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Startup Test Report Pilgrim Nuclear Power Station Cycle 11

Enclosed is our Startup Test Report for Pilgrim Nuclear Power Station Cycle 11. If additional information is required please contact Mr. Robert Haladyna at (508) 830-7904 or Mr. Bruce Hagemeier at (508) 830-7808.

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ETB/RAH/nas/Rap95/Startup

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INTRODUCTION

Technical Specification 6.9.A.1 requires a summary report of plant startup and power escalation testing be submitted to the NRC following installation of fuel that has a *different* design or has been manufactured by a *different* fuel supplier.

The Pilgrim Station Cycle 11 reload batch is based on the General Electric GE 11 fuel type. This fuel type is distinguished by a 9x9 lattice geometry and part length fuel rods. Compared to the 7x7 and 8x8 lattice geometries and full length fuel rods used in previous Pilgrim reloads the GE11 reload batch for Cycle 11 constitutes a *different* fuel design.

General Electric has supplied all fuel loaded at Pilgrim Station since commercial operation began in 1972.

As required by Technical Specification 6.9.A.1 Pilgrim Station has provided the startup test report for Cycle 11 within 90 days of the resumption of commercial power operation on June 6th 1995.

SUMMARY

A reload batch of 136 GE11 fuel bundles with a bundle-average enrichment of 3.78 w/o was loaded in the Pilgrim Cycle 11 core to provide a cycle energy capability of 574 effective full-power days. This reload batch constitutes the first use of GE11 fuel at Pilgrim Station.

As-loaded Cycle 11 core maps showing fuel loading by both bundle type and bundle serial number are presented in Figures 1 and 2. The Cycle 11 core loading is octant symmetric and is generally based on both the low-leakage and control-cell-core design principles. The Cycle 11 core design is documented in the Pilgrim Plant Design Change Package (PDC) 94-33, *Reload 10/Cycle 11 Core Design*.

The final as-loaded core loading was verified to be consistent with the design core loading by Pilgrim personnel on May 8th 1995. Core loading verification following refueling was performed in accordance with the requirements of Station Procedure 4.5, *Reactor Core Fuel Verification*. No core loading errors were identified.

Control rod coupling integrity was verified to be satisfactory consistent with the requirements of Station Procedure 9.13, *Control Rod Sequence and Movement Control*.

Control rod scram time testing was verified to be consistent with the requirements of Technical Specifications 3.3.C.1 and 3.3.C.2. As required by Technical Specifications this testing was completed prior to exceeding 40% of rated core thermal power. Control rod scram time testing was performed in accordance with Station Procedure 9.9, *Control Rod Scram Time Evaluation*.

Shutdown margin was demonstrated to be consistent with the requirements of Technical Specification 3.3.A.1 by both the two-rod method and the in-sequence critical method. The two-rod method was performed a total of three times on May 8th and 9th to facilitate control rod friction testing and in-vessel visual inspection. The in-sequence method was performed on June 2nd following initial criticality. These demonstrations were performed in accordance with the requirements of Station Procedures 9.16, *Shutdown Margin Check*, and 9.16.1, *Insequence Critical for Shutdown Margin Demonstration*, respectively.

Calibration of instrumentation important to monitoring core thermal power and core margins to thermal limits was performed as required by station procedures and Technical Specifications. This instrumentation includes APRMs, LPRMs, TIPs and jet pump flow indicators. Calibration of this instrumentation was performed in accordance with the relevant station procedures.

Process Computer data processing checks were completed consistent with the requirements of Station Procedure 9.28, *Process Computer New Cycle Update*.

Margins to thermal limits calculated by P-1 wer compared to margins calculated by PANACEA and margins calculated by 3D-MONICORE. P-1 was generally found to yield less MFLCPR margin than either 3D- MONICORE or PANACEA. P-1 was used as the official thermal limit calculation throughout the power ascension program to demonstrate compliance with Technical Specifications.

Hot excess reactivity of the Cycle 11 core was found to be consistent with the requirements of Technical Specification 3.3.E. Hot excess reactivity was determined in accordance with Station Procedure 9.8, *Reactivity Follow*.

RFO 10 officially ended on June 6th when Pilgrim Station went on line after a refueling outage of 73 days. Rated power was reached on June 19th.

CORE DESIGN

The Cycle 11 core was designed to provide 574 effective full-power days of cycle energy capability as specified by the Pilgrim Station energy utilitzation plan for Cycle 11. This cycle energy capability includes a planned power coastdown of 14 effective full-power days.

The Cycle 11 core design is based on the General Electric GE 11 advance fuel type. The GE 11 fuel type continues the basic trend of earlier advanced General Electric fuel designs by accommodating greater discharge exposures and providing more margin to thermal limits, the principle ingredients to reduced reload fuel costs and higher plant capacity factors. The GE 11 fuel type continues this basic trend through a number of key design features: a 9x9 lattice geometry, 8 part length rods, two large central water rods, 10 atmospheres of helium prepressurization, high performance ferrule spacers, a high pressure drop lower tie plate and a low pressure drop upper tie plate.

