U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Report No.:	99900081/97-01
Organization:	Siemens Power Corporation - Nuclear Division 2101 Horn Rapids Road Richland, Washington
Contact:	Chris M. Powers Director, Quality
Nuclear Industry Activity:	Serves the U.S. nuclear industry by providing boiling- and pressurized-water reactors with fuel assemblies, reload core designs, and safety analyses
Dates:	February 10 through May 13, 1997
Inspection Team:	See Appendix C
Approved by:	Mr. Gregory C. Cwalina, Chief Vendor Inspection Section Special Inspection Branch Division of Inspection and Support Programs Mr. Timothy E. Collins, Chief Reactor Systems Branch Division of Systems Safety and Analysis

Enclosure 2

ABBREVIATIONS, ACRONYMS, and SYMBOLS Continued

SBLOCA	Small-Break Loss-of-Coolant Accident
SDD	Software Design Description
SDR	Software Development Record
SER	Safety Evaluation Report
SLCS	Standby Liquid Control System
SH1	CP&L, Shearon Harris Unit 1
SL1	FP&L, St. Lucie Unit 1
SLMCPR	Safety-Limit Minimum Critical Power Ratio
SPC	Siemens Power Corporation - Nuclear Division
SRLR	Supplemental Reload Licensing Report
SRP	Standard Review Plan
SRS	Software Requirements Specification
SRXB	Reactor Systems Branch (NRR/DSSA)
SSM	Sandvick Specialty Metals Corporation
SVVR	Software Verification and Validation Results
TD	Theoretical Density, g/cm^3
T/H	Thermal-Hydraulic(s)
TREXs	Tube Reduced Extrusions
TS	Technical Specifications
UCL	Upper Confidence Limit
URI	Unresolved Item
U	Uranium
UF_6	Uranium Hexafloride Gas
UO_2	Uranium Dioxide
UT	Ultrasonic Testing
V&V	Verification and Validation
WNP2	Washington Public Power Supply System, Washington Nuclear Plant Unit 2
WP	Work Practice
Zr	Zirconium
zirc	Zircaloy

ABBREVIATIONS, ACRONYMS, and SYMBOLS Continued

HBR2	CP&L, H.B. Robinson Unit 2
INEL	Idaho National Engineering Laboratory
LBLOCA	Large-Break Loss-of-Coolant Accident
LC2	ComEd, LaSalle County Unit 2
LFA	Lead Fuel Assembly
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio
MF	Manufacturing Follower
MSIV	Main Steam Isolation Valve
NAF	Nuclear Absorber Fuel Pellet
NCR	Nonconformance Report
NE	Nuclear Engineering
NFUF	Nuclear Fuel Users Forum
NNEC	Northeast Nuclear Energy Company
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation, Office of (NRC)
OJT	On-the-Job Training
OLMCPR	Operating-Limit Minimum Critical Power Ratio
P/CAR	Preventive/Corrective Action Report
PCT	Peak Cladding Temperature
PLFR	Part-Length Fuel Rod
PLSA	Part-Length Shield Assembly
PME	Product Mechanical Engineering
PNL	Pacific Northwest Laboratory
PO	Purchase Order
PPD	Plant Parameters Document
PPL	Project Parts List
PP&L	Pennsylvania Power & Light Company
PSIB	Special Inspection Branch (NRR/DISP)
PTS-PWR	Plant Transient Simulation Model for Pressurized Water Reactors
PWR	Pressurized-Water Reactor
QA	Quality Assurance
QAP	Quality Assurance Procedure
QC	Quality Control
QCP	Quality Control Procedure
RAI	Request for Additional Information
R&T	Research and Technology
S2	PP&L, Susquehanna Unit 2
SAFDL	Specified Acceptable Fuel Design Limits

ABBREVIATIONS, ACRONYMS, and SYMBOLS

3D	Three-Dimensional
AEOD	Analysis and Evaluation of Operational Data, Office for (NRC)
ANF	Advanced Nuclear Fuels Corporation
AOO	Anticipated Operational Occurrence
ASTM	American Society for Testing Materials
AVL	Approved Vendors List
BNL	Brookhaven National Laboratory
BOC	Beginning-of-Cycle
BWR	Boiling-Water Reactor
CAR	Corrective Action Report
CE	Combustion Engineering
CFR	*Code of Federal Regulations*
CHF	Critical Heat-Flux
CMR	Code Modification Request
CPA	Comanche Peak Unit 1
CPB	Comanche Peak Unit 2
COLR	Core Operating Limits Report
ComEd	Commonwealth Edison Company
CPCo	Consumers Power Company
CP&L	Carolina Power and Light Company
CPL	Component Parts List
CPR	Critical Power Ratio
CSDM	Cold Shutdown Margin
Δ	Delta (Differential)
DISP	Division of Inspection and Support Branch (NRR)
DOE	U.S. Department of Energy
DSSA	Division of Systems Safety and Analysis (NRR)
ECCS	Emergency Core Cooling System
ECP	Engineering Computer Program
EMF	Engineering and Manufacturing Facility
EOC	End-of-Cycle
EOL	End-of-Life
FCTF	Fuel Cooling Test Facility
FP&L	Florida Power & Light Company
FUG	Fuel Users Group
Gd	Gadolinia
GDC	General Design Criterion
GENE	General Electric Nuclear Energy
GG1	Entergy Operations, Incorporated, Grand Gulf Unit 1
GL	Generic Letter
GWd/MTU	Gigawatt-days per Metric Tonne of Initial Uranium Metal

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EXECUTIVE SUMMARY

Siemens Power Corporation - Nuclear Division
NRC Inspection Report 99900081/97-01

The inspection was performed to assess Siemens Power Corporation (SPC) activities with regard to the design and manufacture of nuclear fuel. In conducting this inspection, the team emphasized technically directed observations and evaluations of SPC's activities related to nuclear and mechanical engineering and manufacturing. In so doing, the team's primary objective was to establish a level of confidence that SPC's products will perform their intended safety functions.

During the inspection, the NRC team identified several safety-significant issues. Foremost among these issues was SPC's failure to verify the adequacy of the ANFB critical power correlation and the adequacy of its application to the ATRIUM™-10 fuel assemblies designed for the Pennsylvania Power and Light Company, Susquehanna Unit 2 Cycle 9 reload, the first reload using the ATRIUM™-10 fuel design for an NRC licensee (Nonconformance 99900081/97-01-01, example (1)). Because the results of the ANFB correlation are used as inputs to the safety limit methodology, this finding affects both the safety-limit and operating-limit minimum critical power ratios (SLMCPR and OLMCPR) for the ATRIUM™-10 fuel assemblies used in Susquehanna Unit 2 Cycle 9.

This finding was further exacerbated by the following assertions communicated during SPC's ATRIUM™-10 fuel design presentation to the NRC staff on May 4, 1995:

- SPC evaluated the transient tests and found that they demonstrated acceptable behavior of the ANFB correlation for the ATRIUM™-10 fuel design.

- The specific fuel design analyses (e.g., the mechanical analyses, stability evaluation, and thermal-hydraulic compatibility analyses) complied with the NRC-approved generic boiling-water reactor (BWR) design criteria in ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1, April 1990.

- SPC concluded that no additional NRC review was required.

Subsequently, in January 1997, SPC submitted EMF-97-010, "Application of ANFB to ATRIUM™-10 for Susquehanna Reloads," Revision 0, for NRC review. In that submission, SPC stated that the ANFB correlation may be used in the approved methodologies for the design, safety, and monitoring analyses associated with the ATRIUM™-10 reload at Susquehanna. The purpose of EMF-97-010 was to describe the dry-out testing of the ATRIUM™-10 design and the application of the ANFB correlation to the results of that testing, as they applied to the Susquehanna reload.

Contrary to SPC's assertions during the presentation to the NRC staff on May 4, 1995, and

SPC's conclusions in EMF-97-010, the inspection team identified significant failures in SPC's proposed application of the ANFB correlation to the ATRIUM™-10 fuel design used in Susquehanna Unit 2 Cycle 9 reload, as follows:

- The local peaking factor for this reload fuel design was outside the NRC-approved range for the ANFB correlation.

- The ANFB correlation included a nonconservative flow-bias and, therefore, was outside the NRC-approved SPC methodology.

Resolution of these findings required the NRC to review and approve a cycle-specific change to the Susquehanna Unit 2 Technical Specifications that addressed a critical power ratio uncertainty penalty for Cycle 9 to assure the plant is operated within its safety limits.

On the basis of its findings regarding SPC's failure to verify the adequacy of the ANFB critical power correlation and the adequacy of its application to the ATRIUM™-10 fuel design, the inspection team chose to evaluate the adequacy of the ANFB correlation to the ATRIUM™-9 fuel design. As a result of that evaluation, the team found that SPC failed to develop an adequate number of test points, and failed to test an adequate range of conditions to justify the uncertainty values for the "additive constants" used in determining the SLMCPR for the ATRIUM™-9 fuel design (Nonconformance 99900081/97-01-01, example 2). This finding implies that SPC should have used larger uncertainty values in the SLMCPR determinations, in order to reflect the full operability range of the ATRIUM™-9 fuel design. In addition, this finding had immediate implications regarding the SLMCPR and OLMCPR (and therefore the startup) of the following plants with ATRIUM™-9 fuel assemblies:

- Commonwealth Edison Company: Quad Cities Unit 2 Cycle 15, Dresden Unit 3 Cycle 15, and LaSalle County Unit 2 Cycle 8

- Washington Public Power Supply System: Washington Nuclear Unit 2 Cycle 13.

Resolution of this finding for the affected plants (on an exigent basis for Quad Cities Unit 2 Cycle 15) required the NRC to review and approve a cycle-specific change to each plant's Technical Specifications to assure the plants are operated within safety limits.

The NRC and its licensees must have confidence in the adequacy of new fuel designs because of the need for their compliance with NRC-approved generic design and performance criteria. However, the inspection team found SPC's last two evolutionary BWR fuel designs (specifically, the ATRIUM™-9 and -10 fuel designs SPC marketed to U.S. nuclear plants) unacceptable with regard to the adequacy of the ANFB critical power correlation and the adequacy of its application to these fuel designs. As a result, the two examples of nonconformance cited above have significantly eroded the NRC's ability to rely upon SPC to declare itself compliant with NRC-approved generic design criteria without requiring NRC review of new fuel designs. This is because the team found that SPC failed to comply with generic design criteria despite advising the NRC that it had.

Of the other nonconformances identified by the team, one of the more significant was the finding that SPC had not performed adequate verification and validation (V&V) for any of the loss-of-coolant accident (LOCA) analysis codes evaluated during this inspection (Nonconformance 99900081/97-01-02). This finding calls into question the accuracy of information provided to the NRC about the results of LOCA analyses, the impacts of changes to SPC's approved LOCA codes, and the conclusions drawn therefrom. The staff considers these issues generic *for all licensees that rely on the results of SPC's LOCA analyses*. As a result, the staff also questions the licensees' bases for accepting the accuracy of information provided by SPC concerning the above-mentioned issues and will review SPC actions as they relate to the requirements of Title 10, Part 21, of the Code of Federal Regulations (10 CFR Part 21).

During the inspection, the staff reviewed the original approvals and the records of changes made to SPC's LOCA analysis codes, and has assessed the current implementation of these codes for LOCA analyses for operating plants. The staff also has made use of its own experience in conducting confirmatory LOCA analyses, and in some cases, licensee analyses using LOCA codes other than SPC's. Based on these reviews and on other steps mandated by the staff, described below, the staff believes that SPC's LOCA evaluation models, as currently implemented, meet the requirements in Appendix K to 10 CFR 50, and thus, that there is reasonable assurance that licensees using these models meet the regulatory requirements and acceptance criteria related to LOCA evaluation models in Part I of Appendix K to 10 CFR 50 and 10 CFR 50.46(b). Nevertheless, the staff expects SPC to take corrective action to ensure that it is meeting the commitments in its approved QA program related to code documentation and V&V, as required by Criteria V, VI, VII, and XVII of Appendix B to 10 CFR 50. Furthermore, since licensees using or referencing SPC's LOCA codes and analyses are required to comply with code documentation and V&V requirements in Part II of Appendix K to 10 CFR 50 and 10 CFR 50.46(a)(1)(i), the staff strongly recommends that SPC review those regulatory requirements with affected licensees to ensure that SPC's code documentation and V&V records are adequate to establish that those licensees are in compliance.

On April 4, 1997, the staff issued Information Notice 97-15, "Reporting of Errors and Changes in Large-Break Loss-of-Coolant Accident Evaluation Models of Fuel Vendors and Compliance with 10 CFR 50.46(a)(3)," to remind licensees and fuel vendors of their responsibilities to ensure compliance with regulatory requirements concerning information provided to the NRC on LOCA codes and results of analyses. The NRC staff has also sent letters to all licensees who use SPC's LOCA codes informing them of key issues arising from this inspection.

The inspection team also identified several unresolved items where more information is required to determine whether the issue in question is an acceptable item, a nonconformance, or a violation. Of these, the most significant item was the lack of documentation to substantiate that the fuel cooling test facility (FCTF) reflood heat transfer correlation produces conservative results. The team determined that SPC failed to analyze the original data acquired using a 17 x 17 test bundle with a verified code under appropriate quality controls. The NRC staff views this issue as Unresolved Item 99900081/97-01-08, requiring additional information.

Additionally, the inspection team found that SPC had not adequately fulfilled a commitment to verify the methodology used to scale its 17 x 17 FCTF reflood heat transfer correlation for 15 x 15 geometry. This issue is Unresolved Item 99900081/97-01-07, requiring additional information and constitutes an unreviewed modification to an NRC-approved large-break loss-of-coolant accident (LBLOCA) methodology, per 10 CFR 50.46 and Appendix K to 10 CFR Part 50. The staff identified the information required to address these unresolved items in its letter from J.E. Lyons (NRC) to H.D. Curet (SPC) conveying a "Request for Additional Information (RAI) Regarding Reflood Heat Transfer," dated May 7, 1997.

The FCTF reflood heat transfer correlation has been the subject of extensive discussion between the NRC staff, SPC, and licensees using SPC's PWR LBLOCA evaluation model, predating this inspection. The staff has reviewed and accepted SPC's modification to the original FCTF correlation which, in the staff's judgment, produces conservative results for the peak clad temperature (PCT) as calculated by the computer code. The staff's judgment is based on experience with other, similar models, such as those based on data from the FLECHT tests cited in Appendix K to 10 CFR 50. Notwithstanding these previous actions, the staff requires that SPC address these unresolved issues. SPC must provide documentation to show that the 17 x 17 data used to develop the correlation were, in fact, analyzed using a valid data analysis code. In the absence of such evidence, the staff expects SPC to perform a complete reanalysis of the data in order to confirm that the reflood heat transfer model, as currently implemented, produces conservative results. The staff also expects SPC to complete the verification of the scaling methodology. The staff acknowledges receipt of SPC's response to the RAI included in the staff's May 7, 1997, letter; this information is currently being reviewed to determine its adequacy to resolve the above issues.

1 INSPECTION SUMMARY

From February 10 through May 13, 1997, representatives of the U.S. Nuclear Regulatory Commission (NRC) conducted a performance-based inspection of the activities at Siemens Power Corporation - Nuclear Division (SPC), in Richland, Washington.

In conducting this inspection, the team emphasized technically directed observations and evaluations of SPC's activities related to nuclear and mechanical engineering and manufacturing. In so doing, the team's primary objective was to establish a level of confidence that SPC's products will perform their intended safety functions. As the technical bases for the inspection, the team relied upon the following:

- Part 21, "Notification of Failure to Comply or Existence of a Defect," as defined in Title 10 of the *Code of Federal Regulations* (10 CFR)
- 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems [ECCs] for Light-Water Nuclear Power Reactors"
- General Design Criterion (GDC) 10, "Reactor Design," and GDC 12, "Suppression of Reactor Power Oscillations," 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants"
- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"
- 10 CFR Part 50, Appendix K, "ECCS Evaluation Models"
- NUREG-0800, "Standard Review Plan" (SRP), Rev. 2, July 1981, Section 4.2, "Fuel System Design," and Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," Rev. 0
- SPC topical report EMF-1A, "Quality Assurance Program for Nuclear Fuels and Services," Rev. 27, May 26, 1994, (prepared by SPC's Engineering and Manufacturing Facility (EMF), and approved by the NRC on June 2, 1994, as meeting the requirements of Appendix B to 10 CFR Part 50)
- SPC topical report EMF-1, "Quality Assurance Program for Nuclear Fuels and Services," Rev. 28, February 10, 1995 (approved by the NRC on January 16, 1996, as meeting the requirements of Appendix B to 10 CFR Part 50)
- XN-NF-75-21, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," Rev. 2, January 1986

- XN-NF-79-71, "Exxon Nuclear Plant Transient Methodology for Boiling-Water Reactors [BWRs]," Supplements 1 - 3, Rev. 2, March 1986

- XN-NF-74-5, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized-Water Reactors (PTS-PWR)," and Supplements 1 - 6, Rev. 2, October 1986

- XN-NF-84-73, "Exxon Nuclear Methodology for Pressurized-Water Reactors: Analysis of Chapter 15 Events," Rev. 2, March 1989

- XN-NF-80-19, "Advanced Nuclear Fuels [ANF] Methodology for Boiling-Water Reactors," Volume 1, Supplement 3 (including Appendix F) and Supplement 4, November 1990

- ANF-89-98(P), "Generic Mechanical Design Criteria for BWR Fuel Designs," Rev. 1, (approved by the NRC on April 20, 1995)

During this inspection, the team reviewed SPC's corrective actions taken to address NRC's findings during its 1994 inspection of SPC. The results of this review are discussed in Section 2 of this report.

Also during this inspection, the team identified several instances in which SPC failed to conform to the NRC's requirements or approved methodologies. The team also identified several unresolved items where more information is required to determine whether the issue in question is an acceptable item, a nonconformance, or a violation. These nonconformances and unresolved items are discussed in Section 3 of this report.

In addition, in Section 3 of this report, the team also discusses certain weaknesses and other observations regarding aspects of SPC's activities that affect product quality. However, neither the weaknesses nor the observations described in this report require any specific action or written response from SPC. That Section of the report also describes the team's evaluation of SPC's 10 CFR Part 21 program; QA program; the development, verification, use, and maintenance of the safety analysis and thermal-hydraulic (T/H) codes; the ATRIUM™-10 fuel design; the reload design and safety analysis process; and certain fuel fabrication activities.

Section 4 of this report describes the entrance and exit meetings. At the end of this report, Appendix A gives a partial listing of the SPC staff members who were contacted during the inspection, and Appendix B lists the items opened and closed during this inspection.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

During this inspection, the team evaluated the current status of findings that remained open from the previous NRC inspection of SPC. That inspection was conducted from February 28 through May 13, 1994, as documented in Inspection Report 99900081/94-01, dated August 29, 1994. SPC responded to this inspection and associated Notice of Nonconformance in a letter to the NRC dated September 28, 1994, regarding "Siemens Power Corporation Reply to Notice of Nonconformance (Inspection Report 99900081/94-01." The NRC responded to SPC in a letter dated November 9, 1994, stating that SPC's reply to the Notice of Nonconformance, and the corrective actions taken by SPC appeared to be adequate and that NRC would review the implementation of SPC's corrective actions to prevent recurrence during a future inspection.

As a result of its review during this inspection, the team closed Nonconformances 99900081/94-01-01, -02, and -03, as described in Sections 2.1 - 2.3 of this report.

2.1 Nonconformance 99900081/94-01-01 (CLOSED)

Revision 26 of SPC's "Quality Assurance Program" topical report, was approved by the NRC on February 17, 1993, as meeting the requirements of Appendix B to 10 CFR Part 50. Section 16 of that report defined the procedures for use in implementing and ensuring the quality of corrective actions. In addition, QA Procedure (QAP) 16, "Corrective Action Systems," Revision 7, dated July 8, 1992, augmented the guidance provided by EMF-1A. Contrary to that guidance, however, the 1994 inspection team found that SPC failed to properly close Audit 91:67, Corrective Action Report (CAR) 962, on the basis that the corrective action was complete. In addition, the team found that SPC failed to promptly close Audit 93:38, CARs 1200 and 1261, because the plant manager did not respond to the initial CAR for almost 8 months in spite of numerous followup requests by QA personnel.

During the current inspection , the team verified that SPC has since closed CARs 962, 1200, and 1261. In addition, the team noted that SPC has taken the following steps to correct the procedural deficiencies that led to this nonconformance:

- revise the QAPs to prevent recurrence
- establish and execute quality sensitivity training for all SPC employees
- establish a "tickler" system for open nonconformance reports (NCRs) and preventative/corrective action reports (P/CARs) to apprise management of open issues and timeliness
- devote greater attention to NCRs and P/CARs
- write and implement new procedures to revise the existing corrective action program controls

The team reviewed SPC's revised corrective action program and determined that the revisions should ensure that senior management will frequently review the status of the open issues, and the newly modified[1] quality department will conduct proactive monitoring. For example, the team verified that EMF-P00,066, QAP-13, "Control of Nonconformances," Revision 0, August 22, 1996, and EMF-P00,067, QAP-14, "Corrective Action," Revision 0, August 22, 1996, were appropriately written and implemented to preclude problems similar to those identified during the previous NRC inspection. The team also reviewed SPC's new quality training outline documents and attendance records that had been used to conduct training for SPC's Richland facility personnel and its procedures for the new "NCR-P/CAR Reporting & Quality Engineering Monthly Report" program. No adverse findings were identified.

The team concluded that SPC appears to have implemented appropriate modifications to correct the earlier nonconformance regarding the corrective action program. Additionally, the team concluded that SPC's new "NCR-P/CAR Reporting & Quality Engineering Monthly Report" strengthens the QA program and shows management's support of both programmatic quality and product quality, while alerting management to potential trouble areas before they become extensive or prolonged.

2.2 Nonconformance 99900081/94-01-02 (CLOSED)

During the 1994 inspection, the team noted that SPC failed to adhere to the requirements defined in Section 5, "Instructions, Procedures, and Drawings," of QA topical report EMF-1A, Revision 26. Specifically, SPC failed to prescribe certain activities that affect quality; ensure that activities that affect quality are accomplished in accordance with instructions, procedures, or drawings; and ensure that instructions, procedures, or drawings include the appropriate acceptance criteria. In practice, the 1994 inspection team noted the following instances of this nonconformance:

(a) Procedure EMF-P66,767, "High-Pressure End Closure Welder Station J," Revision 8, May 24, 1993, governed the welding operations for this work station. However, at the time of the 1994 inspection, this procedure was not available at welder station J. In addition, the procedure failed to prescribe test welds that, in part, verify the operability of the welding equipment and ensure adequate quality of production welds. Since 1994, SPC has revised the procedure to require a test weld at startup of the welding equipment. Revision

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[1] In the 1994 inspection, the team found that the quality control functions were the responsibility of the Plant Manager, while the QA functions were handled by a separate entity. SPC modified that structure to ensure that both functions are overseen by the Director, Quality, who reports to senior management instead of Plant Operations.

12, dated December 26, 1996, was available for review at the time of the recent inspection. In addition, the procedure was available at end closure welding station J.

(b) The supplier's process outline, approved by SPC, included the temperatures at which fabrication operations were performed for fuel clad tubing supplied in compliance with specification EMF-S35,055, "Zircaloy Tubing for Fuel Rod Cladding," Revision 7, February 21, 1992. However, the outline failed to specify an upper-limit for the quench temperature that would be necessary to meet the requirements of specification EMF-S35,055. Since 1994, the supplier's process outline for fuel clad tubing had been revised to specify the upper quench temperature limits. The team reviewed the revised outline during the current inspection and found it to be adequate.

(c) Quality control procedure (QCP) EMF-P66,521, "Fuel Assembly," Revision 28, dated December 28, 1993, and QCP EMF-530,571, "Fuel Assembly for 14 x 14 CE-Type PWRs," Revision 10, prescribed the procedure for assembling bundle components. However, these QCPs were not in use during the assembly of fuel bundles for Millstone Unit 2, a Northeast Nuclear Energy Company (NNEC) plant. In addition, QCP EMF-P66,521 failed to prescribe spring clip installation instructions. Since 1994, SPC has revised EMF-P66,783 to specify the requirements for spring clip installation. The team reviewed that procedure during the recent inspection, and found it to be acceptable. In addition, the team conducted a walk-through to verify that a mini-library is now located in the vicinity of the fuel cage and bundle assembly area.

(d) The supplier's process outline, approved by SPC, included the temperatures at which fabrication operations were performed for zirconium (Zr) alloy (zircaloy-4 or zirc-4) spacer side plate strip material supplied in compliance with SPC specification EMF-S35,001, "Spacer Components Fabricated from Zircaloy-4 Sheet," Revision 12, March 24, 1993. However, the outline failed to specify an upper-limit for the quench temperature that would be necessary to meet the requirements of specification EMF-S35,001. Since 1994, the supplier's process outline has been revised to specify the required quench temperature ranges. The team reviewed the revised outline during the current inspection and found it to be adequate.

The team concluded that SPC had taken adequate corrective action to resolve the identified nonconformance and prevent its recurrence. Additionally, the team concluded that SPC has begun to monitor this area to ensure the effectiveness of program compliance and implementation.

