U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 89-04

Docket No. 50-352

License No. NPF-39

Licensee: Philadelphia Electric Company
Correspondence Control Desk
P.O. Box 7520
Philadelphia, Pa 19101

Facility Name: Limerick Generating Station, Unit 1

Inspection Period: January 24, 1989 - February 26, 1989

Inspectors: T. J. Kenny, Senior Resident Inspector
L. L. Scholl, Resident Inspector
T. F. Dragoun, Senior Radiation Specialist

Reviewed by: for James Linville 3/17/89
J. H. Williams, Project Engineer — Date

Approved by: James Linville 3/17/89
James Linville, Chief, Projects Section 2A — Date

Summary: Routine daytime (183 hours) and backshift/holiday (32 hours) inspections of Unit 1 by the resident inspectors consisting of (a) plant tours, (b) observations of maintenance and surveillance testing, (c) review of LERs and periodic reports, (d) review of operational events and (e) system walkdowns.

During this inspection period:

The licensee:

- Performed core off-loading (section 2.3.1).
- Investigated the cause of fuel failures (section 2.3.2).
- Submitted several licensee event reports (section 7.0).
- Performed a causal review of 1988 and 1989 LERs (section 7.1).
- Briefed NRR staff members on various issues (section 8.0).

The inspectors:

- Monitored core off-loading activities (section 2.3.1).
- Reviewed licensee actions relative to the February 15 Unusual Event (section 2.3).
- Observed licensee actions taken to identify the cause of fuel failures (section 2.32).
- Closed an open item dealing with the ALARA program (section 3.0).
- Reviewed LERs and special reports (sections 6.0 and 7.0).
- Observed PORC LER Review Meeting (sec.ion 7.1).

DETAILS

1.0 Persons Contacted

Within this report period, interviews and discussions were conducted with members of licensee management and staff as necessary to support inspection activity.

2.0 Operational Safety Verification

2.1 Documents Reviewed

- Selected Operators' Logs
- Shift Superintendent's Log
- Temporary Circuit Alteration Log
- Radioactive Waste Release Permits (liquid and gaseous)
- Selected Radiation Work Permits (RWP)
- Selected Chemistry Logs
- Selected Tagouts
- Health Physics Log

2.2 The inspector conducted routine entries in the protected areas of the plant, including the control room, reactor enclosure, fuel floor, and drywell (when access is possible). During the inspection, discussions were held with operators, technicians (HP & I&C), mechanics, security personnel, supervisors and plant management. The inspections were conducted in accordance with NRC Inspection Procedure 71707 and affirmed the licensee's commitments and compliance with 10 CFR, Technical Specifications, License Conditions and Administrative Procedures.

2.3 Inspector Comments/Findings

On January 24, the NRC was notified, via the Emergency Notification System (ENS), that the Main Control Room Emergency Fresh Air System (CREFAS) actuated as designed due to a "main control room radiation monitor isolation signal" on channels B and D. This occurred twice, during two different steps in the procedure when the signal generator was turned on during the equipment setup for a functional test on the radiation monitor.

Subsequent investigation determined that the voltage spike is a normal operating characteristic of the test equipment and the resultant radiation monitor response was expected by the technician performing the test. Additionally, the Chief Operator had been informed prior to releasing the equipment for testing that the Main Control Room isolation/CREFAS initiation would occur during performance of the test. This actuation of an Engineered Safety Feature (ESF) was determined to be nonreportable. A followup call was made on February 13 to amend the original notification to the NRC Duty Officer.

On February 4, a refuel floor isolation occurred due to low ventilation differential pressure. The standby gas treatment system started and operated properly to reestablish the proper negative differential pressure. The loss of differential pressure occurred when auxiliary steam was lost to the ventilation system heating coils and the exhaust fans could not remove the added volume of air which results when the cold air expands as it mixes with the warmer air inside the enclosure.

On February 8, the licensee commenced off-loading the reactor fuel. Fuel bundle sipping to determine which bundles contain leaking fuel pins in addition to visual inspections and bundle reconstitution also began. These activities identified that, in addition to finding failed fuel pins in the initial core fuel, leaks were found in the reload one fuel which was exposed to only one cycle of operation. All of the reload one fuel was fabricated with heat-treated cladding and was previously believed not to be susceptible to crud induced localized corrosion which appears to have been the failure mechanism in the leading bundles. In addition to the failed pins, various other fuel pins exhibited accelerated corrosion. See Attachment C for a summary of inspection results. The licensee is working with General Electric (GE), the fuel manufacturer, in an effort to determine the root cause for the cycle one fuel accelerated corrosion and associated failures. The results of this investigation, reload plans and the cycle 3 operating strategy were presented to NRC management in a meeting on March 15, 1989.