The Cycle 11 core loading pattern is based on the low-leakage and control-cell-core design principles in use at Pilgrim since Cycle 5. The control-cell-core design principle designates selected rods in the core for reactivity control and power shaping at rated power and restricts fuel loading in the adjacent fuel cells to once- or twice-burned fuel. By avoiding rod withdrawals at power in the vicinity of fresh fuel, the control-cell-core design simplifies operation, improves fuel reliability, increases operating thermal margin and improves capacity factors.

The low-leakage design principle preferentially loads twice- and thrice-burned low-reactivity fuel on the core periphery to reduce radial neutron leakage, thereby yielding improved fuel cycle efficiency and reduced reload fuel costs. Reduced radial neutron flux also yields a reduced fast-neutron flux at the reactor pressure vessel wall, the reactor shroud and other core internals.

The Cycle 11 core design provides the cycle energy capability specified in the Pilgrim energy utilization plan for Cycle 11 with a 136 bundle reload batch of GE11 fuel at a bundle-average enrichment of 3.78 w/o. With this reload batch the inventory of fuel in the Pilgrim Cycle 11 core is:

Number of Bundles	Bundle Type	Cycle Loaded
136	GE7B-P8DRB300-5G5.0/2G4.0-80M-145-T	8
168	GE8B-P8DQB323-10GZ-90M-4WR-145-T	9
140	GE10-P8HXB355-11GZ-100M-145-T	10
136	GE11-P9HUB378-15GZ-100T-141-T	11

Figures 1 and 2 present the as-loaded Cycle 11 core maps showing fuel loading by both bundle type and bundle serial number.

The Cycle 11 core is loaded to be octant symmetric by both fuel type and, with a small allowance for variance, bundle exposure.

The Cycle 11 core design is documented in Pilgrim Plant Design Change Package (PDC) 94-33, *Reload 10/Cycle 11 Core Design*.

The Cycle 11 core design meets all licensing criteria specified in Revision 10 of NEDE-24011-P-A and NEDE-24011-P-A-US, the General Electric Standard Application for Reactor Fuel (GESTAR-II).

CORE VERIFICATION

The final as-loaded Cycle 11 core loading was verified on May 8th 1995 consistent with the requirements of Station Procedure 4.5, *Reactor Core Fuel Verification*. Three separate criteria were verified: bundle orientation, bundle seating and bundle location.

Bundle seating was verified by observing the channel fasteners of adjacent bundles in each fuel cell were vertically aligned.

Bundle orientation was verified by observing the channel fasteners of adjacent bundles in each fuel cell were oriented toward the center of the cell.

Bundle location was verified by observing bundle serial numbers in the core were consistent with bundle serial numbers in the final fuel loading plan.

Verification of the final as-loaded Cycle 11 core loading identified no core loading errors.

CONTROL ROD COUPLING INTEGRITY

Control rod coupling integrity was verified whenever a control rod was fully withdrawn for the first time following refueling. Coupling integrity was established by observing a discernible response of nuclear instrumentation during the rod withdrawal and, upon withdrawal to the full-out position, observing the rod would not reach its over-travel position.

Control rod coupling integrity is governed by Station Procedure 9.13, *Control Rod Sequence and Movement Control*.

CONTROL ROD SCRAM TESTING

Single rod scram time testing on all 145 control rods was successfully completed on June 9th prior to exceeding 40% of rated core thermal power as required by Technical Specification 4.3.C.1. Results of this testing are presented in Table I-A and Table I-B.

SHUTDOWN MARGIN

Shutdown margin (SDM) was demonstrated using both the local two-rod subcritical method and the in-sequence critical method. Both methods demonstrated adequate SDM although the margin demonstrated by the local two-rod subcritical method was substantially less than predicted by design.

The actual critical position observed during the in-sequence critical SDM was in excellent agreement with the estimated critical position.

Local Two-Rod Method

A local SDM demonstration was aborted by Pilgrim Operations personnel on May 8th 1995 when the reactor core approached criticality. This action was performed consistent with the requirements of Station Procedure 9.16, *Shutdown Margin Check*. It is noteworthy that the intent of this procedure is to demonstrate SDM while maintaining the reactor subcritical.

An evaluation of this aborted SDM demonstration found a SDM of 0.53%Δk was demonstrated assuming the reactor to be critical when the demonstration was aborted. While 0.53%Δk meets the requirements of Technical Specifications for a 0.25%Δk minimum SDM this value is 0.96%Δk below the design value of 1.49%Δk. The magnitude of difference between the design and demonstrated SDM was cause for concern and provoked a root cause evaluation.

The root cause evaluation determined the difference between the demonstrated and design SDM is largely a consequence of the increased uncertainty associated with a local two-rod SDM demonstration. In particular the local SDM demonstration is more sensitive to uncertainty in the control blade depletion and core exposure distribution. For the SDM demonstration performed on May 8th 1995 a more accurate accounting of control blade depletion was found to reduce the predicted SDM by approximately 0.1%Δk. Accounting for exposure using P1 instead of PANACEA was found to reduce the predicted SDM by 0.25%Δk. Together these two effects account for 0.35%Δk of the observed difference between the design and demonstrated SDM.

An additional 0.3%Δk of the difference between the demonstrated and design SDM was accounted for when the cold target eigenvalues used for the design SDM calculation were adjusted to reflect the cold cross section libraries actually used for the Cycle 11 design calculations.