2.3 Nonconformance 99900081/94-01-03 (CLOSED)

During the 1994 inspection, the team noted that SPC failed to adhere to the requirements defined in Section 2, "Quality Assurance Program," of QA topical report EMF-1A, Revision 26. Specifically, only one of the six people assembling fuel bundles for Millstone Unit 2 had passed a written test on the standard operating procedures for the work stations in the fuel bundle assembly area. Moreover, SPC used temporary employees to perform critical fuel bundle assembly operations, and these temporaries were trained by observing lead technicians who were working from memory rather than referring to steps in a procedure to confirm that activities were being accomplished in accordance with written procedures. In addition, SPC's training records failed to document evidence that personnel were aware of procedures for work affecting quality.

SPC responded that its system of maintaining training records made it difficult to trace operator training; however, SPC's investigation identified that the operators in question did in fact, receive training for the duties that they were performing. Therefore, SPC concentrated its corrective action on modifying the method used to maintain the training records. Subsequent SPC audits and improved utility audits confirmed the effectiveness of the revised method. Additionally, the team determined during the recent inspection that the adequacy of the components fabricated during the time period in question were acceptable. SPC concluded that the fuel bundles were acceptable since the subject fuel bundle assembly attributes were verified by an independent quality control team.

During the recent inspection, the team determined that SPC personnel had revised and reviewed procedure EMF-1527, "Rod/Bundle Operations: Work Station Training and Operator Qualification Guide." to preclude recurrence of similar events in the rod/bundle operations area. Revision 1, dated February 13, 1995, delineated that the area supervisor was ultimately responsible for conducting personnel training and maintaining the related training records. Additionally, to prevent the same type of problems from developing in other areas, SPC has initiated a program to use a different type of training notebooks to make it easier to observe compliance with the procedural requirements. Using these notebooks in each of the six SPC facility operations areas should prevent recurrence of the nonconformance in different SPC facility manufacturing areas.

In addition to placing responsibility on the area supervisors, SPC has made the Secretary of plant operations responsible for maintaining the training files and entering the training data in the individual files. The SPC staff indicated that documentation and training responsibilities were still not clearly defined for some work station personnel; however, the team noted that SPC has started and continues to overhaul this area by adding more concise details to the procedure, requiring consolidation of training records, assigning ownership and responsibilities, and so forth. At the time of the interim exit meeting for this portion of the inspection, the team determined that more than 700 SPC employees have received the new EMF-1 orientation and indoctrination for SPC's QA program.

The team concluded that SPC had taken adequate corrective action to resolve the identified nonconformance and prevent its recurrence. Additionally, the team noted that SPC has begun to monitor this area to ensure the effectiveness of program compliance and implementation.

3 FINDINGS FROM THIS INSPECTION

SPC performs core reload design analyses, fuel development engineering, safety and transient analyses, and other fuel-related services. This inspection included an evaluation of SPC activities related to reload core design, neutronics and safety analysis, and fuel rod/bundle fabrication. This evaluation emphasized the adequacy of SPC's 10 CFR Part 21 program; QA program; the development, verification, use, and maintenance of the safety analysis and T/H codes; the ATRIUM™-10 fuel design; the reload design and safety analysis process; and certain fuel fabrication activities. Sections 3.1 through 3.6 discuss the team's findings in each of these areas.

3.1 10 CFR Part 21 Program

a. Inspection Scope

During this inspection, the team reviewed SPC Policy Guide 10.2, "Nuclear Safety Hazard Reporting," EMF-1713, Revision 4, December 15, 1995. This guide defined the policies and procedures that SPC had adopted to implement the provisions of 10 CFR Part 21.

b. Observations and Findings

In reviewing procedure EMF-1713, the team determined that it addressed the major attributes of 10 CFR Part 21, including the requirements contained in 10 CFR 21.21, "Notification of Failure to Comply or Existence of a Defect and Its Evaluation." The team also determined that SPC Policy Guide 10.2 correctly delineated the documents required to be posted in accordance with 10 CFR 21.6, "Posting Requirements," and that each of those documents was conspicuously posted in the SPC facility.

However, the team noted that EMF-1713 contained certain inconsistencies regarding use of the terms "defect" and "deviation," as defined in 10 CFR 21.3, "Definitions," and required clarification regarding the delineated responsibilities of SPC sub-tier safety-related component suppliers. The team discussed these inconsistencies with the Quality Engineering Manager to clarify EMF-1713.

c. Conclusions

The team determined that SPC had established and implemented a satisfactory procedure to ensure compliance with the provisions of 10 CFR Part 21, with the exception that SPC had incorrectly applied certain terms in a few sections of the procedure. The SPC Director, Quality, committed to resolve the inconsistencies within 60 days of the interim exit meeting on April 4, 1997.

3.2 Quality Assurance Program

During this inspection, the team observed that SPC had not yet fully implemented the requirements of QA topical report EMF-1, Revision 28, although the NRC had approved that revision on January 16, 1996. Consequently, the team evaluated certain aspects of SPC's activities in relation to QA topical report EMF-1A, Revision 27. To evaluate the QA Program, the team reviewed training, quality audits, and self-assessments. The following Sections summarize the results of this review.

3.2.1 Training

a. Inspection Scope

The inspection team reviewed SPC QA topical report EMF-1, Revision 28, as well as SPC's QAPs, and conducted interviews with SPC employees to evaluate the completeness and effectiveness of the training program. The team also evaluated the experience of the SPC staff members, as well as the training they received initially (as a new hire) and on an ongoing basis.

b. Observations and Findings

To evaluate SPC's training program, the team reviewed the training provided for Nuclear Engineering (NE) and safety analysts, and the programmatic requirements for training. The following paragraphs summarize the results of this review.

b.1 *Nuclear Engineer Analysts*

To evaluate the adequacy of training in the Nuclear Engineering units, the inspection team primarily relied on discussions with SPC management and staff. During the course of these discussions, SPC provided a list of personnel showing each individual's years experience at SPC, as well as the total years of professional experience for each. In reviewing that list, the team noted that for the SPC NE organization, the professional staff had on average, more that 9 years of experience at SPC. However, through discussions with SPC staff, the team learned that SPC has not provided any formal special training or classes. Instead, the majority of the staff's training is provided through routine section meetings, specific topic forums, and on-the-job training (OJT).

An SPC new hire from college is given reports and guidelines to read for methodology training. The inexperienced engineer is assigned a mentor who oversees the analysis being performed, and a notebook from a prior calculation is used as the template for the new analysis. Interviews with the least experienced engineers (those with 3 to 5 years of experience at SPC) convinced the inspection team that these engineers had good knowledge of the analysis process being performed and the mechanics involved in performing the analysis, but lacked indepth understanding of the rationale or basis for the analysis. By contrast the team found that, on average, the more experienced engineers on the SPC staff have an unusually high number of years of experience, and are very knowledgeable in the rationale or basis for the analyses they conduct.

During interviews with SPC employees regarding the technical aspects of SPC computer codes, the inspection team became concerned that some of the NE analysts lacked the necessary indepth knowledge of code detail. QA topical report EMF-1, Revision 28, QAPs, and good practices require SPC management to define, implement, and document appropriate training for all employees. Consequently, the team reviewed SPC's stated training policy for the NE, Product Mechanical Engineering (PME), and Research and Technology (R&T) groups.

Specifically, EMF-1560, "Nuclear Engineering and Product Mechanical Engineering Training and Proficiency Assessment Guideline," Revision 1, provided guidance regarding SPC's philosophy for training and proficiency assessment in NE and PME. According to that document, SPC's training policy is to rely on OJT, supplemented by brief in-house seminars. The inspection team found no such document for the basis of R&T training and considered this a weakness.

The inspection team questioned whether SPC had a legitimate basis for its practice of relying on OJT as its principle training mechanism. In particular, the team had concerns related to the problems identified with SPC documentation (Section 3.3 of this report presents additional discussions concerning SPC's lack of code documentation), and the practice of using copies of previous work without review (i.e., copying data from earlier calculation notebooks into new analyses), as discussed later in this report.

b.2 ***Safety Analysts***

In conducting this inspection, the team found that some of SPC's safety analysts appeared to lack sufficient indepth knowledge of the computer codes they were using. For example, the team interviewed two accident analysts who could not explain how the water level in the reactor vessel was modeled in the RELAX code used for loss-of coolant accidents (LOCAs) and in the CONGEN/COTRANSA2 code used for transients. This was a concern, since the water level is one of the most significant parameters affecting the transient dynamics. In addition, one of the two analysts could not explain how the separator was modeled or how some of the options were used in his input deck. The team also found it noteworthy that neither of these two analysts had complete sets of code documentation. (Section 3.3 of this report presents additional discussions concerning SPC's lack of code documentation.)

The inspection team determined that SPC lacked systematic written guidelines for code nodalization[2] or for the selection of the various code options used in LOCA analyses for BWRs. The nodalization and many of the code options have been unchanged (frozen) since 1975, and the newer analysts use them without adequate training or understanding. Neither analyst interviewed by the team could explain or justify the nodalization. The team considered this a weakness and a concern, since core and fuel design changes may affect the adequacy of the selected options; without an understanding of their bases, analysts will not be able to identify the situations where these models are not adequate. Notably, of the engineers interviewed by the team, the designated code custodian was the only one with detailed knowledge of the code. (The code custodian appeared to be a very experienced engineer with extensive institutional memory.) As a result, the team was concerned that, when the custodian is not available, SPC may not be able to adequately respond to situations requiring a detailed knowledge of the code and indepth analysis of the calculated results. In addition, the analysts performing the LOCA safety analyses did not appear to be knowledgeable of the codes as they should; this may, in part, be a result of the lack of complete and up-to-date code manuals.

b.3 *Programmatic Requirements*

EMF-1560 instructs management to "encourage" employees to attend the lectures given during routine section meetings and specific topic forums, but only requires special training when it is "necessary due to defect frequency." The team determined that this was not an adequate program to provide indoctrination and training of personnel performing activities affecting quality in order to ensure that they achieve and maintain suitable proficiency. The team also determined that the requirement to provide indoctrination and training for employees has appeared in a QA topical report

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[2]In general, nodalization is the manner in which an input deck for a computer code specifies how the physical system being analyzed is "broken up" into discrete volumes or nodes. The nodalization thus defines how the fluid properties and relevant parameters are averaged or otherwise specified when the controlling differential equations (e.g., mass, momentum, or energy) are solved by the computer code.

for many years, but SPC did not write a QA guideline until a Pennsylvania Power & Light Company (PP&L) audit required SPC to do so. Even after PP&L imposed that requirement, R&T group did not develop a program.

In its internal memorandum regarding the "Response Package for Issues from NRC Inspection Week 1," dated March 14, 1997, SPC informed the team that a training guideline for all engineering groups (including R&T) is under development.

In interviews with SPC employees, the team determined that significant portions of the OJT program relied upon the SPC code documentation. This is adequate for codes with good documentation; however, the team found that this approach did not provide acceptable training for some of the poorly documented computer codes. (See Section 3.3 of this report.)

The team determined that the current training program, as outlined in EMF-1560, was inadequate because it lacked any established method or procedure for determining the knowledge and skills needed to perform the engineering-related tasks associated with a given job category. Moreover, EMF-1560 failed to establish a method or procedure for evaluating the effectiveness of the training program. In addition, the team found that SPC failed to establish an appropriate indoctrination and training program as required by Criterion II, "Quality Assurance Program," of Appendix B to 10 CFR Part 50, and Section 18, "Training," in QA topical report EMF-1, Revision 28. The team identified these deficiencies as Nonconformance 99900081/97-01-05.

c. **Conclusions**

For the NE analysts interviewed, the team concluded that SPC provided minimal direct, formal technical training. However, the team did not observe any instances in which the lack of formal training caused errors of any significance in the reload analysis.

On the basis of the interviews, the team also concluded that SPC's LOCA safety analysts were weak in the theory and application of the analysis codes. This weakens SPC's ability to adequately respond to situations requiring a detailed knowledge of the code and indepth analysis of the calculated results.

In addition, the inspection team found that the lack of an adequate training program predicated on documented job requirements was a pervasive problem within SPC and, therefore, SPC can not show that its engineers have the knowledge and skills needed to perform the engineering-related tasks associated with a given job category. These findings resulted in a nonconformance.

3.2.2 Quality Audits

a. Inspection Scope

To evaluate the effectiveness of the required internal audit program implemented by SPC, the team conducted interviews with SPC's QA personnel and reviewed QA topical report EMF-1A, Revisions 27 and EMF-1, Revision 28, as well as QAPs and SPC's internal and external audit reports and CARs.

b. Observations and Findings

In its acceptance of QA topical report EMF-1, Revision 28, the NRC agreed that it is acceptable for SPC to rely on the numerous audits performed by the licensee customers and the Nuclear Fuel Users Forum (NFUF). These audits assist in fulfilling the internal audit requirements (defined in Appendix B to 10 CFR Part 50), provided that SPC conducts annual reviews of the audits conducted by the licensee customers and NFUF. However, SPC must supplement these audits, as necessary, with its own internal audits for those areas not addressed in sufficient detail in the audits conducted by licensee customers and NFUF. In addition, areas of weakness identified by the licensee customers and NFUF must receive emphasis in SPC's internal audits and must be audited on an accelerated basis, as warranted.

The team observed, however, that the numerous audits conducted by licensee customers apparently did not identify the problems with code calculations, correlations, and procedures that were identified during this inspection. Therefore, the team questioned the effectiveness and acceptability of using external audits to supplement SPC's internal audits. Moreover, the inspection team found that SPC failed to identify potential generic issues related to the findings from audits conducted by either SPC or its licensee customers.

In one example, SPC audits 96-021 and 96-022 conducted by PP&L in February and March 1996 revealed that the PME group did not have a procedure for defining, implementing, and documenting appropriate training. SPC's corrective action for this finding was to revise EMF-1560. However, SPC failed to make related revisions to training programs for the R&T group.

In another example, internal audit 93:107, which focused on the SPC engineering program, identified a concern related to inadequate QA verification and validation (V&V) of purchased computer codes. The internal audit report stated, in part, "the use of non-QA purchased codes may be a generic problem for other codes which are run on SPC hardware." CAR 1293 resolved the specific problem with the ABAQU code, but did not address the generic issue. (See Section 3.3.2 of this report.)

The team determined that these findings indicated a weakness in SPC's implementation of Section 17, "Quality Audits," of QA topical report EMF-1, Revision 28.

This weakness was discussed with SPC during the course of the inspection. Although SPC acknowledged the weaknesses identified, SPC stated that it will continue to rely on the numerous audits performed by licensee customers and NFUF and strengthen its own self-assessment program. During the fourth inspection week, SPC provided the team with background data concerning the self-assessment program. According to that information, SPC's current intention is for external audits to drive the need for self-assessments. SPC also intends to make the internal audits more performance-based and less programmatic. To further clarify its intent, SPC provided EMF-1928(P), "Nuclear Engineering/Methods & Codes Work Practices," Revision 0, dated March 26, 1997, to the team along with the following work practices (WPs):

- EMF-P104,104, "Engineering Control of Nonconformances and Corrective Action," Rev. 0, March 26, 1997

- EMF-P104,109, "Practice for Engineering Self-Assessment and Tracking," Rev. 0, March 26, 1997

The inspection team reviewed these new WPs, as well as the new quality procedure (EMF-1928(P)) and found that they incorporate some of the needed improvements to the audit process. As a result, the team determined that these practices and procedures have the potential to improve SPC's internal audit program, provided that they are properly implemented, with the necessary training.

c. <u>Conclusions</u>

Because the team observed licensee customer audits that did not identify the problems with code calculations, correlations, and procedures that were identified by the inspection team during this inspection, the team concluded that the effectiveness of using external audits to supplement SPC internal audits appears to be questionable.

In addition, the inspection team found it noteworthy that certain licensees failed to verify that SPC complied with the NRC-approved methodologies required by the plant's Technical Specifications. (See Section 3.3 of this report.)

Through review of SPC's internal audits, the team found examples of ineffective followup actions, as well as examples of technical issues that were not discovered through either external or internal audits. The inspection team concluded that SPC's internal audit process constitutes a weakness.

3.2.3 External and Self-Assessments

a. Inspection Scope

In November 1993, H.B. Robinson Unit 2 (HBR2), which is licensed and operated by the Carolina Power and Light Company (CP&L), experienced an event involving a misconfigured fuel assembly. Following that event and subsequent NRC inspections of SPC, self-assessment became an established process supported by an SPC policy. During the fourth week of this inspection, SPC provided the inspection team with background data on its self-assessment program. According to that information, SPC's current intention is for external audits (and/or assessments) to drive the need for indepth, critical self-assessments. SPC also intends to make the internal audits more performance-based and less programmatic.

b. Observations and Findings

SPC cited its self-assessment performed in November 1996 and documented in EMF-1924, "PWR LB LOCA Methodology Development Process Augmented Assessment," as a water-shed event, in that, for SPC the results of the self-assessment showed the strength and value of performing critical self-assessments versus performing more programmatic internal audits. Therefore, to evaluate SPC's position and reliance on self-assessments the inspection team evaluated several self-assessments and 2 external assessments that contributed to SPC's position.

The inspection team observed that SPC's self-assessment program is strongest in the manufacturing and QA areas. In addition, the team noted that the NE and the Methods and Codes group are making increased use of self-assessments; however, these groups are more reactive than proactive in their approach.

To evaluate SPC's external and self-assessments, the team reviewed self-assessment EMF-94-198; the external- and self-assessments associated with SPC's PWR LBLOCA methodology; and self-assessment 96-68. The following paragraphs summarize the results of this review.

b.1 *Self-Assessment EMF-94-198*

Beginning in February 1994, SPC established the PWR and BWR Nuclear Design Self-Assessment teams to define, critically examine, and recommend changes to the processes employed within PWR and BWR NE to produce products for internal and external customers and to address the misconfigured fuel bundle assemblies that SPC supplied to HBR2. The results of these self-assessments are documented in EMF-94-198, "PWR/BWR Nuclear Design Self-Assessment Team Final Report," dated October 1994. That report included final recommendations and highlighted key implementation elements, along with the associated benefits, and the basis for the recommendations. This discussion was presented in the following five areas:

- work process management
- work process execution
- personnel development
- product preparation and delivery
- general

As a result of this inspection, the team determined that the recommendations were appropriate to the issues identified during past NRC inspections. In addition, the team found that many of the recommendations presented in EMF-94-198 had already been implemented as part of SPC's "SucessFuel" program; however, the team did not review the SuccessFuel program in detail.

b.2 *PWR Large-Break Loss-of-Coolant Accident Methodology*

In October 1996, the NRC rejected portions of SPC's PWR large-break loss-of-coolant accident (LBLOCA) methodology. This rejection led to a progressive decrease in the NRC staff's confidence regarding SPC's engineering capabilities. As a result, SPC and certain of its licensee customers initiated the following internal and external assessments:

- On November 4 - 7, 1996, - Consumers Power Company (CPCo) and Florida Power & Light Company (FP&L) conducted a "Joint Utility Special Assessment" NPAD/P-96-16, which generated 20 NCRs.

- During November 1996, SPC conducted an internal "PWR LBLOCA Methodology Development Process Augmented Assessment," EMF-1924, which generated 30 NCRs.

- On December 9 - 13, 1996, CP&L led four other SPC customer utilities in performing the "SPC Fuel Users Group (FUG) Technical Assessment," DAA:97:013, which identified two potential open issues and five areas needing review for completeness.

SPC subsequently determined that the findings and recommendations of these three assessments overlapped one another, and specifically noted that EMF-1924 was intended to be as comprehensive as possible. SPC therefore established an integrated corrective action plan utilizing the NCR reporting and P/CAR system to track commitments. As a result of this plan, SPC identified the following significant deficiencies (among others):

- SPC's code documentation is inadequate, as is the guidance provided for V&V. In addition, SPC's assessment of the downstream impact in licensing applications has been incomplete following revisions of the codes.

- SPC responds to immediate problems and encourages engineered solutions, but has been inconsistent in permanently solving recurring problems.

- Historically, SPC has had no formal problem tracking mechanism and less-than-adequate commitment to self-critical assessments; however, according to SPC, this deficiency is partially offset by recent improvements in the QA program and the ongoing implementation of an improved code problem identification system.

- Technical decisions are typically made by line management, without benefit of independent review.

- NRC product licensing relationships have had minimal peer review and senior management involvement.

- SPC's engineering practices and personnel training are inconsistent within and among departments.

The team noted that several of these self-assessment findings were also identified as a result of this inspection, as reported in greater detail elsewhere in this report.

b.3 *Self-Assessment 96-68*

The inspection team found that SPC's Self-Assessment 96-68 originated as a recommendation from a earlier SPC evaluation of an error in CP&L's Shearon Harris Unit 1 Cycle 7 reload calculations. The earlier evaluation recommended that a self-assessment be performed to examine generic issues regarding inadequate review of code input and output. As a result, SPC organized a self-assessment team, and their assessment yielded 3 findings, 10 recommended immediate corrective actions, and 5 continuous improvement ideas.

The inspection team reviewed the documentation regarding the disposition of the self-assessment team's findings and recommendations. As a result of this review, the team noted that the status lists for the engineering NCRs and P/CARs tracked the NCRs identified in the disposition of the self-assessment items. No adverse findings were observed.

c. Conclusions

The team concluded that the self-assessment program appears to be well established in the manufacturing and QA areas.

The team noted that SPC has corrective actions in process, as indicated by the due dates for the relevant NCRs. The team therefore concluded that no judgment could be made at this time regarding the effectiveness of the corrective action process for the code input and output concerns.

Overall, the team concluded that SPC's self-assessments constituted a strength of the QA program. The team also noted that SPC is further strengthening its self-assessment program, particularly in engineering, to include clear problem reporting, causal analyses, management assessment, and analysis and oversight elements.

3.3 Safety Analysis and Thermal-Hydraulic Codes

The team evaluated SPC's safety analysis and thermal-hydraulic (T/H) codes against the requirements specified in 10 CFR 50.46, as well as Appendices B and K to 10 CFR Part 50:

- 10 CFR 50.46 requires, in part, that an evaluation model must be used as the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. The evaluation model includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedures for treating the program input and output, specification of those portions of the analysis not included in computer programs, parameter values, and all other information necessary to specify the calculational procedure.

- Appendix K to 10 CFR Part 50, item II, "Required Documentation," paragraph 1.a., requires, in part, that a description of each evaluation model shall be furnished. Moreover, this description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters or the procedure for their selection (in accordance with a specified physical law or empirical correlation, for example).

- The QA requirements of Appendix B to 10 CFR Part 50 govern computer codes affecting the safety-related function of structures, systems, and components used in safety analyses for nuclear plants (e.g., the cooling performance evaluation model for the emergency core cooling system (ECCS)). This generally requires following procedures that ensure control over the configuration of the code and complete current documentation of all models, features, and input for the code. It also requires having procedures for modifying the code, either to correct errors as they are discovered, or to add new features, models, and improvements. The procedures define the process for developing error corrections, as well as the requirements for testing and internal review before releasing a new version of the code. The procedures should also specify the acceptance criteria for V&V of code changes.

The definition and requirements related to the evaluation model are only explicitly specified for ECCS evaluation models; however, any reasonable implementation of documentation requirements that meet the standards of Appendix B to 10 CFR Part 50, "Design Control," would also require the same level of documentation for all analytical codes (such as those used for transient analysis). The documentation should be sufficiently detailed such that a knowledgeable engineer can review and understand the work without recourse to the person who performed the original analysis.

- In Generic Letter (GL) 83-11, dated February 8, 1983, the NRC staff informed licensees and vendors of its practice regarding qualification for performing safety analyses in support of licensing actions. The staff's practice, as stated in the GL, included evaluating (a) the acceptability of the QA procedures used for code development, verification, use, and maintenance, (b) competence regarding QA practices, and (c) technical competence regarding their ability to set up an input deck, execute a code, and assurance that the results are properly interpreted. NRC obtains this adherence by reviewing the code verification information, including comparisons performed by the user of the code results to experimental data, plant operational data, or other benchmarked analyses.

The inspection team reviewed SPC's computer codes, methodologies, and analysis documentation including user guidelines and plant calculation notebooks for codes used in ECCS and transient analysis. Specifically, the codes included RELAP4, RFPAC, TOODEE2, ANF-RELAP, S-RELAP, COTRANSA, RELAX, FLEX, MICROBURN-B, and HUXY. The team also interviewed several code developers and analysts as part of the inspection. To evaluate SPC's Safety Analysis and T/H Codes, the team reviewed SPC's control of engineering computer code programs (ECPs), V&V, BWR and PWR safety analysis and T/H codes, and the ANFB critical power correlation. The following Sections summarize the results of this review.

3.3.1 Engineering Computer Code Programs

a. Inspection Scope

In this portion of the inspection, the team reviewed the adequacy of SPC's procedures governing the use of ECPs, as well as SPC's conformance with those procedures. Specifically, procedures governing the use of ECPs were prescribed in EMF-608, "Computer Code Control Requirements — Engineering," Revisions 12 - 14, dated December 27, 1995; February 7, 1997; and March 12, 1997, respectively. Revision 12 was in effect at the start of this inspection. SPC subsequently issued Revisions 13 and 14 during the inspection to address the team's findings. The team also reviewed SPC's recently implemented QAP-21, "Computer Software Control," Revision 0, dated August 22, 1996. In addition, the team interviewed SPC management and staff of the ECP development groups and the end-user (engineering) groups, in order to evaluate the adequacy of EMF-608 and SPC's conformance to the established procedures and/or methodologies.

b. Observations and Findings

The inspection team reviewed SPC's software development records (SDRs) for the MICROBURN-B ECP. In particular, the team focused on the umar96 and usep96 SDRs, which addressed 30 code modification requests (CMRs) and 39 CMRs, respectively. While most CMRs involved administrative changes that did not affect results, a number of CMRs involved code errors.