On February 15, the licensee reported (via the ENS) that an engineering review had determined that the feedwater maintenance isolation valve, HV-41-1F011B, may be unavailable in the event of a design basis fire. Inadvertent closure of the valve due to fire damage of cabling would block the flow path for Reactor Core Isolation Cooling (RCIC) system injection to the reactor vessel. RCIC is required to support two of the safe shutdown methods described in the Fire Protection Evaluation Report (FPER). The inspector will review the corrective actions to be taken in a future report upon receipt of the licensee event report (LER).

On February 15, an Unusual Event was declared at 10:07 a.m. at Limerick Unit 1. A contractor working in the drywell area collapsed due to apparent heat exhaustion. The employee was wearing plastic protective clothing. The Goodwill Ambulance Company transported the injured employee and a health physics technician to the Pottstown Memorial Medical Center. The Unusual Event was declared since the injured worker was not declared to be free of radioactive contamination prior to transport. The Unusual Event was terminated at 11:35 a.m. when it was determined the injured worker had not been radioactively contaminated.

On February 16, a refuel floor isolation occurred and the standby gas treatment system started due to a low differential pressure signal caused by a loss of heating steam to the ventilation system. Problems were encountered with both trains of the standby gas treatment system and fuel handling operations were suspended in accordance with technical specification requirement 3.6.5.1.2. Repairs were made to a temperature transmitter in the 'A' train and the low flow switch on the 'B' train to return both trains to an operable condition. The NRC duty officer was notified via the ENS and the resident inspectors performed a followup inspection which verified the licensed action.

During the inservice inspection (ISI) the licensee identified a defect in the recirculation system piping. The defect is located at one of the jet pump inlet nozzle welds. To date all 10 jet pump nozzles have been inspected with only one defect identified. Altogether 17 of 23 planned ISI inspections have been completed. The licensee briefed the NRC staff at a meeting held at NRC Headquarters. Region I, Division of Reactor Safety, has been informed and will review the licensee actions in a future inspection.

On February 22, several technical specification surveillance requirements were not accomplished due to the failure to perform the required instrument channel checks. The checks were missed for two shifts when the performance of ST-6-107-591-1, Daily Surveillance Log-OpCons 4 and 5, was suspended upon completion of the core offload as directed by procedure GP 6.1, Shutdown Operations-Refueling, Core Alteration and Core Off-Loading. The direction provided in GP 6.1 was incorrect in that various channel checks were still required after exiting OpCon 5 since irradiated fuel was being handled in the spent fuel pool.

The missed channel checks were performed satisfactorily upon discovery of the error and procedure revisions are being prepared to prevent recurrence. In that this violation of plant technical specifications was licensee identified, of minor safety significance, and was promptly corrected, a notice of violation will not be issued per the provisions of paragraph G of 10CFR50, Part 2, Appendix C (50-352/89-04-01).

On February 28, while performing reconstitution of the fuel bundles, a fuel pin was dropped during the transfer of the bundle to the inspection area. In anticipation of a radioactive release that may have accompanied a broken pin, the operators manually isolated the refueling floor HVAC. All parts of the system functioned normally. However, the pin was not damaged. The pin was returned to the fuel bundle after inspection. This pin will not be returned to the reactor. The HVAC was returned to normal following recovery of the pin.

The licensee is conducting an investigation to determine the cause for dropping the pin. Initial review indicates the cause was due to human error and not a result of any pin handling tool problems. The inspectors will review the results of the investigation.

2.3.1 Defueling Operations

The inspectors observed the core off-loading operations and maintenance of plant shutdown conditions. The following procedures controlling these activities were also reviewed:

GP-6.1, Shutdown Operations-Refueling, Core Alteration and Core Off-loading

GP-6.2, Shutdown Operations-Shutdown Condition Technical Specification Actions

FH-105, Core Component Movement-Core Transfers

S53.0.B, Alternate Level Control and Cleanup During Refueling Operations

Several equipment problems were encountered with the fuel handling bridge and were corrected in accordance with station procedures. The fuel handling was controlled well and was completed without incident.

The inspector identified a concern relative to the use of temporary submersible pumps in the reactor cavity. Two submersible pumps are lowered into the reactor cavity and take a suction just above the reactor vessel flange. The discharge of the pumps was directed into the fuel pool cooling and cleanup (FPCC) system skimmer surge tank inlets. This promotes mixing and cleanup of the water in the cavity thus improving water clarity so that fuel handling can be observed more readily.

Section 9.1 of the Final Safety Analysis Report (FSAR) described the design of the spent fuel pool and the fuel pool cooling and cleanup system. The basis of the design is to prevent inadvertent draining or siphoning of the spent fuel pool. Also the low level alarm on the skimmer surge is one indication of a loss of inventory from the FPCC system.

With the temporary pumps discharging to the skimmer surge tank inlet, the potential exists for a loss of inventory from the reactor cavity (and the spent fuel pool when the gates are removed) in the event of a leak in the FPCC system. Also the pump makeup to the skimmer surge tank may prevent the activation of the low level alarm during a loss of water inventory in the FPCC system.