The 0.35%Δk difference due to control blade depletion and exposure differences together with the 0.3%Δk target eigenvalue difference account for 0.65%Δk of the 1%Δk difference between the demonstrated and design SDM. The remaining difference of 0.35%Δk was attributed to the inherent uncertainty of the General Electric modeling methodology.

The results of this evaluation are documented in Pilgrim Station Problem Report 95.9270.

On May 9th 1995 the local two-rod subcritical SDM test was repeated and successfully demonstrated a SDM of 0.42%Δk without approaching criticality. This test used the same margin and object rod but positioned the margin rod at Notch 18 instead of the more conservative Notch 22 used in aborted demonstration. A less conservative temperature correction term was used as well. This temperature correction term applied to moderator temperatures less than or equal to

90°F. The first SDM demonstration had used a temperature correction term applicable for temperatures of 100°F or less.

May 9th also saw two additional local two-rod subcritical SDM demonstrations successfully performed. These SDM demonstrations were performed on the next two strongest worth control rods in the core to provide assurance SDM had in fact been demonstrated with the strongest rod fully withdrawn. This assurance was considered prudent in view of the magnitude of difference observed between the design and demonstrated SDMs. The SDMs demonstrated by these tests were 0.42%Δk and 0.43%Δk.

In-Sequence Method

A SDM of 1.23%Δk was demonstrated on June 2nd using the in-sequence critical method of Station Procedure 9.16.1, *Insequence Critical For Shutdown Margin Demonstration.* The agreement between this value and the design SDM of 1.49%Δk provided independent confirmation the root cause evaluation had identified correctly the causes of the discrepancy between the local two-rod SDM and the design SDM.

Estimated Critical Position

The estimated critical position for control rods was found to be in excellent agreement with the actual critical position. Initial criticality was estimated to occur when the 7th rod in Group 2 was pulled from Notch 12 to Notch 48 in an A-2 Sequence. Moderator temperature was assumed to be 180°F. This estimated critical position was based on a critical eigenvalue calculated for the core configuration observed during the aborted local two-rod SDM demonstration on May 8th.

Initial criticality was actually realized when the 7th rod in Group 2 was pulled to Notch 28. Control rods were withdrawn in an A-2 Sequence and the average moderator temperature was 180°F. The reactor period was 208 seconds.

INSTRUMENT CALIBRATIONS

APRMs

Average Power Range Monitors (APRMs) were calibrated as required during the power ascension to maintain the APRM gain adjustment factors (AGAFs) between 0.87 and 1.00. AGAFs are the ratio of the desired APRM reading to the actual relative reactor power reading as determined from a reactor heat balance.

APRMs were first calibrated in Cycle 11 on June 6th at a power level of 20% of rated. Subsequent calibrations were performed on June 8th at power levels of 22% and 38%, on June 10th at a power level of 50%, on June 11th at a power level of 87%, on June 14th at a power level of 96%, on June 15th at a power level of 94%, and twice on June 19th at a power level of 100%. APRMs have since been calibrated as required.

All APRM calibrations were performed consistent with the requirements of Station Procedure 9.1, *APRM Calibration*.

The initial APRM calibration was based on a hand heat balance. All subsequent APRM calibrations were based on reactor power values from OD-3, the Process Computer *Core Thermal Power and APRM Calibration* on-demand program.

LPRMs

Local Power Range Monitors (LPRMs) were calibrated as required by Station Procedure 9.5, *LPRM Calibration*. LPRMs are calibrated to maintain gain adjustment factors (GAFs) between 0.95 and 1.05. GAFs are the ratio between the desired LPRM console readings and the actual LPRM console readings.

LPRMs were first calibrated in Cycle 11 on June 10th. Subsequent calibrations were performed on June 22nd and August 2nd.

TIPs

The Traversing incore probe (TIP) system was used as needed to update the P-1 BASE array and calibrate LPRMs. Update of the P-1 BASE array is required following significant changes in core power distribution. Significant changes in core power distribution manifest themselves by a large number of BASE CRITs when P-1 executes and an increasing uncertainty in the P-1 calculation of thermal limits.

Update of the P-1 BASE array is effected by execution of the Process Computer OD-1 on-demand program. Through August 1st OD-1 was executed as indicated in Table II.

All OD-1's were executed consistent with the requirements of Station Procedure 9.5.1, *Operation of TIP Machines for Process Computer Updating*.

Accurate axial alignment of TIP machines is required for accurate updating of the P-1 BASE array and accurate LPRM calibrations. Station Procedure 9.20, *TIP Axial Alignment*, is used to assess the degree of alignment between TIP machines. Application of this procedure in Cycle 11 found TIP Machine A to be aligned between 4 and 5 inches too low for the 8 channels involved.

Axial alignment of Machine A was returned to specification on July 27th. Axial alignment of the remaining TIP machines was consistent with procedural requirements.

Jet Pumps

Jet pumps and recirculation drive flow were calibrated as required during the power ascension consistent with the requirements of Station Procedure 9.17, *Core Flow Evaluation*.