The inspection team observed that SPC had no procedures for the ECP end user to review and determine the impact of existing code errors. The team determined that SPC's control of ECPs was weak in that it allowed up to 6 months to correct known code errors, without imposing requirements to notify the end user of the errors.

Code development and modifications are usually requested through CMRs transmitted by code users to the code developers at SPC, and are documented in SDRs. The team reviewed several SDRs covering many different types of modifications, and found a wide variation in the structure and content of these reports. The team determined that this variation most likely resulted from a lack of guidance in EMF-608 with regard to preparation of SDRs. In addition, the team identified a significant weakness in this area, in that SPC had no requirements for tracking CMRs in SDRs that were prepared in response to CMRs. Although some SDRs did, in fact, note which CMRs were addressed by the modifications in the report, this appeared to be at the discretion of the SDR preparer and was not always the case.

The inspection team also determined that SPC had no procedures for the end user to review known code errors. In particular, the team found that Revisions 12 and 13 of EMF-608 were lacking in the areas of notification of code errors, procedures or guidance for correction of code errors, procedures for end-user .eview of the impact of existing code errors, and procedures that require non-conservative code errors to be recorded and reported.

In preparing Revision 13 of EMF-608, SPC made one specific change to introduce two classes of code modifications, "major" and "minor," which primarily differed in how the modification was documented, and in requirements for V&V after the code modification was completed. However, the team found that EMF-608 still had no criteria for determining whether a modification was "major" or "minor"; this determination was left to the judgment of the custodian for the computer code being modified.

In addition, the team identified the following areas in which Revision 13 of EMF-608 was found to be deficient:

- no guidance on followup or tracking with authors of affected documents
- no procedures for code error notifications (i.e., warning messages)
- no procedures for end-user review of the impact of existing code errors
- no procedures requiring non-conservative code errors to be recorded and reported

Finally, the team found that Revision 13 of EMF-608 provided little guidance on updating code documentation as the codes evolve. As a result, SPC's existing code manuals are, in many cases, old, confusing, and inadequately updated. This is a particularly acute problem for older SPC codes.

Taken together with other ECP use and access control issues in Revisions 12 and 13 of EMF-608, these pervasive and wide-ranging deficiencies constitute a nonconformance because SPC failed to comply with Section 4, "Design Control," of QA topical report EMF-1, Revision 28. In particular, paragraph 4.5.3, "Design Errors," requires that design errors and deficiencies must be dispositioned in accordance with sub-tier QA and engineering procedures (i.e., EMF-608). However, the team determined that EMF-608, Revision 12 and 13, failed to address notification of code errors and evaluation of the errors' effect on the end user. These findings constitute example (2) of Nonconformance 99900081/97-01-04.

SPC responded to this nonconformance in an attachment to SPC's internal memorandum, "Responses to NRC Comments," dated March 13, 1997. In reviewing that response, the team found that SPC had partially addressed the nonconformance (i.e., lack of procedures for notification of code errors). Specifically, SPC revised Section 10 of EMF-608, Revision 14, to include guidance on code errors, stating that codes with known errors should be locked by the code custodian as soon as possible. (Locking the code prevents execution of the code.) In the event that a code is unable to be locked, an error file should be used to notify the end user of code errors.

These revisions partially remediated the team's original findings regarding Revisions 12 and 13, but some deficiencies still were not adequately addressed.

With Revision 14 of EMF-608, SPC implemented EMF-P00,066, QAP-13, "Control of Nonconformances," Revision 0, dated August 22, 1996, as well as EMF-P00,067, QAP-14, "Corrective Action," Revision 0, dated August 22, 1996. (Although, SPC had approved these QAPs in August 1996, they were not implemented before this inspection). These QAPs state that code errors are examples of nonconformances and are therefore required to be reported under the NCR process. SPC stated that one goal of the NCR process was to improve tracking and correction of ECP errors, and SPC staff provided examples showing how the NCR and P/CAR process was being implemented. However, at the time of the inspection, SPC had not yet written the implementing procedures for QAP-13 and QAP-14. During the team's last week of inspection activities, SPC was developing draft Revision 15 of EMF-608 to address the team's findings.

c. **Conclusions**

On the basis of these findings, the team concluded, through Revision 13, that EMF-608 exhibited a lack of procedures for code error notifications and for end user review of the impact of existing ECP errors. These findings resulted in example (2) of Nonconformance 99900081/97-01-04 which was partially addressed by Revision 14.

During the team's last week of inspection activities, SPC was developing draft Revision 15 of EMF-608 to address the team's findings. Additionally, SPC committed to provide specific guidance in new WPs.

3.3.2 Verification and Validation

a. **Inspection Scope**

To evaluate SPC's software V&V procedures, the inspection team reviewed QA topical report EMF-1, Revision 28; EMF-608, Revisions 12 - 14; and numerous SDRs. Specifically, the team investigated the following areas:

- V&V of code development activities
- adequacy of code documentation
- code input preparation control

The goal of code verification is to ensure that changes installed in the code are properly implemented and perform the desired function, preferably without introducing new errors into the code. This is accomplished by developing a suite of standard test cases that exercise the main features of the code and as many of the options as practical. An independent reviewer then evaluates the code changes to ensure correct implementation of the changes. Specifically, this verification involves running the full suite of test cases (including those test problems that specifically exercise the code changes) whenever modifications are made in the code. The objective of this verification is to determine that there are no differences in the results, or that any differences are consistent with the expected effects of the code changes.

By contrast, the goal of code validation is to demonstrate that the models in the code predict results that are consistent with the physical behavior being simulated, or that they are at least conservative with respect to limiting parameters of interest (e.g., parameters such as temperature, pressure, or boiling transition behavior). Adequate validation needs a larger set of test cases than verification, and includes comparisons with relevant experimental data, separate effects tests, and (where possible) analytic solutions.

b. Observations and Findings

To evaluate SPC's [illegible] V&V activities, the team reviewed non-SPC developed codes, code changes, certain SDRs, and changes to the RELAX and FLEX codes. The following paragraphs summarize the results of this review.

b.1 *Non-SPC Developed Codes*

During the course of this inspection, the team determined that many SPC LOCA and transient analysis ECPs had evolved from codes developed by the NRC and the U.S. Department of Energy (DOE), which SPC then modified for its own use. This is a concern because the QA standards defined in Appendix B to 10 CFR Part 50 apply to ECPs used in T/H analyses; however, the NRC and DOE codes used by SPC were neither developed nor maintained according to those QA standards.

Therefore, SPC should have verified these codes to ensure that the modified codes are accurately documented and that they perform as expected. However, the team determined that SPC did not perform this verification when these codes were purchased from outside sources for SPC's use. Furthermore, the team determined that SPC had no documentation available to show that SPC tested the code.

Similarly, code validation must be performed to show that a code will give accurate results under the conditions in which the code is used. T/H system codes (such as ANF-RELAP and S-RELAP) are large and have many models requiring validation for accident and transient analysis. Validation is usually performed by comparing code results to experimental data or analytical solutions. However, the team determined that, in many cases, SPC either neglected to perform and/or document the code validation, or simply performed minimal validation and referenced validation runs from NRC code documentation for phenomena important to the analysis for which the code was being used.

The team discussed these deficiencies with the code custodian for ANF-RELAP and S-RELAP. The code custodian said he was confident that the codes were of high quality because he was a developer of RELAP5 at the DOE laboratory.

These findings constitute example (1) of Nonconformance 99900081/97-01-02.

This finding calls into question the accuracy of information provided to the NRC about the results of LOCA analyses, the impacts of changes to SPC's approved LOCA codes, and the conclusions drawn therefrom. The staff considers these issues generic *for all licensees that rely on the results of SPC's LOCA analyses*. As a result, the staff also questions the licensees' basis for accepting the accuracy of information provided by SPC concerning the above-mentioned issues. Therefore, in addition to evaluating your response to this nonconformance, the staff will monitor your compliance with the NRC's requirements in Title 10, Part 21, of the *Code of Federal Regulations* (10 CFR Part 21).

b.2 ***Code Changes***

In evaluating SPC's V&V procedures (as defined in EMF-608 and QAP-21, "Computer Software Control," Revision 0, dated August 22, 1996), the team found that the procedures lack sufficient detail and guidance for code changes. For instance, the definition of a *minor* change was not clear. Notably, codes used in the LOCA methodology have undergone 52 revisions in 10 years, and SPC classified every one of these revisions as *minor*. The inspection team determined that SPC had never quantified the total effect of all these changes, and the V&V procedures do not provide any guidance in this area. Additionally, SPC neglected to assess the overall effect of each change using standard assessment cases (such as a suite of standard assessment test runs for each code modification), and the code assessment procedure was not adequately defined. In particular, the process did not require an assessment of code changes against appropriate phenomenological tests, separate effects tests, and/or integral system tests. In addition, the documentation of the V&V did not cover the evaluation of the results, acceptance criteria, or a discussion of discrepancies.

Consequently, contrary to Criterion III, "Design Control," of Appendix B to 10 CFR Part 50, and QA topical report EMF-1, Revision 28, the team found that SPC did not establish an acceptable procedure for V&V of code development and modifications. Specifically, the team determined that SPC failed to (a) adequately define a *minor* change or error and (b) establish an adequate code assessment process that included an assessment of code changes against appropriate phenomenological tests, separate effects tests, and/or integral system tests. These findings constitute example (1) of Nonconformance 99900081/97-01-04.

b.3 *SDRs 124-1 and 301-V5*

In this portion of the inspection, the team was particularly concerned with SPC's lack of acceptance criteria for code modifications. The team specifically noted that SPC did not distinguish between acceptance criteria for *major* verse *minor* code modification, or between changes in computer platforms and compilers compared to changes in method alone.

In reviewing the SDRs, the inspection team found that this lack of acceptance criteria led to the following discrepancies:

- SDR-124-1 (TOODEE2, uaug95) resulted in a change in the peak cladding temperature (PCT) of 63 °F because of a change in RELAP5 boundary conditions.

- SDR-301-V5 (RELAP5M2, udec94) resulted in a PCT change of 49 °F because of a change in computer platforms.

SPC considered these modifications acceptable because they resulted in only a *minor* effect on PCT. However, *significant* is defined in 10 CFR 50.46(a)(3)(i) as follows:

> *"a significant change or error is one which results in a calculated peak fuel cladding temperature [PCT] different by more than 50 °F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of absolute magnitudes of the respective temperature changes is greater than 50 °F."*

Moreover, the team concluded that even though the PCT change in SDR-301-V5 was less than 50 °F, SPC's treatment of the change was non-conservative.

The team therefore found that SPC failed to comply with 10 CFR 50.46 by not considering a change in PCT of 63 °F a significant change. This finding constitutes Nonconformance 99900081/97-01-03.

In addition, on April 4, 1997, the NRC issued Information Notice (IN) 97-15, "Reporting of Errors and Changes in Large-Break Loss-of-Coolant Accident Evaluation Models of Fuel Vendors and Compliance with 10 CFR 50.46(a)(3)." In that IN, the NRC staff made the following statements.

The SPC LBLOCA ECCS model, TOODEE2, was approved by the NRC staff to meet the requirements of 10 CFR 50.46 in a letter dated July 8, 1986. In 1991, SPC had made changes to the NRC-approved fuel cooling test facility (FCTF) reflood heat transfer coefficient correlation used in TOODEE2.

During August 1995, the NRC met with SPC about the LBLOCA ECCS evaluation model. As a result of that meeting, the staff sent a letter to SPC, dated November 13, 1995, that identified problems concerning changes in the TOODEE2 computer code specifically related to the 1991 changes to the NRC-approved FCTF reflood heat transfer coefficient correlation and the significance of the code changes. The staff then requested in a letter dated March 13, 1996, that SPC formally submit to the staff for its review and approval all model revisions and corrections implemented in TOODEE2 since the staff's approval of the code in July 1986.

On June 2, 1996, SPC submitted topical report XN-NF-82-20, "EXEM/PWR Large Break LOCA ECCS TOODEE2 Updates," Revision 1, Supplement 5, which described the updates made in the TOODEE2 computer code between 1986 and 1991. TOODEE2 is part of the evaluation model used by SPC for pressurized-water reactors. The staff has completed its review of this report and has concluded that the proposed LBLOCA-ECCS model (i.e., the 1991 model) is not acceptable and the previously approved model (i.e., the 1986 model) contains an unacceptable error. This information was formally communicated to SPC in a safety evaluation enclosed in a letter dated November 29, 1996.

After concluding that the 1991 model was unacceptable, the staff met with SPC and those licensees using SPC's LBLOCA evaluation model on October 16, 1996, to discuss the unacceptable error in the 1986 model. The staff also requested and received information from licensees that demonstrated that they were in compliance with 10 CFR 50.46 (see meeting summary dated November 5, 1996).

The IN also notes that although the LOCA analyses are performed by the fuel vendors, licensees are responsible for compliance with the regulations related to the LOCA analysis, that is, 10 CFR 50.46(a). The staff's recent interactions with the licensees using the SPC's LBLOCA methodology (the review experience of the SPC LOCA evaluation model changes) indicate that licensees may not be closely monitoring the work of their respective fuel vendor. Licensees may not be performing adequate assessments of errors when they are aware of them.

Furthermore, licensees' audits of SPC's evaluation model changes appear to have been ineffective in identifying the technical inadequacy of the changes.

b.4 ***SDR-103-15***

SDR-103-15 documents a change to the COTRANSA code (controlled by EMF-608), which is used to perform plant anticipated operational occurrence (AOO) analyses. In reviewing this SDR, the inspection team discovered that one of the V&V cases was not technically adequate. Upon questioning the SPC employee responsible for COTRANSA maintenance, the team determined that an input error had been discovered and was corrected in the identified case, and SPC reanalyzed the case with satisfactory results. However, the documents available to the inspection team did not discuss this input error or its effect on the analysis.

The team therefore determined that SPC failed to document the input error and its effect on the analysis. These findings constitute example (2) of Nonconformance 99900081/97-01-02.

b.5 ***SDR-106-10***

In this portion of the inspection, the team evaluated SDR-106-10, which documents the re-compilation of the FLEX code (controlled by EMF-608) for use on SPC DEC UNIX platforms, as well as the release of the UNIX version of the code that replaced the earlier version designed for Cray Supercomputers. In this case, the inspection team discovered several small differences in the results between the UNIX and Cray versions of the code, which could lead to a reduction in the calculated PCT. The team discussed these differences with SPC employees, who claimed that it was their understanding that this difference resulted from differences in the floating point precision of the two computer systems. However, SDR-106-10 neither discussed nor confirmed this causal factor.

The team therefore determined that SPC failed to document and confirm its assumed causal factor, including consideration of variables such as compiler options. These findings constitute example (3) of Nonconformance 99900081/97-01-02.

b.6 ***RELAX and FLEX Codes***

In this portion of the inspection, the team evaluated SPC's V&V procedures related to modification of the RELAX and FLEX codes, the documentation of the modified code versions, as reported in ANF-91-048(P)(A), published in 1993. The team also compared this report to the primary documentation of these codes, XN-NF-980-19(P)(A), published in 1982.

The documentation of the modifications to the RELAX and FLEX codes showed that SPC altered the code solution procedure, substantially changed important T/H models, and added a number of new input options. The reported V&V of these changes consisted of a single systems calculation. No comparisons with experimental data were included, nor were there any test cases to show that models unaffected by the modification gave the same results as the previous version of the code.

The extent of the changes to the codes, together with the scant amount of V&V of these changes, led to the team's finding that SPC had implemented major modifications to the RELAX and FLEX codes without adequate V&V of the changes. These findings constitute example (4) of Nonconformance 99900081/97-01-02.

c. **Conclusions**

All codes and methodologies examined suffered from inadequate V&V. Therefore, the team characterized this problem as generic in nature. SPC attributed part of the code validation problem to the availability of adequate data. The team concluded that this may have been a problem when some of the codes were first developed in the 1970s, but confirmatory data currently exists for all major phenomena that occur in LOCA analysis. The team also found that SPC did not maintain an extensive set of test and validation problems that could be run every time a new code version was implemented, and the acceptance criteria were not specified. (Example 1 of Nonconformance 99900081/97-01-02)

The team concluded that since the codes and documentation were not developed under a QA program compliant with Appendix B to 10 CFR Part 50, SPC lacks verification that equations have been correctly programmed and that the documents are accurate. The team characterized this problem as generic in nature and the staff considers these issues generic *for all licensees that rely on the results of SPC's LOCA analyses*. As a result, the staff also questions the licensees' basis for accepting the accuracy of the results of SPC's LOCA analyses. Therefore, in addition to evaluating your response to this nonconformance, the staff will monitor your compliance with the NRC's requirements in Title 10, Part 21, of the *Code of Federal Regulations* (10 CFR Part 21).

The team concluded that EMF-608 provides inadequate V&V guidance, and placed too much burden upon the SPC employees performing the V&V reviews. In addition, EMF-608 did not clearly define management's expectations as to what constitutes a minimally acceptable review. The team determined that this condition directly contributed to the discovered instances in which the available documentation failed to demonstrate that modified codes performed as intended. More importantly, the team concluded that this condition indicates that the V&V process documented in EMF-608 failed. (Example 1 of Nonconformance 99900081/97-01-04)

During this portion of the inspection, the team identified several examples that constitute Nonconformance 99900081/97-01-02.

SPC informed the team of their intent to replace EMF-608 with a new WP P104,121, "Control of Computer Codes." The new WP was scheduled to be issued by June 30, 1997.

3.3.3 BWR Safety Analysis and Thermal-Hydraulic Codes

During this inspection, the team evaluated SPC's development and documentation of codes, the reflood in BWR LOCA analyses, and code input decks.

3.3.3.1 *Development and Documentation of Codes*

a. Inspection Scope

In this portion of the inspection, the team evaluated the SPC code development procedures specified in QA topical report EMF-1, Revision 28; QAP-21, "Computer Software Control," Revision 0, dated August 22, 1996; and EMF-608, "Computer Code Control Requirements," Revision 12, dated December 1995. The related code development documents included the required software requirements specification (SRS), software design description (SDD), and/or software verification and validation results (SVVR).

b. Observations and Findings

On the bases of this inspection, the team determined that the only ECP code developed after SPC implemented the recently revised QAPs was a fuel mechanical code, TRULOAD (not a safety analysis code). The inspector's review of this code did not yield any significant findings. It is noteworthy, however, that SPC has computer software procedures to control the development and verification of new codes, but does not have any procedures in place to bring the older codes (some of which were obtained from outside organizations) into compliance with present QA standards (i.e., there is no "backfit" provision). The team considered this to be a weakness of the procedures.

SPC had also recognized the problem in self-assessment EMF-1924 that recommended that a program be put in place to review and update documentation of codes and user guidelines on a prioritized basis.

The team also found that SPC had not adequately updated the code manuals and, in many cases, comprehensive code manuals did not exist. The completeness of the code development documentation appeared to depend upon the individual developer.

For example, the team found that the RELAX code documentation, which is one of the most important codes in the LOCA analysis (for both BWRs and PWRs), was scattered in three volumes of manuals and many SDRs (software development records). While the code changes were documented in the SDRs, these changes were not incorporated into the code manuals in a timely manner. Specifically, the publication dates of the three volumes were 1975 (WREM, NUREG-75/056), 1982 (XN-NF-80-19(P)(A): Volume 2A), and 1991 (ANF-91-048(P)(A)). It was difficult to determine the details of these codes, since no systematic records exist with regard to the SRS, SDD or SVVR, which QAP-21 requires for recently developed code.

In the BWR LOCA methodology, the analyst was responsible for determining the reflood onset time. The analyst did this by examining the plots of relative and absolute entrainment of droplets at the peak temperature node in the fuel bundle, but *each analyst had his own criteria for determining this time*. This results in the prediction of the peak cladding temperature (PCT) being dependent on the analyst even though the computer codes and base input would be the same. SPC is in the process of developing and implementing user guidelines to remove arbitrary user judgement from the process. In addition, the analysts performing the LOCA safety analysis did not appear to be as knowledgeable about BWR LOCA methodology as they should; this may, in part, be a result of the lack of complete and up-to-date code manuals.

The team also noted that the entrainment model in the FLEX code contained inconsistent documentation between SDRs and the original reference. In addition, SPC had no documentation or assessment of how well the code predicted entrainment, even though it could have a large impact on PCT.

The team also determined that certain references cited in code manuals were not maintained on site.

A review of SPC's code input guidelines indicated that they were not up to date. For example, one option in aused96 in the COTRANSA input deck was not documented in the input manual at all. Similarly, the team found that one of the options an analyst had selected (option-3 for the critical power correlation, card group 9 of the CONGEN/COTRANSA2 code) was not listed in the input guidelines manual. When interviewed by the team, the analyst could not explain what this specific option meant.

c. <u>Conclusions</u>

The team concluded that the lack of code documentation and its need to be brought up to date had resulted in the lack of use of the code manuals by the SPC staff. In addition, there are multiple input options for many BWR T/H code inputs, but there are no user guidelines given for what options should be used in the analysis.

All codes and methodologies examined by the team exhibited inadequate documentation of the analytical code and selection of input parameters for the code, although some codes were better documented than others. Therefore, the inspection team characterized the documentation problem as a generic weakness.

SPC had also recognized the problem (in self-assessment EMF-1924) and had recommended a program to review and update, on a prioritized basis, code documentation and user guidelines.

3.3.3.2 *Reflood in BWR LOCA Analyses*

a. <u>Inspection Scope</u>

In this portion of the inspection, the team reviewed SPC's analysis guideline, as presented in EMF-868(P), "BWR LOCA — Core Reflood Analysis," Supplement 1, Revision 5, March 14, 1997. This document indicated that the time of reflood is determined on the basis of results from the FLEX code.

b. <u>Observations and Findings</u>

SPC's analysts use the entrainment calculated by FLEX to establish the time of reflood, which is then used as an input to the HUXY code. The original criterion for determining the time of reflood included the use of a relative entrainment fraction. However, the team found that no guidance was given to the analyst to ensure consistency in the selection of the appropriate entrainment fraction to establish the time of reflood.

SPC then changed the process for determining the time of reflood, requiring not only a relative entrainment fraction, but also an "absolute" criterion determined on the bases of the mass velocity of entrained liquid. During the inspection, SPC provided the team with proposed corrective actions to several of the issues identified by the team. In an internal memorandum, "Response Packages for Issues from NRC Inspection Week 1," March 14, 1997, SPC provided new guidance for analysts to use in selecting the reflood time on the bases of the "absolute" entrainment criterion. Specifically, the new guidance stated that reflood was established the first time that the entrainment mass velocity reached the specified value. SPC proposed to incorporate the new guidance as Section 6.7, "Reflood Criteria," in EMF-868(P), Supplement 1, Appendix C.3, Revision 1.

The team found, however, that the analysts still have no guidance for establishing the time of reflood using the "absolute" criterion. This is inadequate because the reflood mass velocity is not a smooth, monotonically increasing function of time. Instead, the reflood mass velocity is subject to significant oscillations, spikes, and other unsteady behavior. The team concluded that the new guidance was still not adequate in that it did not appear to account for the possibility of a "spike" in the entrainment mass velocity (during which the value would momentarily exceed the specified value and then fall below that value for an extended time).

Analysts require guidance to establish the criteria for the steadiness of the entrainment mass velocity and for a minimum duration over which the mass velocity exceeds the "absolute" value used to determine the onset of reflood. The time of reflood is an important parameter in BWR LOCA analyses, and can have a significant impact on the predicted PCT.

In addition, the team noted that Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50 requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings. As such, the team concluded that the absence of guidance regarding the selection of the appropriate entrainment fraction to establish the time of reflood constituted example (3) of Nonconformance 99900081/97-01-04.

c. <u>Conclusions</u>

The team identified a nonconformance involving the absence of guidance regarding the selection of the appropriate entrainment fraction to establish the time of reflood.

3.3.3.3 *Code Input Decks*

a. <u>Inspection Scope</u>

During this portion of the inspection, the team evaluated SPC's preparation of the input decks for the LOCA and transient analyses, including notebooks, sources of plant data, and analysis verification.

b. <u>Observations and Findings</u>

As a result of this evaluation, the team found that SPC had thoroughly documented and reviewed the preparation of plant input decks for the LOCA and transient analyses. The notebooks were standardized, well written, and easy to follow. Sources of plant data were adequately referenced; and the analyses were verified by an independent engineer. In addition, the review remarks were clear and properly discussed with the preparer of the data when significant disagreement occurred.

However, the team noted significant overlap between the plant input decks for the various safety analyses. For example, in many cases, the same input data is required for both the LOCA and plant transient analyses. To compound the problem, this common input data may be independently prepared by two (or more) different analysts, as in the case of the Susquehanna Unit 2 Cycle 9 reload core design. The team determined that this practice was a weakness in SPC's input deck preparation.

c. Conclusions

The team identified a weakness in the practice of preparing input decks because it created a potential source of inconsistency and confusion that could (and should) be avoided.

3.3.4 PWR Safety Analysis and Thermal-Hydraulic Codes

To evaluate the PWR safety analysis and T/H codes, the team reviewed the development and documentation of codes, the reflood in PWR LOCA analyses, and code input decks. The following paragraphs summarized the results of this review.