The licensee has reviewed the temporary pump installation and suspended the use of pumps until the discharge hoses can be secured in a position as to not discharge directly to the surge tank inlet. This will allow the pump to be used to improve water clarity without the potential for resulting in a low level condition in the reactor cavity or spent fuel pool.

The inspector also noted that procedure GP-6.1 utilized the Temporary Circuit Alteration (TCA) procedure to install and remove several jumpers required during the procedure. The inspector questioned why the installation and removal of jumpers could not be totally controlled within the GP-6.1 procedure which would aid in the TCA reduction program. Licensee progress in reducing the number of temporary circuit alterations will be reviewed in subsequent inspections. Unresolved item 87-19-03 documented this concern and remains open.

2.3.2 Fuel Cladding Degradation Meeting

On February 15, the plant manager convened a meeting of site and corporate managers and technical personnel as well as representatives from GE fuels division to review factors that may have contributed to the fuel cladding degradation. A manager familiar with the Kepner-Tragoe (K-T) method of problem analysis was the facilitator during the discussion. The inspector observed the meeting to be an open and thorough exchange of information relevant to the fuel cladding problems. Although a root cause for the fuel cladding failures could not be determined as a result of the meeting, several task forces were formed to further review several areas such as the effects of chemistry control, the fuel fabrication process, and thermomechanical nuclear design.

The results of the task group reviews including root cause determination and recommendations to prevent fuel failures during the next operating cycle will be presented to NRC Region I and NRR management prior to the plant restart.

3.0 Update of Open Items

a. (Closed) 50-352/87-04-01. Weakness in the ALARA program includes poorly defined responsibilities and relationships, lack of top management support and failure to follow the Corporate Radiation Protection Manual. The licensee has completed the following actions:

- A Limerick ALARA Council was instituted consisting of top site management in accordance with new procedure A-103.2. The first quarterly meeting was held in July 1988.

- A corporate level Executive ALARA Council was formed and held its first bi-monthly meeting in June 1988. Minutes of these meetings indicate that substantial issues are addressed and support by senior management was excellent.

- The responsibility for ALARA reviews of Plant Design Modifications is assigned to the Corporate Nuclear Engineering Department with assistance from the Corporate Radiological Engineering Group. This was accomplished by policy NGAP-XXR07 issued in August 1988.

- A concise but well written ALARA manual providing guidance for the conduct of the program was issued in April 1988.

- A review by the inspector found the policies to be excellent and program implementation to well under way. This item is closed.

4.0 Surveillance/Special Test Observations (61726, 64704)

During this inspection period, the inspector reviewed in-progress surveillance testing as well as completed surveillance packages. The inspector verified that surveillances were performed in accordance with licensee approved procedures and NRC regulations. The inspector also verified that instruments used were within calibration tolerances and that qualified technicians performed the surveillances.

The following surveillances were reviewed:

ST-6-107-591-1	Daily Surveillance Log - OpCons 4 and 5
ST-4-0934-1	Division IV 125VDC Safeguards Battery Test

5.0 Maintenance Observations (62703)

The inspector reviewed the following safety related maintenance activities to verify that repairs were made in accordance with approved procedures, and in compliance with NRC regulations and recognized codes and standards. The inspector also verified that the replacement parts and quality control utilized on the repairs were in compliance with the licensee's QA program.

- 8900236 1D RHR Pump Seal Cooler Cleaning
- 8900992 Wide Range Accident Monitor Cable Separation Correction

5.1 Maintenance Team Inspection

A two week maintenance team inspection was conducted from January 30 through February 10. The team was composed of specialist inspectors from Region I, King of Prussia and the Office of Nuclear Reactor Regulation, Washington, D.C. The results of the inspection will be documented in inspection report 50-352/88-80.

6.0 Review of Periodic and Special Reports (90713)

Upon receipt, the inspector reviewed periodic and special reports. The review included the following: inclusion of information required by the NRC; test results and/or supporting information consistent with design predictions and performance specifications; planned corrective action for resolution of problems, and reportability and validity of report information. The following periodic report was reviewed:

- Unit 1 Monthly Operating Report - January 1989
- Safeguards Event Report No. 89-S01

Prior to entering the Protected Area, a contract employee discovered three .22 caliber bullets while emptying his pockets during the access search process. The contract employee brought these items to the attention of security personnel. A subsequent pat down search of the contract employee revealed a small amount of Marijuana in his possession. The employee's access was terminated and a local law enforcement agency was notified. The NRC was notified via the ENS.

The inspector had no questions regarding this report.

7.0 Licensee Event Report Followup (90712, 92700)

The inspector reviewed the following LERs to determine that reportability requirements were fulfilled, that immediate corrective action was taken, and that corrective action to prevent recurrence was accomplished in accordance with technical specifications.

