U.S. NUCLEAR REGULATORY COMMISSION

REGION V

Report Nos.	50-206/91-08, 50-361/91-08, 50-362/91-08
Docket Nos.	50-206, 50-361, 50-362
License Nos.	DPR-13, NPF-10, NPF-15
Licensee:	Southern California Edison Company Irvine Operations Center 23 Parker Street Irvine, California 92718
Facility Name:	San Onofre Units 1, 2, and 3
Meeting at:	Region V Office, Walnut Creek, CA
Conference Date:	March 1, 1991
Prepared by:	David L. Proulx, Project Inspector

Approved By: [signature] For 3-29-91

P. H. Johnson, Chief — Date Signed

Reactor Projects Section 3

Summary

A management meeting was held on March 1, 1991 to discuss Unit 1 design basis documents, single failure analysis, and restart issues; on-line maintenance and its implications; mid-loop operations; and the results of the recent NRC setpoint methodology team inspection.

DETAILS

1. Meeting Participants

Nuclear Regulatory Commission

J. B. Martin, Regional Administrator
R. P. Zimmerman, Director, Division of Reactor Safety and Projects
S. A. Richards, Chief, Reactor Projects Branch
D. F. Kirsch, Chief, Reactor Safety Branch
P. H. Johnson, Chief, Reactor Projects Section 3
C. W. Caldwell, Senior Resident Inspector
C. W. Townsend, Resident Inspector, San Onofre
D. G. Acker, Reactor Inspector
D. L. Proulx, Project Inspector

Southern California Edison Company

H. B. Ray, Senior Vice President, Nuclear
B. Katz, Manager, Nuclear Oversight
J. T. Reilly, Manager, Nuclear Engineering and Construction
R. W. Waldo, Operations Manager
M. P. Short, Technical Manager
K. L. Johnson, Controls Discipline Manager
F. R. Nandy, Manager, Nuclear Licensing
L. D. Brevig, Supervisor, Onsite Nuclear Licensing

2. Background

On March 1, 1991, a management meeting was held at the Region V office in Walnut Creek, California. The purpose of this meeting was to discuss Unit 1 design basis documents, single failure analysis, and restart issues; on-line maintenance and its implications; mid-loop operations; and the results of the recent NRC setpoint methodology team inspection. The meeting was convened at 12:30 p.m. A copy of slides used during the licensee's presentations is enclosed.

3. Introduction

Mr. Zimmerman opened the meeting by explaining the purpose of the meeting and went over the agenda briefly. He also stated that the issues to be discussed during the meeting were of interest to the NRC and expressed a desire for open dialogue between the NRC and the licensee on the issues.

4. Discussion of Unit 1 Design Basis Documents (DBD)

Mr. Reilly opened the licensee's presentation with a discussion of SCE's progress in updating the design basis documentation for Unit 1. He stated that six DBD updates had been completed for Unit 1 in 1990 (Nuclear Instruments, Nitrogen System, Load Sequencing, Component Cooling Water, Salt Water Cooling, and Environmental Qualification).

The discussion then focused on SCE's efforts to update the DBD for the Safety Injection System (SIS). Mr. Reilly summarized the progress made to review the SIS. He noted that SCE had reperformed the small break loss of coolant accident (LOCA) analysis and had found that the original analysis was very conservative. The calculated peak clad temperature fell from about 2150 degrees F under the old analysis to about 1450 degrees F with the new analysis. Mr. Reilly stated that the completion of the SIS DBD effort was coordinated with system modifications scheduled for the cycle 12 refueling outage, however SCE considered that their review of the system was sufficiently complete to provide a high level of assurance that no significant deficiencies remain undetected. Mr. Reilly completed the discussion in this area by outlining SCE's schedule for the completion of other DBDs.

5. Discussion of Unit 1 Main Feed Pump (MFP) Restart Delay

SCE had recently identified a timing delay in the restart of the MFPs under certain accident conditions. The MFPs also function as the safety injection pumps. Mr. Reilly stated that the analyses performed for the Unit 1 accidents that the MFP delay could affect consisted of (1) a simultaneous break and loss of power (LOP) for a large break loss of coolant accident (LOCA), (2) a loss of power at the time of reactor trip, and (3) a simultaneous break and LOP for a small break LOCA.

Mr. Ray stated that the 11-second time delay associated with the restart of the MFP's was a known factor. Mr. Ray's opinion was that SCE's error was in not considering the order in which the LOP or the SIS actuation could occur.

Mr. Caldwell then asked the licensee to explain what SCE considered to be the worst case scenario for Unit 1.

Mr. Ray explained that the actual worst case accident consisted of a break with SIS actuation, with the loss of power occurring at just the proper time frame (including all time delays) which gives the worst-case results. He referred to this scenario as a SIS with a "smart LOP". He contended that although the SIS with a smart LOP had the worst outcome of any possible situation, SCE is not required to analyze for this type of accident. Mr. Ray then stated that SCE considers that the intent of the regulation for the SIS/LOP scenarios involves a loss of power that is dependent upon other perturbations that logically follow in the sequence of events. He summarized his discussion by stating that SCE had made the necessary design change to eliminate this problem and that SCE has a high level of confidence that no significant weaknesses exist in their analyses at this time that could hinder Unit 1 restart.

Mr. Townsend then commented that the analysis takes credit for the charging pumps during the injection phase of the design basis accident. He perceived this as an apparent weakness in SCE's DBD because the charging pumps have lockout relays that prevent restarting of these pumps until the logic is reset. Mr. Reilly responded by stating that this was true, but the analysis only takes credit for the charging pumps after the logic is reset. He further added that the charging pump lockout relays were being removed.

Mr. Ray then summarized some preliminary results of the Main Steam Line Break analysis, and stated that the final results should be available for the NRC in March 1991.

Mr. Martin then concluded the discussion in this area by expressing general satisfaction with the licensee's efforts to update DBDs and to find and correct engineering problems. Mr. Martin questioned whether the licensee's efforts to update their understanding of the plant design was of value and Mr. Reilly indicated that it was.

6. Discussion of Single Failure Analysis for Unit 1

Mr. Reilly discussed the licensee's efforts in completing the Single Failure Analysis (SFA) for Unit 1. He stated that SCE had noted nine modifications that needed to be performed as a result of the SFA, and that eight of these were already complete. The remaining modification (dealing with Vital Bus Transfers) will be completed during the cycle 12 refueling outage. He added that once all of the modifications are complete, there will be a 2% decrease in the calculated core melt frequency.

Mr. Caldwell agreed that the Unit 1 SFA was a significant effort, but he felt that some items used in contemporary SFAs were not considered by SCE in their analysis. He then stated that check valves and relief valves are considered by the industry to be active components, and that SCE's SFA may not be valid without considering these components.

Mr. Ray replied that they considered check valves and relief valves to be passive components because SCE assumed that Unit 1 was licensed by an analysis that considered these valves to be passive.

Mr. Caldwell pointed out that the modification that installed valve 1100-E downstream of the Volume Control Tank (VCT) left the charging system in a state in which it still has the potential for a common mode failure. He stated that the two VCT discharge valves limit shut rather than torque shut so there is a potential for a common mode failure of both valves if there is significant leakage. He then questioned if the VCT modification actually meets SCE's intent.

Mr. Reilly agreed that a torqued valve would work better, but explained that this has minimal safety significance so it does not need to be remedied in an expeditious manner. He added that SCE would probably make these minor modifications during the cycle 12 refueling outage.

7. Discussion of the Misalignment of CV-518

Mr. Reilly discussed the role of CV-518 in the Unit 1 integrated Emergency Core Cooling System (ECCS) response to a LOCA. He then presented the results of SCE's investigation into the cause of the valve misalignment, along with the status of SCE's planned corrective actions for the event, per the attached slides. Mr. Reilly, along with Mr. Ray, noted a missed opportunity to have discovered that CV-518 was out of alignment when CV-517 was found similarly misaligned. It was also pointed out that the CV-518 event had some similarities to the event in which the steam trap for the turbine driven auxiliary feed pump in Unit 2 was isolated for an extended period of time.

Mr. Reilly then presented the licensee's position on the safety significance of the CV-518 event per the attached slides. SCE considers this event to be of minimal safety significance because operators would have noted the anomalies in flow during the recirculation phase of an accident and concluded that CV-518 was misaligned. They would, however, depend on non-safety grade instrument air to provide the motive force for any desired repositioning of the valve.