3.3.4.1 *Development and Documentation of Codes*

a. Inspection Scope

During this portion of the inspection, the team evaluated SPC's code development activities and the related procedures specified in QA topical report EMF-1, Revision 28; QAP-21, "Computer Software Control," Revision 0, dated August 22, 1996; and EMF-608, "Computer Code Control Requirements," Revision 12, dated December 1995. The related code development documents included the required SRS, SDD, and/or SVVRs.

b. Observations and Findings

In conducting this evaluation, the team discovered numbers in calculation notebooks that had been corrected with "whiteout" and written over without any indication of who made the change to the notebook, or why.

In addition, the team found that SPC had not adequately updated code manuals and, in many cases, comprehensive code manuals did not exist at all. Moreover, the team noted that the completeness of the code development documentation appeared to depend upon the individual developer. In one case, a reviewer questioned an unreferenced hydraulic diameter in an ANF-RELAP transient deck calculation notebook but then accepted it because the same number was used in the ANF-RELAP small-break loss-of-coolant accident (SBLOCA) input deck. The team later found that the value in the SBLOCA input deck was incorrect.

In some cases, the team also found that SPC's input decks used flow areas that did not correspond to the physical flow areas as work-a-rounds to overcome computer code problems; however, SPC failed to document either the use of these discrepancies as work-a-rounds, or the basis for the chosen value. In addition, SPC's user guidelines did not address such situations.

To evaluate the development and documentation of codes, the team reviewed code modeling, code documentation, and the application of the QA program to codes and modifications. The following paragraphs summarize the results of this review.

b.1 *Code Modeling*

In order to perform the plant safety analyses, NRC expects licensee's analyses to be conservative and in compliance with applicable regulations and code use restrictions, and that the results of those analyses are reported accurately to the NRC, per the requirements of 10 CFR 50.46. Consequently, during this portion of the inspection, the team conducted a detailed evaluation of the plant models and codes used in the PWR safety analyses. This evaluation yielded the following observations with respect to the code modeling:

- SPC used a two-phase pump model derived on the basis of Semiscale pump data. While this was probably the only data available at the time the model was developed and approved by the NRC, the model does not strictly satisfy the requirements of Appendix K to 10 CFR Part 50, which state that the pump model for the two-phase region shall be verified by applicable two-phase pump performance data.

  While this NRC-approved model appeared to be conservative, the team determined that the LOCA evaluation model did not comply with Appendix K to 10 CFR Part 50. Since this issue will require further consideration by the NRC staff, the team is treating this issue as Unresolved Item 99900081/97-01-06.

- SPC used multiple codes with different models for the same phenomena. The team therefore determined that this was a source of inconsistency and confusion, and, in general, a weakness in the code modeling system.

- The LOCA code has reflood heat transfer correlations based on 17 × 17 fuel cooling test facility (FCTF) data. However, the team determined that SPC had not developed the basis for the application of these correlations to different fuel designs involving changes in pressure, reflood velocity, initial PCT, peaking, spacer and tie plate design, and hydraulic diameter. Consequently, the team considered this portion of the issue a weakness in the code modeling. (See Section 3.3.4.2 of this report, as this observation relates to Unresolved Item 99900081/97-01-07.)

b.2 ***Code Documentation***

During this portion of the inspection, the team found that SPC evaluated the performance of safety systems and determined reactor conditions during design basis accidents using a complex system of computer codes designed for simulating reactor transients. The documentation of these codes was expected to include a detailed description of the model formulation, constitutive relationships, limitation of the models and numerics.

In LBLOCA applications, SPC used a suite of codes following the modeling requirements in Appendix K to 10 CFR Part 50. The team made the following observations with respect to the documentation of these codes:

- Several codes and their associated documentation were obtained from outside sources. SPC had made many subsequent code modifications including error corrections, model changes, and changes in computing platforms, as documented in the SDRs. In the case of EXEM/PWR LOCA, there have been 52 revisions of the model. The large number of revisions indicated a need for periodic upgrade of the code documentation to include all of the modifications. However, the format of the code documentation makes it very difficult to determine what is in the code without going back to the original documentation and tracking each of the SDRs describing the subsequent changes. This became especially clear during interviews with the SPC staff when the new staff was observed to have substantial difficulty in identifying the status of the code. The team considered this to be a weakness in the code documentation.

The team attempted to review the documentation of the various LOCA codes; however, many reports were not readily available. Table 1, "Missing Documentation," list reports requested during the inspection and the missing supporting documentation that SPC could not retrieve within twenty-four hours, as required in QAP-16, "Quality Records and Archive Samples," Revision 0, dated August 22, 1996. The team concluded that the required code documentation was not readily available and was not being used by the analysts. The team therefore considered this to be a weakness in the code documentation.

Table 1 — "Missing Documentation"

Code	Development Source	Requested Document	Comments
LOOPT	SPC	XN-NF-83-107(P), May 1984	no documentation of code or V&V
BLOCK	SPC	from guideline document EMF-1238(P)	no documentation of code or V&V
SHAPE	SPC	EMF-1238, EMF-CL-065, May 1996	no documentation of original code or V&V
RODEX2	SPC	available V&V documentation	no documentation of original code; V&V provided in new SDRs
RELAP4/ MOD3	INEL	available V&V documentation	no V&V performed when received in house
RELAP4/ HOTCHANNEL	INEL	available V&V documentation	no V&V performed when received in house
FISHEX	SPC	XN-NF-941, January 1981	verified in 1984
SIS	INEL	available V&V documentation	no documentation of original V&V
ICECON	SPC	XN-NF-942, June 1977	no documentation of model or V&V
PREFILL	SPC	XN-NF-CC-44, December 1977	no documentation of original code or V&V
REFLEX	SPC	EMF-78-30, May 1979	no documentation of original code or V&V
TODEE2	SPC	EMF-CC-072P, January 1997	documentation of code available but no record of V&V

- SPC used a complex suite of interfaced codes to perform the LOCA analyses. The transfer of information between these codes is through an automated file transfer process, which minimizes the possibility of introducing errors. The team considered this a strength.

- The available code documents, reviewed by the team did not include the complete code documentation describing the models, assumptions, range of applicability, and uncertainty in the correlations used in the models required by EMF-608. The team considered this to be a weakness in the code documentation.

b.3 *QA of Codes and Modifications*

After reviewing both SPC's QA for the codes and the SDRs documenting the code changes, the team made the following observations:

- The EMF-608 procedure only applied to code development performed since the procedure was established; it did not include any backfit provision. The team considered this a weakness of the ECP codes.

- The code modifications are made under SPC's QA procedures, which include line-by-line verification of the coding changes by an independent reviewer. The team considered this a strength.

- The codes acquired from sources external to SPC did not undergo V&V to ensure that the codes have the models described in the documentation and that they have been correctly implemented. SPC does not have a QA procedure for codes acquired from external sources. See the nonconformance cited in Section 3.3.2.b.1 of this report.

c. Conclusions

The team concluded that the lack of code documentation and the need to update existing documentation had resulted in the lack of use of the code manuals by analysts.

In addition, there are many code inputs with multiple input options, but there are no user guidelines given for what option(s) should be used in the analysis.

All codes and methodologies examined by the team exhibited inadequate documentation of the analytical code and the selection of input parameters for the code. Therefore, the inspection team characterized the documentation problem as a generic weakness.

SPC had also recognized the problem (in self-assessment EMF-1924) and had recommended that a program be put in place to review and update documentation of codes and user guidelines on a prioritized basis.

3.3.4.2 *Reflood in PWR LOCA Analyses*

a. Inspection Scope

In this portion of the inspection, the team reviewed substantial documentation concerning PWR reflood heat transfer testing in the FCTF. This testing was performed to support development of correlations for use in SPC's EXEM/PWR ECCS evaluation model for analysis of LBLOCAs.

b. Observations and Findings

To evaluate the reflood in PWR LOCA analyses, the team reviewed the FCTF scalability and the FCTF data analysis. The following paragraphs summarize the results of this review.

b.1 *FCTF Scalability*

On July 8, 1986, the NRC issued a safety evaluation report (SER) in which the staff approved SPC's PWR LBLOCA evaluation model. (At that time, SPC was Exxon Nuclear Company). The approved model included a reflood heat transfer correlation derived through FCTF testing in a bundle with a 17 x 17 rod array geometry. It also included a scaling methodology to apply the results of the correlation to other geometries. As a condition of its approval, the staff confirmed SPC's commitment to perform additional FCTF testing on a 15 x 15 rod array geometry bundle to verify the applicability of the scaling methodology.

In its letter to the NRC dated October 21, 1992, SPC informed the NRC staff that it had attempted to run two test programs in 15 x 15 bundles, but had not succeeded in obtaining valid data. SPC requested that the NRC agree to an alternative approach, in which SPC would assess the scaling methodology using data from other testing programs that were available in the open literature. The NRC responded in a letter dated August 2, 1993, stating that SPC's proposal was acceptable, as follows:

> *We agree that data from the [alternative test programs] can be used as a means to verify extension of 17 x 17 FCTF-based correlations and find that the [alternative] verification that [SPC has] proposed satisfies the [SPC] commitment identified in the NRC SER of July 8, 1986.*

To determine the intent of this statement, the team discussed the wording of the letter with the NRC staff member who originally drafted the letter. In particular, the team asked whether the staff expected SPC to submit the results of the scaling verification for review, or if the verification was simply to be performed but not submitted to the NRC. In the latter case, SPC would retain the documentation for possible future NRC review or audit. The team learned that the intent of the letter was, in fact, the second option, and there was no implied requirement for SPC to submit the verification study.

The team found, however, that SPC had not promptly performed the scalability verification after receiving the letter of August 2, 1993. Rather, SPC appeared to have undertaken the verification more than *3 years* after receiving the letter, beginning in late 1996; SPC completed its draft scalability verification in January 1997.

The team reviewed SPC's draft report on the scalability of the FCTF reflood heat transfer, and found that it did not contain sufficient information to support the use of the 17 x 17 rod array correlation for other geometries. Therefore, the inspection team and NRC staff considers this issue an unreviewed modification to an NRC-approved LBLOCA methodology, per 10 CFR 50.46 and Appendix K to 10 CFR Part 50. The staff has requested additional information (see the discussion under "Conclusions" of this section) to demonstrate that the FCTF reflood heat transfer correlation, as currently implemented in SPC's LOCA model for fuel designs other than the 17 x 17 rod array geometry, produces conservative results when compared to appropriate, applicable data. Consequently, the team is treating this issue as Unresolved Item 99900081/97-01-07.

b.2 *FCTF Data Analysis*

During this portion of the inspection, the team reviewed a series of internal memoranda prepared in 1987 through 1989, when SPC was called Advanced Nuclear Fuels (ANF). Specifically, these memoranda discussed ANF's efforts to adapt the computer code used for data analysis in the original FCTF 17 x 17 rod array tests for use in the planned 15 x 15 rod array tests. The code, called LEPER, contained algorithms to derive heat transfer coefficients from temperature data.

A memorandum from a consultant for ANF regarding "Status of LEPER Code," dated May 18, 1987, included the following statement:

> *...code documentation should identify the governing equations, and provide sufficient detail to follow through a deviation (sic) of the equations to be programmed. It should be possible to check the code to verify that programming was done correctly. With such documentation, code changes could be made with confidence that the validity of the code would remain intact. Nothing approaching this exists for LEPER.*
>
> *As a result of inadequate documentation, it was necessary to read FORTRAN coding to gain an understanding of how LEPER works. This was not entirely successful. It was not possible to derive the equations of the solution algorithm by this process. Furthermore, within the algorithm, there are summations of terms that appear to have inconsistent units. This and other similar details make it impossible to have faith that the code is error free.*

The team did not attempt to ascertain what QA procedures were in place when this code was originally used to analyze the FCTF 17 x 17 rod array data. However, the heat transfer coefficients derived using this code were the basis of the reflood heat transfer correlation used in the NRC-approved PWR LBLOCA evaluation model. Consequently, the team determined that this issue raises questions regarding the adequacy and validity of the NRC's bases for approving the PWR LBLOCA evaluation. In fact, additional information is necessary to confirm the adequacy of the heat transfer coefficients used as the basis of the reflood heat transfer correlation.

A second memorandum from a consultant to ANF dated June 1, 1987, addressed the subject of "Document Differences in Results Obtained from the Current Version of LEPER and the Results Obtained in 1983." In particular, that memorandum described a reanalysis of selected data from three tests and comparison of the heat transfer coefficients thus derived to those obtained in the original data analysis. In general, the results of the new analysis showed that the "new" heat transfer coefficients exceeded the "old" ones. The author reached the conclusion, therefore, that "there is no reason to believe that the 1983 results and the 17 x 17 rod array correlation are non-conservative." However, at the time of this inspection, SPC had no evidence that a comprehensive reanalysis of the original data had been performed.

More than a year later, an internal memorandum dated October, 26, 1988, discussed the "(1) Verification of the new FCTF Data Processing Code; [and] (2) Benchmarking of the 17 x 17 rod array FCTF Heat Transfer Coefficient Data Against Corresponding Heat Transfer Coefficient Output from the New FCTF Data Processing Code for Scaling Verification Program." In particular, that memorandum included two specific recommendations regarding analysis of FCTF data.

First, the memorandum recommended that whatever new code is chosen for the planned tests "needs to be verified to conform to ANF's Quality Assurance program." Secondly, the memorandum recommended that selected data from four of the original 17 x 17 rod array tests should be reanalyzed to benchmark the new code. During this inspection, the team determined that ANF implemented the first recommendation (see below), but there is no similar evidence that ANF even followed the second recommendation.

SPC subsequently gave the team another memorandum dated March 20, 1989, which discussed the "Incorporation of XRAD Algorithm into FCTFRED." In particular, that memorandum indicated that ANF had completed the new FCTF data processing code and placed it in a controlled directory. The team interpreted this statement as indicating that the new code had been placed under the oversight of ANF's QA program. The new code was "benchmarked" against the original 17 x 17 rod array test data, using three data points from one test.

The team determined that SPC's failure to re-analyze the original 17 x 17 rod array data, using a verified code under appropriate QA controls, was a significant technical issue. The team also found that there was no systematic evidence that the FCTF reflood heat transfer correlation, either as originally developed or as modified and currently implemented, produced conservative results. Consequently, the team considered these issues as Unresolved Item 99900081/97-01-08.

To address this issue, the NRC expects SPC to provide documentation that the data used to develop the 17 x 17 rod array correlation were, in fact, analyzed using a valid data analysis code. In the absence of this evidence, the NRC expects SPC to perform a complete re-analysis of the data to demonstrate that the reflood heat transfer correlation, as currently implemented, produces conservative results. The NRC staff will review SPC's response to determine whether this issue has been properly resolved.

c. Conclusions

The lack of documentation that the FCTF reflood heat transfer correlation produces conservative results constitutes a significant issue for which the staff requires additional information. The NRC staff will handle both the scaling verification issue and the FCTF data analysis issue as unresolved items. A letter from J.E. Lyons (NRC) to H.D. Curet (SPC) dated May 7, 1997, conveyed the staff's "Request for Additional Information (RAI) Regarding Reflood Heat Transfer." That RAI identified the specific information required to address these unresolved items.

3.3.4.3 *Code Input Decks*

a. Inspection Scope

To evaluate the code input decks, the team reviewed SPC's preparation of the input decks for the LOCA and transient analyses, including notebooks, sources of plant data, and analysis verification. The following paragraphs summarize the results of this review.

b. Observations and Findings

In addition to the development and modification of computer codes, the preparation of the input decks is another critical aspect of safety analyses. The team found that SPC developed input decks in cooperation with the licensee. This process involved the following four steps:

- collecting plant data
- processing the data to obtain relevant parameters such as volumes and areas

- developing the input deck with the plant-specific data and relevant code options
- testing the deck and nodalization

To evaluate SPC's performance of these four steps, the team reviewed the following input decks, which SPC developed in cooperation with other licensees:

- L&SE-814 E-5146-868-1 and 2, dated September 5, 1989
- L&SE-665 E-7525-963, dated July 31, 1989
- L&SE-929 E-0111-2-29, dated November 9, 1989

The team noted that the input deck preparation had been well documented, checked, and signed by an independent engineer. The team, therefore, considered this a strength. However, the team also identified instances in which SPC used "whiteout" to change the numerical values. This violated SPC's established QA procedures, which stated that the proper procedure for correcting errors is to strike the error with a line and rewrite it.

The team also noted that CP&L had performed a QA review and commented (L&SE-814 E-5146-868-1 and 2, September 5, 1989) about SPC's use of a constant accumulator flow instead of a mechanistic determination of the accumulator flow. However, SPC did not provide any justification for this assumption. The team therefore concluded that this assumption constitutes a weakness in SPC's application of the QA program for input deck preparation.

c. <u>Conclusions</u>

SPC's input deck preparation, documentation, and review was considered a strength. However, the team identified a weakness associated with SPC's failure to provide justification for the use of constant accumulator flow.

3.3.5 Critical Power Correlation

In this portion of the inspection, the team reviewed SPC's analyses using ANF-1125(P)(A), "ANFB Critical Power Correlation," Supplements 1 and 2, dated April 19, 1990. The team focused on evaluating the applicability of the critical power correlation to the BWR ATRIUM™-9 and ATRIUM™-10 fuel designs[3]. According to SPC, its developmental goal was to establish a generic dry-out correlation for current and new SPC fuel designs and reloads with coresident fuels supplied by other

[3] ATRIUM™ is a trademark used by SPC to refer to fuel designs with a large, square internal water channel/canister (typically replacing a 3 x 3 fuel rod array). Before this inspection, however, SPC had used this design for NRC licensees with various 9 x 9 fuel rod arrays identified by various designations that may or may not include the word ATRIUM™.

vendors. To achieve this goal, SPC's approach was to separate the generic T/H effects from the design-specific local geometry effects. Nonetheless, the ANFB critical power correlation is dependent on the fuel design parameters. The resulting correlation is used to determine whether the safety-limit and operating-limit minimum critical power ratio (SLMCPR and OLMCPR) for a particular fuel reload meet the Technical Specifications (TS) for the plant. The acceptance criterion is that only 0.1% or less of the fuel rods may experience boiling transition, as specified in NUREG-0800, the NRC's Standard Review Plan (SRP).

The team found that the ANFB correlation constituted an empirical fit to the relevant set of boiling transition data obtained in electrically heated test bundles modeling a particular nuclear fuel design. SPC used statistical methods to determine the uncertainty of the fit between the correlation and the data set, and to quantify the contribution of that uncertainty to the SLMCPR.

In addition, the team examined the applicability of the ANFB correlation to different fuel geometries. To do so, the team evaluated the manner in which SPC obtained the empirical coefficients of the correlation for the ATRIUM™-9 and ATRIUM™-10 fuel designs. (The NRC-approved methodology for developing the base ANFB correlation[4] was described in ANF-1125(P)(A), Supplements 1 and 2.)

After evaluating these reports in detail and holding numerous conferences and discussions with SPC staff, the team developed the following generic concerns and observations related to the SPC data sets obtained to support application of the ANFB correlation to new fuel designs:

(a) The data set(s) modeling a new design must span the intended range of application of the new fuel design, including mass, velocity, pressure, inlet subcooling, axial power profiles, and radial power peaking.

(b) SPC should examine the data for significant biases with respect to each of the independent parameters used in the correlation, and for sub-regions the correlation where the fit between the correlation and the data sets shows greater variance than the variance in the fit over the data set as a whole.

[4] The ANFB correlation has the following sets of additive constants, for the fuel designs included in the original correlation database, as documented in ANF-1125(P)(A). The database for each set of additive constants contains 1185 data points for 8 x 8 fuel array; 1336 data points for 9 x 9 fuel array (with small water holes); and 320 data points for 9 x 9 fuel array (with 3 x 3 central water hole).

(c) SPC has exhibited a practice of taking only a small amount of data (i.e., generally less than 20% of the total data set) at low flow rates and pressures other than 6.89 MPa (1000 psia). This practice introduces significant inconsistencies in the coverage of the database, which can affect the validity of statistical analyses performed using the parameters (such as the "additive constants") derived from the fit between the correlation and the data set. This practice raises questions concerning the validity of the safety-limit derived using such statistical parameters and methodologies.

To evaluate SPC's BWR critical power correlation, the team evaluated the adequacy of the ANFB critical power correlation and the adequacy of its application to the ATRIUM™-10 fuel assemblies designed for the PP&L, Susquehanna Unit 2 Cycle 9 reload.

On the basis of its findings regarding SPC's failure to verify the adequacy of the ANFB critical power correlation and the adequacy of its application to the ATRIUM™-10 fuel design, the team chose to evaluate the adequacy of the ANFB correlation to the ATRIUM™-9 fuel design. The following sections discusses the team's evaluation of the application of the ANFB critical power correlation to the ATRIUM™-9 and -10 fuel designs.

3.3.5.1 *Application of ANFB to ATRIUM™-10*

a. <u>Inspection Scope</u>

In this portion of the inspection, the team evaluated the methodology used to develop the ANFB correlation, as described in ANF-1125(P)(A), Supplements 1 and 2. In addition, the team assessed SPC's description of the correlation and its application to the ATRIUM™-10 fuel design[5], as presented in EMF-97-010(P), "Application of ANFB to ATRIUM™-10," Revision 0, dated January 1997. That report described a series of critical power tests and the adjustments that SPC made to the ANFB correlation in order to establish the basis for application of the correlation to the ATRIUM™-10 fuel bundle design.

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[5] The database for the set of "additive constants" to extend the ANFB correlation to ATRIUM™-10 fuel consisted of 620 data points.

b. **Observations and Findings**

On May 4, 1995, during its ATRIUM™-10 fuel design presentation to the NRC staff, SPC made the following assertions:

- SPC evaluated the transient tests and found that they demonstrated acceptable behavior of the ANFB correlation for the ATRIUM™-10 fuel design.

- The specific fuel design analyses (e.g., the mechanical analyses, stability evaluation, and thermal-hydraulic compatibility analyses) comply with the NRC-approved generic boiling-water reactor (BWR) design criteria in ANF-89-98(P), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1, April 1990.

- SPC concluded that no additional NRC review was required.

Subsequently, in January 1997, SPC submitted EMF-97-010, "Application of ANFB to ATRIUM™-10 for Susquehanna Reloads," Revision 0, for NRC review. The purpose of EMF-97-010 was to describe the dry-out testing of the ATRIUM™-10 design and the application of the ANFB correlation to the results of that testing, as they applied to the Susquehanna Unit 2 Cycle 9 reload.

In reviewing EMF-97-010, Revision 0, the team found that the ANFB critical power tests were performed for a cosine axial power shape, and included a set of 12 radial power shapes with a range of power, flow, pressure, and inlet subcoolings. On the bases of available industry data and the results from these initial tests, SPC performed additional tests for both an upskewed and downskewed axial power distribution. The report concluded that the ANFB correlation may be applied to the ATRIUM™-10 reloads at Susquehanna for use in approved methodologies for the design, safety, and monitoring analyses.

Contrary to SPC's assertions during the presentation to the NRC staff on May 4, 1995, and SPC's conclusions in EMF-97-010, however, the inspection team identified significant failures in SPC's proposed application of the ANFB correlation to the ATRIUM™-10 fuel design used in Susquehanna Unit 2 Cycle 9 reload, as follows:

- The approved range of the ANFB correlation, as described in SER condition 3.3(1) of ANF-1125(P)(A), Supplement 1, and in the test data prescribed by EMF-97-010(P) Revision 0, limits the application of the ANFB correlation to local pin peaking factors (F_L) of ≤ 1.3. However, for PP&L's Cycle 9 reload at Susquehanna Unit 2, SPC's reload design had peaking for certain bundles in excess of the limit of $F_L = 1.3$. (The Susquehanna ATRIUM™-10 reload was the first reload application where the ANFB correlation was applied to a fuel assembly with part-length fuel rods (PLFRs).)

- As formulated, the ANFB correlation does not provide an acceptable method for predicting critical power in the ATRIUM™-10 fuel design. This is because the ANFB correlation nonconservatively over-predicts the upskewed critical power test data at low flows. This systematic critical power over-prediction or bias at low flows is substantial and is outside the NRC-approved SLMCPR methodology, as described in ANF-524 (P)(A), "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," Revision 2, dated April 19, 1989.

- The correlation contains a flow-dependent bias that is not allowed by the NRC-approved SLMCPR determination methodology.

- The mean and standard deviations of the correlation's fit to its data depend on an axial power shape, which is not allowed by the NRC-approved SLMCPR determination methodology.

- The set of "additive constants" were modified by *adhoc* corrections to achieve the desired fit. This approach is not permissible with the approved methodology, as described in ANF-1125(P)(A).

The inspection team also evaluated recent upskew critical power test results on the ATRIUM™-10 fuel design. On the bases of that evaluation, the team concluded that the ANFB critical power correlation would need to be re-compiled to more accurately predict the fuel bundle critical power attributable to the PLFRs used in the ATRIUM™-10 design.

On the basis of findings from the review of the ANFB correlation and its applicability to the ATRIUM™-10 design, and after reviewing all of the additional data and presentations provided by SPC, and considering several detailed discussions with the SPC staff, the team reached the following conclusions:

- The application of the ANFB correlation to ATRIUM™-10 fuel assemblies used in the Cycle 9 reload at Susquehanna Unit 2, with local peaking $F_L > 1.3$, was outside the approved range of the ANFB correlation.

- The use of the ANFB correlation, including the nonconservative flow-bias, in determining the SLMCPR and OLMCPR for the ATRIUM™-10 fuel assemblies used in the Cycle 9 reload at Susquehanna Unit 2, was outside the approved SPC methodology.