The acceptance criteria for jet pump calibration are agreement within ±5% between panel C-905 indicated core flow and calculated core flow. The acceptance criteria for recirculation drive flow calibrations are agreement within ± 2% between the two process computer loop flows and agreement within ± 5% between the indicated APRM loop flows and the calculated loop flows.

Jet pump and recirculation drive flow calibrations were first performed in Cycle 11 on June 11th at a core thermal power of 72% of rated. Subsequent jet pump and recirculation drive flow calibrations were performed on June 20th and July 14th at 100% power.

Jet pump flows were monitored on July 6th, July 27th, and August 3rd to demonstrate consistency with the acceptance criteria of Station Procedure 9.17.

PROCESS COMPUTER DATA PROCESSING CHECKS

The P-1 Process Computer databank was updated consistent with the requirements of Station Procedure 9.28, *Process Computer New Cycle Update*.

A number of checks are specified by Station Procedure 9.28 to verify the new Process Computer databank is consistent with the reload core design and has been correctly loaded into the Process Computer. These checks were completed satisfactorily by August 15th.

Consistency between the official databank transmittal and the reload core design was verified by a general review of the relevant core design documents against the official databank transmittal. This general review was completed on May 11th.

Before startup a number of checks were made to verify the new Process Computer databank had been correctly loaded:

1. Differences between the old and new databank identified by the Process Computer were verified to be consistent with the differences identified in the official databank transmittal.

2. The bundle loading identified by the Process Computer was verified to be consistent with the bundle loading specified by the core design documentation.

3. At least one node in each bundle was verified to be shuffled correctly into the exposure (EXF) and void history (EXVF) arrays.

4. All new LPRMs and control blades were verified to have zero exposures in their relevant arrays (CICEX and TCREX).

5. The isotopic compositions for at least one bundle in each batch were verified to be unchanged from their values in the old databank.

These checks were completed on May 22nd.

Following startup a number of additional checks were performed. These checks include:

1. Verification P-1 is calculating symmetric thermal limit and power distributions given a symmetric core loading and core control rod pattern.

2. Verification control rod positions are consistent with those indicated in the control room.

3. Verification LPRM readings are consistent with those indicated in the control room.

The last of these checks was completed on August 15th.

THERMAL HYDRAULIC LIMITS AND POWER DISTRIBUTION

Thermal Limits Calculated by P1

The maximum fraction of limiting critical power ratio (MFLCPR), the maximum fraction of limiting power density (MFLPD) and the maximum average planar linear heat rate (MAPRAT) were monitored throughout the startup using the General Electric P1 NSS core monitoring software. Margins to thermal limits were maintained as required by Technical Specifications.

The P1 power distribution was updated as required during the power ascension using the traversing incore probe (TIP) system during the ascent to rated power. The core thermal power, rated flow and thermal limits obtained from selected updates are presented in Table II.

Two features of Table II are of particular note. One is the MFLCPR value of 1.015 observed on June 13th at 21:32 hours. This MFLCPR resulted from executing OD-1 to clear BASE CRITICALS and update the BASE array in P-1. A P-1 executed before this OD-1 at 18:10 hours on the 13th showed MFLCPR to be 0.982. Power and flow at this time were 92% and 85% respectively. Following execution of OD-1 the control rod pattern was adjusted to restore MFLCPR to a value less than 1.0 as required by Technical Specifications. The P1 executed at

21:47 hours on the 13th showed a MFLCPR of 0.983. Core power and flow were 90% and 86% respectively.

The second feature of note in Table II is the drop in MFLCPR from a value of 0.988 on the 22nd to 0.967 on the 23rd. As discussed in the section that follows this drop is the result of corrective action taken to address a large TIP asymmetry introduced by a unique TIP instrument tube for the electro-chemical potential (ECP) probe.

Thermal Limits Calculated by PANACEA

P1-calculated thermal limits were compared with off-line thermal limits calculated by the General Electric's PANACEA design code throughout the startup. Selected results from this comparison are presented in Tables III-A, III-B and III-C. Table III-A shows PANACEA generally underestimated the P1-calculated MFLCPR by between 0.05 to 0.09. Tables III-B and III-C show generally excellent agreement between PANACEA and P1 for MFLPD and MAPRAT.

Tip Asymmetry

Part of the large difference between the P-1 calculated and PANACEA calculated MFLCPR was attributed to a significant TIP asymmetry between locations 28-37 and 36-29. An investigation of this TIP asymmetry revealed it to be a consequence of the unique design of the instrument tube at location 28-37. This instrument tube contains probes used to measure the electrochemical potential of reactor water. Incorporation of these probes into the instrument tube required a larger diameter and thinner wall for the outer tube sheath. As a result of this geometry difference the instrument tube at location 28-37 was surrounded by more water than would be the case with a standard tube. More water yields increased neutron thermalization and a greater LPRM reading for a given power level in adjacent bundles.

The conclusion that the TIP asymmetry between locations 28-37 and 36-29 was a consequence of the unique design of the instrument tube at location 28-37 and not a real core power asymmetry is consistent with the fact the Cycle 11 core was designed to be octant symmetric. This conclusion was confirmed by results from the General Electric 3D-MONICORE core monitoring software which was running in parallel to the official NSS (P-1) core monitoring software throughout the startup. 3D Monicore showed no significant power asymmetry.