Several NRC representatives at the meeting questioned whether operators could exhibit such troubleshooting under the duress of an accident. Mr. Reilly noted that if the operators could not determine the status of the valve, they would probably alternate running the Refueling Water Pumps, which would provide adequate flow to mitigate the event.

Mr. Ray stated that there is no certainty that a LOCA coupled with this event would have been mitigated by operator action, however SCE concluded that the training operators receive made operator intervention very likely. He then stated that a full root cause analysis will be completed shortly for this issue.

8. <u>Discussion of Unit 1 Sodium Silicate Blockage of Containment Spray Rings</u>

Mr. Reilly first presented background information and a brief explanation of the problem with sodium silicate blockage of the Unit 1 containment spray rings, per the attached slides. He stated that this event had no safety significance because an evaluation by SCE has determined that with a conservative evaluation, only 39 of 70 nozzles needed to be available. (Only seven nozzles were actually blocked.) Mr. Ray added that SCE will write an informational Licensee Event Report (LER) for the sodium silicate blockage problem.

9. <u>Discussion of Performing Maintenance During Plant Operation</u>

Mr. Waldo opened the discussion about performing maintenance during plant operations (on-line maintenance) by reiterating the lessons learned and corrective actions resulting from the recent escalated enforcement action that dealt with the mispositioned containment sump valve and the inoperability of the turbine driven auxiliary feed pump. Mr. Ray added that SCE is committed to performing less on-line maintenance, which will require extending outages in the future. Mr. Waldo then outlined SCE's new policy for work on safety-related systems, per the attached slides. Mr. Ray stated that SCE was not presently in agreement with the INPO guidelines for minimizing outage time by performing maintenance while the plant is operating.

Mr. Martin stated that if SCE had a good Reliability Centered Maintenance (RCM) program, there would be better reliability of safety-related systems and less corrective maintenance would be required. He then requested that SCE briefly explain the status of the SONGS RCM program.

Mr. Ray responded that the full analysis of RCM at the site has not yet been completed, but SCE will issue a report of the results of the analysis. Mr. Katz added that about 50% of the analyses were currently complete.

Mr. Martin requested that SCE provide the data on the unavailability of the High Pressure Safety Injection (HPSI) system, the Auxiliary Feedwater (AFW) system, and the Emergency Diesel Generators (EDGs). Mr. Waldo presented the requested system unavailability data, which is enumerated in the attached slides.

Mr. Martin commented that the original intent of the technical specifications action statements was not to provide for routine on line maintenance to allow for shorter refueling outages. Mr. Martin encouraged the licensee to ensure that the performance of on line maintenance was well thought out when performed.

Mr. Ray then briefly discussed SCE's boundary of the week program, and stated that this program can be a useful tool in planning work if properly employed.

10. Discussion of SCE's Probabilistic Risk Assessment (PRA)

Mr. Katz discussed SCE's PRA efforts with particular emphasis placed upon its impact on performing on-line maintenance. He also gave a brief presentation of the licensee's uses of PRA, per the attached slides. He added that SCE plans to use on-line risk assessments using a computer code prepared in Great Britain (with NRC approval). In addition SCE would like to employ risk-based technical specifications in the future.

Mr. Martin stated that he was pleased that SCE is using PRA in rendering decisions about plant operations and maintenance. He added that the industry must have a rational basis for removing safety-related systems from service; and he expressed concern that the general trend in the industry of minimizing outage time by maximizing on-line maintenance may not be based on sound safety analysis. He then recommended that SCE maintain an open dialogue with NRR regarding the use of PRA.

11. Discussion of Mid-Loop Operations

Mr. Martin stated that the use of mid-loop operations contains a high risk factor, especially when coupled with complicated work items.

Mr. Ray stated that a better mode of operation for an outage would be to use nozzle dams. He added that SCE is planning to off-load the entire core during the next refueling outage, and does not currently intend to exercise the mid-loop operation option. He noted that the tendency in the nuclear industry of late is to forego mid-loop operation in favor of off-loading the core. He added that a full safety analysis needs to be done prior to widespread use of this mode of operation.

Mr. Martin reiterated the increased risk associated with shutdown operations, and cautioned the licensee that the NRC will be focusing on this issue in the near future. He explained that hydraulic phenomena associated with boiling media during mid-loop operations was much worse than previously thought. In addition, he said, if utilities off-load the entire core, much flexibility can be attained in planning an outage. However, he stated that if it is unreasonable to off-load the core, (e.g. brief outage for steam generator tube repair) then licensees will be obligated to go to mid-loop operation. He noted, however, that special care must be taken, particularly during periods of high decay heat load.

12. Discussion of the Setpoint Methodology Team Inspection

Mr. K. Johnston discussed the SCE position on the findings and conclusions of the NRC team inspection of the Unit 2 and 3 instrument setpoint methodologies. Mr. Ray noted that SCE did not agree with some of the conclusions reached by the team, and did not agree that some of the NRC's findings were representative of recent work.

Mr. Richards replied that he attended the exit meeting and that the licensee attendees were substantially in agreement with the findings of the team. He then reiterated the team's findings and conclusions and stated that these findings appeared to be technically justified.

Mr. Ray stated that since SCE and the NRC could not reach a consensus on the team inspection issues at this time, SCE would like to meet with the NRC concerning these issues at another date.

Mr. Kirsch agreed, and stated that arrangements will be made for another meeting concerning the Setpoint Methodology Team Inspection.

13. Discussion of the Control of Switchyards

Mr. Martin discussed a problem at another nuclear power plant in which an outside organization working in the switchyard caused an event. He then requested that SCE describe how switchyard work is handled at SONGS.

Mr. Waldo replied that the policy at SONGS is that switchyard workers contact the control room prior to entering the switchyard. He added that one half of the switchyard is owned by SCE, and the other half is owned by SDG&E. He noted that there have been isolated incidents in which SDG&E personnel have entered the switchyard without the knowledge of the control room operators, but there have been no problems associated with these minor events. However, he stated, SCE will closely monitor switchyard activities to ensure no major problems arise.

Mr. Martin concluded this discussion by cautioning the licensee that they should have a high degree of confidence regarding switchyard work, to prevent problems which could adversely affect the plant.

14. Discussion of SCE Erosion-Corrosion Modeling

Mr. Martin questioned whether comtemporary methods issued by the Electric Power Research Institute (EPRI) would have predicted certain pipe failures at SONGS. He also stated that Region V was discussing this issue with NRR management. He questioned whether SCE planned to meet with EPRI and make a presentation to NRR to demonstrate the techniques SCE is employing in evaluating erosion-corrosion.

Mr. Ray replied that SCE will endeavor to meet with EPRI and will meet with INPO and NRR on this issue in the near future.

15. Closing Remarks

Mr. Martin thanked everyone, and adjourned the meeting at 4:45 p.m.

DESIGN BASES DOCUMENTATION PROGRAM STATUS

DBDs COMPLETED

SIX DBDs COMPLETED IN 1990. ALL UNIT 1.

- NUCLEAR INSTRUMENTATION SYSTEM
- AIR/BACKUP NITROGEN SYSTEM
- SAFEGUARD LOAD SEQUENCING
- COMPONENT COOLING WATER
- SALTWATER COOLING
- ENVIRONMENTAL QUALIFICATION

ATTACHMENT

1

DESIGN BASES DOCUMENTATION PROGRAM STATUS

DBD SCHEDULE

	1991	1992	1993	1994	1995
UNIT 1 DBDs	3	7	8	4	5
UNITS 2&3 DBDs	11	6	7	5	2
UNITS 1, 2&3 DBDs	8	6	4	-	-
TOTAL	22	19	19	9	7

ATTACHMENT

FEEDWATER PUMP RESTART DELAY FOLLOWING LOP/SIS

BACKGROUND

UNIT 1 SAFETY ANALYSIS EVENT SEQUENCE CONSIDERED:

- SIMULTANEOUS BREAK AND LOP FOR LARGE LOCAs FOLLOWED RAPIDLY BY A SIS SIGNAL.
- LOP AT THE TIME OF REACTOR TRIP FOR SMALL BREAKS (INCLUDING SGTR).
- SIMULTANEOUS BREAK AND LOP FOR SLB.
  <u>NOTE</u>: SLB WAS EVALUATED WITH AND WITHOUT LOP.

IN SAFETY ANALYSIS, LOP IS A CONSEQUENCE OF EITHER THE EVENTS (LOCA, SLB, OR SGTR) OR REACTOR TRIP. THAT IS, LOP IS NOT AN INDEPENDENT EVENT.

