APR 07 1987

Virginia Electric and Power Company
ATTN: Mr. W. L. Stewart, Vice President,
Nuclear Operations
P. O. Box 26666
Richmond, VA 23261

Gentlemen:

SUBJECT: MEETING SUMMARY - SURRY, DOCKET NOS. 50-280 AND 50-281

This refers to the meeting conducted at our request in Richmond, Virginia on March 25, 1987. This meeting was held to discuss use of probabilistic risk assessment (PRA) results for NRC inspections. A list of attendees at the meeting is shown in Enclosure 1. Some of the details of the meeting are provided in Enclosure 2 and Enclosure 3 contains the meeting handouts.

It is our opinion that this meeting was beneficial in that it enabled us to better understand your concerns related to the use of PRA information.

In accordance with Section 2.790 of NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

Should you have any questions concerning this letter, we will be pleased to discuss them.

Sincerely,

Virgil L. Brownlee

for Luis A. Reyes, Director
Division of Reactor Projects

Enclosures:
1. Meeting Attendees
2. Meeting Summary
3. Meeting Handouts

cc w/encls:
R. F. Saunders, Station Manager
R. J. Hardwick, Manager - Nuclear
Programs and Licensing
R. Wright, Idaho National Engineering
Laboratory

(bcc w/encls: See page 2)

IE45

APR 0 7 1987

bcc w/encls:
NRC Resident Inspector
Document Control Desk
Commonwealth of Virginia

RII	RII	RII	RII	RII
RCroteau	FCantrell	TPeebles	FJape	VBrownlee
3/ /87	3/ /87	3/ /87	3/3 /87	3/ /87
4/2	4/2	4/2		4/6

ENCLOSURE 1

MEETING ATTENDEES

VIRGINIA POWER/NRC MEETING
MARCH 25, 1987

SUBJECT: USE OF PRA RESULTS IN THE NRC INSPECTION PROGRAM

Name	Affiliation
R. P. Croteau	NRC
J. L. Caldwell	NRC
L. E. Nicholson	NRC
S. Tingen	NRC
D. Sommers	VA PWR
R. Berryman	VA PWR
R. Calder	VA PWR
D. VandeWalle	VA PWR
R. J. Hardwick	VA PWR
S. A. Ahmed	VA PWR
N. E. Clark	VA PWR
E. S. Grecheck	VA PWR
M. Smith	VA PWR
J. Ahlados	VA PWR
W. E. Holland	NRC
F. Jape	NRC
R. Wright	Idaho Nat. Engineer Lab.

ENCLOSURE 2

MEETING SUMMARY

Licensee: Virginia Electric and Power Company (VEPCO)

Facility: Surry

Docket Nos: 50-280 and 50-281

SUBJECT: USE OF PRA RESULTS FOR NRC INSPECTIONS

The topics discussed included PRA applications for NRC inspection of Surry power station and future NRC inspection plans.

The opening remarks were made by Frank Jape followed by a presentation on the PRA applications program for inspection of Surry Power Station given by Ron Wright of the Idaho National Engineering Laboratory. Frank Jape ended the meeting with a discussion of the future NRC inspection plans.

Enclosure 3 contains more details on the areas of discussion.

ENCLOSURE 3

AGENDA

VIRGINIA ELECTRIC AND POWER COMPANY

MARCH 25, 1987

INTRODUCTION	ALAN R. HERDT
PRA APPLICATIONS PROGRAM FOR INSPECTION OF SURRY POWER STATION	RON WRIGHT
FUTURE INSPECTION PLANS	FRANK JAPE
FINAL DISCUSSIONS AND QUESTIONS	ALL