88-030 (Revision 2)

This LER addressed the fact that three ventilation fire damper access doors located in the Reactor Enclosure Building had not been structurally upgraded. This was discussed in previous monthly report (88-26). The current revision addresses the licensee's analysis of the consequences of a steam line break in the vicinity of these access doors. The licensee has concluded, through engineering judgment and worst case analysis, that most of the safety related equipment located in the affected areas would not have experienced environments that would have exceeded their environmental qualifications. The licensee has decided, in light of the findings and the prohibitive cost involved in a detailed analysis, not to perform a detailed analysis. New doors have been installed by the licensee and will be operational for plant startup. The resident inspector had no further questions regarding this LER.

88-042

This LER addresses the cable separation problem that was identified by the licensee in December 1988. The sequence of events leading up to the problem is delineated in inspection report 88-26. The licensee plans to utilize this LER number to collect all of the data regarding the cable separation findings. The inspector has received and reviewed up to revision three. The inspector will close out this LER when all of the problems have been identified. The licensee is continuing to evaluate the cable separation.

89-001

This LER describes a missed surveillance test. The licensee identified the late surveillance test on January 4, 1989 (ST-6-012-206-0, Schuylkill River Makeup to Spray Pond ISI Test) which had been overdue since September 10, 1988. The reason for the missed surveillance was identified as an improper entry into the scheduling computer. The licensee has modified the computer programs to detect and list on a periodic report overdue tests. The surveillance was subsequently successfully performed. The inspector had no further questions regarding this LER.

89-002

This LER describes additional Appendix R fire protection problems (also described in report 88-26) that are a result of the licensee's ongoing review into their Appendix R program. These problems are being reported to the NRC through the LER system and will be collectively resolved with a supplement to LER 88-031. The licensee is taking compensatory measures as these problems are identified. The inspector had no further questions at this time.

89-003

This LER describes the event and licensee actions as a result of a Reactor Enclosure Secondary Containment Isolation. Personnel had improperly propped open an exhaust plenum access door during maintenance in the area. This open door caused a differential pressure to result, which caused the isolation. The licensee has posted signs on this door and other similar doors to warn personnel against opening these doors without the knowledge of the shift supervisor. The inspector had no further questions regarding this event.

89-004

This LER describes improper undervoltage protection on the 4160 V Buses. This event was described in inspection report 88-26. After review of the LER and discussions with the licensee, the inspector learned that the modification to the relays will be completed by the end of the refueling outage and a technical specification change will be submitted to NRR. The inspector had no further questions at this time.

89-005

This LER is a special report delineating an iodine spike that was detected during the unit shutdown on January 11, 1989. After the manual scram was initiated, Reactor Coolant Dose Equivalent Iodine-131 (DEI) exceeded the technical specification limit of 0.2 uci/gram and remained above this level for 14 hours and 30 minutes, requiring the special report. This was not unexpected due to the fuel pin leaks. The inspector had no further questions regarding this LER.

89-006

This LER describes an Engineered Safety Feature Actuation of the Residual Heat Removal System which is described in inspection report 88-26. After review of the LER the inspector had no further questions.

89-007

This LER describes a Control Room Isolation which is described in inspection report 88-26. After review of the LER the inspector had no further questions.

89-008 and 89-009

These LERs describe additional cable separation problems identified during the licensee's evaluation of cable separation and Appendix R. The systems affected were secondary containment isolation, standby gas treatment, reactor enclosure recirculation and the nuclear steam supply shutoff systems. These particular separation problems were corrected by sleeving the appropriate cables and are now corrected. The inspector had no further questions at this time.

7.1 Plant Operations Review Committee

On January 26, the inspector attended a Plant Operations Review Committee (PORC) meeting which was held to review causal factors for 1988 LERs and the 1989 LERs up to the time of the meeting. The meeting was also attended by the Chairman of the Nuclear Review Board, Mr. E. Kistner.

One area of focus was those LERs associated with control room isolations. The predominant cause of these isolations was the wetting of the chlorine detectors during rainstorms which has been remedied by relocation of the probes. The PORC concluded that approximately two isolations per year could be expected to occur due to other detection system failures, i.e. toxic gas and radiation monitors. Based on the concern expressed by Mr. Kistner, that any isolation results in operators having to don self contained breathing apparatus, thus making plant operation difficult, PORC commitments were issued to members of the plant staff to review the toxic gas analyzer design and to present an analysis of radiation monitor detector problems in an effort to determine if the probability of future isolations could be further minimized.

In the area of personnel error LERs, inattention to detail was the predominant cause. There was also an increased rate of personnel error in the 1989 LERs. PORC has conveyed the results of this review to plant management and is also requesting that the Independent Safety Engineering Group (ISEG) do an evaluation of the 1989 personnel error events using the human performance evaluation system (HPES) to gain additional insight to the root causes.

8.0 NRC Site Visit

On February 23, NRC staff members from NRR visited the site to be briefed by the licensee on the progress being made in the reconstitution of the damaged fuel. The staff toured the site including the refueling floor and attended a meeting where the licensee delineated their program, including plans to present to the NRC on March 15, 1989 the analysis of why the fuel failed and a program for preventing its recurrence during the next fuel cycle.