ATTACHMENT

FEEDWATER PUMP RESTART DELAY
FOLLOWING LOP/SIS

PROBLEM

LOP/SIS TEST IDENTIFIED TWO UNANTICIPATED INTERACTIONS:

- OBSERVED DELAY IN FEEDWATER PUMP START.

- LOCK-OUT OF CHARGING PUMPS.

ATTACHMENT

FEEDWATER PUMP RESTART DELAY FOLLOWING LOP/SIS

SIGNIFICANCE

- LOP/SIS UNDERVOLTAGE CIRCUITRY
  - NOT APPLICABLE FOR DESIGN BASIS LARGE AND SMALL BREAK LOCAs SINCE LOP DOES NOT PRECEDE SIS BY MORE THAN 11.5 SECONDS.
  - WILL NOT IMPACT VERY SMALL LOCA OR STGR
  - MSLB PRELIMINARY ASSESSMENT--REMAINS WITHIN DESIGN BASIS.
- CHARGING PUMP LOCKOUT DURING LOPSIS CONDITIONS HAS NO IMPACT ON SAFETY ANALYSIS.
  - NO IMPACT--CHARGING PUMPS ARE NOT CREDITED.

ATTACHMENT

FEEDWATER PUMP RESTART DELAY
FOLLOWING LOP/SIS

CORRECTIVE ACTION

UNDERVOLTAGE AND LOCKOUT FEATURES BYPASSED BY SIS.

FINALIZE MSLB EVALUATION.

ATTACHMENT

SINGLE FAILURE ANALYSIS
RELATIVE TO UNDERVOLTAGE CIRCUITRY

- SFA CORRECTLY IDENTIFIED UNDERVOLTAGE PROTECTION SCHEME, OPERATIONAL FEATURES

- SFA CONSIDERED INDIVIDUAL COMPONENT FAILURE WITHIN THE UNDERVOLTAGE PROTECTION CIRCUIT

- EFFECTS OF THE UNDERVOLTAGE PROTECTION CIRCUITS WERE EVALUATED ON THE BASIS OF SIMULTANEOUS SIS AND LOP CONSISTENT WITH THE DESIGN BASIS AND LICENSING BASIS

ATTACHMENT

7

SONGS 1 ECCS SINGLE FAILURE ANALYSIS
BACKGROUND

PREVIOUS ANALYSIS

- 1976 ECCS SINGLE FAILURE ANALYSIS

- 1987 ESF/RPS SINGLE FAILURE ANALYSIS (PERFORMED IN RESPONSE TO PT-459 FAILURE)

SONGS 1 ECCS SINGLE FAILURE ANALYSIS
BACKGROUND

NEW ANALYSIS

- 1990 ECCS SINGLE FAILURE ANALYSIS (PERFORMED PER COMMITMENT IN CYCLE 10 RESTART REPORT)
- LOW SAFETY SIGNIFICANCE
- CONFIRMS EXPECTATION OF CYCLE 10 RESTART REPORT THAT ANY FUTURE ISSUES WOULD HAVE LOW SAFETY SIGNIFICANCE

ATTACHMENT

SONGS 1 ECCS SINGLE FAILURE ANALYSIS HIGHLIGHTS

SFA METHODOLOGY

- MEETS MODERN SFA STANDARDS
- FAILURE MODES AND AFFECTS ANALYSES
- EVENT SPECIFIC/TIME DEPENDENT EVALUATIONS
- APPROXIMATELY 3,000 INTERACTIONS EVALUATED

NINE MODIFICATIONS

- EIGHT COMPLETED THIS OUTAGE
- ONE (VITAL BUS XFER) SCHEDULED FOR CYCLE 12

ATTACHMENT

SONGS 1 ECCS SINGLE FAILURE ANALYSIS HIGHLIGHTS

<u>VALIDATION OF ASSUMPTIONS (VERIFY PLANT CONFIGURATION ASSUMED IN SFA)</u>

- NEW CALCULATIONS, ADDITIONAL IST, PROCEDURE REVISIONS

ATTACHMENT

<u>OVERALL SAFETY SIGNIFICANCE</u>

- INDIVIDUALLY/COLLECTIVELY LOW SAFETY SIGNIFICANCE

- INCREASED OVERALL SONGS 1 CORE DAMAGE FREQUENCY BY APPROXIMATELY 2%

SONGS 1 ECCS SINGLE FAILURE ANALYSIS SUMMARY

- LOW SAFETY SIGNIFICANCE OF SFA ISSUES, CONSISTENT WITH EXPECTATION IN SCE CYCLE 10 RESTART REPORT
- EIGHT MODIFICATIONS COMPLETED
- ONE MODIFICATION FORMALLY DEFERRED TO CYCLE 12

ATTACHMENT

MISORIENTATION OF CV-518

BACKGROUND

- FOR UNIT 1, HIGH SPRAY FLOW DURING INJECTION AND TRANSITION PHASE, LOW SPRAY FLOW DURING RECIRCULATION PHASE
- CV-517 OR -518 ALONE PASSES FULL HIGH SPRAY FLOW
- CLOSING CV-517 AND -518 REDUCES SPRAY FLOW AT ONSET OF RECIRCULATION PHASE

CAUSE

- OPERATORS CHANGED VALVE POSITION DURING MAINTENANCE WITHOUT ADEQUATE CONTROLS
- NO EXTERNAL VALVE POSITION INDICATION
- DISCOVERED DURING RECENT OUTAGE TESTS