USE OF PRA INSIGHTS FOR NRC INSPECTIONS

PRESENTED TO

VIRGINIA ELECTRIC AND POWER COMPANY

FOR

SURRY POWER STATION

MARCH 25, 1987

Idaho
National
Engineering
Laboratory

PRA APPLICATIONS PROGRAM FOR INSPECTION OF NUCLEAR POWER PLANTS

RON WRIGHT

INTRODUCTION

- PROGRAM HISTORY AND CURRENT STATUS
- PROGRAM REQUIREMENTS AND PRODUCTS
- REGION II PLANTS

PROGRAM HISTORY

- PURPOSE - THIS PROGRAM INTEGRATES PRA INSIGHTS INTO THE NUCLEAR POWER PLANT INSPECTION PROCESS

- PROGRAM ATTRIBUTES

  - IDENTIFIES IMPORTANT PLANT SYSTEMS
  - IDENTIFIES IMPORTANT COMPONENTS FOR RISK SIGNIFICANT SYSTEMS
  - IDENTIFIES COMPONENTS FAILURE MODES
  - IDENTIFIES COMPONENTS TO SPECIFIC INSPECTION MODULES
  - PROVIDES IMPORTANT COMPONENT PLANT SPECIFIC CHECKOFF LIST

ADVANTAGES TO NRC

1. THE PROGRAM IS A DIRECT RESPONSE TO THE COMMISSION'S POLICY AND PLANNING GUIDANCE WHICH CALLS FOR THE USE OF PRA IN SETTING INSPECTION PRIORITIES.

2. INSPECTIONS WILL BE BETTER FOCUSED ON EQUIPMENT WHOSE FAILURE HAS THE GREATEST IMPACT ON PUBLIC RISK.

3. INSPECTORS CAN MANAGE THEIR INSPECTION TIME BASED ON THE IMPORTANCE OF SYSTEMS AND THEIR COMPONENTS. THIS SHOULD RESULT IN THE MORE EFFICIENT USE OF INSPECTION TIME.

IMPORTANT FEATURES OF THE PROGRAM

1. IT IS CONSISTENT WITH AND CONSTRUCTED AROUND THE CURRENT IE MODULES.

2. IT IS STRUCTURED FOR USE BY RESIDENT AS WELL AS REGION BASED INSPECTORS.

3. IT COVERS ALL OF THE COMPONENTS AND ACTIVITIES THAT CONTRIBUTE SIGNIFICANTLY TO PUBLIC RISK.

4. THE PROGRAM CAN BE APPLIED TO ANY FACILITY FOR WHICH A PRA HAS BEEN DEVELOPED.

5. THE PROGRAM CAN BE USED BY NRC INSPECTORS WITHOUT THE NEED TO CONDUCT A DETAILED REVIEW OF THE PRA.

6. A CLEAR DEFINITION OF WHAT CONTRIBUTES TO RISK IS PROVIDED WITHOUT REQUIRING PRA EXPERTISE.

PRA APPLICATIONS PROGRAM
FUNCTIONAL AREAS

- PRA BASED INSPECTION GUIDANCE
- GENERIC BASED INSPECTION GUIDANCE

PROGRAM STATUS

- o INEL
  - INDIAN POINT 2
  - SEABROOK
  - ZION
  - HADDAM NECK (CY), GENERIC
- o BNL
  - LIMERICK
  - INDIAN POINT 3
  - SHOREHAM
  - MILLSTONE
  - GRAND GULF
- o PNL
  - OCONEE

REQUIRED INPUTS

- o PROBABILISTIC RISK ASSESSMENT
- o SYSTEM DESCRIPTIONS
- o TECHNICAL SPECIFICATIONS
- o TESTING PROGRAMS (LIST OF TITLES)
- o MAINTENANCE PROGRAMS (LIST OF TITLES)
- o EMERGENCY OPERATING PROCEDURES
- o CHECK OFF LISTS

BASIC PRA PLANT PROGRAM DESCRIPTION

- INPUT PRA SEQUENCES
- DETERMINE EVENT IMPORTANCES
- SYNTHESIZE SYSTEM IMPORTANCE
- RANK SYSTEMS
- FOR TOP SYSTEMS, PROVIDE COMPONENT AND FUNCTION ANALYSIS

TABLE 16. MOST IMPORTANT SYSTEMS

Code[a]	Name	Importance for Public Health	Importance for Plant Damage
RHR	Residual heat removal	.83	.59
EP	Electric power	.11	.07
CS	Containment spray	.08	.00
AFW	Auxiliary feedwater	.07	.16
RC	Reactor coolant	.01	.07
RP	Reactor protection	.01	.13
SI	Safety injection	.00	.08
CC	Component cooling	.00	.06
MS	Main steam	.00	.03

a. These codes agree with those in Tables 14 and 15.

