (2)\ (10)}{0.5} = 70$$

Thus $Pc = 10^{-8} (0.06) (1440 + 50 (70)) (2) (1) (1)/yr$

$= 5.93 \times 10^{-6}/yr$

As an added measure of conservatism, it can be postulated that the leak is sufficient to damage the pumps in the SW pump room, such that $Pc/Pl = 1.0$. This results in $Pc = 9.88 \times 10^{-5}/yr$.

Since there has been no leak in this system during 16 years of plant operation, a best estimate approach should discount the calculated value.

Sensitivity Calculation

Let $P = 10^{-7}$ Thomas upper bound

$\frac{Pc}{Pl} = 0.1$ Thomas upper bound

Then $Pc = 10^{-7} (0.1) (1440 + 50 (70)) (2)$
$= 9.88 \times 10^{-5}/yr$ upper bound

Then applying the conservatism that $Pc/Pl = 1.0$ yields $Pc = 9.88 \times 10^{-4}/yr$ upper bound. This value also should be discounted for the fact that no leaks have occurred in 16 years.

Application in the Oconee PRA

The Oconee PRA used the following form of the Thomas correlation:

$$Pc = \overline{P} \left(\frac{Pc}{Pl}\right) (Qp + A\ Qw)\ BF$$

where the terms are defined as previously by and the same nominal values as those in NSAC-113 were used for $\overline{P}$, Pc/Pl, A, B and F.

It was also assumed that catastrophic ruptures could be distributed as follows:

P maximum, DE guillotine break	~ 0.1 Pc
P large rupture	~ 0.3 Pc
P medium rupture	~ 0.6 Pc

When the above methodology was applied in the turbine building flood analysis, the total mean annual frequency was calculated to be 2.9 X 10^{-2}/yr. An appraisal of historical data on turbine building floods presented elsewhere in the Oconee PRA indicated a historical frequency of 1.6 X 10^{-2}/yr. Thus, the use of the Thomas correlation along with discounting for break size appears to generate a reasonable estimate for that specific application.

Overall, the Brookhaven evaluation (3) of the Oconee (OPRA) analysis concluded that:

- The above approach yields higher pipe-break frequencies than could be obtained from the use of the mean rupture rates given by the RSS (Reactor Safety Study, WASH-1400) for pipes larger than 3 inches in diameter. The pipe rupture rates used by Thomas are based on an appraisal of the data in References 5 to 8 (of the paper).

- The reviewers do agree that, overall, the Thomas methodology as modified by OPRA to include the break-size-frequency distribution represents a realistic model. While the rupture rates derived by Thomas seem to be on the high side, they are used in the OPRA for the piping of the secondary system which can be anticipated to have rupture rates somewhat higher than those of the primary system.

Alternate Analysis: At the February 23, 1987 meeting with the NRC, it was suggested by D. Ericson, consultant to the NRC, that a 15-ft run of pipe with four welds would be a more appropriate basis for computing the failure probability.

Usi : Thomas' nominal values yields

$$Pc = 10^{-8} (0.06) [Qp + 50 Qw] (2) (1) (1) (1)$$

where $Qp = \dfrac{10 (15 \times 12)}{(0.5)^2}$

$Qw = \dfrac{4 (1.75)(10)}{0.05}$

and

$$Pc = 10^{-8} (0.06) [7200 + 50 (140)] (2)$$

$$Pc = 1.70 \times 10^{-5}/yr$$

If $Pc/Pl \approx 1.0$, then $Pc = 2.84 \times 10^{-4}/yr$

Using Thomas' upper bounds

$F = 10^{-7}$

$Pc/Pl = 0.1$

$$Pc = 10^{-7} (0.1) [14200] (2)$$

$$= 2.84 \times 10^{-4}/yr$$

If $Pc/Pl = 1.0$, $Pc = 2.84 \times 10^{-3}/yr$.

References

1. H. M. Thomas, "Pipe and Vessel Failure Probability," Reliability Engineering, 2 (1981), p. 83ff.

2. Oconee PRA, NSAC-60, June 1984, p. 9-183ff.

3. N. A. Hanan et al., "A Review of the Oconee-3 Probabilistic Risk Assessment," NUREG/CR-4374, BNL-NUREG-5197, Vol. 2, p. 2-6ff.