MISORIENTATION OF CV-518

CORRECTIVE ACTION

- CV-518 ALIGNED PROPERLY

- PROPER POSITION INDICATOR ALIGNMENT VERIFIED FOR SIMILAR VALVES

- INSTALLED EXTERNAL POSITION INDICATOR

ATTACHMENT

MISORIENTATION OF CV-518

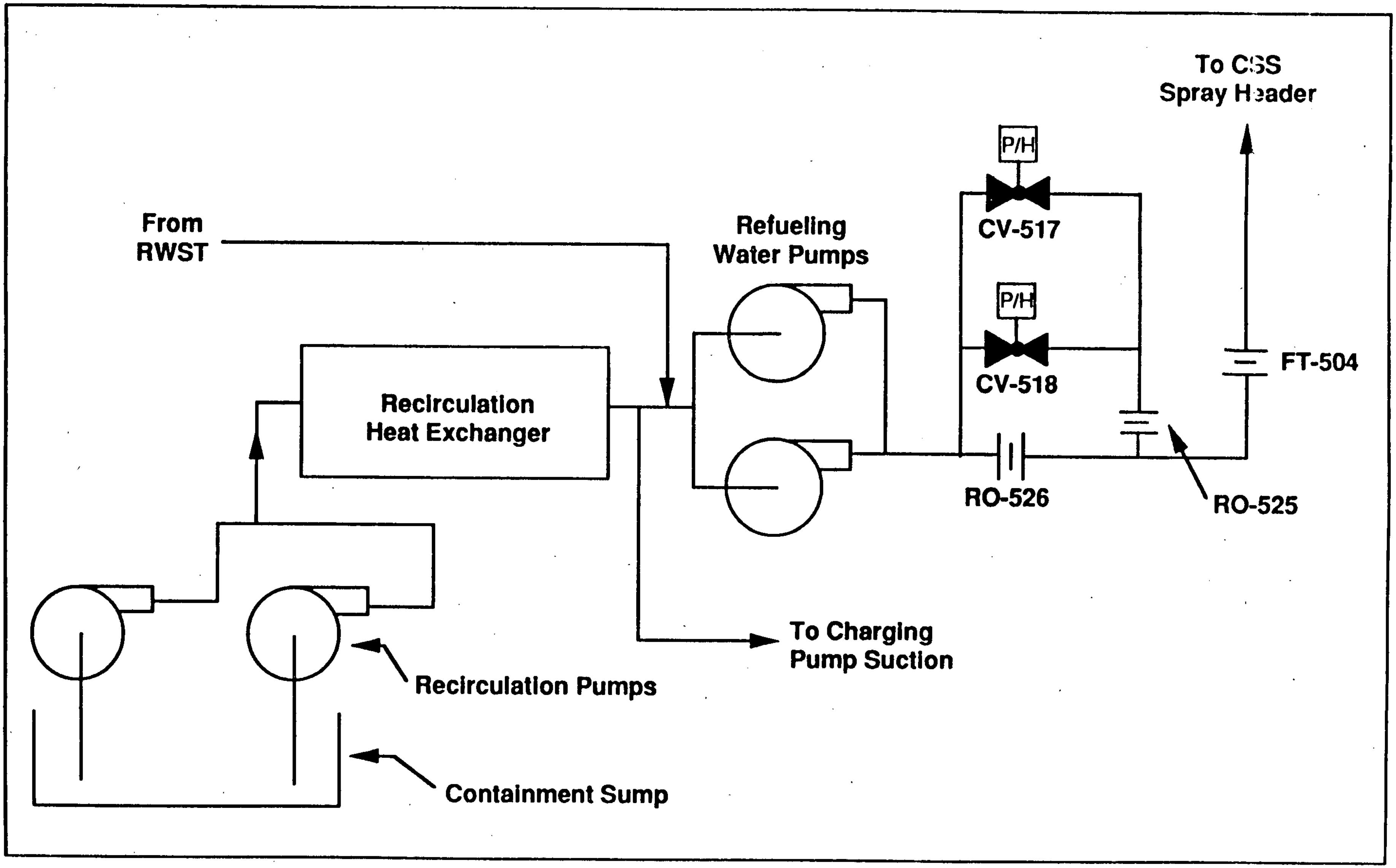
CONSEQUENCES

CONTAINMENT SPRAY FLOW RATE DURING THE INJECTION AND TRANSITION PHASES WAS UNAFFECTED

CONTAINMENT SPRAY FLOW DURING THE RECIRCULATION PHASE WOULD HAVE BEEN GREATER THAN EXPECTED AND WOULD HAVE PLACED AN OVER-POWER DEMAND ON THE SPRAY PUMP

CONTROL ROOM OPERATORS WOULD HAVE BEEN ALERTED TO THIS CONDITION BY THE ABSENCE OF EXPECTED SPRAY FLOW DECREASE WHEN SHIFTING TO RECIRCULATION, OR BY THE TRIPPING OF THE SPRAY PUMP DUE TO OVERCURRENT.

SAFETY SIGNIFICANCE WAS CONSIDERED TO BE MINIMAL AS THE OPERATORS WOULD HAVE OBSERVED THAT EXCESSIVE FLOW EXISTED AND TAKEN ACTION TO CORRECT IT



Spray Flow Limiter System During Recirculation

ATTACHMENT

POTENTIAL BLOCKAGE OF CONTAINMENT SPRAY RINGS

ATTACHMENT

BACKGROUND

- SODIUM SILICATE COATING APPLIED TO SPRAY HEADER MID-70s AS CORROSION PROTECTION
- SURVEILLANCE TESTING OF SPRAY RING PERFORMED EVERY OTHER REFUELING OUTAGE
- THERMOGRAPHY TESTING PERFORMED IN PREVIOUS THREE SURVEILLANCES 6/80, 11/83, 12/88 (NO BLOCKAGE FOUND)

PROBLEM

- SEVEN CONTAINMENT SPRAY NOZZLES DETERMINED TO BE PLUGGED DURING MOST RECENT SURVEILLANCE

POTENTIAL BLOCKAGE OF CONTAINMENT SPRAY RINGS

CONSEQUENCES

- 39 OF THE TOTAL 71 NOZZLES REQUIRED TO BE AVAILABLE BY DESIGN BASIS CRITERIA (PRESSURE/TEMP/DOSE)

EVALUATION OF AS-FOUND CONDITION

- SEVEN PLUGGED NOZZLES/64 NOZZLES AVAILABLE
- PIPING RESTRICTION HAS NO EFFECT
- DESIGN BASIS SATISFIED

ATTACHMENT

POTENTIAL BLOCKAGE OF CONTAINMENT SPRAY RINGS

<u>WORST CREDIBLE ANALYSIS</u>

- 39 NOZZLES REQUIRED OPERABLE TO SATISFY DESIGN BASIS (PRESSURE, TEMP, DOSE)
- 10 NOZZLES REQUIRED TO MEET REALISTIC CONDITIONS (PRESSURE, TEMP, DOSE)
- TEST RESULTS SHOW PLUGGED NOZZLES REOPEN WITH DIFFERENTIAL PRESSURE OF 15-30 PSID
- EXPECTED 130 PSID ΔP IN SPRAY RINGS WITH 10 NOZZLES OPERABLE
- AT LEAST 10 NOZZLES WOULD UNBLOCK, SATISFYING BEST ESTIMATE CONDITIONS

POTENTIAL BLOCKAGE OF CONTAINMENT SPRAY RINGS

SIGNIFICANCE

- AS FOUND CONDITION ACCEPTABLE
- WORST CREDIBLE ANALYSIS REALISTICALLY ACCEPTABLE
- IN OUR JUDGMENT AT LEAST 39 NOZZLES WOULD BE AVAILABLE
- MINIMAL SAFETY SIGNIFICANCE

ATTACHMENT

POTENTIAL BLOCKAGE OF CONTAINMENT SPRAY RINGS

CORRECTIVE ACTION

- ALL NOZZLES CLEARED
- LOOSE MATERIAL IN HEADER/RINGS REMOVED
- HEADER/RING COATING REMOVED
    - MINIMAL RUST POTENTIAL (55% RH, 108°F)
    - CARBON STEEL RISER PREVIOUSLY REPLACED WITH STAINLESS STEEL
    - UNCOATED AREAS FOUND TO BE RUST FREE
- EVALUATING NON-PLUGGING NOZZLES IN CYCLE 12

ATTACHMENT

POTENTIAL BLOCKAGE OF CONTAINMENT SPRAY RINGS

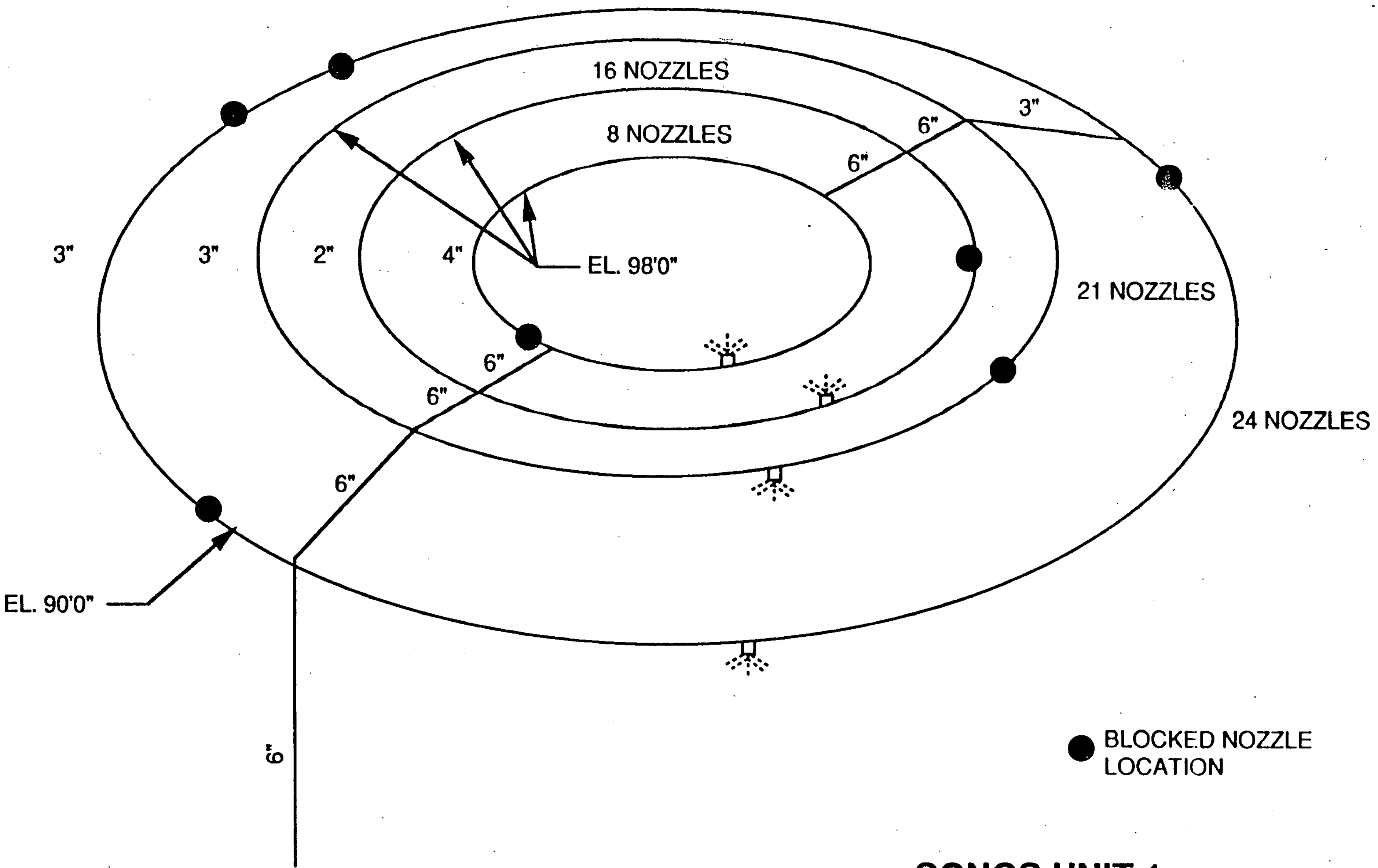
CONCLUSION

- DETERMINISTIC ANALYSIS TO ESTABLISH PRECISE NUMBER OF NOZZLES WHICH COULD PLUG DUE TO LOOSE MATERIAL NOT TRACTABLE

- CONTAINMENT SPRAY EVALUATED TO HAVE BEEN ADEQUATE

- INFORMATIONAL LER TO BE SUBMITTED FOR INDUSTRY AWARENESS

ATTACHMENT



SONGS UNIT 1
CONTAINMENT SPRAY PIPING

ATTACHMENT

ONLINE MAINTENANCE
OPERATOR ATTENTIVENESS, UNITS 2/3

ATTACHMENT

PROBLEMS:

- HIGH ATTRITION LED TO REDUCED AVAILABLE MANPOWER
- AGGRESSIVE IMPLEMENTATION OF A PREVENTATIVE MAINTENANCE PROGRAM RESULTED IN INCREASED ONLINE WORK LOAD
- OPERATORS BELIEVED SUPPORTING SCHEDULED WORK HAD A HIGH PRIORITY AND CONCENTRATED EFFORTS HERE
- EXTENSIVE ROUND SHEETS CREATED A LARGE DEMAND ON PLANT EQUIPMENT OPERATOR TIME

ONLINE MAINTENANCE
OPERATOR ATTENTIVENESS, UNITS 2/3

ATTACHMENT

SOLUTIONS:

- SIGNIFICANT PAY INCREASES HAVE BEEN AWARDED AT ALL LEVELS REVERSING PREVIOUS ATTRITION TREND
- LARGE NUMBERS OF OPERATORS IN TRAINING PIPELINE WILL BE AVAILABLE IN FALL 1991
- OPERATORS HAVE BEEN RECEIVING REPEATED COACHING ON ACTUAL MANAGEMENT PRIORITIES
- INCREASED MONITORING REQUIREMENTS HAVE BEEN PROCEDURALIZED
- DETAILED TAILBOARDING OCCURS AT THE START OF EACH SHIFT
- OVERALL MAINTENANCE WORKLOAD REDUCED
- EXTRA OPERATOR SUPPORT SCHEDULED FOR HIGH WORK EVOLUTIONS

SCE WORK APPROVAL POLICY

(UNDER DEVELOPMENT FOR SAFETY RELATED SYSTEMS)

- PMs NORMALLY DONE DURING OUTAGES
- EQUIPMENT OUTAGES NORMALLY LIMITED TO ACTIVITIES NEEDED TO MAINTAIN SYSTEM OPERABILITY AND RELIABILITY
- OPTIONAL WORK IS APPROVED IF IT DOES NOT APPRECIABLY INCREASE SYSTEM OUTAGE TIME
- RETEST COMMITTEE REVIEW OF RETEST REQUIREMENTS
- HIGH PRIORITY GIVEN TO MINIMIZING SYSTEM OUTAGE TIME
- RESTRICTIONS PLACED ON APPROVAL OF CROSS-TRAIN OR MULTIPLE SYSTEM OUTAGES

ATTACHMENT

SAFETY SYSTEM PERFORMANCE

SYSTEM	UNIT 1		UNIT 2		UNIT 3		IND MED	
	P	UP	P	UP	P	UP	P	UP
HPSI	.0001	.0019	.0263	.0021	.0123	.0034	.0062	.0010
AXFDWTR	.0289	.0062	.0147	.0315	.0258	.0009	.0090	.0015

ATTACHMENT

	SITE		IND MED	
	P	UP	P	UP
DGEN	.0087	.0023	.0064	.0075

DATA IS A 24 MONTH AVERAGE FOR THE YEARS OF 1989-1990.

P = PLANNED
UP = UNPLANNED

OPTIMIZING SAFETY USING PRA

OVERVIEW

OBJECTIVE

Mission Statement of the Nuclear Safety Group:

To provide plant management with timely and credible advisories regarding:

> whether existing and proposed equipment, operation and maintenance are within design basis assumptions and acceptably safe, and cost-effective measures to reduce plant risk.

ATTACHMENT

CHALLENGE

Maintaining "objectivity" and "perspective".

MECHANISM

Probabilistic Risk Assessment.

OPERATIONAL USES OF PRA

Outage Safety Assessments

Quarterly Assessments of Plant Risk

Design Change Safety Assessments

Risk-Based Technical Specifications

ATTACHMENT

OUTAGE SAFETY ASSESSMENTS

WHAT

Risk-based FMEA of shutdown condition.

WHY

Identify and address high consequence / high probability accident scenarios during refueling outages.

ATTACHMENT

EXAMPLE

RHR SYSTEM WITH SINGLE (A-TRAIN) COMPONENT AVAILABILITY
FULL CAVITY AND SINGLE FAILURES
10 DAYS

Component	Single Failure Mode	Failure Prob/Day	Compen-satory Actions	Comp Action Failure Prob/Demand	Consequences of Failed Compen-satory Action	Current Consequence Prob/Period
RHR Pump 14A	Mechanical Failure RHR Pump	7.0E-4	Start Pump B or alt RHR w/in 20 hrs	2.0E-1	Boiling	1.4E-4

UNIT 1 CYCLE 11 OUTAGE
Safety Assessment Results

Residual Heat Removal System

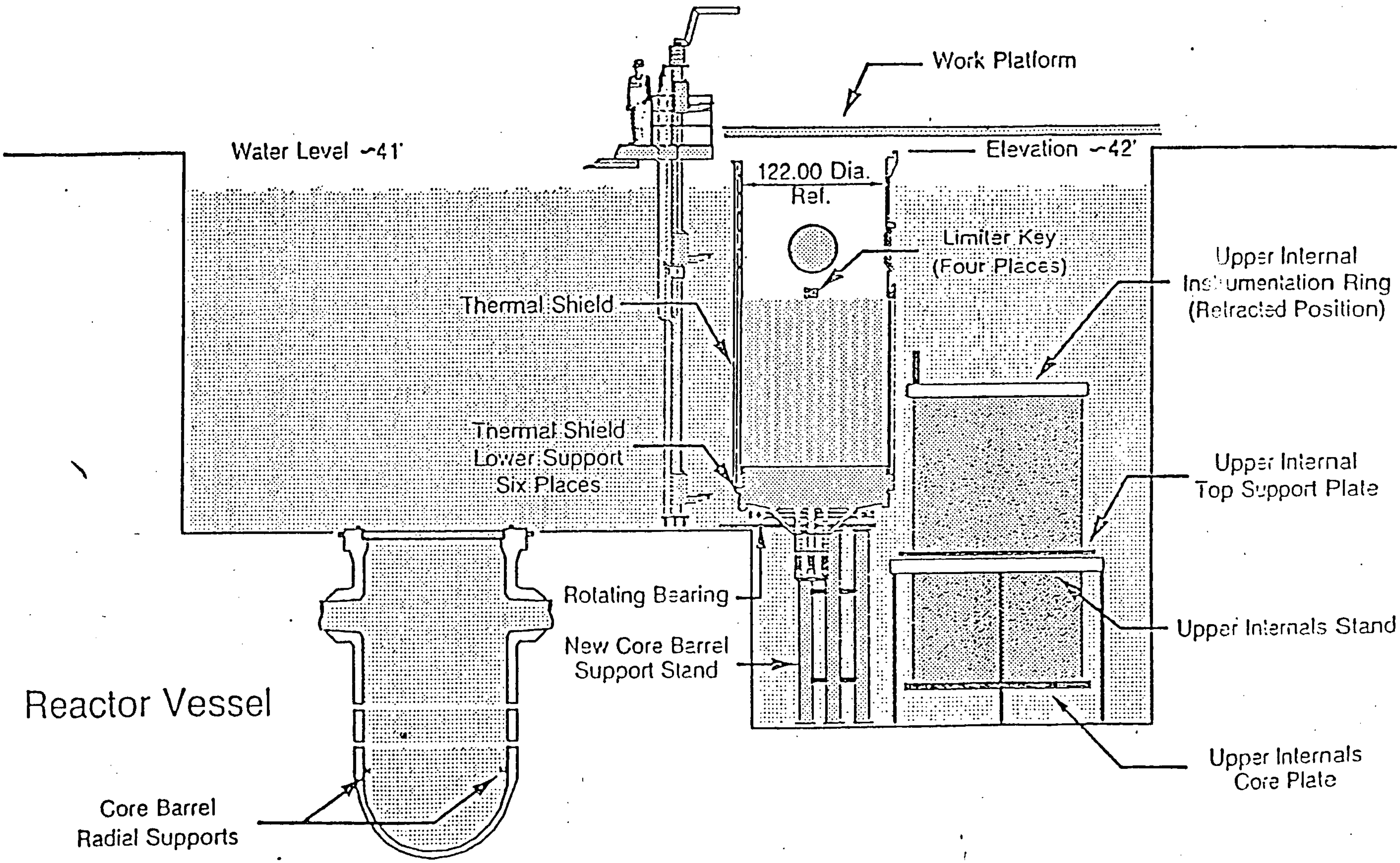
	Worst Case Probability	Modified Probability
o CORE MELT	5E-6	3E-6
o BOILING	4E-3	1E-5
o PERSONNEL EXPOSURE	2E-3	7E-4
o REACTIVITY CONTROL	3E-8	3E-8
o SPILLAGE	2E-4	2E-4

