10 CFR 50.73

Boston Edison

Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

E. T. Boulette, PhD
Senior Vice President — Nuclear

January 15, 1996
BECo Ltr. #96-005

U.S. Nuclear Regulatory Comission
Attn: Document Control Desk
Washington, D.C. 20555

Docket No. 50-293
License No. DPR-35

The enclosed Licensee Event Report (LER) 95-012-00, "Core Thermal Power Exceeded Technical Specification Limit Due to Omission in Calculation" is submitted in accordance with 10 CFR 50.73.

In this letter, the following commitment is made:

- Revise the Core Thermal Power Calculation to account for Reactor Recirculation Pump seal purge flow.

Please do not hesitate to contact me if there are any questions regarding this report.

ET Boulette
E. T. Boulette, PhD

JPC/dmc/9501200

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
U.S. Nuclear Regulatory Comission
475 Allendale Road
King of Prussia, PA 19406

Sr. NRC Resident Inspector - Pilgrim Station

Standard BECo LER Distribution

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NRC Form 366 (5-92)

U.S. NUCLEAR REGULATORY COMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for number of digits/characters for each block)

APPROVED BY OMB NO.3150-0104
EXPIRES 5/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE(3)
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TITLE (4)
Core Thermal Power Exceeded Technical Specification limit due to Omission in Calculation

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER 05000
12	14	95	95	012	00	01	15	96	FACILITY NAME N/A	DOCKET NUMBER 05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)

OPERATING MODE (9)	N		20.402(b)		20.45(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)(D)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)	x	50.73(a)(2)(i)(B)		50.73(a)(2)(viii)(A)		(specify in Abstract
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		below and in Text,
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jeffrey P. Calfa - Senior Compliance Engineer	508-830-8108

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE(15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On December 14, 1995, engineers of the Regulatory Affairs Department determined that the reactor had not been operated in accordance with Technical Specification 3.11.D in that the reactor had operated slightly above 1998 MWT in excess of the time allowed by Technical Specifications. An omission existed in the calculation for determining Core Thermal Power by heat balance in that the Reactor Recirculation seal purge flow to the Reactor Vessel from the Control Rod Drive System was not accounted for as an energy input to the energy balance. The maximum power attained was less than 0.1% above the Technical Specification limits described in the Core Operating Limits Report and Section 3.A of the Facility Operating License. Immediate corrective action was to issue an administrative standing order limiting power to below the Technical Specification limits. Additional corrective action planned includes revising the core thermal power calculation to account for the Reactor Recirculation seal purge flow.

The condition was discovered with the plant operating at 100 percent power with the reactor mode selector switch in the RUN position. The Reactor Vessel pressure was 1034 psig with the Reactor Vessel water temperature at saturation temperature for the reactor pressure. This condition posed no threat to public health and safety.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

BACKGROUND

Technical Specification 3.11.D and the Core Operating Limits Report (COLR), Revision 11B, Section 3.4 collectively state that the plant power-to-flow relationship shall not exceed the limiting value shown on Figure 3.4-1 of the COLR. The maximum core thermal power (CTP) allowed by Figure 3.4-1 is the rated CTP of 1998 megawatts-thermal (MWT). Technical Specification 3.11.D states that any time plant personnel determine the power-to-flow relationship of the COLR is exceeded, action shall be initiated within fifteen minutes to restore operation within the prescribed limits. If the power-to-flow relationship is not returned to within the prescribed limits in two hours, the reactor shall be shut down to the Cold Shutdown condition within thirty-six hours.

Reactor Engineering personnel use an energy balance methodology to calculate CTP using the Nuclear Steam Supply Software of the Emergency Plant Information Computer (EPIC) or manual calculations in accordance with Procedure 9.3, "Core Thermal Power Evaluation". In both cases, an energy balance is made on the energy inputs and outputs to the set of reference components comprised of the Reactor Vessel, the Reactor Recirculation System loop piping, and the piping to and from the demineralizers of the Reactor Water Cleanup (RWCU) System. The CTP is equal to:

  Radiative losses of reference components
+ Energy of RWCU water leaving Reactor Vessel for RWCU demineralizers
+ Energy added to the Control Rod Drive (CRD) System flow to Reactor Vessel to change it to steam
+ Energy added to Feedwater System flow to Reactor Vessel to change it to steam
- Energy of RWCU water returning to Reactor Vessel from RWCU Demineralizers
- Energy added to Reactor Recirculation flow from Reactor Recirculation Pump work.

The CRD System flow input for the CTP calculation is measured from the output of Flow Transmitter FT-302-55. FT-302-55 is located in the CRD System piping between pump discharge drive water filters and the flow control valves, FCV-302-6A/B. The CRD System pump flow enters the Reactor Vessel and associated piping through the Hydraulic Control Units, the reference leg fill for Reactor Vessel level condensing chambers, and the Reactor Recirculation Pump seal purge flow. The Reactor Vessel reference leg fill tap is located downstream of FT-302-55 and is accounted for in the heat balance calculation.