As a result of these issues, the team concluded that SPC failed to verify the adequacy of the ATRIUM™-10 reload design for Susquehanna Unit 2, Cycle 9. These findings are example (1) of Nonconformance 99900081/97-01-01.

In addition, the team determined that the data and analyses presented in EMF-97-010, Revision 0, show that the ANFB correlation needs additional development in order to be applicable to the ATRIUM™-10 fuel design. In particular, SPC needs to more accurately address the impact of PLFRs on bundle critical power behavior. This work will incorporate data from the ATRIUM™-10 tests reported in EMF-97-010, Revision 0, and any additional data SPC may obtain in further testing of this fuel design.

However, the team's findings caused the NRC staff to require Susquehanna Unit 2 to develop and obtain NRC approval of a critical power ratio (CPR) penalty and corresponding TS change before the Cycle 9 startup.

SPC has since revised the ANFB/ATRIUM™-10 methodology in EMF-97-010(P) in order to address these concerns. The NRC subsequently reviewed and approved this revision (Revision 1, March 1997) specifically for application to Susquehanna Unit 2, Cycle 9. SPC has also indicated that it will provide additional submittals to resolve these concerns generically.

c. <u>Conclusions</u>

The team identified a nonconformance regarding the failure of the ANFB correlation to adequately support the ATRIUM™-10 fuel design. In particular, the team cited SPC's failure to verify the adequacy of the ATRIUM™-10 reload design for Susquehanna Unit 2 Cycle 9.

SPC's interim resolution of this issue has been incorporated in a Susquehanna cycle/TS-specific report (EMF-97-010, Revision 1).

3.3.5.2 ***Application of ANFB to ATRIUM™-9***

a. <u>Inspection Scope</u>

On the basis of the findings regarding the applicability of the ANFB critical power correlation to the ATRIUM™-10 fuel design (discussed in Section 3.3.5.1 of this report), the team recognized the need to evaluate the applicability of that correlation to the BWR ATRIUM™-9 fuel design.

Consequently, the team then evaluated the methodology used to develop the ANFB correlation, as described in ANF-1125(P)(A) Supplements 1 and 2. In addition, the team considered how SPC used that methodology to develop a set of "additive constants[6]" that would render the correlation applicable to the ATRIUM™-9 (9 x 9-IX) fuel design with ULTRAFLOW™ spacer grids, as reported in EMF-93-075(P).

b. Observations and Findings

The amount of data required to validate a new application of the ANFB correlation depends on the extent to which the correlation will be applied. The ANFB correlation reflects an empirical fit to data. As such, the correlation cannot be extrapolated to conditions outside its database. The NRC-approved methodology for developing the ANFB correlation used the common assumption that the coefficients of the correlation are constants with no functional dependence on the range of operating conditions. SPC therefore fitted the correlation to the data set using standard statistical techniques, and used the variance of the fit to assess the uncertainty of the predictions resulting from the correlation.

The team found that this process typically involved obtaining a completely new set of coefficients for a given database. The fit of the correlation therefore should then be examined over the full range of the database, to evaluate the "goodness of fit" and the validity of the underlying assumption that the coefficients can indeed be treated as essentially constant over the database. (In other words, the objective is to determine that there are no significant biases or trends with independent parameters, such as flow, pressure, or enthalpy.)

However, the team's evaluation revealed that SPC's methodology for developing the ANFB correlation differed from the NRC-approved approach. Specifically, SPC assumed that some of the coefficients (namely, the empirical coefficients on flow, pressure, and bulk enthalpy) would remain the same from one fuel design to the next. Therefore, SPC treated these coefficients as known constants in subsequent applications of the correlation to data sets representing new fuel designs. In particular, the team determined that SPC had assumed that all effects of the new fuel design on boiling transition behavior would be purely local, and would be captured by the "local conditions" term through the weighted local rod energy balance factor (F_{effo}) and the local corrections to this term, the "additive constants."

[6] The database for the set of "additive constants" to extend the ANFB correlation to ATRIUM™-9 fuel with ULTRAFLOW™ spacers consisted of 125 data points.

The team determined that the main advantage of this approach was that it made it easier for SPC to derive a new form of the ANFB correlation for a new fuel design. In essence, SPC initially determined a set of values for the coefficients of the independent parameters (i.e., flow, pressure, and bulk enthalpy — the A, B, and C terms reported in ANF-1125(P)(A)) for the base ANFB correlation. Then, in fitting the ANFB correlation to a new data set for a new fuel design, SPC simply used the same coefficients (without modification). The team found that SPC simply assumed that all of the differences between the original database and the new data set were captured by the process of fitting the correlation to the new data set to determine the set of "additive constants" for that fuel design.

The team determined that SPC's approach was technically sound. However, such approach requires that SPC validate the underlying assumptions about the character of the different coefficients for each new fuel design to which the correlation is applied. Instead, SPC assumed that coefficients related to the nominal operating conditions (i.e., flow, pressure, and bulk enthalpy), would remain constant and invariant despite changes in geometry and power distributions. As a result, SPC must closely examine the correlation to identify biases related to these parameters each time the correlation is applied to a new data set modeling a new fuel design. The NRC staff will followup on the results of SPC's examination of the correlation as Unresolved Item 99900081/97-01-09

Information subsequently presented to the team by SPC in response to this issue shows that significant variation may exist in the fit of the correlation with these independent parameters within subregions of the data set and for different fuel designs. The team found that this variation was particularly noticeable on the fringes of the data set (i.e., with low flow rate or high pressure). The team also determined that it was reasonable for SPC to construct data sets that are weighted to the nominal operating region of the core; however, it is also necessary to obtain sufficient data on the fringes of the data set to ensure that predictions derived from the correlation will also exhibit comparable accuracy in these regions. The team concluded that SPC's practice of taking only a small number of data points, usually in only one bundle, artificially minimized the effect of the data in these regions. Consequently, the team concluded that it was possible for SPC's correlation to significantly under- or over-estimate any lack of fit in these regions.

The team found that another deficiency of SPC's approach was that it was not possible to separate out the different components of the uncertainty in the fit to data. For a given data set, some of the uncertainty will arise from a lack of fit in the A, B, and C terms, and some of it will result from a lack of fit of the weighted local rod energy balance term (F_{eff0}) and the "additive constants" for the specific data set.

In addition, the team found that SPC's method of determining the "additive constants" deliberately induced conservative biases into the fit between the correlation and the specific data set. The team found that this was not inherently bad, but it confounded the determination of the true variance between the fit and the data set, which was the significant parameter in determining the uncertainty of the predictions derived from the correlation.

Because SPC assumed that coefficients related to the nominal operating conditions (i.e., flow, pressure, and bulk enthalpy), would remain constant and invariant despite changes in geometry and power distributions, the team determined that it was absolutely necessary for SPC to derive the set of "additive constants" for each new fuel design over the full range of intended applications in the operating reactor. Moreover, since the A, B, and C terms and the coefficients of F_{eff0} are not optimized for each specific correlation, any lack of fit relative to flow, pressure, and enthalpy that is not already captured by the generic correlation must be reflected in the fit to the new values of the "additive constants." The team also noted that SPC's response to this issue demonstrated statistically significant variations in the fit. These variations showed conclusively that these "generic" coefficients do not perfectly capture the effects for the ATRIUM™-9 fuel design, and clearly illustrated the need for full coverage of the data set.

Despite SPC's assertion of its "generic" nature, the team determined that the ANFB methodology was essentially an *adhoc* empirical critical heat-flux (CHF) correlation. As such, the data set for a specific fuel design must span the full range of applications of the ANFB correlation for that design. In addition, the correlation must adequately cover the entire data set in order to ensure proper characterization of the uncertainty of correlation predictions over the intended range of applications. In turn, such characterization is essential to ensure an accurate assessment of the "goodness of fit," and to ensure that the MCPR safety-limit actually meets the acceptance criteria. Consequently, the team determined that SPC had no tenable basis for its contention that a smaller data set was sufficient to characterize the statistical behavior of the correlation for a given application.

The team also noted the following generic observations concerning SPC's treatment of dry-out data when evaluating the applicability of the ANFB correlation:

- In the test series for the 8 x 8 rod array and the 9 x 9 rod array (pre-ULTRAFLOW™ spacer) fuel designs, the data sets did a fairly adequate job of spanning the full range of the correlation (in accordance with Table 1.1, of ANF-1125(P)(A), Supplement 1).

- The database had a severe shortcoming in its lack of upskew and downskew power profiles in any test bundles modeling the 9 x 9 rod array fuel designs, and in any such bundle modeling the 8 x 8 rod array fuel designs.

- The STS12 series data set (for the 9 x 9-IX (the -IX designation identifies the number of water rods; in this case, the 9 x 9 rod array has 9 water rods) rod array fuel design with ULTRAFLOW™ spacer) had serious shortcomings in terms of the range of data and the performance of the correlation over the tested range. These shortcomings raised certain questions concerning the basis for SPC's conclusion that the ANFB correlation was fully applicable to the ATRIUM™-9 fuel design.

On the basis of its evaluation of the ANFB correlation methodology used to develop a set of "additive constants" for the ATRIUM™-9 fuel design, the team determined that SPC used an insufficient number of test points and an inadequate range of conditions tested for the 9 x 9 rod array fuel designs with an internal water channel. As a result, the methodology failed to justify the uncertainty values for the "additive constants" used in determining the SLMCPR.

Lacking adequate justification for these values, SPC should have used larger uncertainties in the SLMCPR determinations. Specifically, the uncertainty values should have reflected the full operability range of the ATRIUM™-9 fuel design. Moreover, the team concluded that this finding may affect the SLMCPR of certain operating plants, and the resulting SLMCPR error would have affected the OLMCPR prescribed in the Core Operating Limits Report (COLR). Thus, the team documented example (2) of Nonconformance 99900081/97-01-01, which specifically addresses SPC's failure to use a sufficient number of test points and to test an adequate range of conditions to justify the uncertainty values for the "additive constants" used in the SLMCPR determination for the ATRIUM™-9 fuel design.

According to SPC, this finding may immediately affect the startup and TS-approved methods for the following plants operating with ATRIUM™-9 fuel:

- Commonwealth Edison Company
  - Quad Cities Unit 2 Cycle 15
  - Dresden Unit 3 Cycle 15
  - LaSalle County Unit 2 Cycle 8

- Washington Public Power Supply System
  - Washington Nuclear Unit 2 Cycle 13

- Taiwan Power
  - Kuosheng Units 1 and 2

On April 1, 1997, SPC issued NCR 6057, Revision 1, which stated that safety-limit evaluations for 9 x 9 internal water channel fuel designs may be nonconservative. In addition, NCR 6057 required customer notification for those instances in which SPC performed the safety-limit evaluation or provided information that may have been used by other vendors in performing the safety-limit evaluation. The team also asked SPC if it would perform an evaluation in accordance with 10 CFR Part 21.

On May 22, 1997, SPC issued "10 CFR [Part] 21 Notification of Defect in Critical Heat-Flux (CHF) Database Range," which included the following statement

> *. . . NRC determined that the CHF data base for the ATRIUM™-9 fuel design and other 9 x 9 fuel designs with internal water channels was not extensive enough to adequately estimate the uncertainties for the additive constants used in SPC CHF correlation. A statistical treatment of the existing relevant CHF data was developed to estimate the uncertainties beyond the original CHF data ranges. These estimated uncertainties are larger than the original additive constant uncertainties. Revised safety limit calculations with the larger additive constant uncertainties indicate that certain plants may have operated with incorrect safety limits.*
>
> *The affected licensees have been informed of the increase in additive constant uncertainties and of the possibility that previously calculated safety limits may have been impacted by the larger uncertainties. Safety limit calculations for current reactors with ATRIUM™-9 fuel design and other 9 x 9 fuel designs with internal water channels have been performed with the inclusion of projected higher additive constant uncertainties.*

c. <u>Conclusions</u>

The team concluded that SPC's ANFB correlation failed to adequately support the ATRIUM™-9 fuel design. In addition, the team issued a nonconformance citing SPC's failure to use a sufficient number of test points, and to test an adequate range of conditions to justify the uncertainty values for the "additive constants" used in determining the SLMCPR for the ATRIUM™-9 fuel design.

SPC has undertaken remedial action to assess the impact of increased uncertainties on operating plants. SPC has also committed to develop a supplement to the ANFB methodology to statistically establish the uncertainties consistent with the amount of test data.

The NRC staff is currently reviewing the cycle/TS-specific effects of this finding for each of the NRC-licensed plants before plant startup for cycles operating with ATRIUM™-9 fuel assemblies.

3.4 ATRIUM™-10 Fuel Design

The ATRIUM™-10 fuel design consists of a 10 x 10 rod array of fuel rods with a central water channel/canister, which also provides the structural support for the assembly by attaching to the upper and lower tie plates. This design differs from previous SPC BWR fuel designs, as illustrated by the following examples:

- smaller diameter fuel rods with higher heat flux

- large central water channel/canister that provides the structural support for the assembly, rather than 2 to 10 small water rods with 8 tie rods as the structural support

- part-length fuel rods (PLFRs)

- additional spacer grid at the bottom ends of the fuel rods to provide additional lateral restraint (because the fuel rod end caps did not engage into the lower tie plate as in previous BWR fuel designs)

In addition, a large compression spring supports the upper tie plate and water channel/canister in the ATRIUM™-10 design. (This differs from previous SPC BWR fuel designs, which used a compression spring at the top of each fuel rod for tie plate support.) Also, smaller flow holes in the lower tie plate grid help reduce the flow of debris through the assembly, which causes debris fretting that can result in failed fuel rods.

The team noted that ATRIUM™-10 represents a significant change from previous SPC BWR fuel designs. In particular, the team noted the changes made to the assembly configuration (i.e., the 10 x 10 rod array, central water channel/canister, and PLFRs), and the assembly structural support which impacts fuel rod and assembly performance.

a. Inspection Scope

The focus of this inspection was to examine SPC's thermal-mechanical analyses to verify that the ATRIUM™-10 design meets the applicable requirements, as defined in SPC topical report ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1, dated April 1991, and the related safety evaluation report (SER) prepared by the NRC.

General Design Criterion (GDC) 10 in Appendix A to 10 CFR Part 50 requires and Section 4.2 of the NRC's Standard Review Plan (SRP) specifies that a given fuel design must satisfy specified acceptable fuel design limits (SAFDLs). Specifically, the fuel design criteria in ANF-89-98(P)(A) identify the SAFDLs that apply to the ATRIUM™-10 fuel design.

The team also examined SPC's fuel surveillance program (i.e., use and post-irradiation examination of lead assemblies). In particular, the SRP specifies that the lead assemblies in a new fuel design should be subjected to a post-irradiation fuel surveillance plan. The objective of this surveillance, according to the SRP, is to determine the performance characteristics of the new design and/or to verify that the performance is not significantly different from that of previous designs. SPC used lead assemblies to verify their new designs and referred to them as lead fuel assemblies (LFAs).

In addition, the team noted that the ATRIUM™-10 fuel design had certain u[illegible] [illegible]el performance characteristics, including T/H performance and assembly growth. Section 3.3 of this report discusses the team's evaluation of SPC's T/H analytical methods and analyses used to determine the thermal performance margin of this new SPC fuel design.

b. <u>Observations and Findings</u>

The focus of the team's review of the ATRIUM™-10 fuel design was in the areas of SPC's thermal-mechanical analyses and LFA surveillance program. The following paragraphs summarize the results of this review.

b.1 *Thermal-Mechanical Analyses*

During this part of the inspection, the team evaluated whether SPC's thermal-mechanical analyses use appropriate NRC-approved models and methods to demonstrate that the ATRIUM™-10 design satisfies the SAFDLs in ANF-89-98(P)(A). Specifically, the team examined SPC's thermal-mechanical analyses and associated analytical methods related to cladding stress and strain, cladding collapse, fretting wear, corrosion, axial fuel rod and assembly growth, rod internal pressure, rod bowing, assembly liftoff, fuel melting, and seismic-LOCA loads. The team determined that SPC's analytical models and methods can be divided into three groups, as follows:

- those that depend on out-of-reactor and/or in-reactor test methods to demonstrate satisfactory performance

- those that use analytical models that may change with a design change (i.e., in-reactor data is needed from LFAs to develop a new analytical model, if necessary, to verify that the model applies to the new design)

- those that are generally independent of the fuel design

The results of SPC's thermal-mechanical analyses were discussed in EMF-95-52(P), "Mechanical Design Evaluation for Siemens Power Corporation ATRIUM™-10 BWR Reload Fuel," Revision 0, July 1995, in relation to the requirements in ANF-89-98(P)(A). However, that report did not discuss the analytical models and methods used to obtain the documented results. The lack of this information in the topical report made it difficult for the team to verify that SPC used NRC-approved models and methods in the ATRIUM™-10 analyses. Consequently, the team considered the lack of information a weakness in SPC's documentation.

On the basis of the results documented in EMF-95-52(P), the team found that the analytical models that require out-of-reactor testing included fretting wear tests (in-reactor visual data from LFAs were also used to verify satisfactory fretting performance), flow tests to support assembly liftoff analyses, and spacer grid failure load tests to determine seismic-LOCA load limits. The team also determined that the out-of-reactor testing in these areas demonstrated satisfactory performance of the ATRIUM™-10 fuel design (i.e., the performance of the new design was equal to or better than that of previous SPC BWR designs).

For those analytical models that require in-reactor data from LFA tests to verify the applicability of the model to the ATRIUM™-10 fuel design, the team determined that the analytical models included cladding corrosion, axial rod and assembly growth, and rod bowing. SPC provided to the team LFA fuel rod data from a 10 x 10-8 rod array LFA that had dimensions and cladding similar to that of the ATRIUM™-10 fuel design. According to SPC, these data demonstrated that the ATRIUM™-10 design exhibited cladding corrosion, rod growth, and rod bowing similar to those of previous 9 x 9 rod array and 8 x 8 rod array SPC fuel designs. On the basis of its evaluation, the team determined that SPC's performance models (which rely on data from the SPC 9 x 9 rod array and 8 x 8 rod array fuel designs) appear to be applicable to the ATRIUM™-10 fuel design

The team also determined that SPC used a model they received from Siemens Europe to calculate the axial assembly growth of the fuel channel for the ATRIUM™-10 fuel design. In addition, SPC assumed that the axial growth of fuel channels from the 9 x 9 rod array and the 8 x 8 rod array assemblies were the same for the growth of the ATRIUM™-10 assembly because both components (earlier fuel channels and the ATRIUM™-10 water channel/canister) have the same material (fully recrystallized zirconium (Zr) alloy (zircaloy-2 or zirc-2)) and operating conditions (stress and fast flux). SPC therefore applied the European assembly growth model to the

ATRIUM™-10 design in a manner similar to how it was applied to the 9 x 9 rod array and the 8 x 8 rod array axial assembly growth models for calculating fuel rod-to-tie plate clearances and assembly upper nozzle clearances with reactor internals at end-of-life (EOL).

However, the team determined that the assembly axial growth models that SPC used for the earlier 9 x 9 rod array and the 8 x 8 rod array designs were not applicable to the ATRIUM™-10 fuel design. The distinction arose because the design of the structure changed (from tie rods to central water channel/canister) and the materials used for the structure changed (from cold worked/stress-relief to fully recrystallized zirc-2) to cause the axial growth to be significantly less in the ATRIUM™-10 design than in the previous SPC BWR fuel designs.

In its SER for topical report ANF-89-98(P)(A), the NRC stated that if the NRC-approved axial growth models were not applicable because of a particular design change, SPC must submit a revised growth model for NRC review and approval. However, the team determined that SPC failed to submit for NRC review and approval the new assembly growth model revised to address the ATRIUM™-10 fuel design as stated in the SER.

In an internal memorandum (AR:97:016), "Growth Models Used by SPC in the ATRIUM™-10 Evaluation," dated February 21, 1997, SPC explained that it had not implemented a model change because its analysis of the ATRIUM™-10 fuel design used methodology that was previously applied to other NRC-approved fuel designs. After reviewing SPC's position, the team determined that SPC's position was not valid because there was a distinct difference between the axial growth model and the analysis methodology that applied this model. The team also determined that SPC was required to submit a new model for ATRIUM™-10 assembly growth, as stated in the SER, because none of SPC's previously approved axial assembly growth models were applicable to the ATRIUM™-10 fuel design.

Consequently, the team found that SPC's failure to submit the revised assembly growth model for NRC review and approval constituted a failure to comply with the SER requirements for ANF-89-98(P)(A). The team therefore identified this finding as a nonconformance.

In response to this finding, SPC submitted a letter (HDC:97:019) to the NRC, "ATRIUM™-10 Irradiation Growth Evaluation," dated February 24, 1997. In that letter, SPC explained how the new axial growth model was developed, identified the data on which it was based (including LFA ATRIUM™-10 data from the German Gundremmingen Block B plant to confirm applicability of the new model), and discussed how the model was applied to the ATRIUM™-10 design.

The team found that axial growth data from an ATRIUM™-10 LFA was necessary to confirm applicability of the new assembly growth model to the ATRIUM™-10 fuel design because this model was initially developed using growth data from the European 8 x 8 rod array and the 9 x 9 rod array assembly fuel channels. SPC assumed that this data would apply to ATRIUM™-10 assembly growth because both structures are similar for those parameters that control axial growth, including material type, fast neutron flux, and axial stresses. The team therefore agreed with SPC's assumption on the basis of the apparent agreement between the ATRIUM™-10 LFA axial growth data and the 8 x 8 rod array and the 9 x 9 rod array fuel channel growth data from Europe.

The team also examined the ATRIUM™-10 thermal-mechanical analyses for the Susquehanna Unit 2 Cycle 9 reload of ATRIUM™-10 fuel, and determined that those analyses meet the requirements of ANF-89-98(P)(A). The specific analyses examined included cladding stress and strain, cladding collapse, fretting wear, corrosion, axial fuel rod and assembly growth, rod internal pressure, rod bowing, assembly liftoff, fuel melting, and seismic-LOCA loads.

b.2 *Surveillance Program*

During this part of the inspection, the team examined SPC's fuel surveillance program for the ATRIUM™-10 design to verify that SPC obtained the necessary data to demonstrate (a) acceptable fuel performance and (b) applicability of design-dependent models up to the range of exposure (burnup) intended for commercial fuel reloads (i.e., burnup, measured in gigawatt-days per metric ton of initial uranium metal (GWd/MTU)).

The initial SPC fuel surveillance program provided to the team for review showed more than 100 LFAs in 8 plants. However, of the 8 plants, only one plant, Gundremmingen Block B, would produce burnup data on ATRIUM™-10 LFAs before the first ATRIUM™-10 fuel assemblies at Susquehanna Unit 2 would be discharged following Cycle 9 operation. At the time Susquehanna Unit 2 would load the ATRIUM™-10 fuel for Cycle 9, the 8 ATRIUM™-10 LFAs in Gundremmingen Block B would have burnup data of approximately 38 GWd/MTU. That burnup data would be the most ATRIUM™-10 burnup performance data available before the first reload quantity of ATRIUM™-10 fuel is loaded in an NRC licensee (Susquehanna Unit 2 Cycle 9). SPC plans to obtain additional data near the discharge burnups expected for the first ATRIUM™-10 reload.

The team identified the lack of burnup data as a weakness of SPC's surveillance program. For major design changes (e.g., the ATRIUM™-10 fuel design) the team expected SPC to follow accepted industry practice, that is to have ATRIUM™-10 LFAs in three or four plants, from which burnup data would be available near the discharge burnup values for the first fuel reload *before* discharge burnups were achieved in the first reload. In addition, and of greater concern to the team was that the Gundremmingen Block B ATRIUM™-10 LFAs did not have the same fuel clad tubing material that was prototypical of the ATRIUM™-10 reload for Susquehanna Unit 2 Cycle 9. (That is, the Gundremmingen LFAs used fully recrystallized zirc-2 fuel clad tubing, rather than the cold-worked/stress-relieved zirc-2 fuel clad tubing used in the Susquehanna Unit 2 Cycle 9 reload.) Therefore, the team determined that the fuel rod data (e.g., cladding corrosion, rod bow, rod axial growth, and cladding creep-down data) from the Gundremmingen Block B plant LFAs was not directly applicable to the Susquehanna Unit 2 Cycle 9 ATRIUM™-10 fuel reload.

The team discussed these concerns with SPC and, as a result, SPC offered the team additional burnup data in an internal SPC memorandum (AR:97:023), "Information Requested by NRC on ATRIUM™-10," dated April 4, 1997. Specifically, that memorandum provided fuel rod data on rod growth, rod bow, corrosion, and cladding creep-down from an LFA in the Gundremmingen Block C plant, which used a 10 x 10-8 rod array design. (The fuel design had 8 small water rods with tie rods as the assembly structural support.) SPC stated that this 10 x 10-8 rod array design had cladding that was cold-worked/stress-relieved zirc-2 fuel clad tubing identical to that used in the ATRIUM™-10 Susquehanna Unit 2 reload. In addition, SPC stated that except for compression springs between the upper end cap and the tie plate the fuel assembly configuration, including dimensions, was similar to the ATRIUM™-10 fuel design.

c. <u>Conclusions</u>

The team identified a nonconformance related to SPC's failure to submit the new ATRIUM™-10 assembly axial growth model for NRC review, as required by the SER associated with ANF-89-98(P)(A). However, during the course of this inspection, the team reviewed the new assembly growth model provided in a letter (HDC:97:019) from SPC to NRC, dated February 24, 1997, as well as the supplemental information SPC provided to the NRC in a letter (HDC:97:038) dated May 1, 1997. On the bases of that review, the team concluded that SPC's use of this new assembly growth model for the ATRIUM™-10 designs was acceptable and closed the nonconformance previously cited.

