PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION

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SANATOGA, PENNSYLVANIA 19464

(215) 327-1200 EXT. 2000

March 28, 1991
Docket No. 50-352
License No. NPF-39

J. DOERING, JR.
PLANT MANAGER
LIMERICK GENERATING STATION

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Licensee Event Report
Limerick Generating Station - Unit 1

This voluntary LER concerns an error in the General Electric Company supplied process computer databank which provides the cycle specific data necessary to calculate and analyze reactor core performance. As a result of this error, non-conservative calculations of the Maximum Fraction of Limiting Power Density (MFLPD) at certain locations in the core may have caused an under prediction for the actual value of MFLPD. This may have resulted in violation of Technical Specifications Section 3.2.2 since MFLPD may have been greater than the Fraction of Rated Thermal Power and the associated TS Action Statement was not implemented within the required time period. A review has verified that no thermal limits were exceeded as a result of this error.

Reference:	Docket No. 50-352
Report Number:	1-90-036
Revision Number:	00
Event Date:	July 9, 1990
Report Date:	March 28, 1991
Facility:	Limerick Generating Station P.O. Box A, Sanatoga, PA 19464

Very truly yours,

J Doering

JKP:rgs

cc: T. T. Martin, Administrator, Region I, USNRC
T. J. Kenny, USNRC Senior Resident Inspector, LGS

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U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

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TITLE (4) Error in the General Electric Company Supplied Process Computer Databank which provides Cycle Specific Data necessary to Calculate and Analyze Reactor Core Performance.

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 NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 7	0 9	9 0	9 0	— 0 3 6 —	0 0	0 3	2 8	9 1		0 5 0 0 0

		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
OPERATING MODE (9)	1	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	1 0 0	20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	X OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	Voluntary Report
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
G. J. Madsen, Regulatory Engineer, Limerick Generating Station	AREA CODE 2 1 5 — 3 2 7 - 1 2 0 0

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 fifteen single-space typewritten lines) (16)

On July 9, 1990, General Electric (GE) Company informed Philadelphia Electric Company (PECo) that an error had been made by GE concerning data supplied in the databank for the Limerick Generating Station (LGS) Unit 1 third operating cycle. This databank provides the cycle specific data necessary to calculate and analyze reactor core performance, including the Linear Heat Generation Rate (LHGR) and the Maximum Fraction of Limiting Power Density (MFLPD). GE personnel incorrectly entered control cell location information that is used to account for Control Blade History (CBH) effects. As a result of this error, MFLPD may have been greater than the Fraction of Rated Thermal Power (FRTP) with reactor power less than 100%. LGS Unit 1 Technical Specifications (TS) Section 3.2.2 requires to adjust the Average Power Range Monitor (APRM) gain or the APRM setpoints when MFLPD exceeds FRTP. Although no actual instances of MFLPD exceeding FRTP have been identified from historical data, the databank error may have caused a violation of TS section 3.2.2 in that MFLPD may have exceeded FRTP and the associated TS ACTIONs were not taken. As a result, this voluntary LER is being submitted to report a condition that may have been a condition prohibited by TS. A root cause analysis was performed and it was concluded that personnel error, procedure deficiency, lack of verification by GE and PECo, and inadequate communications were primary causal factors of this event. A PECo task force is following up on recommended corrective actions for both GE and PECo.

Background

The General Electric (GE) Company is the Nuclear Steam Supply System (NSSS) vendor for the Limerick Generating Station (LGS). The reactors at LGS are of the GE Boiling Water Reactor (BWR) type. BWRs use the manipulation of selected patterns of control rods to perform power distribution shaping and reactivity control within the reactor core during operation of the reactor. The control cell core (CCC) reactor fuel bundle loading pattern methodology used for LGS Unit 1 operation is designed to improve reactor fuel reliability and simplify reactor core operations. The reactor core is configured using 1/8 core symmetry as shown in Figure 1, with control cells consisting of low reactivity fuel bundles located in the A-2 control rod sequence locations in the reactor core. The A-2 sequence control rods are quadrant mirror symmetric, and A-2 is usually the only control rod sequence to be used during a CCC cycle. The term control rod sequence is used in defining the group of control rods used for reactivity control and neutron flux shaping while the reactor is at power. The A-1 and A-2 control rod sequences are two of four possible operating control rod sequences. Both the A-1 and A-2 control rod sequence patterns were used during the LGS Unit 1 third cycle of operation (See Figure 1).