During the licensee's presentation (agenda attached in attachment A) the licensee summarized the progress of the reconstitution effort (the results are in attachment B) that had been completed as of February 23, 1989.

The licensee also briefed the NRC about their proposed action plan for condensate filter demineralizers which will be utilized during the cycle III operation (attachment C).

At the conclusion of the site meeting, the licensee proposed an additional meeting regarding the recirculation system piping jet pump nozzle defect (see section 2.3 for further details). This meeting was scheduled for March 7, 1989 in NRC headquarters.

9.0 Region I Concern Regarding Instrument Air Systems

In response to two instrument air supply problems identified at Pilgrim Power Station, the licensee assessed whether similar problems exist at Limerick.

The first deficiency was related to the nonsafety related air supply which maintains the inflatable inner seals on the truck bay air lock. Limerick does not have a pneumatic seal.

The second deficiency was related to the air supply to the accumulators for the torus to the reactor building vacuum breaker block valves. These valves are for a Mark I containment. Limerick which has a Mark II containment, has no such valves.

The inspector had no further questions regarding this concern.

10.0 Region I Temporary Instruction 87-06

The inspector reviewed Region I Temporary Instruction 87-06 regarding lubrication of General Motors Electro-Motive Division diesel generators with Ingersoll Rand air start motors. The Limerick Unit 1 diesel generators are manufactured by Colt-Fairbanks Morse and do not utilize Ingersoll Rand air start motors. Rather, an air distributor delivers starting air to the engine cylinders in the firing order. This Temporary Instruction is not applicable.

11.0 Assurance of Quality

During this period the licensee has demonstrated assurance of quality in that:

- plant management was very involved in efforts to determine the root cause of the fuel cladding degradation (section 2.3).

- the PORC performed a detailed causal analysis of 1988 LERs and an early review of 1989 LERs to determine whether a negative trend was evident in the number of personnel error LERs and what actions could be taken to reverse this trend (section 7.1).

- controls for an orderly operability verification for ESW and RHR SW systems have been established (section 4.1).

An area where assurance of quality was not demonstrated was in the installation of temporary pumps in the reactor cavity in a manner which was not consistent with the design objectives stated in the FSAR.

12.0 Exit Interview

The NRC resident inspectors discussed the issues in this report throughout the inspection period, and summarized the findings at an exit meeting held with the Plant Manager, Limerick Generating Station, on February 28, 1989. No written inspection material was provided to licensee representatives during the inspection period.

Attachment A

NRC Site Visit
February 23, 1989

NRC Personnel:

Roger W. Woodruff - NRR
Frank J. Witt - NRR
Angelo Munno - NRR
Richard J. Clark - NRR
James C. Linville - Region I
William V. Johnston - Region I
E. Harold Gray - Region I
Larry L. Scholl - Resident Inspector
Thomas J. Kenny - Senior Resident Inspector

Schedule

9:00 a.m. NRR Personnel arrive at VIP parking lot.

9:15 a.m. NRC Personnel arrive at Admin. Bldg. Lobby. Woodruff, Witt and Munno processed as visitors.

9:30 a.m. Leave for Refuel Floor.

11:00 a.m. NRC and PECo Personnel assemble in Fourth Floor Conference Room.

PECo Presentation:

- Fuel Inspection - M. Gallagher
  - Inspection Results
  - Video Tap
- Reload Projections - L. Rubino
- Efforts to reduce Feedwater Copper - R. Scholz
  G. Barley

1:30 p.m. Return to Fourth Floor Conference Room

- NRR Personnel depart Unit 1 for Unit 2 tour
- Service water discussion - R. Scholz

2:30 p.m. - Jet Pump Riser Nozzle Repair - William Texter

LIMERICK 1 FUEL INSPECTION AND RECONSTITUTION SUMMARY

SIPPING RESULTS

- 296 BUNDLES SIPPED
- 5 INITIAL CORE LEAKERS
- 13 RELOAD 1 LEAKERS

INSPECTION RESULTS

- INITIAL CORE
  - 26 BUNDLES EXAMINED
  - 12% OF PINS ARE VISUAL 4,5,6
  - ALL INITIAL CORE BUNDLES BEING RECONSTITUTED HAVE ALL PINS INSPECTED
- RELOAD 1
  - 6 INSPECTED
  - 2 LOW EXPOSURE BUNDLES HAD VISUAL 1,2 PINS
  - 4 HIGHER EXPOSURE CONTAINED SIGNIFICANT NUMBERS OF VISUAL 4,5,6 PINS

LIMERICK 1 FUEL INSPECTION AND RECONSTITUTION SUMMARY (cont.)