The team determined that SPC's LFA surveillance program for the ATRIUM™-10 fuel design was significantly weak. In particular, the team was concerned that the only LFA burnup data that will be available before the discharge burnups of the first ATRIUM™-10 fuel reload (Susquehanna Unit 2 Cycle 9) were from LFAs for the German Gundremmingen Block B and C plants, neither of which were completely prototypical of the ATRIUM™-10 fuel reload for Susquehanna Unit 2. The team noted, however, that SPC plans to have prototypical LFA data from four to five plants near maximum discharge burnups within a year or two *following* the first discharge of the first reload from Susquehanna Unit 2.

3.5 Reload Design and Safety Analysis Process

The team found that the licensee initiated the reload design process when it released the Preliminary Schedule Delivery Date notice and SPC released the Preliminary Licensing Analysis Report. In addition, the reload contract defined the reload-specific work scope and responsibilities. After several reviews by SPC and the licensee, the licensee issued the Final Schedule Delivery Date notice, along with the energy and operating strategy requirements (e.g., the number and type of fuel bundles, spectral shift operation, batch split, scatter loading, and number of gadolinia rods).

After obtaining the licensees' energy and operating requirements, SPC issued the preliminary Plant Parameters Document (PPD), which included the data input required for the plant's licensing analysis (e.g., core burnup, channel type(s), OLMCPR, SLMCPR, transient limiting exposure points, maximum flows, and scram delays). The team found that the PPD was reviewed by SPC and ultimately verified by the licensee as the primary reload data interface between SPC and the licensee. SPC's projects manager was the primary customer interface, and position functions were prescribed in EMF-P00,057, QAP-3, "Quality in Marketing and Project Management," Revision 0, dated August 22, 1996, and EMF-P00,058, QAP-4, "Design Control," Revision 0, dated August 22, 1996.

The team's review consisted of examining selected reload packages and other specific topics by following the flow of information/analysis through the reload analysis and licensing processes. The reload package inspection process consisted of the following steps:

- The team began by holding discussions with SPC's project personnel concerning the contract and the intended process and flow of information. The team then discussed the reload project with SPC's project manager. On that basis, the team determined that the key documents were the contract's Schedule of Recurring Cycle-Specific Key Interactions and the Calculation Plan (a contract deliverable).

- Next, the team discussed the reload packages with engineers in the NE neutronics and safety analysis organizations, and reviewed the process for creating and controlling design inputs. The team also reviewed calculation notebooks and discussed them with preparers and reviewers to confirm compliance with EMF-954, "Procedure for Preparation of Calculation Notebooks," Revision 9, dated March 1996. In addition, the team reviewed published reload reports and compared them with data in the calculation notebooks.

- Finally, the team reviewed the applicability of SPC's reload methodology for new fuel designs. To assist in this process, SPC's lead safety engineer provided the team with a list of the relevant topical reports and SERs, with itemized lists of the SER restrictions and limitations. In addition, the team reviewed earlier topical reports, SERs and other supporting data, and discussed the findings with SPC personnel.

The SPC reload analyses were defined in the Calculation Plan and the Index of Calculations, which were formal SPC documents. The Calculation Plan determined the analyses to be performed, as well as the associated methods and schedule. The analyses performed for a given reload depend on the specific reload work scope and can vary significantly. The team found that typical analyses included the following factors:

- fuel cycle design
- neutronics licensing input (e.g., rod worth, delayed neutron fraction)
- fuel mechanical design
- thermal-hydraulic design
- AOOs and accident analyses
- stability analysis
- criticality evaluations

The team evaluated SPC's reload core design and safety analysis process by reviewing the performance, interfaces, and documentation of the reload process. As part of this evaluation, the team reviewed SPC's design inputs, design processes and controls, interface controls, and process documentation and reports. The team also examined the application of the approved neutronics and T/H methodologies to the reload design and safety analysis process. In addition, the team examined the applicability of the approved methodology to the current reload fuel designs relative to the fuel designs in place at the time the methodology was approved.

The focus of the team's evaluation of the reload design and safety analysis process was in the areas of SPC's BWR and PWR reloads. This evaluation also included a detailed review of PWR as-built resinter pellet density and thermal-mechanical analyses.

3.5.1 BWR Reload Design and Safety Analysis

a. Inspection Scope

The team observed that the engineering analysis to support current reloads was performed either by SPC in its entirety or as a shared responsibility between SPC and the licensee. The team also observed that current reload designs applied to reactor cores that consisted exclusively of SPC fuel, or mixed cores where SPC fuel was coresident with another fuel vendor's fuel. In addition, the team observed that new SPC fuel designs (i.e., ATRIUM™-9 and -10) were being loaded into reactor cores for the first time. On the bases of these observations, the team selected for evaluation a subset of BWR reloads that exhibited these circumstances.

The team found that SPC had documented the individual calculations in calculation notebooks according to EMF-954, as required by QAP-4. In addition, SPC documented the results of the analyses in the fuel cycle design report, startup and operation report, transient analysis report, LOCA reports, and the fuel mechanical design report. As required by QAP-4 and EMF-868, "Guidelines for BWR Design and Safety Analysis," Revision 3, Supplement 1, dated January 31, 1995, SPC subjects the overall reload design analysis to a formal fuel design review. In particular, the objectives of that review are to ensure the completeness and validity of the design, and to verify that all fuel design criteria were satisfied.

b. Observations and Findings

The team reviewed eight different SPC guidelines and observed that the quantity and quality of the guidelines varied from scant to in-depth detail. In particular, the team noted that EMF-868, Appendix A.7, "POWERPLEX-II Input Preparation," dated August 14, 1992, had substantial depth and provided comprehensive guidance for the preparation of input for the online monitoring system. The team found that this guideline contained checklists and steps to verify the reasonableness of parameters as part of the process. The team noted, however, that this level of detail was missing in the other guidelines reviewed during this inspection. In addition, the team observed that the Table of Contents of EMF-868 contained an incomplete list of the set of guidelines, and that the engineering guidelines did not address all aspects of the reload design and safety analysis process. The team concluded that SPC had previously identified this weakness, and a plan was in place to address the underlying causes.

To evaluate the BWR reload design and safety analysis process, the team reviewed the reload packages for LaSalle County Unit 2, Washington Nuclear Plant 2 Grand Gulf Unit 1, and Susquehanna Unit 2. The following paragraphs summarize the results of this review.

b.1 *LaSalle County Unit 2 Cycle 8*

As part of the evaluation of the SPC reload design analysis process, the team conducted an indepth review of the plant parameters, calculation notebooks, and ATRIUM™-9 fuel design for Commonwealth Edison (ComEd) LaSalle County Unit 2 Cycle 8 (LC2-8) reload. Cycle 8 was the first reload analysis for the ATRIUM™-9 fuel design.

For LC2-8, SPC shared the reload responsibility with ComEd, the licensee. ComEd had the responsibility for fuel management, and also determined both the bundle design and the core loading pattern. Additionally, ComEd performed some of the neutronics aspects of the reload safety evaluation and prepared the inputs for the online monitoring system (POWERPLEX).

The team found that the fuel contract specified the shared reload licensing responsibilities. Contract tables of key initial reload interactions and recurring cycle-specific interactions specified the activity, schedule, and responsibility. The team also found that the subsequent Calculation Plan for the reload specified the analysis to be performed, the methodology and computer codes to be used, and other specific details for the analysis to be performed by SPC, as well as the analysis to be performed by ComEd. The following paragraphs summarize the results of this review.

Plant Parameters

SPC informed the team that since LC2-8 was the first reload provided by SPC for this reactor, the initial effort focused in gathering plant design data and benchmarking the SPC analysis models to plant data and/or earlier design calculations. The team observed that the plant specific data gathering was an informal and loosely controlled process in that SPC did not use check lists, steps to verify the reasonableness of individual parameters, or sign-offs for individual parameter verification. The team identified this concern as a weakness in SPC's reload design process.

Following the initial data gathering, SPC documented the plant specific data in the neutronics and transient analysis model building calculation notebooks controlled by SPC procedures. The team observed that the PPD documented the plant parameters that may change from cycle to cycle, and that this document was updated for each new reload. To further address the weakness identified by the team, SPC provided a newly issued Work Practice (WP) document, which addresses the external interface control aspects for the PPD, as well as the requirement for licensee verification of the data in the PPD before use by SPC in the safety analysis. The team also observed that, before performing the Cycle 8 reload analysis, SPC benchmarked its models to previous plant specific operating data.

The team noted that ComEd performed the core design and the steady-state neutronics aspects of the safety evaluation using SPC's guidelines and computer code set installed at ComEd. In addition, the team noted that SPC performed confirmatory calculations to support the ComEd neutronics analysis.

Calculation Notebooks

The team reviewed calculation notebooks for the various neutronics and safety analyses that were within SPC's scope of responsibility. The team also held discussions with SPC personnel concerning the preparation of the calculation notebooks, the methodology applied, and the V&V review process. The team found that the calculation notebooks were easy to read and understand, frequently discussed background concerning the methodology, and clearly documented the analysis performed.

However, the team found that the neutronics and safety analysis methodology used in the reload analysis was propagated via the calculation notebook process. (That is, the new reload analysis was derived on the bases of either similar analyses from another reload or the initial analysis performed for LaSalle during the analytical model building and V&V process.) The team's discussions with the notebook preparers substantiated the team's finding that the preparers rely on earlier notebooks for guidance. Although, the calculation notebooks referenced the SPC methodology found in NRC-approved topical reports and guidelines, the as-found practice was that the source of the methodology used by the analysts was the notebook. The team identified this as a weakness and raised the concern with SPC, noting that the specific engineering analysis procedures were not being used as part of the reload process and that the potential could exist for the propagation of errors by repeating the analyses from earlier notebooks.

ATRIUM™-9 Fuel Design

The inspectors reviewed the applicability of SPC's reload methodology to the new ATRIUM™-9B fuel design used in the LC2-8 reload. During this review, SPC provided a list of the relevant topical reports and associated SERs, with itemized lists of the SER restrictions and limitations. The team reviewed this list relative to the licensing documents and design features of the new reload. The team also reviewed the previous topical reports and their SERs, and discussed their findings with SPC personnel. In addition, the team reviewed the fuel designs currently used by SPC, as well as the designs used over the past several years. Except for the issues raised with regard to the ANFB critical power correlation, as discussed in Section 3.3.5 of this report, the information reviewed showed that the Cycle 8 analysis satisfied the neutronics methodology limitations specified in the SERs.

b.2 ***Washington Nuclear Plant Unit 2 Cycles 10 and 11***

As part of the evaluation of the SPC reload design analysis process, the team conducted an indepth review of the energy requirements, calculation notebooks, cold shutdown margin, and safety analysis for Washington Public Power Supply System, Washington Nuclear Plant Unit 2 (WNP2) Cycles 10 and 11 (WNP2-10 and -11) reloads. In particular, the team selected these reloads because the unit has had SPC fuel reloads using the 9 x 9-9X rod array fuel assembly (the predecessor of the ATRIUM™-9B) since Cycle 7, and had a earlier history of fuel design and operation problems (i.e., the instability event during the Cycle 8 startup). In addition, although the cycle was short (1 year), there have been frequent design modifications to accommodate changes in energy requirements associated with weather-related capacity factor adjustments. The following paragraphs summarize the results of this review.

Energy Requirements

SPC informed the team that a challenge to the WNP2 reload designs resulted from several changes in energy requirements specified by the licensee, Washington Public Power Supply System. These changes change end-of-cycle (EOC) n-1 and beginning-of-cycle (BOC) n reactivity parameters and distributions, since burned fuel reactivity is increased and then compensated for by a reduced number of fresh reload assemblies. The team reviewed the related correspondence, calculation notebooks, and design reports and found that interactions concerning reload design were clearly documented. These documents indicated that, for Cycle 10, SPC included an expanded stability analysis scope and initially planned a power-up rate but then deleted it from the requirements.

Calculation Notebooks

The team held discussions with the SPC engineers responsible for calculation notebook preparation and review. In particular, these discussions focused on the preparation of the calculation notebooks, the methodology applied, and the verification/review process. The team found that the calculation notebooks were easy to read and understand, frequently discussed background concerning the methodology, and clearly documented the analysis performed.

The team also reviewed a number of calculation notebooks for the neutronics and follow on transient and safety analyses, to assess compliance with the requirements of EMF-954. Not all of Revision 9 to EMF-954 applied to the WNP2 calculation notebooks, however, since the notebooks were prepared before the issue date. For example, Section 3.1.10, "Quality Assurance Review," which required a lead engineer's signature indicating review of the QA review plan, was not yet in effect.

Cold Shutdown Margin

The team conducted an indepth examination of the cold shutdown margin (CSDM) neutronics analysis for WNP2-10 and -11. The team selected this aspect since there had recently been an indication of industry problems in both the calculational method and type of measurement made for BWR shutdown margin verification (i.e., in-sequence or local critical measurement and analysis).

The team found that the target k (multiplication factor) for the WNP2 CSDM calculations was well documented on the basis of previous BOC measurements at burnup points during these cycles. The scatter in the measured k compared to calculated k data was such that a target cold critical k could be established, with confidence that the margin to the TS limit of 0.38% $\Delta k/k$ would be maintained. The calculation margin to the TS requirements was approximately 1% $\Delta k/k$, which was sufficient in Cycle 11 to accommodate energy requirement changes from Cycle 10. In addition, all of the CSDM measurements for WNP2 were in-sequence measurements.

Safety Analysis

The team concluded that the neutronics and safety analysis methodology was founded on similar analyses for previous WNP2 reloads. Discussions with the notebook preparers substantiated their reliance on earlier calculation notebooks. The team raised the concern that specific engineering analysis procedures were not part of the reload process, and that the potential exists that preparers will propagate errors through the notebooks and might fail to communicate problems of a generic nature among the designers for different units.

b.3 ***Grand Gulf Unit 1 Cycle 7***

As part of the evaluation of the SPC reload design analysis process, the team conducted an indepth review of the neutronics fuel design, safety analysis, and fuel mechanical design for Entergy Operations, Incorporated, Grand Gulf Unit 1 Cycle 7 (GG1-7). The following paragraphs summarize the results of this review.

Neutronics Fuel Design Analysis

The GG1-7 fuel loading consisted of 288 fresh 9 × 9-5 rod array fuel bundles with average enrichment ranging from ~3.0 to 3.5 weight percent (w/o) U_{235} including both axial enrichment and gadolinia zoning. The GG1-7 design specifies a cycle length of 18 months, and SPC's scope of work for the GG1-7 reload design included the full core neutronics design analysis. This involved the CASMO-3G fuel bundle neutronics calculations and the MICROBURN-B three-dimensional (3D) core performance analysis. SPC also performed neutronics analyses for the control rod withdrawal event, control rod drop accident, fuel bundle mislocation/misorientation error, spent fuel storage rack criticality, standby liquid control system (SLCS) reactivity, and loss of feedwater heating. The GG1-7 analyses were documented in the SPC Series E-5479 calculation notebooks.

The team conducted and indepth review of the GG1-7 fuel assembly design, POWERPLEX-II analysis, and spent fuel storage rack criticality analysis. The fuel assembly design analysis was included in calculation notebook E-5479 No. 7-1, dated

July 6, 1994. The team reviewed this analysis with the GG1-7 lead neutronics engineer, and found that the analysis was complete and well-documented. The notebook also included special approval letters for the use of the new versions of the MICRO-B/ajan94 (cross-section fitting) and the CAZAM (CASMO-3G output processing) codes, as required by EMF-608. The approved-use codes that were employed included MICBURN-B (gadolinia depletion), CASMO-3G (bundle neutronics), and FUELRQ (uranium separative work unit (swu) requirements).

The REPROC-E program is a peripheral-code and the calculation notebook included the necessary model and V&V documentation required by EMF-608. In addition, as required by EMF-954, the calculation notebook included the major assumptions used in the fuel bundle analysis, as well as the analysis verification.

Calculation notebook E-5479 No. 9-1, dated May 1, 1995, documented the preparation of the POWERPLEX-II core monitoring input decks for GG1-7. The team reviewed the POWERPLEX-II analysis with the GG1-7 lead neutronics engineer, and found that the analysis described in the calculation notebook was complete and well-documented. SPC also used the peripheral PPLXPES code (for processing fuel assembly weights), and the notebook included the method and V&V documentation, as required by EMF-608. The notebook also documented the POWERPLEX-II analysis assumptions, including the details of the core loading, reactor set points, and local power range monitor (LPRM) replacements. In addition, the notebook included verification of the analysis, and SPC followed a special POWERPLEX-II input deck checklist.

Safety Analysis

The GG1-7 reload design was a full-scope safety analysis for operation within the maximum extended operating domain. The SPC analysis included the evaluation of the limiting AOOs (i.e., load rejection without bypass, loss of feedwater heater, control rod withdrawal error, feedwater controller failure without bypass, and the fuel loading error) at the bounding power and flow conditions. The GG1-7 safety analysis also included a determination of the safety- and operating-limit MCPRs. The core stability analysis was performed by GG1.

For the GG1-7 reload, the team reviewed the safety analysis and reload documentation with SPC's lead safety engineer.

SPC performed the GG1-7 safety analysis on the basis of PPD ANF-86-133, "Principle ECCS and Plant Transient Analysis Parameters Grand Gulf-1," Revision 4, dated June 1991. This data listing was originally prepared for the GG1-4 reload design, and the licensee verified its applicability for GG1-7. The plant input preparation was included in calculation notebook E-5749-593-1, "COTRANSA2

DECKPL Update," and SPC used COTRANSA2/uapr91 and CONGEN/uapr91 approved-use codes in the preparation of the input deck, as required. In addition, the calculation notebook included the required analysis documentation and verification.

Calculation notebook E-5479-595-1 documented the GG1-7 load rejection without bypass analysis SPC performed over a range from 40% to 104% of rated power and over a range of EOC burnup (GWd/MTU). The COTRANSA2, XCOBRA, and XCOBRAT approved-use codes were used in the analysis. The CTZFLOW peripheral code was used to calculate the initial junction flows for COTRANSA2, and the required code documentation and verification were provided in SPC report E-5086-1.

During the review, the team noted an improvement in the GG1-7 analysis of the feedwater controller failure transient. In order to maximize the overcooling and the transient reactivity insertion, the analyst reduced the initial separator water level to the low-level alarm setpoint. The team noted that this modification substantially improves the accuracy of the calculation.

However, SPC had no mechanism in place to ensure timely communication of this type of improvement to the SPC safety analysis staff, and the team considered this a weakness. The team also noted that SPC had not updated EMF-868, which defined the procedures for performing plant transient analyses, since December 1994.

Fuel Mechanical Design

The criteria for the fuel mechanical design were defined in ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1, April 1991, which the NRC approved in April 1995. To a significant degree, the GG1-7 fuel mechanical design was predicated on the generic or previous cycle mechanical design analyses. Specifically, the GG1-7 power history was bounded by the generic design for 9 × 9-5 rod array fuel bundles, and the seismic-LOCA analysis was bounded by the GG1-4 reload predictions. Previous cycle analyses of cladding collapse, AOO temperature/strain, tie plate and spacer design, and fuel assembly compatibility also bound the GG1-7 fuel reload.

Together with SPC's lead fuel mechanical engineer, the team reviewed the specific fuel mechanical analyses performed for GG1-7. Because of changes in the pellet diameter and concerns associated with pellet "cocking," the inspectors specifically evaluated the fuel pellet dimensions and density analysis for the GG1-7 reload. As a result of this evaluation, the team found that this analysis, which was documented in calculation notebook E-5479-337-124, included the required documentation and verification.

During the review of this documentation, the team noted that the pellet dimensional changes, which required the GG1-7 update of the pellet dimension and density analysis, were actually introduced before the GG1-6 reload. Nonetheless, a review of the GG1-6 design analysis indicated that SPC failed to perform any pellet dimension/density reanalysis for GG1-6; consequently, the GG1-6 fuel mechanical design analysis did not reflect the actual pellet design and dimensions.

When the team brought this finding to the attention of the SPC staff, they indicated that an earlier internal SPC audit had identified this problem and the GG1-7 reanalysis was, in fact, documented in audit CAR 1288. Specifically, CAR 1288 identified insufficient review of generic calculations as the root cause, and required that the outdated generic analyses be updated using the latest dimensions. The team agreed with the root-cause identified in CAR 1288, but they found that the corrective action (i.e., correcting the specific generic calculation used in the GG1-7 analyses) did not ensure that future generic analyses will not be used outside their range of applicability. The team considered this a weakness.

b.4 ***Susquehanna Unit 2 Cycle 8***

As part of the evaluation of the SPC reload design analysis process, the team conducted an indepth review of Susquehanna Unit 2 Cycle 8 (S2-8) neutronics fuel design analysis, safety analysis, and fuel mechanical design. The following sections summarize the results of this review.

Neutronics Fuel Design Analysis

The S2-8 reload consisted of 312 fresh ATRIUM™-10 fuel bundles and 448 second- and third-cycle 9 $\times$ 9-2 rod array fuel bundles, that will operate over a 24 month cycle. The reload core also includes four General Electric Nuclear Energy (GENE) lead test assemblies for which GENE and PP&L share responsibility. The ATRIUM™-10 bundles have an average enrichment of $\sim$4 w/o U_{235} and include axially zoned gadolinia rods. Under the contract for the S2-8 reload, PP&L had responsibility for the core neutronics design; however, SPC performed selected neutronics calculations in support of the SPC transient analyses. The team reviewed these neutronics analyses with SPC's lead neutronics engineer.

SPC performed a series of fuel bundle neutronics calculations to provide input to the LOCA analyses, as documented in calculation notebook E-5773 No. 6-3, Revision 1. The analysis was performed with a variety of approved-use codes, including MICROBURN-B, SDM-VD (shutdown margin-variable dimension), CAZAM, CASMO-3G, MICRO-B, and FIT (Doppler coefficient fitting). Consequently, SPC performed calculations for the Doppler coefficient, neutron lifetime, and capture-to-fission ratio for input to the LOCA analysis. The analysis assumptions and methods were clearly documented and extensively verified by an independent engineer.

Safety Analysis

Under the S2-8 contract, SPC was responsible for analyzing the AOOs under single-loop operation, and the licensee was responsible for the two-loop analyses. The team therefore reviewed the single-loop safety analyses with SPC's responsible lead safety analysis engineer. At the time of this review, SPC was in the process of reevaluating the AOOs using the recently revised set of ANFB correlation additive constants. (See Section 3.3.5 of this report for a discussion on the ANFB critical power correlation.)

The analysis of S2-8 load rejection without bypass was performed using the COTRANSA2, XCOBRA, and XCOBRA-T approved-use codes, as documented in calculation notebook E-5773-878-1, Revision 0. As a result of this review, the team found that SPC's application of these codes in this analysis was consistent with the conditions of the code SERs.

SPC performed the load rejection calculations at EOC and over a range of power and flow. SPC also determined both core-wide and hot-bundle gap conductance with RODEX2 (as a function of the linear heat generation rate (LHGR)) for a selected set of assumed single-loop operating histories. Specifically, SPC determined the core-average gap conductance on the bases of RODEX2 calculations for each of the five S2-8 fuel batches using batch-specific power histories. The analysis received a detailed verification, which followed Plan-C of EMF-954. The review plan was approved by the lead engineer and included responses to the reviewers' comments. In addition, the COTRANSA2 SER requires verification of the selection of time-steps, and the reviewer identified this requirement, which was subsequently carried out by the safety analyst.

The overpressurization transient, included in calculation notebook E-5773-878-4, was organized using the COTRANSA2 approved-use code, assuming that both the main steam isolation valve (MSIV) closure-scram and turbine bypass are inoperable. SPC performed the transient analysis according to the guidelines of EMF-868, which include requirements concerning the methods and code SERs. In addition, SPC verified the analysis as documented in the calculation notebook.

Fuel Mechanical Design

The S2-8 reload included 312 bundles of the new SPC ATRIUM™-10 fuel design. Together with the SPC staff, the team reviewed SPC's analysis of the fuel rod internal gas pressure for the full-length and part-length ATRIUM™-10 fuel rods, as documented in calculation notebooks Q729-337-6C and Q729-337-6F, dated March 1995. The calculations were performed using the NRC-approved methods with the approved-use code RODEX2. In addition, the calculations employed conservative fuel rod power histories and demonstrated that the fuel rod internal gas pressure remained within the NRC-approved limit. The part-length rod calculation accounted for the increased plenum volume. As required, the analysis documentation included a description of the modeling assumptions and the record of verification.

c. Conclusions

After reviewing correspondence, calculation notebooks, and reports, the inspectors found that ComEd had successfully achieved the transition to SPC fuel, and the shared responsibility for the process proved adequate.

Several of the calculation notebooks could be considered excellent with assumptions stated, the QA review plan described, and indepth review comments and responses provided where required. Specifically, these notebooks reported analysis results for the control rod drop accident, control rod withdrawal error analysis, feedwater controller failure analysis, and neutronics input for the licensing transient analysis. The team concluded that the quality of the calculation notebooks was a strength of SPC's reload process.

The team also concluded that SPC had performed adequate reload analyses for LC2, WNP2, GG1, and S2.