The Linear Heat Generation Rate (LHGR) is the heat generation per unit length of a fuel rod. The LHGR can be calculated for each segment of each fuel rod in a fuel bundle. The maximum individual fuel rod LHGR for a node (a six inch segment of a fuel bundle) in the reactor core is calculated by multiplying the local peaking factor at a node by the Average Planar LHGR (APLHGR) at that node. APLHGR is the power in a given node of a specified fuel bundle divided by the total number of fuel rods in the fuel bundle, and is determined for each node. The operating thermal limit for LHGR specified in the Core Operating Limits Report for LGS, Unit 1 is 13.4 kw/ft for GE fuel type BP/P8X8R and 14.4 kw/ft for GE fuel types GE8x8EB and GE8x8NB, and is a conservative limit to ensure that 1% plastic strain on the fuel cladding is not exceeded anywhere in the core during steady state operation of the reactor. The plant process computer (PC) divides the actual LHGR at a node by the LHGR limit. This ratio is called the Fraction of Limiting Power Density (FLPD). If the maximum value of the FLPD (MFLPD) is less than 1.0, the LHGR limit is not exceeded.

The Fraction of Rated Thermal Power (FRTP) is actual reactor core Thermal Power divided by the reactor core Rated Thermal Power. The values of FRTP and MFLPD are compared at least once per twenty four hours during power operation. The Average Power Range Monitor (APRM) system continuously indicates reactor core average neutron flux level and initiates trips at specified trip setpoints to prevent excessive power which may cause fuel cladding damage. Technical Specifications (TS) Section 3.2.2 states that when the value of MFLPD is greater than the FRTP, the APRM setpoints must be adjusted or the APRM amplifier gain may be adjusted to raise the APRM signal to greater than or equal to MFLPD.

Unit Conditions Prior to the Event

Unit 1 was at various Operational Conditions (OPCONS) and power levels since this event covers all of cycle 3 operation.

There were no structures, systems or components out of service which contributed to this event.

Description of the Event

On July 9, 1990, GE informed Philadelphia Electric Company (PECo) personnel that an error had been made by GE concerning data supplied in the plant PC databank for the LGS Unit 1 third operating cycle. This databank provides the cycle specific data necessary for the PC to calculate and analyze reactor core performance, including LHGR and MFLPD.

GE personnel incorrectly entered control cell location information that is used to account for Control Blade History (CBH) effects. CBH effects occur in fuel pins around control rods which are inserted for significant periods of time. The inserted control rod reduces the local flux spectrum by decreasing the number of thermal (i.e., low energy) neutrons. This reduces the burnup rate of adjacent fuel pins while increasing the local conversion ratio of uranium to plutonium. When the control rod is then withdrawn, these adjacent fuel pins operate at a higher power due to the lower uranium burnup and increased plutonium content than if the fuel pins had been previously uncontrolled. The greater the amount of controlled burnup, the larger the CBH effect. As a result, non-conservative calculations of MFLPD at certain locations in the reactor core caused an under-prediction of MFLPD by as much as 5.5%. However, the non-conservatism of the value of MFLPD only affects those locations in the reactor core which receive significant periods of controlled exposure accumulation as seen in Figure 2 (noted by the white slashed blocks). A verification check was performed by July 12, 1990 of the values of MFLPD during the Unit 1 third cycle of operation when CBH effects were significant. This verification revealed that MFLPD was never greater than or equal to 1.0 with the non-conservative error taken into account. Therefore, the actual LHGR had not exceeded the TS allowable limits. The time periods when CBH effects were significant were during the time in which control rod sequence exchanges took place. These time periods are identified below.