RECONSTITUTION UPDATE

- 27 ~~24~~ INITIAL CORE 2.48 e BUNDLES RECONSTITUTED
- NON HEAT TREATED MATERIAL LEFT IN IF VISUAL 1 OR 2
- RECONSTITUTED BUNDLES WILL CONTAIN ONLY VISUAL 1 OR 2 MATERIAL
- SOME BUNDLES HAVE REQUIRED A LARGER THAN EXPECTED NUMBER OF RODS TO BE REPLACED
- NEED AN ALTERNATE PLAN IN CASE INSUFFICIENT INITIAL CORE BUNDLES CAN BE RECONSTITUTED

IMPACT ON CORE DESIGN

ORIGINAL PLAN
(PRIOR TO DISCOVERY OF RELOAD 1 PROBLEM)

- LOADING PATTERN
  - 268 RELOAD 1 BUNDLES
  - 148 BUNDLES FROM THE POOL (PREINSPECTED)
  - 84 RECONSTITUTED INITIAL CORE BUNDLES
  - 40 0.94 e LGS 2 FRESH
  - 224 RELOAD 2 BUNDLES

- INITIAL CORE BUNDLE RECONSTITUTION REPLACED NON HEAT TREATED PINS ONLY

- INTERIOR OF CORE FULLY HEAT TREATED

- POOL BUNDLES RESTRICTED TO LOW POWER REGION OF THE CORE

IMPACT ON CORE DESIGN (cont.)

NEW BASE PLAN
(ELIMINATION OF RELOAD 1)

- LOADING PATTERN
  - 268 LGS 2 FRESH (TO REPLACE RELOAD 1)
  - 148 BUNDLES FROM THE POOL
  - 84 RECONSTITUTED INITIAL CORE BUNDLES
  - 40 0.94 e LGS 2 FRESH
  - 224 RELOAD 2 BUNDLES

- INITIAL CORE BUNDLE RECONSTITUTION ALLOWS NON HEAT TREATED VISUAL 1,2 PINS AS NOTED PREVIOUSLY

- 268 LGS 2 FRESH CONSISTS OF A MIX OF 2.48 e AND 1.63 e BUNDLES
  - SPLIT TO BE DETERMINED

Current plan

IMPACT ON CORE DESIGN (cont.)

ALTERNATE PLAN
(IN CASE OF INSUFFICIENT RECONSTITUTION)

- LOADING PATTERN
  - 268 LGS 2 FRESH
  - 148 BUNDLES FROM THE POOL
  - 48 RECONSTITUTED INITIAL CORE BUNDLES
  - 40 0.94 e LGS 2 FRESH
  - 224 RELOAD 2 BUNDLES
  - 36 LOW EXPOSURE RELOAD 1