IV. Plant Operations (Recovery)

IV.1 Recovery options. What are the specific procedures and training which support the recovery estimates? How was the Success Likelihood Index (SLI) established? Have the Time Reliability Correlations been tested or verified in simulator exercises?

Response: The specific procedures and training will be discussed at the March 31, 1988 meeting.

SLIs were not specifically established for operator actions in NSAC-113. Probabilities for operator failure to feed and bleed were presented in terms of various SLI values as a sensitivity study.

The Time Reliability Correlations (TRCs) used in NSAC-113 and referenced from NUREG/CR-1278 have been confirmed by simulator data. More details on simulator data and TRCs will be available at the meeting.

IV.2 Plant and personnel conditions at recovery point in accident sequence. What assumptions are made about personnel location and functionality under earthquake conditions? How close to fire are staff assumed to function?

Response: The principal recovery actions subsequent to an earthquake are:

- providing AFW water supply from the service water system if a loss of main feedwater occurs coincident with CST failure,
- starting the diesel generators after station battery failure during a station blackout,
- providing HPSI or changing water supply from a source other than the RWST after a small-small LOCA.

The first recovery action can be implemented from the control room. For the other recovery actions, the EPRI/WOG study assumes access to equipment is not restricted by jammed doors or damaged equipment, etc. Operators expected to perform the actions have keys for access to the room in case the plant security system has failed. In the event of RWST failure, the effects of flooding have been considered.

Regarding fires, two scenarios must be considered--fire in the 4160 volt switchgear or the AFW pump room. In either case, the steam admission valves for the turbine-driven AFW pump could be opened from the control room, if DC power is available, or opened locally, if DC power is unavailable. These valves are located in a separate building. No immediate operator action is required in the AFW pump room, since the discharge valves are throttled and locked, and the turbine starts automatically, when steam is supplied.

IV.3 Tools and equipment availability. Where are tools and extra equipment stored? How fast can these items be retrieved under adverse conditions, e.g., fire, earthquake aftershock, etc.?

Response: The only recovery action credited in NSAC-113 which requires "tools or equipment" is refilling the CST using the fire water system for a long term station blackout scenario. This recovery action can be performed in any one of four ways. Two of the means for refill require the use of a connector which makes the fire water system fittings compatible with the fitting at the base of the CST. Both of these also require fire hoses. The preferred means is to use the connector to attach to a fire hose drawn from a hose station located about 40 feet away. The other means is to connect to fire hoses drawn from a hydrant outside. There may be insufficient hoses nearby for this latter method. Both the connector and additional hoses can be obtained from the nonnuclear room or fire hose storage lockers located outside the turbine hall. This room is a few minutes walk from the CST.

The other two means for refilling the CST require inserting a fire hose in the top of the CST. This action requires unbolting the top of the CST manways or vent and using either of the two fire water sources quoted above. Unbolting the manways or vents of the CST requires a wrench, available from turbine operators work station on the floor beneath.

The Two Creeks Fire Station is less than two miles from the plant and could easily connect a pumper from the Pump House Forebay (Lake Michigan) to the CSTs.

IV.4 Further information concerning the value impact analysis of two of the suggested alternatives will be necessary. As agreed to at the February meeting, more details on two alternatives, the Diesel-Driven Auxiliary Feedwater Pump and the Intake Structure Shield Wall Extension, are desired. Particularly more details on the assumptions made and the costs of specific equipment and activities associated with the installation of these alternatives would be helpful. We are also attempting to provide more definitive documentation on these two modifications.

Response: The details of the WEP cost analysis of the two alternatives identified will be presented at the March 31, 1988 meeting.

Attachment C (#4)

Figure 4.2 Location of the Sample Sites in the EUS

POINT BEACH SEISMIC HAZARD CURVE

o PRESCRIBED BASED ON:

- SLOPE OF ZION HAZARD CURVE TAKEN FROM SSMRP

- SCALED TO SET POINT BEACH SSE (0.12g) AT EXCEEDANCE PROBABILITY 2.5E-4/yr

- CURVE CORRECTED FOR SITE AMPLIFICATION

ASSUMPTION:

SEISMIC HAZARD CURVES FOR SITES IN SIMILAR TECTONIC REGIONS HAVE SIMILAR SLOPES.

BRAIDWOOD HAZARD CURVES

PGA	EPRI	LLNL*	RATIO
0.10g	5.3E-5	6.8E-4	≃ 13
0.25g	4.5E-6	5.2E-5	≃ 12
0.50g	3.5E-7	5.2E-6	≃ 15
1.00g	7.2E-9	3.0E-7	≃ 42

* UCID-20421, Vol. 1.