The corrective action plan developed to address this TIP asymmetry changed the P-1 data bank to effectively substitute the TIP data from the instrument tube at location 36-29 for the TIP data from location 28-37. Due to the safety significance of MFLCPR this change was implemented only after a safety evaluation concluded this change could be effected consistent with the criteria of 10CFR50.59. FRN 94-44-07 documents this evaluation.

The corrective action plan to address the TIP asymmetry between locations 28-37 and 36-29 was implemented on the morning of June 23rd. Following implementation of this plan MFLCPR dropped from 0.98 to 0.96 with the limiting MFLCPR of 0.96 at another core location.

Thermal Limits Calculated by 3D-Monicore

3D-Monicore has been used to monitor thermal limits in parallel with P-1 since the start of Cycle 11. 3D-Monicore is GE's latest core monitoring software and is generally considered to provide a more accurate calculation of thermal limits than either PANACEA or P-1.

Table IV presents selected 3D-Monicore cases during the course of the Cycle 11 power ascension. These cases were selected to correspond as closely as possible to the times of the P-1 cases listed in Table II. A comparison of thermal limits in Tables II and IV shows 3D-Monicore generally provided 0.03 to 0.06 more MFLCPR margin than P-1 when rated power was approached. Both 3-D Monicore and P-1 provided substantial MFLPD and MAPRAT margin.

HOT EXCESS REACTIVITY

The actual control rod notch inventory (adjusted to reflect rated reactor dome pressure, rated core inlet flow rate and nominal core inlet subcooling) was verified to be consistent with the design notch inventory on June 23rd and July 17th. Table V presents both the actual and design control rod notch inventories for these dates. The acceptance criteria for this comparison is an actual control rod notch inventory that differs from the design notch inventory by no more than 270 notches.

Monitoring of hot excess reactivity is governed by Station Procedure 9.8, *Reactivity Follow*.

ADDITIONAL TESTING

The GE11 fuel loaded in Cycle 11 required no modifications to plant systems or components. Accordingly the first reload of GE11 fuel at Pilgrim requires no testing during startup beyond that normally performed to assure compliance with Technical Specifications. These test results have been presented in the sections above as required by Technical Specification 6.9.A.1.