Reactor Vessel Cover Plate Operations

	Worst Case Probability	Modified Probability
o PERSONNEL EXPOSURE	3E-2	3E-3
o SPILLAGE	2E-2	2E-2

ATTACHMENT

San Onofre Unit 1 Reactor Cavity With Core Barrel and Upper Internals Removed



ATTACHMENT

QUARTERLY ASSESSMENT OF PLANT RISK

WHAT

Trend in core melt frequency as a function of safety equipment unavailabilities.

WHY

Identify and address equipment or activities which have significant risk consequences.

ATTACHMENT

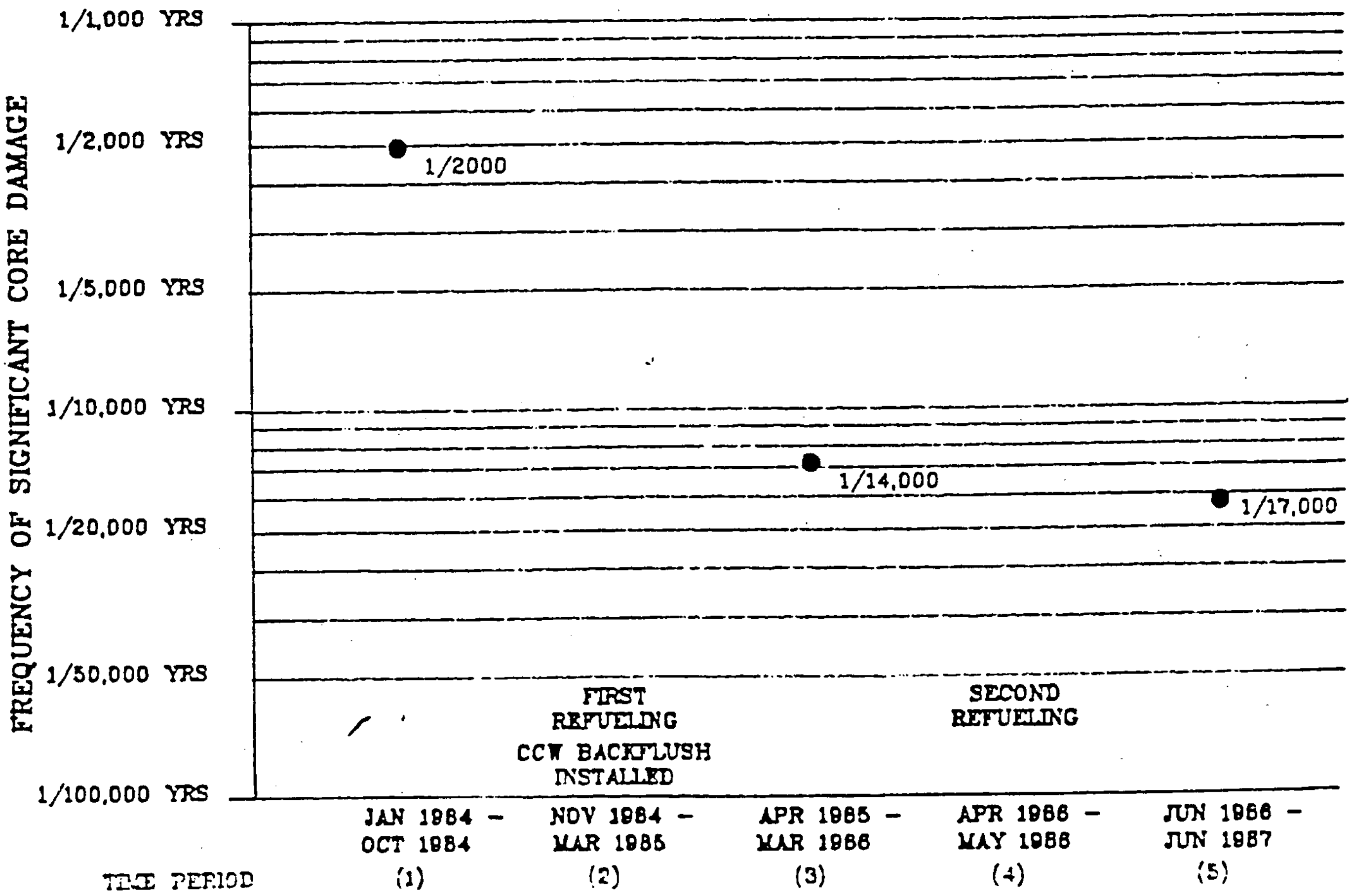
ATTACHMENT

ESTIMATED PROBABILISTIC RISK
UNIT 2

Shown below is the trend in the risk of core damage as calculated using probabilistic risk assessment (PRA) techniques. These results are based upon assessments of plant response to loss-of-coolant and loss-of-offsite-power events.

Time period (3) shows a significant reduction in estimated risk and reflects the impact of Design Change Package 6204 which provided the capability to backflush the CCW heat exchangers.

Period (5) saw a further reduction in risk due primarily to a continued increase in CCW heat exchanger availability. The minimum achievable risk of 1/24,000 years was not reached is this period due primarily to small unavailabilities of the CCW heat exchangers and HPSI trains resulting from maintenance activities.