Nov. 02, 1989 - Nov. 09, 1989 - A-1 to A-2 Control Rod Pattern

Jan. 23, 1990 - Jan. 30, 1990 - A-1 and A-2 Control Rod Pattern (mixed)

March 16, 1990 - March 23, 1990 - A-1 and A-2 Control Rod Pattern (mixed)

June 1, 1990 - June 8, 1990 - A-2 Control Rod Pattern

On September 13, 1990, we concluded that during the above defined time periods, MFLPD may have been greater than the FRTP with reactor power less than 100%. LGS Unit 1 TS Section 3.2.2 requires that, with the APRM flow biased neutron flux - upscale scram trip setpoint (S) and/or flow biased neutron flux - upscale control rod block trip setpoint (SRB) less conservative than the value described in the TS, adjust the APRM gain or the APRM setpoints when MFLPD exceeds FRTP. As a result of the PC databank error, APRM gains or the setpoints were not adjusted if required by TS Section 3.2.2. This may have caused a violation of TS Section 3.2.2 in that MFLPD may have exceeded FRTP and the associated TS

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

ACTION to initiate corrective action with 15 minutes to adjust S and/or SRB values to be consistent with TRIP Setpoint Values or to adjust APRM gains to Core MFLPD or to reduce reactor thermal power to less than 25% of Rated Thermal Power was not taken. Although no actual instances of MFLPD exceeding FRTP have been identified from historical data, there is sufficient reason to conclude that MFLPD may have exceeded FRTP. As a result, this voluntary LER is being submitted to report a possible condition that may have been prohibited by TS.

Analysis of the Event:

The consequences of this event were minimal and there was no release of radioactive material as a result of this event. The PC databank error for CBH effect did not result in any thermal limits being exceeded; however, operation in a condition prohibited by TS may have occurred for a portion of the third operating cycle. There was no fuel damage in the core as a result of this event.

If a reactor power transient had occurred when the MFLPD may have been greater than the FRTP, individual fuel pin power may have increased above the expected level before the SCRAM reduced reactor power. However, the extent of the increase would not have been significant since reactor power was less then 100% during the periods of concern and the error was limited to 5.5% in less than 52 CCC locations. Additionally conservatism in the thermal limit calculations provided additional safety margin. Therefore the possibility for fuel damage following a transient was low.

The generic implications pertaining to this and any other databank errors which may exist and go undetected will be addressed by a PECo database error task force.

Cause of the Event:

The cause of this event is determined to be personnel error resulting in the incorrect entering of databank data in the GE Process Computer Information Transmittal (PCIT) form by GE personnel during design of the core reload for the LGS Unit 1 third cycle of operation. The incorrect information was incorporated into the final PC databank and then was submitted to PECo.

During February and March of 1989, changes to the LGS Unit 1 third cycle core design were being performed by GE personnel. The large number of fuel failures which occurred during the second cycle of operation caused the third cycle CCC fuel bundle loading to be abnormal. Thus, an unusual fuel bundle loading pattern and non-standard analyses resulted from the special reactor core fuel bundle loading requirements. The IRJR array is a type of coordinate system used to identify control cell locations in the PC. As part of the IRJR data entry on the PCIT form, the GE engineer defined the control cell locations for that cycle. Normally, the CCC uses A-2 sequence control rods only, so the control cells do not change from operating cycle to cycle, and the control cell input essentially remains the same. However, the third cycle fuel bundle loading also required the use of A-1 sequence control rods, which were then also required to be inputted in the IRJR array. Normally, the control rod coordinates for the