Attachment D (#1)

APPENDIX D: Insights Gained From Industry-Sponsored Study of Point Beach

As part of a nuclear-industry-sponsored effort regarding DHR-related risk, a reanalysis was performed for one of the limited-scope PRAs considered in the A-45 case studies (Reference 1). Discussions held between industry representatives and the NRC staff regarding similarities and differences between the two analyses are summarized in Reference 2. Considerable detail regarding those discussions is presented in the several enclosures to Reference 2. This appendix summarizes the results of those discussions.

In terms of predicted DHR-related core damage frequency, the A-45 case study for Point Beach calculated a value of 3E-04 per reactor-year, whereas the industry-sponsored study calculated 1E-05 per-reactor year (i.e., a factor of 30 lower). The following summary identifies the major differences in assumptions and methods and their resulting contributions to the total difference. In addition, areas where agreement was possible are indicated. The NRC staff believes that the approximate core damage frequency that would result from use of these agreements in a revised staff-sponsored analysis would be about 9E-05 per reactor-year. As given below, a small fraction (less than 20%) of the revision is due to changes that have been made in the plant, and the remainder is due to changes in the methods, assumptions, and data.

It should be cautioned that the revised values quoted below and the specific methods and correlations discussed were examined only in the context of the Point Beach analyses discussed in this Appendix. Their applicability to other plants would have to be determined by specific analyses of the other plants since dominant sequences, plant equipment, and operating procedures could be different, and the "revised" values quoted and methods discussed may not be directly applicable.

The bases for this revised result of 9E-05 per reactor-year are given below. In summary, it was considered reasonable to accept a lower frequency for the SBLOCA and to allow more credit for the presence of new batteries and the lack of dependence of the SI pumps on the availability of the CCW system (the SBLOCA frequency change is the dominant one). It was not considered prudent to allow more credit for many of the operator recovery actions as proposed in the EPRI/WOG study.

Comparison of Point Beach Studies

Sequence*	Core Melt Frequency per Reactor Year From:		
	NRC Case Study	EPRI/WOG Study	Revised NRC value**
$S_2MH_1H_2$	4.7E-05	5.8E-07	7.0E-06

The staff accepted the 3E-03 per reactor-year value proposed by EPRI/WOG for the initiating SBLOCA event after considering operational data presented by EPRI/WOG. However, the staff believes that the 1E-04 proposed for operator action failure per demand is too optimistic since operating data (though limited) do not appear to support the lower value, and the NRC value of 1E-03 was not changed.

T_1MLE	6.7E-06	7.7E-07	7.7E-07

The staff accepted the initiating event frequency proposed by EPRI/WOG based on plant specific data. The staff also tentatively accepted credit for new batteries (since they are now installed and operational), but information is needed to verify the quantitative credit given.

$T_3QH_1H_2$	2.5E-05	0	3.6E-06

The staff used a new value of Q = 0.01, which is below the Q=0.07 value previously used in the NRC studies but still above the (believed optimistic) value of zero proposed by EPRI/WOG (EPRI/WOG contends that for transient T_3, reactor/turbine trip, it is not possible to cause opening of a PORV, therefore Q is zero. The staff believes the probability is small but non-zero).

$T_2MQH_1H_2$	3.5E-06	1.9E-07	5.0E-07

Because of uncertainty in the operational data presented (it varied greatly from year to year), the staff does not recommend the EPRI/WOG-proposed credit for PORVs being available (i.e., unblocked) a portion of the time. The staff therefore continues to endorse the conservative assumption that PORVs are not available to prevent SRV opening. The staff did however agree with a reduction in the probability of inadvertent opening of an SRV from the previously used 0.07 to 0.01 per demand (based on operational data) with a 0.01 per demand probability that an SRV will fail to reclose once open.

---

* Sequences are described in narrative form on the last page of this Appendix.

** This represents the likely value that would be used if a "revised" NRC-sponsored study were to be conducted for Point Beach as of the date of this writing. The narrative below each entry summarizes the bases.