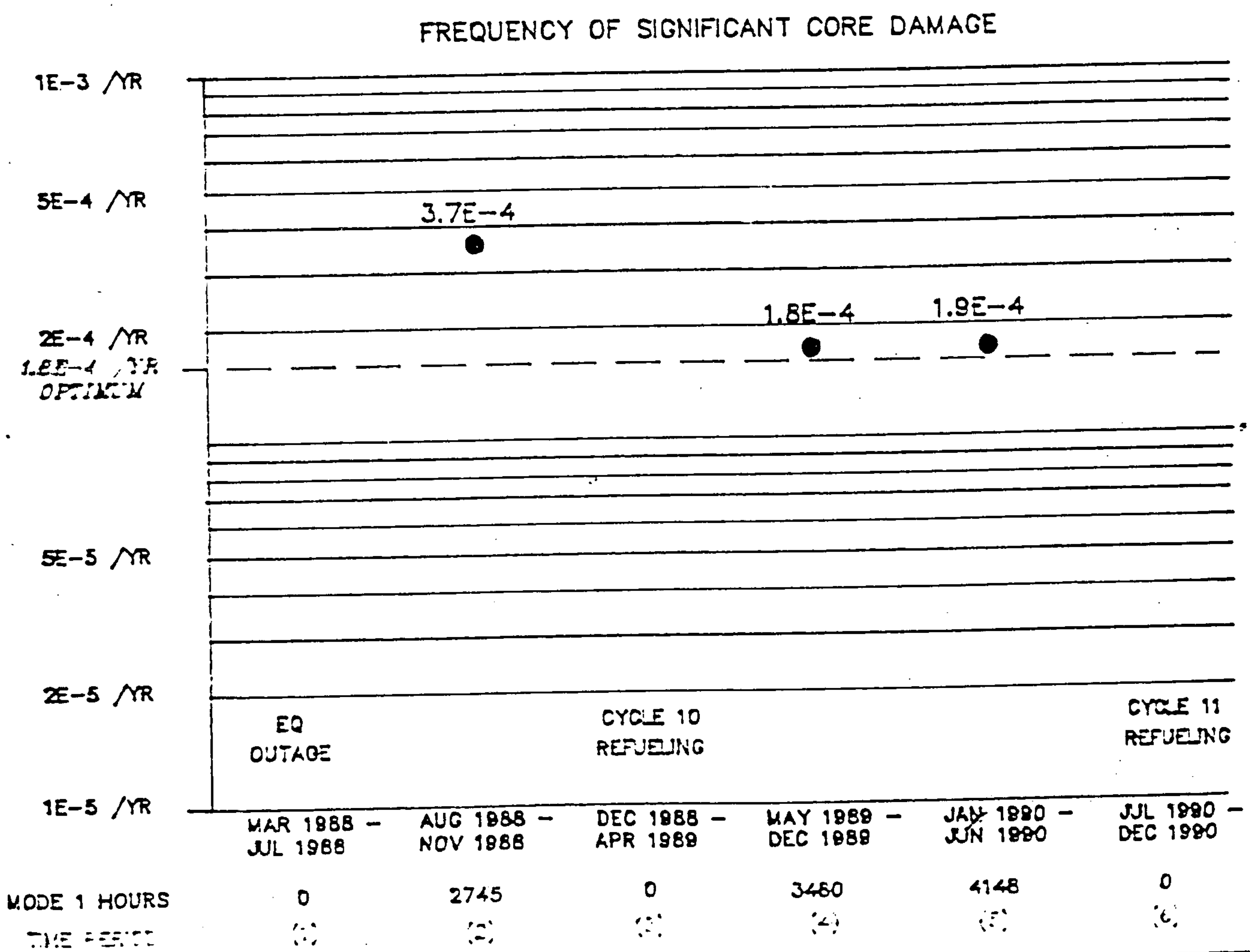


TREND IN ESTIMATED PROBABILISTIC RISK
UNIT 1

The trend in core damage risk appears in the chart below. These estimates derive from probabilistic models (fault trees and event trees) of plant accidents. The models portray the risks of loss-of-offsite power and loss-of-coolant accidents, and reflect actual equipment unavailabilities due to random failures, testing, or maintenance.

Unit 1 was shutdown for the entire fourth quarter of 1990.

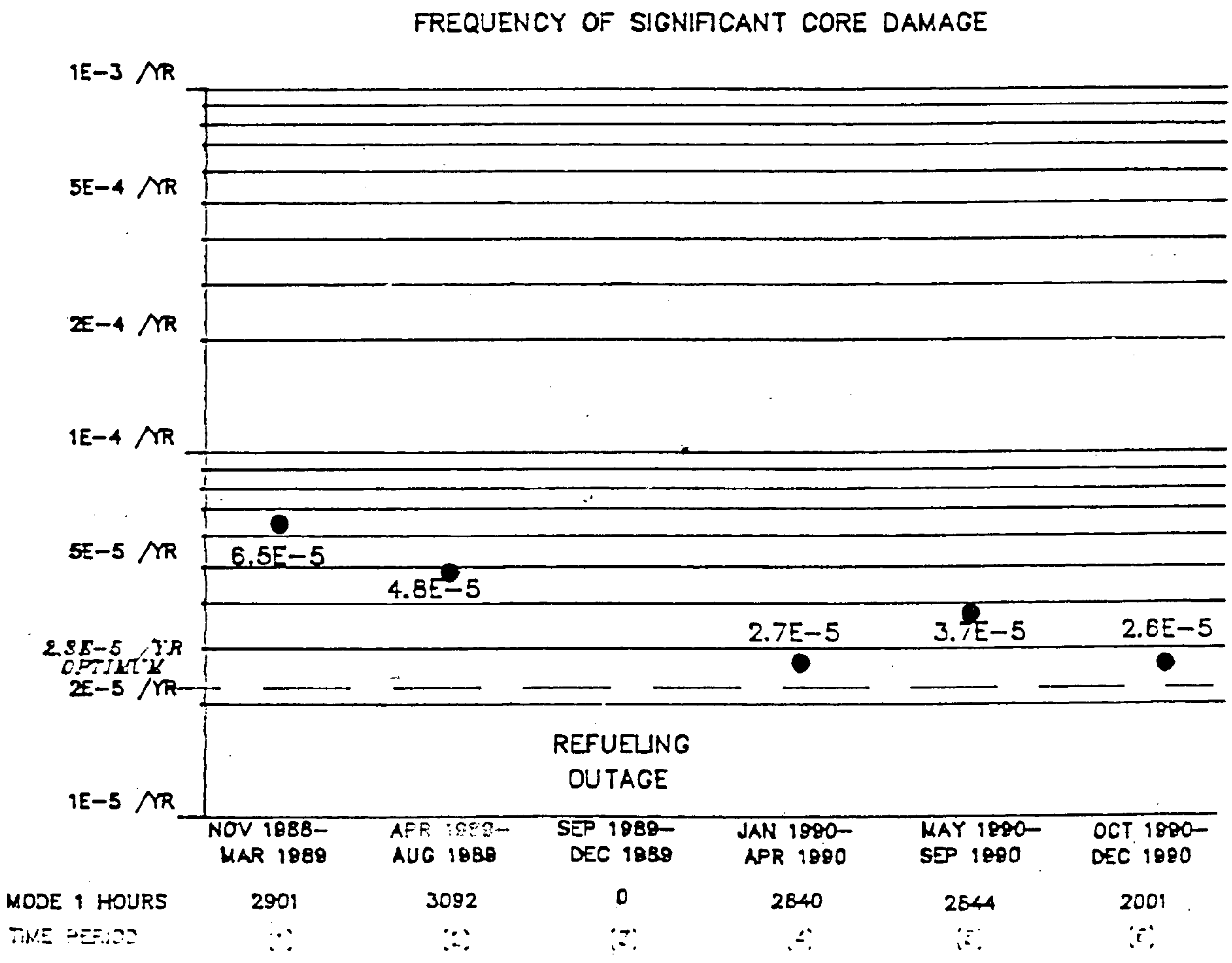
The reduction of risk following the Cycle-10 refueling was caused primarily by the addition of proceduralized actions following a loss-of-coolant accident to ensure isolation of the main feedwater pump miniflow to the condenser (SO1-1.0-10, "Reactor Trip or Safety Injection") and proceduralized actions following a loss of recirculation to establish alternative recirculation cooling (SO1-1.0-25, "Loss of Recirculation Flow").



TREND IN ESTIMATED PROBABILISTIC RISK
UNIT 2

The trend in core damage risk appears in the chart below. These estimates derive from probabilistic models (fault trees and event trees) of plant accidents. The models portray the risks of loss-of-offsite power and loss-of-coolant accidents, and reflect actual equipment unavailabilities due to random failures, testing, or maintenance.

One key factor influencing plant risk during Periods 5 and 6 was the status of the turbine-driven AFW pump P-140. The steam trap on the turbine steam supply line was inadvertently isolated from August 31 until October 21, 1990. This isolation was assumed to result in a pump overspeed trip, but with a high likelihood of recovery with operator action. Still, the conditional probability of core damage given a station blackout increased because P-140 is the only pumping system available in this condition.