However, the team concluded that SPC's external design input process was weak, and the engineering procedures were minimal and lacked specificity. Nonetheless, the team did not observe any breakdown of the process as a result of these concerns. The team attributed this to the quality of the engineering calculation notebooks and the level of experience of the SPC staff.

During the review of the reload process, the team noted that SPC does not use detailed step-by-step procedures during the reload design. Instead, SPC uses a "template" approach, in which the previous cycle reload is used as a guide for the reload analysis. The SPC methodology in topical reports and guidelines was referenced in the calculation notebooks, but the source of the methodology for an analysis was primarily the preceding notebook. The team considered this lack of detailed reload analysis procedures to be a weakness.

After the team discussed these weaknesses with the SPC staff, SPC pointed out that, as a result of several self-assessments (EMF-1924), SPC was planning to implement formal detailed procedures for performing these analyses. These new procedures will provide a more standardized methodology that would be less error-prone and would ensure greater control of the reload process.

3.5.2 PWR Reload Design and Safety Analysis

a. <u>Inspection Scope</u>

On the bases of the initial reviews and discussions, the team reviewed the reload package relative to the process flow chart provided by SPC. The team also evaluated the process by examining reports, calculations notebooks, and engineering guidelines, and by interviewing various engineers associated with the analysis. The team noted that the engineering analysis to support current reloads is either performed entirely by SPC, or the responsibility is shared between SPC and the licensee customer. The team also noted that current reload designs supplied by SPC can be for reactor cores consisting only of SPC fuel or mixed cores consisting of fuel from SPC and coresident fuel from another vendor. On the bases of these observations, the team selected various PWR reloads that exhibited these characteristics.

b. <u>Observations and Findings</u>

The team found that the PWR engineering guidelines for neutronics and safety analysis were far superior to SPC's BWR guidelines. The PWR guidelines were more comprehensive in subject matter and contained more detailed instructions.

To evaluate the PWR reload design and safety analysis process, the team reviewed the reload packages for Shearon Harris Unit 1 Cycle 8, St. Lucie Unit 1 Cycle 14, and H.B. Robinson Unit 2 Cycle 18. The following paragraphs summarize the results of this review.

b.1 <u>*Shearon Harris Unit 1 Cycle 8*</u>

The team selected the CP&L, Shearon Harris Unit 1 Cycle 8 (SH1-8) reload for review because responsibility for that reload belonged solely to SPC. Cycle 8 is the third reload provided by SPC, and the Cycle 8 core consists entirely of SPC fuel. Cycle 8 was designed for an 18-month fuel cycle.

On the basis of this initial review, the team reviewed the calculation notebooks.

Calculation Notebooks

The team found that the calculation notebooks were very neat and well organized, with the contents easy to read and understand. Guidelines were referenced, and the calculation notebook frequently discussed background concerning the methodology. In addition, the team reviewed various neutronics and safety analysis notebooks, and found that they clearly documented the analysis performed. The team also noted that the calculation notebooks were deliverable items to CP&L, the licensee.

b.2 *St. Lucie Unit 1 Cycle 14*

The team selected the Florida Power and Light (FP&L) St. Lucie Unit 1 Cycle 14 (SL1-14) reload for examination because FP&L and SPC shared the analysis responsibility. Cycle 14 is the ninth reload provided by SPC for St. Lucie Unit 1, but it is the first reload with shared analysis responsibility. FP&L performed the core reload design and generated neutronics inputs to the safety analysis. SPC then performed the remainder of the analysis.

The main area of concern to the team was the external design interface and the consistency of the methodology applied to the analysis. SL1 is a CE reactor that is fueled by SPC. Nonetheless, FP&L performed the neutronics analysis using Westinghouse methodology, and provided inputs to SPC for the safety analysis consistent with SPC methodology.

As part of the evaluation of the SPC reload design analysis process, the team conducted an indepth review of St. Lucie Unit 1 Cycle 14 design interface and methodology. The following sections summarize the results of this review.

Design Interface

The team reviewed the project correspondence and found that the initialization of the reload design process was adequately documented in terms of design meeting minutes, PPD, and core designs. The team also found that SPC checked the FP&L core design by performing confirmatory calculations using SPC's neutronics methodology. These SPC calculations confirmed that the reload met the applicable design criteria. In addition, the team noted that the third loading pattern determined by FP&L was the final pattern, that each loading pattern was documented in the project correspondence, and that each loading pattern had SPC concurrence regarding the adequacy of the design.

The team observed that mid-way through the reload analysis process, SPC published a report (EMF-95-164) entitled "Design Interface Document Between Siemens Power Corporation and Florida Power & Light Company," dated November 1995. SPC informed the team that this document formalized the process, and that the next reload would be performed in a manner consistent with its guidance. However, the team observed that the Cycle 14 reload process did not meet various requirements in this report regarding deliverables or types of documents to be produced.

The team also examined SPC calculation notebook E-6380-592-1, "St. Lucie Unit 1 Cycle 14, 88-Assembly Loading Pattern Disposition of Events," Revision 1, dated April 12, 1996, to judge the adequacy of the design interface between FP&L and SPC. The team observed that this file evaluated each of the events identified in SRP Chapter 15 relative to the operation of SL1-14. In addition, SPC used the neutronics data calculated by FP&L to determine if the value of the physics parameter was bounded by the analysis of record or if any specific event needed to be reanalyzed for Cycle 14. The team also reviewed this file relative to the various SPC engineering guidelines for calculating neutronics inputs to the safety analysis. During this process, the inspectors raised many questions pertaining to the values reported by FP&L and their interpretation and use by SPC. The team also raised concerns regarding inadequate calculation instructions, insufficient review of data by SPC, and the use of inadequate procedures for this design interface. With subsequent discussion, the responsible SPC engineers resolved the team's questions regarding the calculation notebook.

Methodology

The second area investigated involved the application of approved methodology to the Cycle 14 reload. The team also examined topical reports related to SPC's established methodology, as well as the corresponding SERs prepared by the NRC. The following topical reports are relevant for the SPC safety analysis transient methodology and the steady-state and transient T/H methodology:

- XN-NF-74-5, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)," Supplements 1 - 6, Rev. 2, October 1986.

- XN-NF-84-73, "Exxon Nuclear Methodology for Pressurized-Water Reactors: Analysis of Chapter 15 Events," Rev. 2, March 1989.

- XN-NF-75-21, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," Rev. 2, January 1986

The team also reviewed the SER for the FP&L neutronics methodology documented in NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point and St. Lucie Nuclear Plants," January 1995.

The team raised the concern that the reload analysis methodology was not described in the topical reports and within the constraints specified in the corresponding SERs. FP&L requested and received approval to apply the Westinghouse methodology only to Turkey Point Units 3 and 4. (Westinghouse was the Turkey Point fuel supplier.) The summary and conclusions of the SER limited the application to the range of Westinghouse fuel assembly and core reload design parameters verified in the topical report. This concern remains an unresolved item requiring additional information from SPC and FP&L to demonstrate that the St. Lucie Unit 1 Cycle 14 reload analysis meets NRC-approved methodologies. The team is treating this issue as Unresolved Item 99900081/97-01-10.

The SER for the methodology that addresses SRP Chapter 15 events restricts the use of the topical report to the set of referenced codes, which includes the SPC neutronics code XTG. Nonetheless, the neutronics data used in the Cycle 14 reload originated from FP&L's calculations using the Westinghouse ANC neutronics code. This concern is also part of Unresolved Item 99900081/97-01-10.

b.3 *H.B. Robinson Unit 2 Cycle 18*

The team selected the CP&L, H.B. Robinson Unit 2 Cycle 18 (HBR2-18) reload for review because of the Cycle 16 history of miss-fabrication and subsequent loading of misconfigured fuel assemblies with an asymmetrical gadolinia fuel rod design. The Cycle 18 reload included 52 fresh assemblies, of which 24 have the asymmetric gadolinia design. Cycle 18 also included five fuel assemblies that were *reinserted* from earlier cycles and so had significant decay time in the spent fuel pool, and 12 part-length shield assemblies (PLSAs) to reduce vessel fluence.

Specifically, the team reviewed five neutronics and safety analysis calculation notebooks, and found that they clearly documented the analysis performed. The calculation notebooks were well organized with the contents were easy to read and understand. Guidelines were referenced, and the calculation notebook frequently discussed background concerning the methodology. In addition, the team noted that the calculation notebooks were deliverable items to CP&L, the licensee.

During the Robinson reload review, the team discussed an issue with SPC engineers regarding the treatment of uncertainties in neutronics parameters that are input to the safety analysis calculations. (This issue was also discussed during the course of the St. Lucie Unit 1 Cycle 14 reload review.) Specifically, the calculational approach is to determine the worst core condition for the event, perform the neutronics analysis, and input the results to the safety analysis. However, the team expressed the concern that these worst conditions represent possible actual core conditions, and the neutronics calculations yield "best estimate" values for input to the safety analyses. If uncertainty factors were included, these best estimate values could exceed the values used in the bounding safety analysis. The methodology also does not account for uncertainties in the neutron kinetics parameters such as the delayed neutron fraction (Beta), and reactivity coefficients such as the Doppler coefficient. In response, SPC stated that some safety analysis reactivity events use multiplying factors of 0.8 or 1.2 (whichever is conservative) on the Doppler coefficient. One such analysis is the rod ejection reactivity insertion accident, which uses a multiplier of 0.8 on the Doppler coefficient. However, examination of the rod ejection methodology revealed that a bounding "least negative" value was used rather than the actual cycle-specific calculated value multiplied by 0.8. It was also not clear to the team what other events used the 0.8 or 1.2 multipliers.

Other safety analysis events use the TS limits for power distribution peaking factors and the moderator temperature coefficient, and are thus conservative. However, the team had a concern regarding the lack of a guideline providing uncertainty factor values and an application procedure for purely calculated values such as the Doppler coefficient and neutron kinetics parameters.

c. **Conclusions**

The team observed differences in the reload processes used by SPC's BWR and PWR organizations. A common strength is the calculation notebooks. The team concluded that the PWR analysis methodology and process is better supported by engineering guidelines; but the BWR organizations have recognized this weakness and are in the process of improving the engineering guidelines. For both PWRs and BWRs, however, the team noted instances of weakness in external design interfaces.

For the St. Lucie reload, unresolved items exist pertaining to the application of approved methodology. The team concluded that the St. Lucie reload analysis was adequate. The team concluded that a weakness exists in the SPC process for the control of customer supplied designs. The team noted that reasonableness checks of data received by SPC from FP&L were not adequate. The team further concluded that the potential exists for the misuse of FP&L neutronics data or the use of inappropriate FP&L data by SPC in the SPC safety analysis due to the observed concerns in the design interface process.

The team concluded that the reload packages are adequate overall, and the team did not identify any performance-based issues or concerns that resulted in errors or safety issues.

3.5.3 As-Built Resinter Pellet Density

a. **Inspection Scope**

As part of the inspection of SPC's PWR analysis methods, the team examined the HBR2 submittal to NRC in CP&L's letter (RNP-RA/97-0007) dated January 17, 1997. That letter described the results of CP&L/SPC's reanalysis of LBLOCA PCT that corrected errors in the 1986 evaluation model used by SPC. In addition, it provided HBR2 with a greater PCT margin than provided in a previous reanalysis submitted in a letter to NRC from CP&L dated October 29, 1996. The reanalysis of

January 17, 1997, used as-built fuel resinter[7] densities for the fuel in question (rather than the bounding resinter density used in the previous analyses) to obtain the additional PCT margins for HBR2. In a conference call on January 28, 1997, the NRC requested CP&L and SPC to provide additional information on pellet lot sizes and resinter data. CP&L submitted this additional information to NRC in a letter (RNP-RA/97-0057) dated March 12, 1997. The team then reviewed the SPC application of this resinter data for the HBR2 LBLOCA analysis.

Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, requires that the steady-state temperature distribution and stored energy in the fuel before a hypothetical LOCA be calculated, taking fuel densification into consideration. In-reactor densification (shrinkage) of fuel pellets affects fuel temperature in several ways, as follows:

(a) gap conductance may be reduced because of the decrease in pellet diameter
(b) the linear heat generation rate is increased because of the decrease in pellet length
(c) the pellet-length decreases may cause gaps in the fuel column and may produce local power spikes and the potential for cladding collapse

Dimensional changes in fuel pellets in the reactor do not appear to be isotropic, so axial and radial pellet dimension changes are treated differently. Furthermore, items (a) and (b) above are single-pellet effects, whereas item (c) is the result of simultaneous changes in a large number of pellets. These distinctions must be taken into account in applying analytical models.

Regulatory Guide (Reg Guide) 1.126, "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification," Revision 1, March 1978, provides an analytical model and related assumptions and procedures that are acceptable to the NRC staff for predicting the effects of fuel densification in light-water-cooled nuclear power reactors. Reg Guide 1.126 also describes statistical methods related to product sampling that will provide assurance that this and other approved analytical models will adequately describe the effects of densification for each initial core and reload fuel quantity produced.

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[7]Sintered pellets have been heated in a furnace to bring about agglomeration in the fuel pellet. Since the density of a fuel pellet in the reactor increases with burnup and achieves a maximum value at a relatively low burnup, it is assumed that the same density change would occur outside the reactor in the same sintered pellets during resintering at 1700 °C for 24 hours.

b. Observations and Findings

The inspection team examined SPC's as-built fuel resinter densities and assessed how they were applied in the LBLOCA reanalysis. Specifically, the UO_2 resinter density (often referred to as fuel densification) is used in the SPC RODEX-2 fuel performance code to calculate fuel stored energy for input to LOCA analyses. This is because UO_2 is the fuel stored energy that drives the core heatup and PCT. SPC measured the resinter density on 11 randomly selected pellets (one lot had only 10 pellets measured) from each of the 19 pellet lots fabricated for the HBR2 reload (3.51 million pellets fabricated from the 19 lots) resulting in a total of 208 resinter measurements. The team determined that SPC's resintering process was consistent with Reg Guide 1.126.

SPC estimated the resinter density for the stored energy LBLOCA analysis using a one-sided 95% upper confidence limit (UCL) on the mean of the data from each of the 19 measured resintered pellet lots. The team noted that SPC's methodology did not follow the methods stated in Reg Guide 1.126, as follows:

> *"Analyses of the effect of densification on stored energy and linear heat generation rate must account for pellets that have the greatest propensity for densification. To accomplish this with a resintering-based model such as described in Sections C.1 and C.2 of Reg Guide 1.126, a resintering density change value **/$\Delta\rho_{sntr}$[8] that conservatively bounds 95% of the population $\Delta\rho_{sntr}$[9] values with 95% confidence should be used."*

Therefore, Reg Guide 1.126 recommends use of the upper one-sided 95/95 tolerance limit on the 208 samples for stored energy i.e., analyses where single-pellet effects are important, rather than the 95% UCL on the average of the mean resinter values for all 19 pellet lots as applied by SPC. Using the 95/95 upper tolerance limit as recommended by Reg Guide 1.126 results in a delta-resinter density of 1.11% TD (theoretical density, g/cm^3), while using the 95% UCL as applied by SPC results in a delta-resinter density of 0.87% TD (used by SPC in stored energy analysis for LBLOCA for HBR2) with a mean density of 0.82% TD of the 208 measured resinter samples.

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[8]**/$\Delta\rho_{sntr}$ is the one-sided 95/95 upper tolerance limit for the total population of $\Delta\rho_{sntr}$ values, g/cm^3.

[9]$\Delta\rho_{sntr}$ is the measured density change of a sintered pellet due to ex-reactor resintering, g/cm^3.

SPC noted that, while their resinter density value for the revised stored energy analysis did not follow the 95/95 tolerance limit per Reg Guide 1.126, the NRC approved their use of the 95% UCL on the mean of resinter density data for stored energy analyses in XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model." The team examined this topical report, the supplements to this report including the NRC SER transmitted to SPC by letter dated November 16, 1983, and Reg Guide 1.126. The responses to NRC's questions raised during the staff's technical review, presented as a supplement to topical report XN-NF-81-58(P)(A), indicated that SPC indeed proposed to use the 95% UCL of the mean density change for LOCA analyses. However, this report was not clear about how the 95% UCL would be obtained on as-built resintered pellets.

In addition to its reference to the 95/95 tolerance limit for analyses of single-pellet effects, Reg Guide 1.126 also refers to analyses (cladding creep collapse) where the use of multiple pellet effects and the 95% UCL on the mean densification is appropriate. According to Reg Guide 1.126 for these analyses,

> *"resintering-based densification models should use a density change value* $*/\Delta\rho_{sntr}$[10] *that bounds the selected material population mean with 95% confidence."*

Reg Guide 1.126 further expands on the definition of population for multiple pellets by stating,

> *"the population to be considered is not the core or reload quantity characterized above, but rather the material population (or subset thereof) within that quantity that exhibits the largest mean of the* $\Delta\rho_{sntr}$ *values from the sample."*

Therefore, SPC's use of the 95% UCL on the mean of their total reload pellet population (all 19 pellet lots) for HBR2 also did not follow Reg Guide 1.126, with regard to multiple pellet analyses.

However, the NRC SER for XN-NF-81-58(P)(A) appears to suggest that the RODEX2 densification model follows the methodology described in Reg Guide 1.126 by stating,

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[10] $*/\Delta\rho_{sntr}$ is the one-sided 95% upper confidence limit on the mean of the $\Delta\rho_{sntr}$ values from the selected material population, g/cm^3.

"The RODEX2 densification model for licensing calculations is a burnup dependent Urania densification model that is based on Reg Guide 1.126 ... Reg Guide 1.126 describes how the densification correlations should be obtained from ex-reactor resintering data and how the statistical method applies to these data."

The SER further states,

"In Reference 5, Equation 4.2 shows a maximum densification as a function of initial density at the upper 95 percent confidence limit on the mean. Since the RODEX2 densification model follows the guidance of Reg Guide 1.126. we conclude that this model is acceptable."

The team found that these statements suggests that the SER was approving the 95% UCL on the mean as defined in Reg Guide 1.126, which as discussed above is the 95% UCL on the largest mean of the $\Delta\rho_{snr}$ values from the sample, and *not* the 95% UCL on the average of the mean $\Delta\rho_{snr}$ values of the 19 pellet lots (reload quantity) for HBR2 as interpreted by SPC.

These issues constitute Unresolved Item 99900081/97-01-11, because additional information is needed to (a) describe how XN-NF-81-58(P)(A) and/or the associated SER clearly addresses using the average $\Delta\rho_{snr}$ densification value instead of the largest mean $\Delta\rho_{snr}$ densification value when applying the 95% UCL and (b) address the apparent confusing statements in XN-NF-81-58(P)(A) and the associated SER.

c. Conclusions

The team concluded that SPC's practice of using the multiple-pellet effects method (upper one-sided 95% confidence limit (95% UCL)) on the mean of all their as-built actual densification data to calculate stored energy for LOCA analyses, did not follow the methods in Reg Guide 1.126. In addition, SPC's methodology did not demonstrate an acceptably conservative means of determining resinter effects for fuel densification consideration when calculating the steady-state temperature distribution and stored energy in the fuel before a hypothetical LOCA, as required by Appendix K to 10 CFR Part 50.

Further, the use of the 95% UCL on the largest mean of the $\Delta\rho_{snr}$ densification values (lot 709-02 from the 19 pellet lots measured) results in delta-resinter densification value of 1.11% TD as opposed to the delta-resinter densification value calculated by SPC of 0.87% TD using the 95% UCL of the average of the mean of all 19 pellet lots. These issues constitute an unresolved item.

On April 3, 1997, SPC presented the team with a response to its conclusions in Supplement 1 to "Response to NRC Audit Question Regarding Application of 'As-Built' Resinter Density to H.B. Robinson Cycle 18 LOCA Analyses." The team reviewed Supplement 1 and meet with SPC staff to discuss the response and concluded that SPC's response did not adequately address the team's concerns.

On April 4, 1997, SPC presented the team with a revised Supplement 1; however, that response did not completely address the team's concerns. SPC also provided additional background in a letter (RAC:047:83), to the NRC regarding, "RODEX2 ECCS Comparisons," August 31, 1983.

On April 8, 1997, the NRC staff met to review the team's concerns and the unresolved questions. The meeting resulted in the staff taking the following actions:

(a) The NRC staff technical review will concentrate on whether SPC's topical report XN-NF-81-58(P)(A) permits the use of the average $\Delta\rho_{sntr}$ densification value instead of the largest mean $\Delta\rho_{sntr}$ densification value methodology described in Reg Guide 1.126. If the technical review concludes that the topical report does not address the use of the average $\Delta\rho_{sntr}$ densification value, this issue will become a nonconformance. However, at the time of this writing, the staff's technical review is not complete.

(b) The NRR Project Manager for HBR2 will issue an RAI requesting CP&L to describe how the topical and/or SER addresses the use of the average $\Delta\rho_{sntr}$ densification value instead of the largest mean $\Delta\rho_{sntr}$ densification value when applying the 95% UCL. Subsequently, the following has occurred.

- In its letter to CP&L, dated April 21, 1997, the NRC staff requested CP&L to provide a discussion on why the SPC analysis (SPC topical report XN-NF-81-58(P)(A)) uses the average mean $\Delta\rho_{sntr}$ densification value instead of the largest mean $\Delta\rho_{sntr}$ densification value methodology described in Reg Guide 1.126, when applying the 95% UCL.

- CP&L responded in its letter to NRC (RNP-RA/97-0121) dated May 20, 1997. At the time of this writing, that letter is being reviewed by the staff.

(c) The generic issue regarding how many plants may also be affected by SPC's use of the average $\Delta\rho_{sntr}$ densification value instead of the largest mean $\Delta\rho_{sntr}$ densification value when applying the 95% UCL will be addressed at the same time the NRC staff addresses the confusion between topical report XN-NF-81-58(P)(A) and the associated SER.

3.5.4 Thermal-Mechanical Analyses for PWRs

a. <u>Inspection Scope</u>

The inspection of SPC's thermal-mechanical analyses for PWRs concentrated on the fuel design reload analyses for Texas Utilities Electric Company, Comanche Peak Unit 1 Cycle 6 (referred to as CPA-3 by SPC). These analyses verify that a particular reload fuel design meets the specified acceptable fuel design limits (SAFDLs), as required by GDC 10 in Appendix A to 10 CFR Part 50 and specified in Section 4.2 of the NRC's Standard Review Plan, and that the fuel remains within a defined safety envelope.

b. <u>Observations and Findings</u>

To evaluate the PWR thermal-mechanical analyses, the team reviewed the power histories and specified acceptable fuel design limits. The following paragraphs summarize the results of this review.

b.1 *Power Histories*

During this portion of the inspection, the team examined the thermal-mechanical analyses, including the cladding strain (steady-state and transient), corrosion, creep collapse, rod pressure, assembly liftoff, fuel melting, and assembly growth analyses.

All of these analyses strongly depend on fuel rod power histories and/or maximum exposure level; however (the assembly liftoff analysis has less of a dependence on exposure. The NRC-approved methodology for developing the power histories and exposures for input to these analyses requires SPC to identify the fuel rods with the maximum rod power histories for each cycle of operation. In addition, SPC must identify the rod with the maximum exposure at discharge (with neutronics calculational uncertainties included) in order to bound maximum possible operation for a given reload application.

In most plant-specific analyses, SPC has identified generic bounding power histories for a given plant or more than one plant with similar fuel designs and operation. However, this is not always the case for reloads because SPC has exercised the option to use plant/cycle reload-specific power histories with an uncertainty factor to account for the uncertainties in the neutronics calculations for the plant cycles in question. Therefore, SPC's development and application of the power histories/exposures to thermal-mechanical analyses is very important to verify that SAFDLs are satisfied for each reload and are within a defined safety envelope.

For the CPA-3 reload, SPC used the bounding power histories developed for Comanche Peak Unit 2 Cycle 3 (referred to by SPC as CPB-1) because both plants have the same fuel designs and operation. The reload for this cycle includes fuel rods that will be discharged after two and three cycles of operation and, therefore, SPC neutronics analyses identified the peak power rods with two- and three-cycle operation.

For those reload rods with two cycles of operation SPC neutronics analyses identified peak power histories for cycle n (first cycle of operation) and cycle n+1 (second cycle of operation). Similarly, for three-cycle rods, SPC analyses identified peak power histories for cycles n, n+1, and n+2, plus the rod that has the maximum discharge exposure (burnup). (The three-cycle rod with maximum exposure is the maximum exposure rod for this reload.) Therefore, the CPA-3 reload had 6 different peak power histories, which SPC has compared against the bounding power histories developed for CPB-1. The team verified that the 6 peak power histories and maximum rod exposure identified for CPA-3 are indeed bounded by the CPB-1 power histories and maximum exposure defined in SPC calculational notebook E-7691-337-HG. The team found that SPC had determined that the thermal-mechanical analyses for the CPB-1 reload bound the thermal-mechanical analyses for CPA-3 and the former analyses can be applied to the latter reload.

The team questioned SPC on whether they monitored the reload power histories for changes in operational power for cycles n+1 and n+2 to verify that the power histories assumed by SPC remain bounding for these cycles and their thermal-mechanical analyses will also remain valid. The team noted, however, that the licensee can and does change their operational strategies for cycles n+1 and n+2 for a reload because operational requirements for the plant in question can change and the power histories assumed by SPC are determined more than 18 months to 4 years before the actual operation of these cycles. Such possible operational changes by the licensee for cycles n+1 and n+2 raise questions concerning whether the power histories assumed for cycles n+1 and n+2 remain bounding.