(Table continued from previous page):

Sequence	Core Melt Frequency per Reactor Year From:		
	NRC Case Study	EPRI/WOG Study	Revised NRC value
$S_2MD_1D_2$	8.7E-06	9.5E-08	9.2E-07

The staff agreed with the event initiating frequency of 3E-03 as previously discussed. Based on pump manufacturer's data, the staff also agreed with removal of the SI pumps' dependency on component cooling water, but additional information is needed to quantitatively confirm the risk change due to that removal. The source of the remaining difference could not be identified; therefore, additional information is also needed to consider accepting the remaining difference.

$T_3QD_1D_2$	4.6E-06	0	1.8E-07

The note for a previous "T_3 . . ." sequence also applies here. In addition to that note, more information would be needed to identify the source of and to consider accepting any part of the remaining difference (1.8E-07 vs. 0).

T_2MLE	6.6E-07	1.0E-07	6.6E-07

No changes. The A-45 case study initiating event frequency of 1.0 per reactor year is not significantly different from the 0.91 proposed by EPRI/WOG; and it is not clear how MFW recovery differs in the EPRI/WOG study. Also, additional information would be needed to identify the source of and to consider accepting the remaining difference (6.6E-07 vs. 1.0E-07).

$T_2MQD_1D_2$	6.6E-07	4.1E-08	4.1E-08

The staff used a new value of Q = 0.01, which is below the Q = 0.07 value used in the A-45 studies but above the lower (not directly specified) value used by EPRI/WOG. The staff recommends recognizing the low (but non-zero) probability that a SRV will be lifted during this event. The staff agrees with removal of SI dependency on CCW (per manufacturer's data), but did not change the event frequency, as the EPRI/WOG-suggested frequency of 0.91 is not significantly different from the 1.0 used in the case studies. The above changes resulted in a value that was lower than the EPRI/WOG result. The staff therefore agrees with the EPRI/WOG result.

(Table continued from previous page):

Sequence	Core Melt Frequency per Reactor Year From:		
	NRC Case Study	EPRI/WOG Study	Revised NRC value
S_2MXD_1	5.7E-07	1.0E-08	1.0E-08

As already discussed, the staff, based on data, agreed with the 3.0E-03 initiating event frequency proposed by EPRI/WOG and with removal of the dependency of the SI pumps on the component cooling water system, but information is needed to justify the quantitative credit given.

T_5MLE	9.1E-07	1.3E-08	9.1E-07
T_4MLE	6.2E-07	0	6.2E-07
T_2MLH_1	2.0E-08	1.0E-07	2.0E-08
$T_1QD_1D_2$	1.0E-08	1.0E-07	1.0E-08

No changes were made by the staff for these four sequences. EPRI/WOG credits considerable additional recovery in the form of operator actions that have not been adequately justified. The A-45 study assumed that loss of an AC bus would either trip the plant or lead to a manual trip, and the staff has elected to retain the modest conservatism associated with that assumption (no significant impact).

LTSB	3.6E-05	5.4E-07	9.9E-06

The staff agreed to plant-specific (lower) values for the T_1 frequency and diesel generator local faults, but has not taken additional credit for CST refill and other long-term recovery actions proposed by EPRI/WOG because of uncertainty regarding the operator's ability to recognize the need and perform the actions in the time available. It is considered likely that further reduction could reasonably be justified provided the bases for assuming offsite power recovery within a few hours are sufficient.

TOTAL (Internal Events only)	1.3E-04	2.5E-06	2.5E-05

The above represents the total for all significant "internal" events as listed above. "External" events are listed below.

(Table continued from previous page):

Sequence	Core Melt Frequency per Reactor Year From:		
	NRC Case Study	EPRI/WOG Study	Revised NRC value
$S_2MD_1D_2$	8.7E-06	9.5E-08	9.2E-07

The staff agreed with the event initiating frequency of 3E-03 as previously discussed. Based on pump manufacturer's data, the staff also agreed with removal of the SI pumps' dependency on component cooling water, but additional information is needed to quantitatively confirm the risk change due to that removal. The source of the remaining difference could not be identified; therefore, additional information is also needed to consider accepting the remaining difference.

$T_3QD_1D_2$	4.6E-06	0	1.8E-07

The note for a previous "T_3 . . ." sequence also applies here. In addition to that note, more information would be needed to identify the source of and to consider accepting any part of the remaining difference (1.8E-07 vs. 0).

T_2MLE	6.6E-07	1.0E-07	6.6E-07

No changes. The A-45 case study initiating event frequency of 1.0 per reactor year is not significantly different from the 0.91 proposed by EPRI/WOG; and it is not clear how MFW recovery differs in the EPRI/WOG study. Also, additional information would be needed to identify the source of and to consider accepting the remaining difference (6.6E-07 vs. 1.0E-07).