FIGURE 1: PILGRIM CYCLE 11 CORE LOADING MAP
BY BUNDLE TYPE

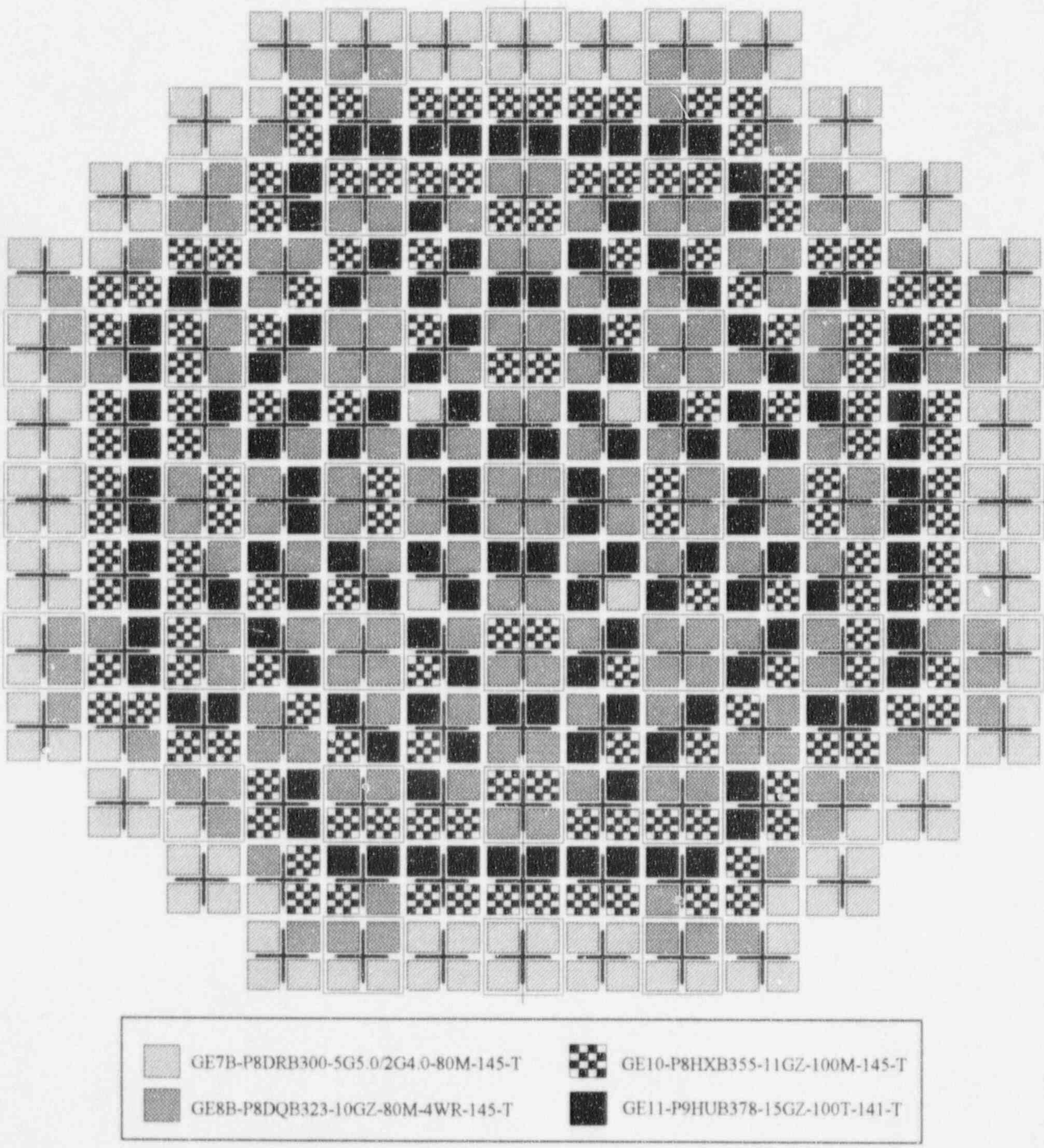

FIGURE 2
PILGRIM CYCLE 11 CORE LOADING MAP
BY BUNDLE SERIAL NUMBER

	01	03	05	07	09	11	13	15	17	19	21	23	25
52							LYF031	LYF090	LYF148	LYF045	LYF055	LYF081	LYF069
50							LYF041	LYZ045	LYZ153	LYZ073	LYF091	LYF054	LYF005
48					LYF018	LYF043	LYF068	YJ4994	YJ4990	LYZ077	YJ4977	YJ4969	YJ4963
46					LYF067	LYF141	LYZ089	YJ4998	YJA645	YJA646	YJA647	YJA648	YJA649
44			LYF079	LYF017	LYF022	LYZ069	YJ5004	YJA655	YJ4974	YJ4951	YJ4986	YJ4976	LYZ029
42			LYF024	LYF042	LYZ053	LYZ097	YJ4953	YJA657	LYZ005	LYZ117	YJA658	LYZ146	YJ4958
40	LYF023	LYF016	LYF006	LYZ057	YJ4959	YJ5014	LYZ161	LYZ157	YJ5008	YJA661	YJ4991	YJA662	LYZ025
38	LYF060	LYZ129	YJ4952	YJ5007	YJA665	YJA666	LYZ021	YJ4967	YJA667	LYZ109	YJA668	LYZ061	YJA669
36	LYF040	LYZ165	YJ4968	YJA675	YJ4950	LYZ113	YJ5016	YJA676	LYZ141	LYZ101	YJ4978	YJA677	LYZ065
34	LYF070	LYZ105	LYZ133	YJA681	YJ4984	LYZ085	YJA682	LYZ137	LYZ001	LYZ013	YJA683	LYZ121	YJ4962
32	LYF044	LYF046	YJ4996	YJA687	YJ5006	YJA688	YJ4997	YJA689	YJ5015	YJA690	LYF030	YJA691	LYZ009
30	LYF058	LYF192	YJ5005	YJA697	YJ4985	LYZ040	YJA698	LYZ149	YJA699	LYZ093	YJA700	LYZ041	YJA701
28	LYF025	LYF061	YJ4995	YJA707	LYZ033	YJ4966	LYZ049	YJA708	LYZ081	YJ4975	LYZ125	YJA709	LYZ017

FIGURE 2
PILGRIM CYCLE 11 CORE LOADING MAP BY BUNDLE SERIAL NUMBER (CONTINUED)

	27	29	31	33	35	37	39	41	43	45	47	49	51
52	LYF115	LYF106	LYF133	LYF143	LYF177	LYF096	LYF153						
50	LYF179	LYF132	LYF097	LYZ074	LYZ154	LYZ046	LYF139						
48	YJ4964	YJ4970	YJ4980	LYZ134	YJ4993	YJ5003	LYF114	LYF190	LYF172				
46	YJA650	YJA651	YJA652	YJA653	YJA654	YJ4999	LYZ090	LYF176	LYF182				
44	LYZ030	YJ4981	YJ4987	YJ4956	YJ4983	YJA656	YJ5013	LYZ070	LYF162	LYF171	LYF103		
42	YJ4961	LYZ147	YJA659	LYZ118	LYZ006	YJA660	YJ4954	LYZ162	LYZ054	LYF140	LYF164		
40	LYZ026	YJA663	YJ4992	YJA664	YJ5009	LYZ158	LYZ098	YJ5019	YJ4960	LYZ058	LYF180	LYF170	LYF163
38	YJA670	LYZ062	YJA671	LYZ110	YJA672	YJ4972	LYZ022	YJA673	YJA674	YJ5010	YJ4955	LYZ130	YJ0793
36	LYZ066	YJA678	YJ4979	LYZ102	LYZ142	YJA679	YJ5017	LYZ114	YJ4957	YJA680	YJ4971	LYZ166	LYF138
34	YJ4965	LYZ122	YJA684	LYZ014	LYZ002	LYZ138	YJA685	LYZ086	YJ4989	YJA686	LYZ078	LYZ106	LYF116
32	LYZ010	YJA692	LYF152	YJA693	YJ5018	YJA694	YJ5000	YJA695	YJ5011	YJA696	YJ5001	LYF144	LYF142
30	YJA702	LYZ042	YJA703	LYZ094	YJA704	LYZ150	YJA705	LYZ037	YJ4988	YJA706	YJ5012	LYF178	LYF134
28	LYZ018	YJA710	LYZ126	YJ4982	LYZ082	YJA711	LYZ050	YJ4973	LYZ034	YJA712	YJ5002	LYF125	LYF165