IRJR array are entered in the South-Southwest (SSW) octant, then reflected to the other seven octants using symmetry (Figure 2). However, the IRJR array data for the third operating cycle was incorrectly specified in the West-Southwest (WSW) octant by GE personnel. As a result, the only information which was reflected was that which was on the common diagonal between the two octants. The result of this error was that, except for those control cells on the diagonal common to the SSW and WSW octant and their symmetric counterparts, the rest of the control cells were not monitored for CBH effects. Because the GE engineer was much more familiar with 3-D simulator code coordinates rather than PC coordinates, he mistakenly recorded the CCC information in the WSW octant instead of the SSW octant. GE did not have pertinent information regarding the PC coordinate system incorporated into a procedure. In addition, the procedure did not specify the use of SSW octant when entering CBH effect data.

This error was detected when another GE engineer was preparing a software databank upgrade for LGS in July of 1990. One of the computer codes used to verify the databank, checks the CCC locations by determining if the cell coordinates entered are in the SSW octant. This verification program was developed in response to another databank error which was identified at another GE BWR plant in February of 1990. In March of 1990, GE initiated a change in the computer program so that the LGS Unit 1 control cell array would be automatically checked for proper input. In early July, 1990, when the GE databank group was preparing a software upgrade for the LGS PC, the LGS Unit 1 control cell array was automatically checked for errors. The error in PC databank was detected at this time. PECo was then notified of the error in a letter from GE dated July 9, 1990.

The root causes of the GE data entry error are:

1) a GE procedural deficiency which resulted in use of a non-standard coordinate system,

2) performance of an infrequently performed and complex task under manpower and time constraints,

3) lack of a consistent coordinate labeling system during communications between GE groups,

4) GE management controls were not effective in ensuring proper procedure implementations,

5) lack of GE design verification guide during review of the PCIT form, and

6) lack of GE supervision review for either the completion or the review of the PCIT form.

The following are contributing causes of this event.

1) The incorrect data was undetected by PECo personnel between the time of databank receipt and the completion of acceptance testing of the PC databank. This was caused in part by incomplete PECo databank review and

unclear databank review responsibilities between various on-site and off-site PECo groups.

2) Previous GE internal audits lacked depth and/or root cause analyses, and were less than adequate.

Corrective Actions:

The LGS PC databank for the CCC array was changed on July 9, 1990, to include the correct coordinates to enable control cell monitoring in all desired locations. A program which automatically checks the databank for changes has already been implemented by GE to avoid a future recurrence of this particular event.

PECo issued a letter to GE on September 10, 1990, which provided a copy of PECo's root cause analysis and recommendations for this event. The letter directed GE to evaluate the recommendations and to take appropriate actions to prevent the recurrence of a similar type of event. A PECo databank error task force has been formed which will follow up on the GE response to our letter.

Additionally, this task force will review GE's list of known databank errors and determine what actions PECo can take to ensure the accuracy of the databank. Possible approaches are:

1) determine which databank data sets most impact plant safety and/or TS compliance, and recommend methods to assure a thorough review by PECo of these data sets, or

2) determine which databank data sets, in whole or in part, are prepared or modified by hand, versus those prepared by controlled computer codes, and develop recommendations for a review of those data sets involving human intervention.

These actions will ensure that the computer programs provided are adequate to prevent recurrence of this event.

In addition, more clearly defined work responsibilities between the affected PECo groups will ensure proper coordination between PECo work groups. This will eliminate any duplicate verification that may be occurring and ensure that all the data that needs to be verified is in fact verified.