$T_2MQD_1D_2$	6.6E-07	4.1E-08	4.1E-08

The staff used a new value of Q = 0.01, which is below the Q = 0.07 value used in the A-45 studies but above the lower (not directly specified) value used by EPRI/WOG. The staff recommends recognizing the low (but non-zero) probability that a SRV will be lifted during this event. The staff agrees with removal of SI dependency on CCW (per manufacturer's data), but did not change the event frequency, as the EPRI/WOG-suggested frequency of 0.91 is not significantly different from the 1.0 used in the case studies. The above changes resulted in a value that was <u>lower</u> than the EPRI/WOG result. The staff therefore agrees with the EPRI/WOG result.

(Table continued from previous page):

Sequence	Core Melt Frequency per Reactor Year From:		
	NRC Case Study	EPRI/WOG Study	Revised NRC value
S_2MXD_1	5.7E-07	1.0E-08	1.0E-08

As already discussed, the staff, based on data, agreed with the 3.0E-03 initiating event frequency proposed by EPRI/WOG and with removal of the dependency of the SI pumps on the component cooling water system, but information is needed to justify the quantitative credit given.

T_5MLE	9.1E-07	1.3E-08	9.1E-07
T_4MLE	6.2E-07	0	6.2E-07
T_2MLH_1	2.0E-08	1.0E-07	2.0E-08
$T_1QD_1D_2$	1.0E-08	1.0E-07	1.0E-08

No changes were made by the staff for these four sequences. EPRI/WOG credits considerable additional recovery in the form of operator actions that have not been adequately justified. The A-45 study assumed that loss of an AC bus would either trip the plant or lead to a manual trip, and the staff has elected to retain the modest conservatism associated with that assumption (no significant impact).

LTSB	3.6E-05	5.4E-07	9.9E-06

The staff agreed to plant-specific (lower) values for the T_1 frequency and diesel generator local faults, but has not taken additional credit for CST refill and other long-term recovery actions proposed by EPRI/WOG because of uncertainty regarding the operator's ability to recognize the need and perform the actions in the time available. It is considered likely that further reduction could reasonably be justified provided the bases for assuming offsite power recovery within a few hours are sufficient.

TOTAL (Internal Events only)	1.3E-04	2.5E-06	2.5E-05

The above represents the total for all significant "internal" events as listed above. "External" events are listed below.

Dominant Sequence Definitions

$S_2MH_1H_2$ - A small break LOCA with subsequent loss of main feedwater and failure of emergency core cooling in recirculation.

T_1MLE - A loss of offsite power transient with failure of auxiliary feedwater and feed and bleed.

$T_3QH_1H_2$ - A transient followed by stuck open relief valve (transient induced LOCA) and failure of emergency core cooling in recirculation.

$T_2MQH_1H_2$ - Loss of feedwater transient followed by a stuck open relief valve (transient induced LOCA) and failure of emergency core cooling in the recirculation mode.

$S_2MD_1D_2$ - Small break LOCA with loss of main feedwater and failure of emergency core cooling in the injection mode.

$T_3QD_1D_2$ - A transient followed by a stuck open relief valve (transiesnt induced LOCA) and failure of the emergency core cooling in the injection mode.

T_2MLE - A loss of feedwater transient with failure of auxiliary feedwater and feed and bleed.

$T_2MQD_1D_2$ - A loss of feedwater transient followed by a stuck open relief valve (transient induced LOCA) and failure of the emergency core cooling in the injection mode.

S_2MXD_1 - Small break LOCA with failure of emergency core cooling in injection mode and failure to achieve secondary blowdown.

T_5MLE - Loss of DC bus transient with failure of auxiliary feedwater and feed and bleed.

T_4MLE - Loss of AC bus transient with failure of auxiliary feedwater and feed and bleed.

T_2MLH_1 - Loss of feedwater transient with failure of auxiliary feedwater and failure of emergency core cooling in recirculation.

$T_1QD_1D_2$ - Loss of offsite power transient followed by stuck open relief valve (transient induced LOCA) and failure of emergency core cooling in injection mode.

LTSB - Long term station blackout caused by loss of offsite power transient and failure to recover offsite power with subsequent failure of diesel generators.