The Reactor Recirculation seal purge supply sub-system piping from the CRD System was installed in May of 1976 under Design Change Request Evaluation Guide (DCREG) 278, Safety Evaluation 353 and Work Request Permits (WRP) 75-716, 75-797, 76-783, and 76-478. Seal purge flow of approximately 3 to 3.5 gpm is provided to each Reactor Recirculation pump. The tap off of CRD flow for the seal purge flow is located downstream of the CRD pump discharge drive water filters and upstream of Flow Transmitter FT-302-55.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

EVENT DESCRIPTION

On December 14, 1995, engineers of the Regulatory Affairs Department determined that the reactor had not been operated in accordance with Technical Specification 3.11.D in that the reactor had been operated slightly above 1998 MWT in excess of the time allowed by Technical Specifications. An omission existed in the calculation for determining CTP by the heat balance method in that the Reactor Recirculation seal purge flow was not accounted for as an energy input to the energy balance calculation. The omission in the calculation has existed since the installation of the Reactor Recirculation seal purge sub-system in 1976. The seal purge flow is diverted from the CRD System prior to FT-302-55. The maximum CTP that could have resulted due the omission of a total seal purge flow of 10 gpm would have been approximately 1999.7 MWT. Since seal purge flow is normally 3 to 3.5 gpm per Reactor Recirculation pump, the maximum power attained was most likely to have been approximately 1999 MWT.

On December 8, 1995, an engineer in the Operations Department noted the omission in the calculation when investigating the applicability of a condition described in the Operating Experience (OE) computer network of the Institute for Nuclear Power Operations (INPO). Apparently, some Boiling Water Reactor plants designed by the General Electric Company are susceptible to the calculation omission. Since the original report on the INPO OE network, additional plants have also reported the omission in CTP calculation.

Upon discovery of the omission in the calculation, the Operations Department Engineer wrote Problem Report 95.9612 to document the problem. The Operations Department Manager immediately issued a Standing Order limiting CTP to 1996 MWT until the heat balance calculation algorithm can be corrected. The condition was discovered while the plant was operating at 100 percent reactor power with the Reactor Mode Selector Switch in the RUN position. The Reactor Vessel pressure was approximately 1034 psig with the Reactor Vessel water temperature at saturation temperature for the reactor pressure.

CAUSE

The cause of the omission in the CTP calculation was utilitiy non-licensed personnel error in not revising the calculation algorithm when the Reactor Recirculation pump seal purge subsystem was installed in 1976. The modification development was a joint effort between personnel of the General Electric Company and the Boston Edison Company. The assumption is that the impact was inadvertently overlooked as the design input review documentation for the Reactor Recirculation pump seal purge modification did not identify any impact on the CTP calculation

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CORRECTIVE ACTION

Upon issuance of Problem Report (PR) 95.9612, the Operations Department Manager issued a Standing Order that limits CTP to 1996 MWT until the heat balance algorithm can be corrected. This limitation compensates for the omission of the CRD flow to the Reactor Recirculation pump seal purge and ensures the limits of Technical Specification 3.11.D and the COLR will not be exceeded.

The calculation in Procedure 9.3 (currently Rev. 16) and the CRD flow computer point within EPIC will be modified to account for the CRD flow to the Reactor Vessel through the Reactor Recirculation pump seal purge sub-system.

There is no need for specific action in regards to the personnel error in the plant design review process. The specific Boston Edison personnel involved in the project are no longer working for Boston Edison. Additionally, the process for plant design changes has been significantly strengthened since 1976. The process is described in Nuclear Operations Procedure (NOP) 83E1, "Control of Modifications to Pilgrim Station". The procedure requires several design impact reviews by Plant and Engineering personnel prior to and during the design document creation. This process is applicable to changes designed by Boston Edison and contractor personnel.

SAFETY CONSEQUENCES

This event posed no threat to the public health and safety.

The maximum CTP that could have occurred would not have exceeded 1999.7 MWT. This deviation is less than 0.1% above the COLR upper limit of 1998 MWT. This deviation is negligible when compared to the uncertainty of approximately 2% in CTP due to measurement inaccuracies assumed by plant analyses. The minor safety significance of the condition is demonstrated in the General Electric (GE) Company evaluation of a potential Loss of Coolant Accident (LOCA) at Pilgrim Nuclear Power Station. This LOCA analysis used GE's SAFER/GESTR-LOCA Application Methodology and was described in document NEDC-31852P dated April 1992 (Revision 1). The GE evaluation assumes an initial power of 2038 MWT or 2% power above the COLR upper limits of 1998 MWT. The SAFER/GESTR-LOCA analysis was performed in accordance with NRC requirements and the plant was shown to meet all licensing requirements related to the analysis. Due to the minor significance of the approximately one MWT deviation above the COLR limits, there were no safety consequences as a result of the omission in calculation methodology of CTP.

LICENSEE EVENT REPORT (LER)

TEXT CONTINUATION

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because the reactor was not operated in accordance with Technical Specification 3.11.D due to operating slightly above 1998 MWT in excess of the time allowed by Technical Specifications.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs). The review focused on LERs involving non-licensed personnel errors and reactor power since 1983. No similar LERs were identified.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Flow Transmitter (FT-302-55)	FT
Computer (EPIC)	CPU

SYSTEMS	
Control Rod Drive System	AA
Reactor Recirculation System	AD
Computer System (EPIC)	ID