Previous Similar Occurrences:

None

Tracking Codes:

A00 - Personnel Error
D02 - Inadequate procedure - did not cover situation
E00 - Management or QA Deficiency

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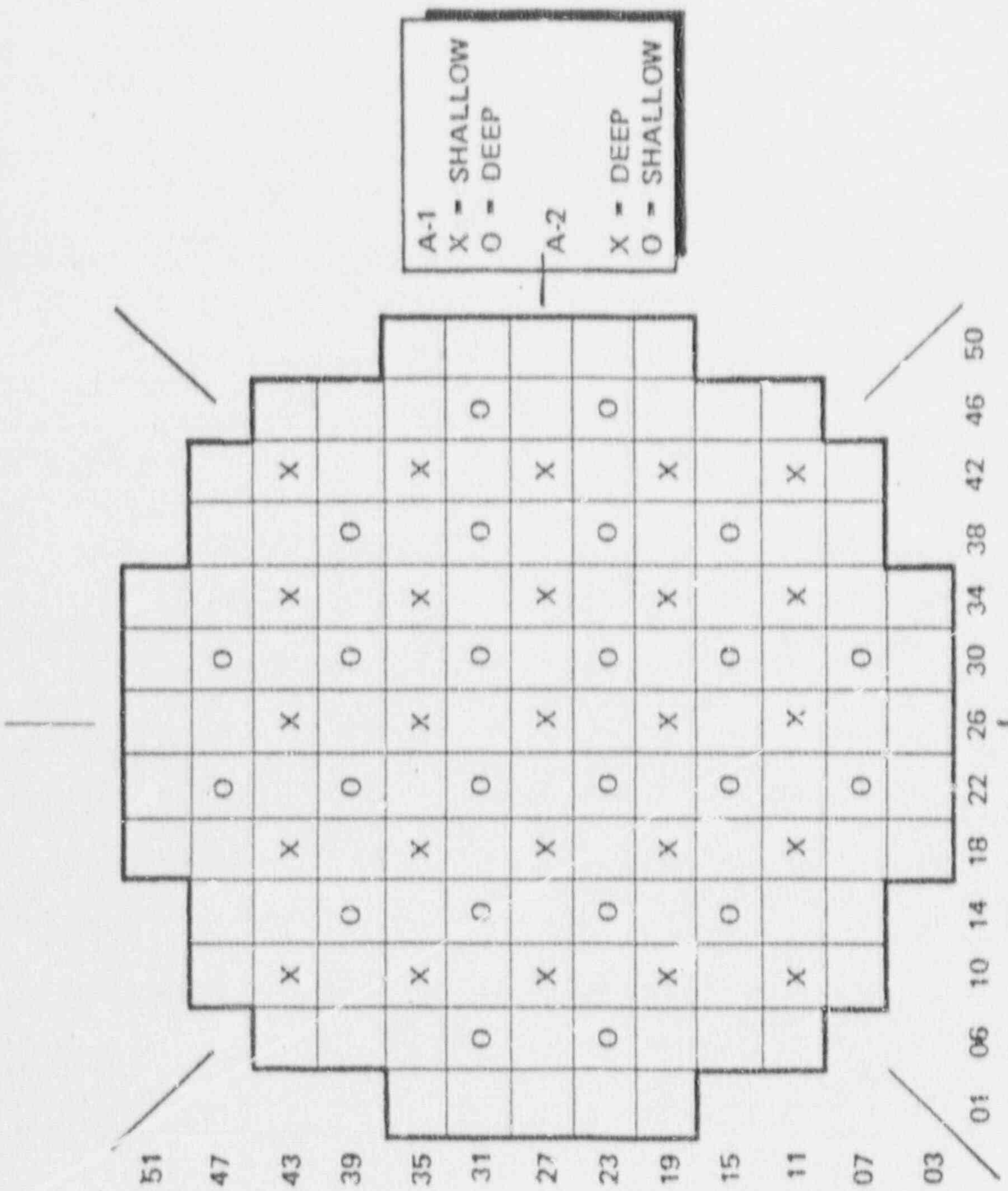

Figure 1 Conventional Control Rod Sequences

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Limerick 1 Cycle 3
Process Computer Databank Error
Affected Cells