- REQUIRES INSPECTION PLAN TO CLEAR 36 RELOAD 1 BUNDLES

- RELOAD 1 FUEL WOULD BE USED IN LOW POWER LOCATIONS ONLY

- SAME RECONSTITUTION CRITERIA FOR INITIAL CORE FUEL AS NEW BASE PLAN

- ~~40~~ 36 POOL BUNDLES MUST BE MOVED TO THE CORE INTERIOR

- SOME REINSPECTION OF THE ~~40~~ 36 POOL BUNDLES REQUIRED

OPERATION OF PLANNED CORE DESIGN

NEW BASE PLAN

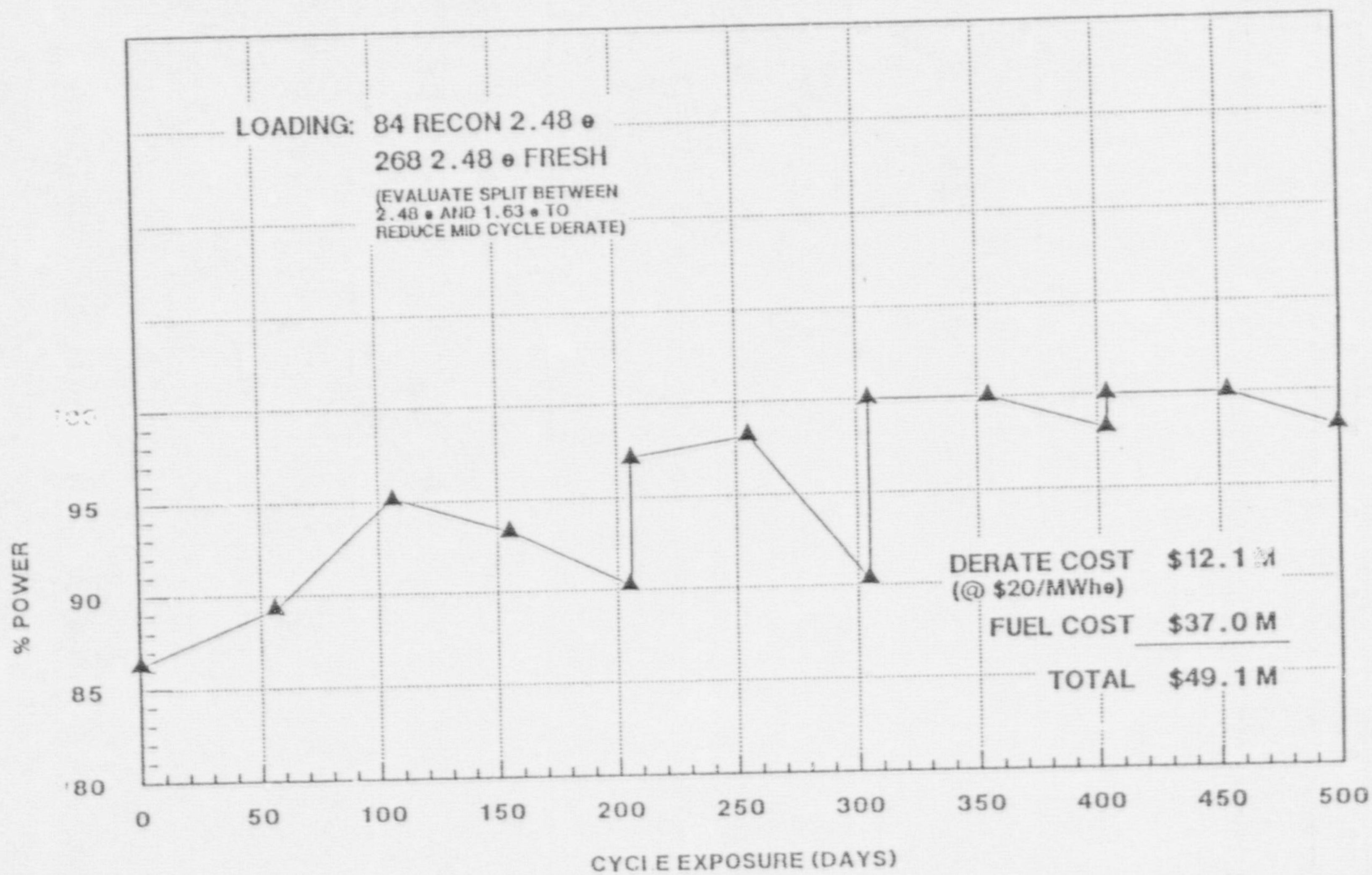
- DERATE TO 85% POWER AT STARTUP DUE TO LOW HOT EXCESS REACTIVITY
- DERATE OF 5-10% AT MID CYCLE DUE TO THERMAL LIMITS
- THERMAL LIMITS CAN BE IMPROVED BY USING A LIMITED AMOUNT OF FRESH 1.63 e, BUT A DERATE TO 75% WILL OCCUR AT STARTUP
- FURTHER EVALUATIONS TO OPTIMIZE THE SPLIT ARE IN PROGRESS

ALTERNATE PLAN

- DERATE TO 80% POWER AT STARTUP DUE TO LOW HOT EXCESS REACTIVITY
- SLIGHT DERATE (≈5%) AT MID CYCLE DUE TO THERMAL LIMITS
- RATED POWER OPERATION DURING SECOND HALF OF CYCLE