ATTACHMENT

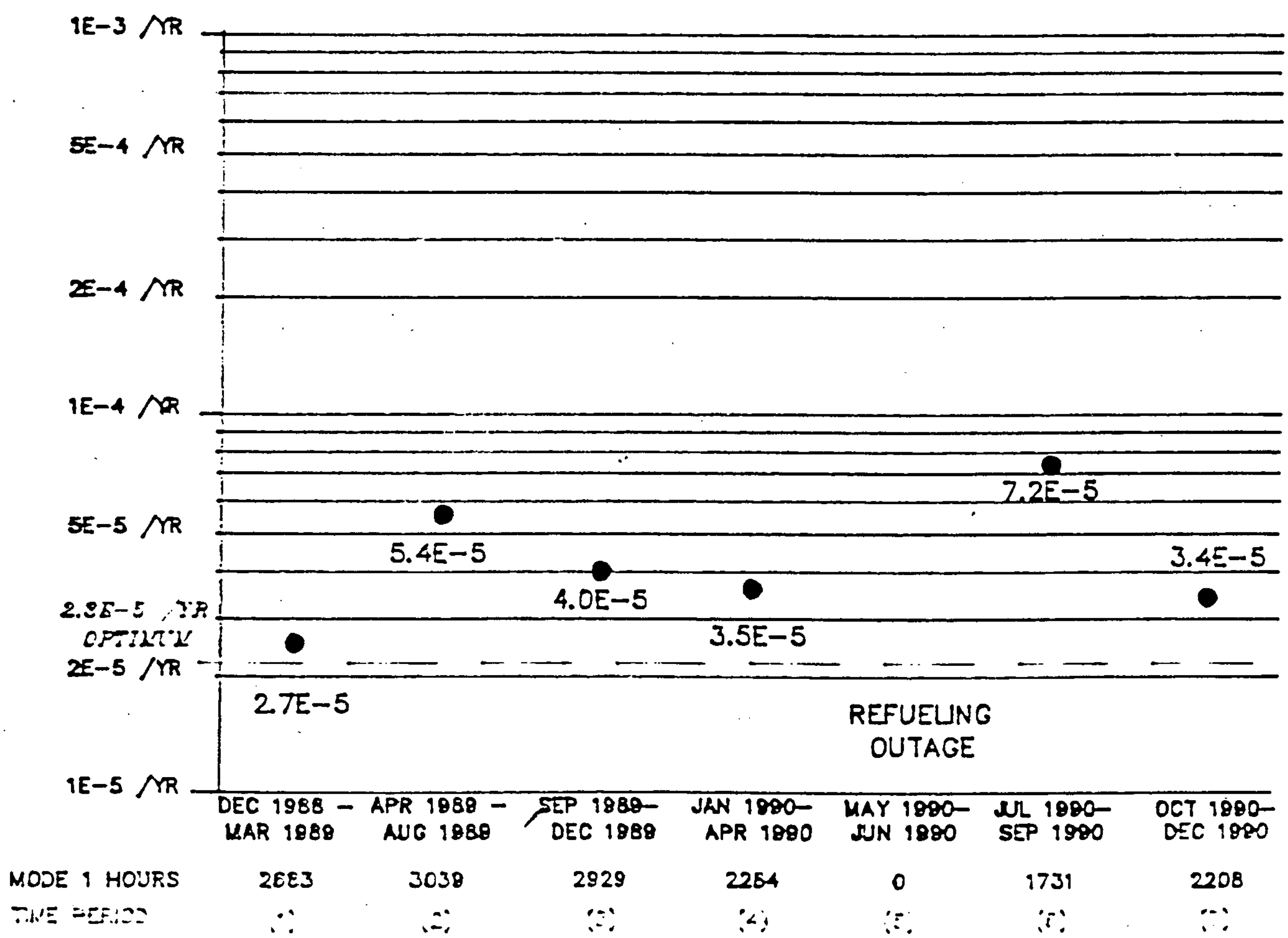
TREND IN ESTIMATED PROBABILISTIC RISK
UNIT 3

The trend in core damage risk appears in the chart below. These estimates derive from probabilistic models (fault trees and event trees) of plant accidents. The models portray the risk of loss-of-offsite power and loss-of-coolant accidents, and reflect actual equipment unavailabilities due to random failures, testing, or maintenance.

Principal contributors to the risk increase during Period 6 were two outages of CCW Heat Exchanger E-001 of about 60 hours each. One outage was due to preventive maintenance ("Boundary of the Week") and the other was required for cleaning. The inadvertent misalignment of ECCS valve 3HV-9302 was shown to have had a very small (2%) impact on core melt risk.

Factors influencing plant risk during Period 7 include preventive maintenance outages for LPSI pump P015, CCW Pump P025, and HPSI pump P018. An extended outage of HPSI pump P017 also ended in November.

FREQUENCY OF SIGNIFICANT CORE DAMAGE



DESIGN CHANGE EVALUATIONS

WHAT

PRAs performed for plant design changes to assess impact on system reliability and overall plant risk.

WHY

10CFR50.59 (NSAC/125)

ATTACHMENT

RISK-BASED TECHNICAL SPECIFICATIONS

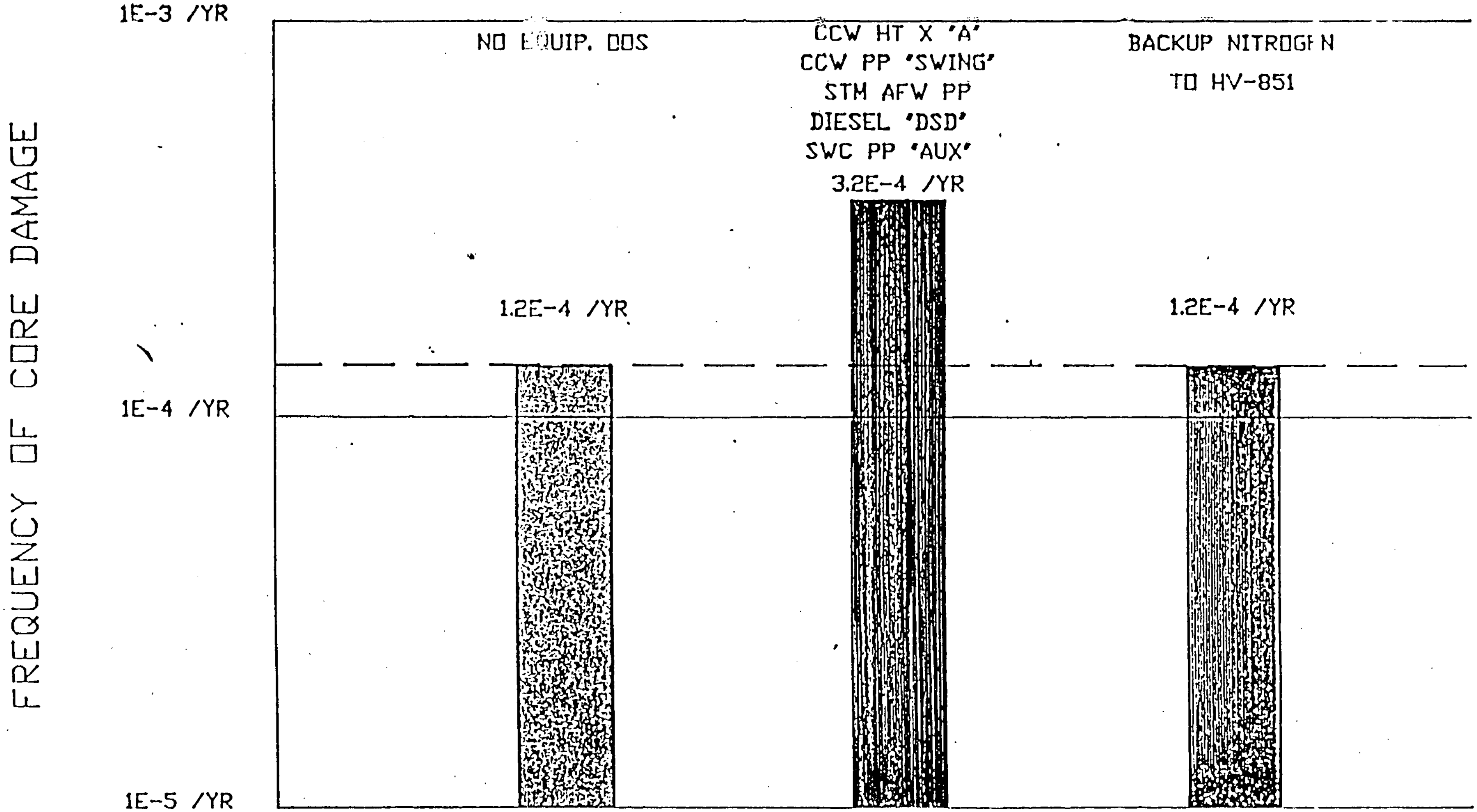
ATTACHMENT

WHAT

Potential pilot plant using Electric Essential Systems Status Monitor (ESSM) in cooperation with NRC's Risk-Based Technical Specifications Working Group.

WHY

Inability of current "deterministic" Technical Specifications to balance plant safety with plant operability.


FREQUENCY OF CORE DAMAGE
1E-3 /YR
1E-4 /YR
1E-5 /YR
NO EQUIP. OOS
1.2E-4 /YR
CCW HT X 'A'
CCW PP 'SWING'
STM AFW PP
DIESEL 'DSD'
SWC PP 'AUX'
3.2E-4 /YR
BACKUP NITROGEN
TO HV-851
1.2E-4 /YR

ATTACHMENT

CONCLUSION

APPLICATIONS OF PRA

- o Licensing
- o Engineering (offsite)
- o Engineering (onsite)
- o Operations
- o Maintenance
- o Training
- o Emergency Preparedness
- o Safety

ATTACHMENT