TABLE 1A. RESIDUAL HEAT REMOVAL SYSTEM FAILURE MODE IDENTIFICATION

The residual heat removal system is important for long term recirculation cooling of the reactor following successful safety injection. The most important system failure is the V sequence, which consists of a loss of coolant accident (LOCA) via an interfacing system. This LOCA thus bypasses reactor containment. Other system failures result in a loss of reactor long term recirculation cooling due to multiple RHR system failures.

Conditions That Lead to Failure

1. Reactor Coolant System to RHR Pumps Isolation Valves 1MOV-RH8701 and 1MOV-RH8702 Fail Open

These valves line up RHR suction from the reactor coolant system. Failure of these valves would expose the RHR piping to RCS pressure, thus creating a leak path which bypasses containment. This accident scenario is several times more important to public health risk than all the other failures in Tables 1 through 9 combined. Maintenance and surveillance of these valves should be observed or reviewed to minimize these failures.

2. Operator Fails to Initiate Switchover from Injection to Recirculation Mode or Fails to Stop Pump at RWST Low Level

This is the dominant failure for the low head recirculation mode and is significant for the high head recirculation mode. It involves the proper interpretation of plant status and proper initiation and completion of full switchover to recirculation. Operator awareness of the criteria for switchover and adherence to emergency procedures are important.

TABLE 1B. IE MODULES FOR RESIDUAL HEAT REMOVAL SYSTEM INSPECTION

Module	Title	Components	Failure[a] Mode
61701	Surveillance(Complex)	RHR Pumps A, B	6
61726	Monthly Surveillance Observation	MOV-RH8701, 8702	1
		MOV-RH8700A, B	3
		RHR Pumps A, B	6
62700	Maintenance	RHR Pumps A, B	4
62703	Monthly Maintenance Observation	RHR Pumps A, B	4
71707	Operational Safety Verification	MOV-RH8701, 8702	1
		MOV-RH8700A, B	2,3
		Containment Sump	5
		RHR Pumps A, B	6
71710	ESF System Walkdown	MOV-RH8701, 8702	1
		MOV-RH8700A, B	3
		Containment Sump	5
		RHR Pumps A, B	6

a. See Table 1A for failure identification.

TABLE 1C. MODIFIED RESIDUAL HEAT REMOVAL SYSTEM WALKDOWN

Component Number	Noun Name	Location	Required Position	Actual Position
	Electrical			
Pump 1A	RHR Pump 1A	Bus149 G34	Racked In	______
	RHR Pump 1A DC and Ckt Bkr Spring Charging Motor Switch		On	______
Pump 1B	RHR Pump 1B	Bus148 G33	Racked In	______
	RHR Pump 1B DC and Ckt Bkr Spring Charging Motor Switch		On	______
1MOV-RH8701	RCS to RHR Pumps Isolation	MCC1391-B3	On	______
1MOV-RH8702	RCS to RHR Pumps Isolation	MCC1381-B6	On	______
1MOV-RH8700A	RHR Pump 1A Suction Isolation	MCC1393C-T5	On	______
1MOV-RH8700B	RHR Pump 1B Suction Isolation	MCC1383A-A4	On	______
	Valve Lineup			
1MOV-RH8700A	RHR Pump 1A Suction Isolation	542' L22	Open	______
1MOV-RH8700B	RHR Pump 1B Suction Isolation	542' M22	Open	______
1MOV-RH8701	RCS to RHR Pumps Isolation	568' Z30	Open	______