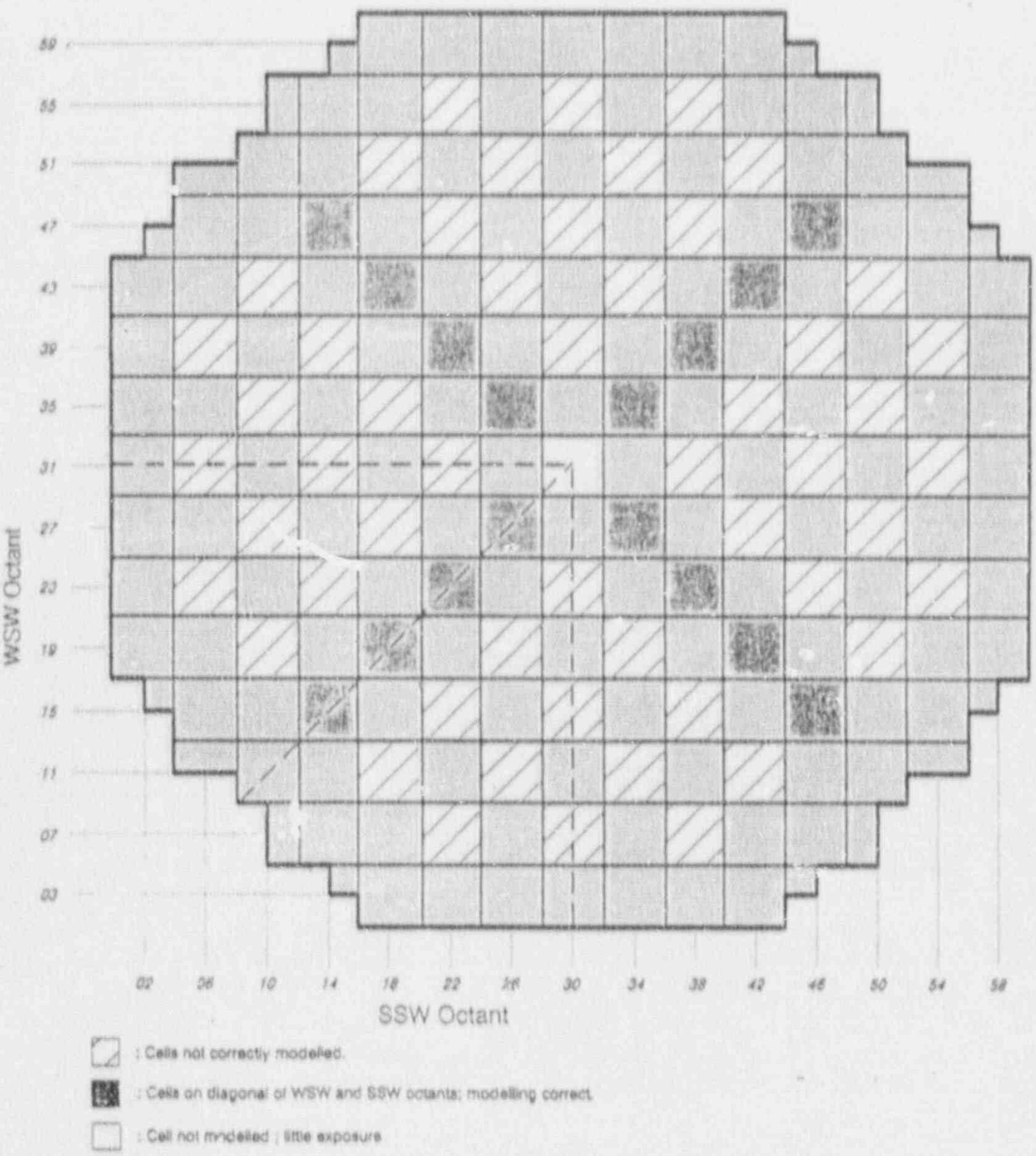

Figure 2