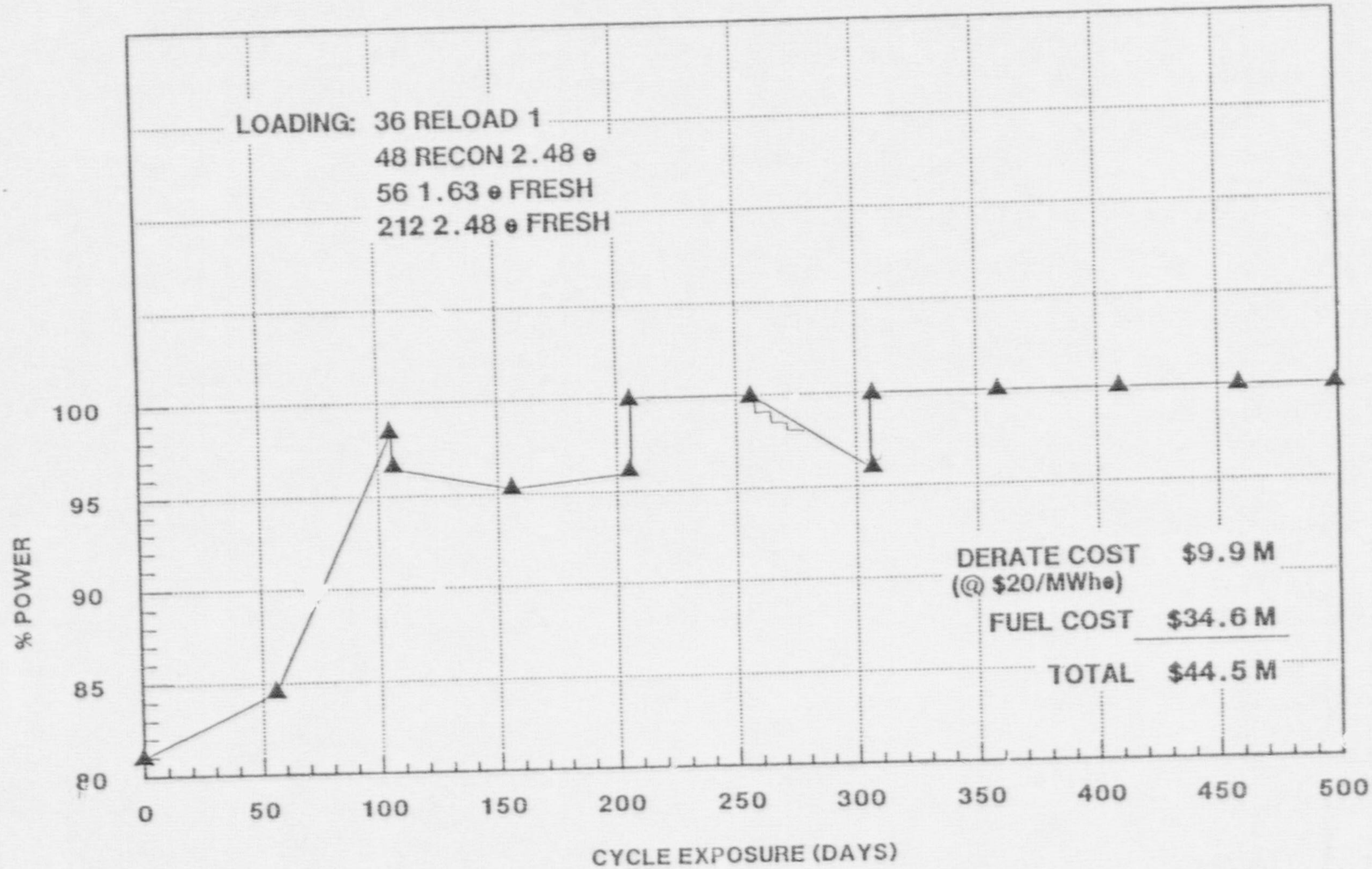
DERATE VERSUS TIME

(ESTIMATED)



DERATE VERSUS TIME

(ESTIMATED)



REPLACEMENT OF LIMERICK 2 BUNDLES

- GENERAL ELECTRIC COMMITTED TO BUILD AND DELIVER REPLACEMENTS TO SUPPORT EARLY JUNE FUEL LOAD

- DOE NOTIFICATION TO BE BASED ON 268 2.48 e BUNDLES

- SOME 1.63 e BUNDLES ARE USED IN BOTH LIMERICK 1 LOADING PLANS

- COST OF EXTRA SWUs WILL BE RECOVERED IN A VARIANCE SETTLEMENT

ROOT CAUSE DETERMINATION OF RELOAD 1 FAILURES

- JOINT PE/GE STUDY TEAMS ESTABLISHED

- AREAS UNDER REVIEW
  - TUBING MATERIAL MANUFACTURING PROCESS
  - ENVIRONMENT (WATER CHEMISTRY)
  - THERMAL/MECHANICAL/NUCLEAR DESIGN REVIEW

- A FOURTH TEAM WILL EXAMINE FEASIBILITY OF FUTURE RECONSTITUTION AND USE OF RELOAD 1 FUEL

PROPOSED ACTION PLAN

FOR CONDENSATE FILTER/DEMINS

DURING CYCLE III

LGS UNIT 1

OBJECTIVE: Feedwater Copper $\leq$0.3 ppb

INITIAL CONDITIONS

1. Use fresh resin for start-up.

2. Load 6 vessels with pre-mixed anion/cation.

3. Load vessels 4 & 8 with Epicarb carboxylic acid resin.

Why not start Carboxylic in all 8 vessels?

- Chloride
- Test experience is with 25%
- Optimization program benefits w/pre-mixed

PROPOSED START-UP OPERATING PARAMETERS

1. All 8 vessels in service, full-flow at equlibrium.

2. Flow during power ascension $<$ 3500 gpm.

3. Change-out precoat on delta P:

   - 20 psid for pre-mixed
   - 25 psid for carboxylic

4. As change-out pre-mixed, replace with Epicarb.

PREPLANNED RESPONSE ACTIONS

1. If conductivity > 0.3 $_u$S/cm
   - stop adding more carboxylic
   - resume when conductivity down

2. Objective for Cu is < 0.3 ppb, but also ALARA

   a. If Cu > 0.2 ppb at equilibrium
      - Assure all 8 vessels in service
      - Maximize carboxylic
      - Increase loading of resin
      - Try Na form of carboxylic

   b. If Cu $\geq$ 0.3 ppb at equilibrium
      - Assure all 8 vessels in service
      - Maximize carboxylic
      - Increase loading of resin
      - Try Na form of carboxylic
      - Recommend consider power reduction

PREPLANNED RESPONSE ACTIONS

3. If chloride spike detected

   - suspend carbox replacement until source identified

4. If carboxylic resin is source of chlorides

   a. If spike $>$15 ppb, suspend use of carboxylic and use pre-mixed.

   b. If spike $<$15 ppb, consider slowing intro of carbox to minimize spike.