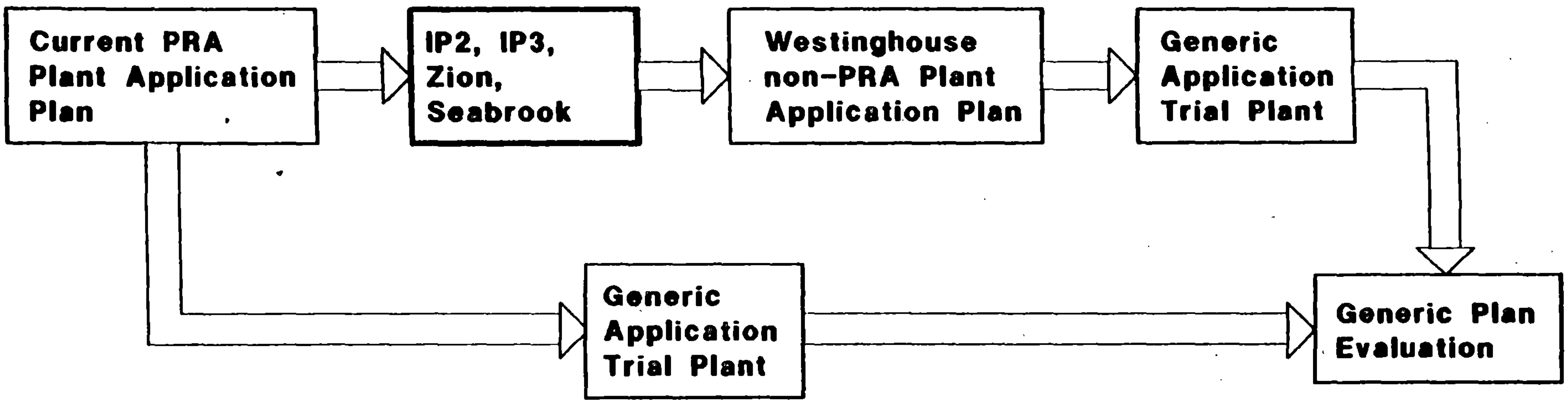
TABLE 5. INDIAN POINT 2 MOST IMPORTANT SEQUENCES

Initiator	Faulted Events	Systems Involved	Public Health Importance
V	A	RHR	.701
ET1	H	EP	.120
ET2	H	EP	.120
ET11B41	F	EP	.003
ET3	A R2	RHR	.003
ET2	E	EP	.002
ET1	A R1	EP, RHR	.002
ET2	A R1	EP, RHR	.002
ET1	E	EP	.002
ET4	A OP41 SL2	PZR, PCS	.001
ET11A	H L1	EP, AFW	.001
ET12A	H L1	EP, AFW	.001
ET7	H L1	EP, AFW	.001
ET1	A LP1	ACC, RHR	.001
FIRE	A FZ1A	FP	.001
FIRE	A FZ14	FP	.001
FIRE	A FZ32A	FP	.001
ET4	A SL1	PCS	.001

Notes:

Initiator	Description
V	Interfacing System Loss of Coolant Accident
ET1	Large Loss of Coolant Accident
ET2	Medium Loss of Coolant Accident
ET3	Small Loss of Coolant Accident
ET4	Steam Generator Tube Rupture
ET7	Soss of Main Feedwater
ET11A	Turbine Trip
ET11B41	Loss of Offsite Power
ET12A	Spurious Safety Injection

PRA Applications to Inspection



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IMPORTANT SYSTEMS FOR EACH PLANT

IP2	IP3	SEABROOK	ZION
RHR	SW	RHR	RHR
SW	EP	SSPS	EP
CCW	CCW	RWST	CS
CF	RPS	PCCW	AFW
EP	HPI	ESFAS	RCS
CS	MS	SW	RPS
SI	RECIRC	EFW	SI
ACC	RCS	EP	CCW
PR	AFW		MS
AFW	LPI		
	SAS		
	ACC		
	CS		
	CF		

GENERIC METHODOLOGY

- o SYSTEM IDENTIFICATION FROM GENERIC DATA AND PLANT FUNCTIONAL REQUIREMENTS

- o IMPORTANT COMPONENTS
  - GENERIC DATA
  - GENERIC FAULT TREE RESULTS (ASEP)
  - SYSTEM FAULT TREE ANALYSIS
  - ENGINEERING JUDGEMENT

REGION II PLANTS

- o SURRY (NUREG/1150)
- o McGUIRE, GENERIC

FUTURE INSPECTION PLANS

- ANNOUNCED INSPECTION

- TEAM OF FOUR OR FIVE INSPECTORS, PLUS TEAM LEADER

- INSPECTION REQUIRES ABOUT TWO WEEKS ON-SITE

- REQUIRES COOPERATION FROM UTILITY:

    OPERATORS

    SUPERVISORS

    CRAFTSMEN