FIGURE 2
PILGRIM CYCLE 11 CORE LOADING MAP BY BUNDLE SERIAL NUMBER (CONTINUED)

	27	29	31	33	35	37	39	41	43	45	47	49	51
26	LYZ019	YJA716	LYZ127	YJ5064	LYZ083	YJA717	LYZ051	YJ5073	LYZ035	YJA718	YJ5044	LYF130	LYF166
24	YJA724	LYZ043	YJA725	LYZ095	YJA726	LYZ151	YJA727	LYZ038	YJ5054	YJA728	YJ5034	LYF187	LYF135
22	LYZ011	YJA734	LYF161	YJA735	YJ5024	YJA736	YJ5042	YJA737	YJ5033	YJA738	YJ5043	LYF145	LYF147
20	YJ5077	LYZ123	YJA742	LYZ015	LYZ003	LYZ139	YJA743	LYZ087	YJ5055	YJA744	LYZ135	LYZ107	LYF121
18	LYZ067	YJA748	YJ5061	LYZ103	LYZ143	YJA749	YJ5023	LYZ115	YJ5089	YJA750	YJ5071	LYZ167	LYF151
16	YJA756	LYZ063	YJA757	LYZ111	YJA758	YJ5072	LYZ023	YJA759	YJA760	YJ5032	YJ5087	LYZ131	LYF131
14	LYZ027	YJA763	YJ5048	YJA764	YJ5031	LYZ159	LYZ163	YJ5025	YJ5080	LYZ059	LYF185	LYF175	LYF168
12	YJ5081	LYZ148	YJA767	LYZ119	LYZ007	YJA768	YJ5086	LYZ099	LYZ055	LYF169	LYF167		
10	LYZ031	YJ5063	YJ5053	YJ5088	YJ5065	YJA770	YJ5035	LYZ071	LYF174	LYF124	LYF112		
08	YJA776	YJA777	YJA778	YJA779	YJA780	YJ5041	LYZ091	LYF189	LYF149				
06	YJ5076	YJ5070	YJ5062	LYZ079	YJ5049	YJ5045	LYF123	LYF191	LYF173				
04	LYF186	LYF137	LYF100	LYZ075	LYZ155	LYZ047	LYF150						
02	LYF122	LYF110	LYF136	LYF146	LYF188	LYF101	LYF160						

FIGURE 2
PILGRIM CYCLE 11 CORE LOADING MAP BY BUNDLE SERIAL NUMBER (CONTINUED)

	01	03	05	07	09	11	13	15	17	19	21	23	25
26	LYF026	LYF066	YJ5037	YJA713	LYZ036	YJ5066	LYZ052	YJA714	LYZ084	YJ5057	LYZ128	YJA715	LYZ020
24	LYF057	LYF013	YJ5027	YJA719	YJ5051	LYZ039	YJA720	LYZ152	YJA721	LYZ096	YJA722	LYZ044	YJA723
22	LYF049	LYF047	YJ5038	YJA729	YJ5028	YJA730	YJ5039	YJA731	YJ5021	YJA732	LYF039	YJA733	LYZ012
20	LYF075	LYZ100	LYZ080	YJA739	YJ5050	LYZ088	YJA740	LYZ140	LYZ004	LYZ016	YJA741	LYZ124	YJ5074
18	LYF053	LYZ168	YJ5068	YJA745	YJ5082	LYZ116	YJ5022	YJA746	LYZ144	LYZ104	YJ5060	YJA747	LYZ068
16	LYF008	LYZ132	YJ5084	YJ5029	YJA751	YJA752	LYZ024	YJ5067	YJA753	LYZ112	YJA754	LYZ064	YJA755
14	LYF028	LYF021	LYF011	LYZ060	YJ5079	YJ5020	LYZ100	LYZ160	YJ5030	YJA761	YJ5047	YJA762	LYZ028
12			LYF027	LYF029	LYZ056	LYZ164	YJ5085	YJA765	LYZ008	LYZ120	YJA766	LYZ145	YJ5078
10			LYF088	LYF009	LYF020	LYZ072	YJ5026	YJA769	YJ5056	YJ5083	YJ5052	YJ5058	LYZ032
08					LYF051	LYF015	LYZ092	YJ5040	YJA771	YJA772	YJA773	YJA774	YJA775
06					LYF019	LYF050	LYF077	YJ5036	YJ5046	LYZ136	YJ5059	YJ5069	YJ5075
04							LYF052	LYZ048	LYZ156	LYZ076	LYF094	LYF059	LYF012
02							LYF038	LYF095	LYF014	LYF048	LYF058	LYF086	LYF076

TABLE I-A
SCRAM INSERTION TIMES FOR AVERAGE OF
ALL RODS IN CORE (TECHNICAL SPECIFICATION 3.3.C.1)

PERCENT INSERTED FROM FULLY WITHDRAWN	MEASURED SCRAM INSERTION TIME, SECONDS	TECHNICAL SPECIFICATION SCRAM INSERTION TIME, SECONDS
10	0.49	0.55
30	0.99	1.275
50	1.50	2.00
90	2.57	3.50

TABLE I-B
SCRAM INSERTION TIMES FOR AVERAGE OF
THREE FASTEST RODS IN EACH GROUP OF FOUR
(TECHNICAL SPECIFICATION 3.3.C.2)

PERCENT INSERTED FROM FULLY WITHDRAWN	MEASURED SCRAM INSERTION TIME, SECONDS	TECHNICAL SPECIFICATION SCRAM INSERTION TIME, SECONDS
10	0.54	0.58
30	1.09	1.35
50	1.62	2.12
90	2.73	3.71

TABLE II
THERMAL LIMITS CALCULATED BY P-1
FOLLOWING EXECUTION OF OD-1

DATE	TIME	% CTP	% WT	MFLCPR	MFLPD	MAPRAT
6-09-95	00:44	38	47	0.671	0.362	0.704
6-10-95	11:46	49	54	0.824	0.458	0.644
6-11-95	15:16	70	57	0.923	0.586	0.691
6-13-95	21:32	96	60	1.015	0.837	0.852
6-18-95	11:53	81	65	0.980	0.649	0.722
6-18-95	18:40	86	71	0.970	0.688	0.803
6-19-95	03:18	96	86	0.984	0.774	0.790
6-21-95	15:26	100	97	0.997	0.794	0.794
6-22-95	16:32	100	98	0.988	0.793	0.794
6-23-95	11:26	100	99	0.967	0.800	0.801
7-12-95	10:27	100	97	0.968	0.791	0.802
8-01-95	13:55	100	93	0.979	0.791	0.809

TABLE III-A
COMPARISON OF MFLCPRS CALCULATED BY PANACEA AND P1

DATE	TIME	% CTP	% WT	PANACEA	P1	DELTA
6-16-95	10:43	94	102	0.87	0.95	0.08
6-21-95	11:59	100	98	0.90	0.99	0.09
6-27-95	9:42	99	100	0.90	0.96	0.06
7-15-95	6:27	100	97	0.90	0.97	0.07
7-22-95	7:40	98	91	0.91	0.96	0.05
7-29-95	8:65	100	92	0.90	0.97	0.07
8-02-95	13:37	100	94	0.90	0.98	0.08

TABLE III-B
COMPARISON OF MFLPDS CALCULATED BY PANACEA AND P1

DATE	TIME	% CTP	% WT	PANACEA	P1	DELTA
6-16-95	10:43	94	102	0.78	0.75	-0.03
6-21-95	11:59	100	98	0.83	0.81	-0.02
6-27-95	9:42	99	100	0.83	0.80	-0.03
7-15-95	6:27	100	97	0.84	0.79	-0.05
7-22-95	7:40.	98	91	0.82	0.78	-0.04
7-29-95	8:65	100	92	0.83	0.78	-0.05
8-02-95	13:37	100	94	0.83	0.79	-0.04

TABLE III-C
COMPARISON OF MAPRATS CALCULATED BY PANACEA AND P1

DATE	TIME	% CTP	% WT	PANACEA	P1	DELTA
6-16-95	10:43	94	102	0.76	0.77	0.01
6-21-95	11:59	100	98	0.80	0.81	0.01
6-27-95	9:42	99	100	0.81	0.80	-0.01
7-15-95	6:27	100	97	0.82	0.80	-0.02
7-22-95	7:40.	98	91	0.80	0.79	-0.01
7-29-95	8:65	100	92	0.81	0.79	-0.02
8-02-95	13:37	100	94	0.81	0.80	-0.01

TABLE IV
THERMAL LIMITS CALCULATED BY 3D-MONICORE

DATE	TIME	% CTP	% WT	MFLCPR	MFLPD	MAPRAT
6-10-95	11:59	48	53	0.762	0.444	0.614
6-11-95	14:59	70	57	0.848	0.591	0.684
6-14-95	02:59	95	100	0.943	0.815	0.821
6-18-95	11:59	81	64	0.900	0.675	0.746
6-18-95	18:59	86	72	0.911	0.701	0.744
6-19-95	05:59	99	92	0.947	0.822	0.801
6-21-95	14:59	100	98	0.935	0.821	0.806
6-22-95	15:59	100	99	0.928	0.818	0.795
6-23-95	11:59	100	99	0.929	0.817	0.793
7-12-95	09:59	100	96	0.931	0.811	0.795
8-01-95	07:59	100	94	0.945	0.809	0.814

TABLE V
HOT EXCESS REACTIVITY
(IN EQUIVALENT NOTCHES ADJUSTED TO RATED
REACTOR DOME PRESSURE, RATED CORE INLET FLOW RATE AND NOMINAL
CORE INLET SUBCOOLING)

DATE	EXPECTED NOTCHES	OBSERVED NOTCHES	DELTA NOTCHES
6-23-95	600	585	-15
7-17-95	580	586	+06