SPC responded that it was the responsibility of the SPC plant project engineer to verify whether the assumed peak power histories for cycles n+1 and n+2 remained applicable and whether a reanalysis was required for a particular reload. The team asked two SPC plant project engineers if they were aware of a reload being reanalyzed in response to changes in plant power operation. These project engineers were aware of instances where a reanalysis had been performed because of changes in cycle n of a reload, but they were not aware of a reanalysis ever being performed for the subsequent cycles n+1 and n+2.

The team therefore determined that SPC did not have a procedure to verify that the assumed power histories used for their thermal-mechanical analyses of a given reload remain bounding when the plant power operation changes for cycles n+1 and n+2.

This can result in a fuel reload operating outside of its safety analysis envelope. The team identified this finding as a nonconformance.

As a result of this finding and nonconformance, SPC developed a procedure, documented in attachments to a letter to NRC (HDC:97:038) dated May 1, 1997. That procedure identifies specific criteria to be applied to each operation cycle to determine if a full reanalysis of the fuel rod thermal-mechanical performance is required for specific reloads. The team reviewed the procedure, proposed for application to PWR reloads to be delivered after January 1998, and found it to be acceptable. SPC also included an assessment of two previous thermal-mechanical safety evaluations as evidence that projections of core specific power history had been properly accounted for but not always documented in the thermal-mechanical safety evaluations for SPC fuel in cycles n+1 and n+2. In response to concerns about interim operation with fuel loaded before January 1998, SPC provided a supplemental letter to NRC (HDC:97:086) dated August 13, 1997, with a commitment to interface with all PWR licensees operating with SPC fuel to inform them of the need to review and document the power history assessments in the thermal-mechanical safety evaluation for the current operating cycles. The letter also provides additional justification, based on conservatisms in the power history and mechanical analysis methodologies and on results of previous power history evaluations, that the probability of exceeding design criteria for current operation while corrective actions are being implemented is very small. The staff finds the SPC response to be acceptable, and therefore, the nonconformance is closed.

b.2 *Specified Acceptable Fuel Design Limits*

The inspection team also examined the thermal-mechanical analyses for CPB-1 to verify that these analyses demonstrate that SAFDLs are satisfied for both the CPA-3 and CPB-1 reloads. The specific analyses examined included the cladding strain analysis for normal operation (calculation notebook E-7691-337-5), cladding strain analysis for AOOs (E-7691-337-7), corrosion analysis (E-7691-337-3), creep collapse analysis (E-7691-337-4), rod pressure analysis (E-7691-337-6), assembly liftoff analysis (E-7691-337-100), fuel melting analysis (E-7691-337-7), and assembly growth analysis (E-7691-337-101). Examination of these analyses demonstrated that SPC used NRC-approved analysis methods, and the SAFDLs were met for reloads CPA-3 and CPB-1 using the peak power histories assumed by SPC for each cycle of operation.

c. <u>Conclusions</u>

The team concluded that SPC's thermal-mechanical analyses for PWR reloads are satisfactory, with the exception of the issue concerning the finding that SPC did not have a procedure to evaluate changes in plant power operation for cycles n+1 and n+2 to prevent a reload from operating outside of its safety analysis envelope. The team also reviewed the new procedure developed by SPC to address this finding, and found it to be acceptable.

3.6 Fuel Fabrication Activities

On the basis that SPC was conducting limited fabrication activities during this portion of the inspection, to evaluate SPC's ongoing fuel fabrication activities, the team reviewed SPC's procurement of fuel assembly components and fuel clad tubing, SPC's assessment of its suppliers of fuel clad tubing, the storage of fuel clad tubing and bar stock, the chemical and ceramic operations, the fuel assembly component fabrication and inspection, fuel rod component fabrication and inspection, the pellet loading, and the QA records of fabrication activities. The following paragraphs summarize the results of this review.

3.6.1 Fuel Assembly Component Procurement

a. Inspection Scope

The team reviewed a sample of the procurement documents for safety-related components that were procured by SPC. The objective of this review was to determine whether applicable regulations were imposed and whether SPC was using approved vendors.

b. Observations and Findings

The team reviewed a sample of SPC's purchase orders (POs) for procurement of ATRIUM™-9 inner "water channels," from CarTech (San Diego, California), and SPC's AG (Brennelementwerk-Hanau, Hanau, Germany), tie plate castings from Wyman Gordan/Sierra Cast (Carson City, Nevada), fuel guards (debris catchers) from Ehrhardt Tool & Machine (Granite City, Illinois), and spacers and spacer components from Caran Precision Engineering and Manufacturing (Paramount, California) and Ehrhardt Tool & Machine. In addition, the team reviewed POs for procurement of ATRIUM™-10 water channel and cage assemblies from Siemens Advanced Nuclear Fuels (GmbH, Lingen, Germany).

In each PO reviewed by the team, SPC had imposed its "Quality Assurance Procurement Clauses," Nos. 1 and 23. Clause 1, "QA Program Requirements," states that the vendor shall provide and maintain a documented quality program commensurate with applicable requirements of Appendix B to 10 CFR Part 50, ANSI N45.2, and 50-C-QA. This program shall be approved by SPC and shall be subject to audit, and revisions shall be promptly submitted to SPC for review. Clause 23, "Reporting of Defects and Noncompliance," states that the vendor shall promptly notify SPC of any defect or noncompliance in the products or services supplied, in accordance with the requirements of 10 CFR Part 21.

A review of SPC's approved vendor list (AVL), EMF-595, Revision 20, dated February 3, 1997, revealed that each of the listed vendors above were on SPC's AVL. The AVL also documented such information as whether the vendor was active (approved) or inactive (approval pending), the date of the last SPC audit of the vendor's facility, and the date that the approval expired. The team did not identify any concerns in the method SPC used to control its procurement documents.

c. <u>Conclusions</u>

The team concluded that SPC's procurement documents were appropriately controlled, imposed the required NRC regulations for the procurement of basic components, and were easily retrievable by SPC staff. No adverse findings were identified by the team.

3.6.2 Fuel Clad Tubing Procurement

a. <u>Inspection Scope</u>

The team assessed whether the fuel clad tubing suppliers provided adequate documentation to ensure that the cladding met the applicable specification requirements.

Advanced Nuclear Fuels (ANF), GmbH, in Duisburg, Germany, and Sandvick Specialty Metals Corporation (SSM) in Finley, Washington, supply the fuel clad tubing for fuel rods. ANF and SSM procure zirc-2 or -4 ingots for manufacturing the fuel clad tubing from either Teledyne Wah Chang (Albany, Oregon) or Compagnie Europeene du Zirconium (CEZUS) (located in France).

b. <u>Observations and Findings</u>

The technical requirements for PWR cladding are specified in paragraphs 4.3.1 through 4.3.9. of the SPC Design Specifications, EMF-S35,055, "Zircaloy Tubing for Rod Cladding," Revision 9, dated February 8, 1996. Those for BWR cladding are specified in paragraphs 3.3.1 through 3.3.9. of the SPC Design Specifications EMF-S35,041, "Zircaloy Tubing with Internal Zirconium Liner for the Cladding Material," Revision 5, dated May 10, 1996. The specifications require seamless tubing to be manufactured by a tube reduction process, with alloy and impurities to conform to American Society for Testing and Materials (ASTM) standard B 811. The General Quality Assurance Requirements of the standard requires the seller to submit a process description including but not limited to hot and cold reduction methods; temperatures at which fabrication operations are performed; cleaning, etching and surface preparation methods; identification marking; heat treating times and temperatures; temperature profile of the furnace used for final product anneal; and standard rework steps.

To evaluate the procurement of fuel clad tubing, the team reviewed the POs described in the following paragraphs.

(a) The team selected SPC PO R-073797, issued to ANF on May 6, 1996, to procure 10,500 cladding tubes manufactured in compliance with EMF-S35,055. These tubes were intended for CP&L's H.B. Robinson Plant Unit 2 (a PWR). The objective of this review was to verify whether the vendor supplied adequate documentation to confirm that the cladding supplied met the specified requirements. To accomplish this, the inspector reviewed the ANF-supplied QA records collected in Release File 51927. These records reflected that the requirements in paragraphs 4.3.1 through 4.3.9 were met, and identified the ingot from which the tubes were manufactured and the manufacturer of the ingot.

(b) The team selected SPC PO R-073799, issued to ANF to procure 21,500 fuel clad tubes with an internal Zr liner manufactured in compliance with EMF-S35,041, Revision 4, dated September 18, 1992. These tubes were intended for CommEd's La Salle County plant (a BWR). The objective of this review was to verify whether the vendor supplied adequate documentation to confirm that the cladding supplied met the specified requirements. To accomplish this, the team reviewed the ANF-supplied QA records in Release File 51497. These records revealed that the seamless tubes were manufactured using a tube reduction process, and the Zr inner liner was bonded to the zirc-2 by coextrusion from tube reduced extrusions (TREXs). The vendor supplied the TREXs certificate, physical and chemical properties of the zirc-2, and documentation regarding the Zr liner TREXs. The team reviewed the Quality Acceptance Certificate for the cladding supplied for La Salle County (typical receipt inspections) in which the SPC receipt inspector documented the results of the receipt inspection. The certificate identified the PO; the product and material specifications; and the quantity of tubes inspected, accepted, and released. It also identified the characteristics inspected, inspection methods used to assess the characteristics, the sample size, and the quantity inspected and accepted. The characteristics included overall dimensions, straightness, deformities, and surface roughness on the inner and outer surfaces. Independently, SPC verified the liner chemistry, the thickness of the liner, and liner bonding.

c. <u>Conclusion</u>

The team determined that fuel cladding suppliers had furnished adequate documents necessary to ensure that the cladding supplied for the H.B. Robinson and La Salle County plants met the SPC specifications. No adverse findings were identified by the team.

3.6.3 Fuel Clad Tubing Suppliers

a. Inspection Scope

The team reviewed the audits that SPC performed on SSM and ANF to determine whether SPC conducted periodic audits, surveillance, and over-checks on its vendors to ensure that they conform to the established QA program.

b. Observations and Findings

To evaluate SPC's assessments of its suppliers of fuel clad tubing, the team reviewed the assessments described in the following paragraphs.

(a) In February 1996, during receipt inspection of cladding (typically 70 tubes per lot) intended for PP & L, Susquehanna Unit 2, SPC detected a zirc-2 tube with inside diameter indications. After the discovery, SPC resorted to a 100% ultrasonic testing (UT) examination.

After being notified by SPC, SSM conducted an investigation and determined that its personnel had identified the subject tube to have an indication during its inspection, but instead of separating the tube and placing it in an area where it would have been cleaned and retested, SSM inadvertently shipped the tube to SPC. During April 2 - 3, 1996, SPC conducted an assessment at SSM and confirmed that incorrect handling at the UT examination station caused the defective zirc-2 tube to be placed in an accepted lot and shipped to SPC. During this audit, SPC also identified adverse findings in other areas. In its letter to SPC dated June 19, 1996, SSM acknowledged SPC's adverse findings, and submitted revisions to correct affected procedures. SPC responded in a letter dated June 26, 1996. That letter commented on SSM's letter, and reminded SSM to respond to the items that were still open regarding corrective actions.

On February 18 - 22, 1997, SPC conducted an audit of SSM to evaluate the implementation of SSM's QA Manual, QAM-2, Revision 3, issued on July 14, 1995, and to determine if SSM was eligible to remain on its AVL. SPC identified several adverse findings, including:

(1) inadequate training of recently-hired personnel
(2) working outside the specification parameters
(3) training to improve the level of understanding of operators
(4) operators being instructed to work outside the defined requirements
(5) process changes and process parameters that were being changed without the benefit of process change authorizations
(6) inadequate SSM management response to SPC's audit findings

These findings were ultimately resolved to SPC's satisfaction such that SSM remained on SPC's AVL.

(b) On September 18 - 22, 1995, SPC conducted an assessment (not an audit, because ANF is a subsidiary of SPC) of ANF (Duisburg, Germany). The assessment covered activities associated with receipt inspection, pilfering[11], annealing, straightening, polishing, testing, and inspection. During the assessment period, ANF was manufacturing zirc-4 fuel clad tubing intended for the fabrication of fuel for Comanche Peak Units 1 and 2. SPC's assessment did not identify any adverse findings.

c. <u>Conclusion</u>

The team concluded that SPC conducted adequate audits of its fuel clad tubing suppliers to control the quality. When SPC identifies quality problems with purchased items during receipt inspection or during manufacture, SPC conducts surveillance activities to investigate and correct the problem. No adverse findings were identified by the team.

3.6.4 Fuel Clad Tubing and Bar Stock Storage

a. <u>Inspection Scope</u>

The team observed the storage of zirc fuel clad tubing and bar stock in the warehouse to determine the adequacy of SPC's storage procedures.

b. <u>Observations and Findings</u>

The team observed that the zirc fuel clad tubing and bar stock materials were identified with appropriate tags in the warehouse. ANF shipped cladding sealed in plastic sheets, which were removed on receipt. The cladding was inspected and stored on shelves with the ends unprotected. Before loading fuel for a specific job, the cladding tubes were retrieved from the warehouse, cleaned ultrasonically, washed in deionized water, and dried before loading fuel. The SPC representative admitted that the practice of removing the protective covers before storing the cladding tubes inadvertently exposed them to the warehouse atmosphere. The SPC representative informed the team that SPC was considering a suitable method by which it could inspect the tubes without removing the protective covers and store them until the covers were removed when the tubes were used.

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[11]Pilfering is the process of cold reduction of the extruded tube shells or TREXs to produce tube hollows. Pilfering is also known as "cold pilfering," "tube reducing," and "rocking."

c. **Conclusion**

With the exception of the weakness regarding the removal of the protective covering, the team determined that SPC's storage and handling of the fuel clad tubing and bar stock was adequate. No adverse findings were identified by the team.

3.6.5 Chemical, Ceramic, and Pellet Operations

To evaluate the chemical, ceramic, and pellet operations, the team reviewed the conversion of uranium hexafluoride (UF_6) gas to uranium dioxide (UO_2) powder; the blending of UO_2 powder for isotopic enrichment; the pelleting operations; the pellet integrity verification activities; and the final inspection, storage, and release of pellets for rod loading. The following sections summarize the results of this review.

3.6.5.1 *Conversion*

a. **Inspection Scope**

The team evaluated the process for conversion of uranium hexafluoride gas to uranium dioxide powder. This evaluation included the review of processes for conversion of recycled scrap material bearing uranium (U). In addition, the team evaluated the process control parameters and system, the sample points, and the interfaces with the laboratory and QA.

b. **Observations and Findings**

The team reviewed procedures which apply to the conversion of UF_6 gas to UO_2 powder documented in procedure EMF-22, "Standard Operating Procedures — Plant Operations," Revision 275, dated December 3, 1996. The process specifications are listed in EMF-268, "Numerical Listing of Process Specifications," Revision 248, dated December 12, 1996. The analytical procedures are in EMF-103, "Analytical Directives and Procedures," Revision 169, dated October 15, 1996.

The process engineer for conversion operations issues a parameter sheet, which covers the parameters to be controlled for a particular lot or project. The team observed the conversion process for the preparation of UO_2 powder for a nominal enrichment of 4.20 w/o U_{235}. In addition, the team examined checklists for the clean-out of equipment between enrichments.

c. Conclusions

The team concluded that SPC adequately controls the processes for the conversion of UF_6 gas to UO_2 powder, through the use of procedures, specifications, process parameter instructions, sampling, and analyses. No adverse findings were identified by the team.

3.6.5.2 *Blending*

a. Inspection Scope

The team evaluated the adequacy of the blending, control, and verification of specified UO_2 powder isotopic enrichments, as well as the homogeneous quality of the powder lots/batches, the process control parameters and systems, the sample points, and the interfaces with the laboratory and QA.

b. Observations and Findings

During this portion of the review, the team examined four documents related to qualification of the blender for blending UO_2 powders:

- Exxon Nuclear Company, Inc., internal memorandum regarding, "Nauta-Mix Blender Qualification Test - UF_6 Expansion," dated March 27, 1978

- Process Test Authorization, Test No. XN-NF-PTA-281, Job Code Q350/R37, "Qualification of ± 1.0% Dry Enrichment Blending Process," dated August 22, 1980

- Exxon Nuclear Company, Inc., internal memorandum regarding, "Qualification of Test Results for Dry Enrichment Blending Within ± 1.0% U_{235} Range," dated December 17, 1980

- EMF-1569(P), "Review of Bases for Enrichment Range and Sample Size for Dry Enrichment Blending," dated January 1994

The use of the "Turbula" mixer for mixing the contents of 45 gallon drums and the use of tumble mixers for mixing the contents of smaller cans are qualified on the basis of proven acceptable results.

The procedures that apply to the blending of UO_2 powders are documented in EMF-22. The process specifications are listed in EMF-268. The analytical procedures are documented in EMF-103.

The process engineer for conversion operations issues a parameter sheet that covers the parameters to be controlled for a particular lot or project. The process engineer for ceramic operations issues a parameter sheet for blending uranium dioxide powders with die lubricants, pore formers, and gadolinia.

The team observed the process of blending UO_2 powder, and examined the checklists for clean-out of equipment between enrichments. The team also observed SPC's storage of drums of blended UO_2 powders, as well as their removal from storage for ceramic operations. In addition, the team observed SPC's blending of pore formers, die lubricants, and gadolinia, interviewed laboratory personnel, and discussed and verified the analytical procedures and calibration of analytical devices.

c. <u>Conclusions</u>

The team concluded that SPC adequately controls the processes for blending UO_2 powder for isotopic enrichment through the use of procedures, specifications, process parameter instructions, sampling, and analyses. No adverse findings were identified by the team.

3.6.5.3 *Pelleting*

a. <u>Inspection Scope</u>

The team evaluated the pelleting processes, the process control parameters, the sample points, and the interfaces with the laboratory and QA used to produce UO_2 fuel pellets and nuclear absorber fuel (NAF) pellets containing gadolinia (Gd).

b. <u>Observations and Findings</u>

The procedures that apply to the production of UO_2 and NAF pellets are documented in EMF-22 and the quality standards are listed in EMF-315.

The process engineer for ceramic operations issues a parameter sheet governing the parameters to be controlled for a particular project or lot. The team also interviewed the manufacturing and inspection personnel, and observed the manufacture of UO_2 pellets for shipment to Germany as well as the manufacture of NAF pellets for Framatome Cogema Fuels in Lynchburg, Virginia. In addition, the team observed the quality control of pellets during manufacture and discussed the laboratory analysis of pellets with analytical personnel.

c. **Conclusions**

The team concluded that SPC adequately controls the processes for production of UO_2 and NAF pellets through the use of procedures, quality standards, process parameter instructions, checklists, sampling, and analyses. No adverse findings were identified by the team.

3.6.5.4 *Integrity*

a. **Inspection Scope**

The team evaluated the methods, process control parameters, and interfaces with the laboratory and QA used to verify pellet integrity on the basis of pellet outside diameter, roughness, green density, sintered density, resintering data, length and dish dimensions, thermal-stability, hydrogen content, isotopic enrichment, chemistry, and other characteristics.

b. **Observations and Findings**

The analytical procedures that apply to the production of pellets are documented in EMF-103. In-process inspection of green pellets, sintered pellets, and resintered pellets are governed by inspection parameter sheets prescribing when and how many samples should be taken, as well as the specifications that apply to the lot or project number.

The team observed SPC's quality control of pellets during manufacture, and interviewed the QC inspectors. In addition, the team interviewed laboratory personnel and discussed and verified the analytical procedures and calibration of analytical devices.

c. **Conclusions**

The team concluded that SPC adequately regulates the methods and processes used to control pellet integrity through the use of analytical procedures and pellet inspection parameter sheets to ensure that the product meets applicable specifications. No adverse findings were identified by the team.

3.6.5.5 *Inspection, Storage, and Release*

a. **Inspection Scope**

The team evaluated the processes for final inspection, storage, and release of the pellets for rod loading, as well as the process control parameters and system, and the interfaces with the laboratory and QA.

b. **Observations and Findings**

The procedures that apply to the final inspection and storage of pellets are documented in EMF-22. In addition, every contract has an inspection plan that specifies the frequency and number of samples to be taken and the acceptance criteria. Pellets leaving the grinder which are deemed acceptable by the grinder operator are loaded onto trays; from which the QC inspectors choose pellets at random for verification. A traveler on the pellet stack identifies the top tray and bottom trays by number, the net weight of the stack, the nominal enrichment, the lot number, and the project number. A complete clean-out of equipment takes place between enrichments. When rod loading requests fuel from storage, they are sent complete stacks of pellets. Those pellets not used are returned to storage for further disposition.

c. **Conclusions**

The team concluded that SPC adequately controls the processes for final inspection, storage, and release of pellets for rod loading through the use of established procedures and inspection plans. No adverse findings were identified by the team.

3.6.6 Fuel Assembly Components

a. **Inspection Scope**

The team observed QC inspectors performing component inspections, discussed inspection methodology with area QC inspectors, and reviewed selected component drawing and specification requirements to verify that the critical attributes were correctly identified in applicable QC inspection procedures. Specifically, the team observed spacer assembly as well as optical comparator and welding activities; conducted discussions with the area technicians; and reviewed applicable specifications, drawings, and procedures.

b. **Observations and Findings**

The team conducted discussions with final and in-process QC inspectors regarding the scope of inspection activities in the spacer assembly area and the receipt inspection area. In both areas, the team asked the QC inspectors to go through inspection attributes for specific spacer components. For example, the team requested a QC inspector to explain inspection steps for an "ULTRAFLOW™" ATRIUM™-9 spacer assembly. The team there by determined that the "ULTRAFLOW™" spacer assembly comes in assembled from the sub-tier manufacturer, Caran (Paramount, California), with the exception of the spacer side plates which are attached and welded at the SPC facility. Another QC inspector was asked by the team how the angularity of the "ULTRAFLOW™" vanes was verified, and the QC inspector showed the team the particular drawing angular requirement, the specified drawing tolerance for the vanes, and the QC inspection procedure that contained the drawing requirement inspection verification. For example, SPC drawing EMF-308,532, "Inner Strip 9 X 9-IX," Revision 3, approved on November 12, 1993, delineates the inner strip vane angularity requirements and SPC QC inspection procedure EMF-P69116, "Procedure for Inspection of ULTRAFLOW™ Spacer Parts and Spacers," Revision 0, dated January 29, 1997, contained the QC inspection verification requirements.

Spacer Assembly

The team also reviewed applicable spacer assembly inspection and test plans for both the SPC facility and for the source inspection that is performed on each spacer assembly lot at the Caran facility. The QC inspector stated that SPC has a QC contractor that they use for inspecting each lot of the spacer assemblies at Caran, where they perform over-checks and inspection verifications during receipt, in-process and final inspection activities. For example, EMF-P69,116 requires the QC inspectors to check the orientation of formed features including mixing vanes, tabs, anti-hang-up tabs and strip offsets. The specific requirement was 20° (± 5°). The team did not identify any concerns in this area.

The team observed an SPC technician assembling spacer assemblies at an assembly workstation, and noted that the workstation had assembly aid diagrams for the applicable spacer that was being assembled. The workstation had numbered bins containing the specific spacer strips and springs for the particular spacer type that was being assembled. The team noted that the technician visually verified correct assembly and configuration — as he was assembling the spacer assembly, and again after the assembly was completed — by means of SPC's system of letters/numbers that were etched/stamped on different locations on the spacer strips. After the assembly was completed, but before the assembly went to the welding operation, the technician verified correct assembly by observing that the etched letters/numbers were at the correct locations, and by means of an optical comparator. The technician stated that the optical comparator verification after assembly was required for all spacer assemblies.

Spacer Welding

The team followed the spacer assembly to the welding operation and held discussions with the welding operators. The team also observed the welding operators performing welding operations. Discussions with the operators identified that the operators perform a final configuration verification before they place the spacer assembly side plates onto the spacer assembly tabs and prepare the assembly for welding. Each spacer assembly with side plates is placed in a welding aid fixture before it is put into the semi-automatic welding machine. The operator observes each spacer strip junction weld to verify that each weld is being appropriately performed.

The team observed that all of the process controls appeared to be adequately controlled and executed.

Training

The personnel were knowledgeable about the jobs they were performing, had the applicable procedures available at their workstations, were able to show the team specific requirements when queried, and exhibited pride and ownership of the product for which they were responsible. The team subsequently reviewed training and certification records of the technicians who were interviewed during this inspection, and identified no concerns.

c. Conclusions

The team concluded that the QC inspectors were competent, knowledgeable, and well-versed in their job responsibilities, drawings, requirements, and the required inspection attributes. The team also concluded that the spacer manufacturing area was well controlled and staffed by knowledgeable, competent personnel who displayed ownership of the product being fabricated. No adverse findings were identified by the team.

3.6.7 Fuel Rod Components

a. Inspection Scope

To evaluate the adequacy of inprocess inspections, the team observed the manufacturing of various fuel rod components, such as upper-end caps for fuel rods intended for FP&L's St Lucie Plant, and upper-end caps and retaining nuts for boron rod cluster assemblies intended for Texas Utilities Electric Company, Comanche Peak plant.

b. Observations and Findings

SPC uses a "manufacturing follower" (MF) to prescribe the operating steps for manufacturing a component and the inprocess inspections that are to be performed. The team reviewed several MFs and determined that they identified the component and its release number (for documentation traceability), and contained relevant information such as the PO number for the bar stock raw materials, the ingot from which the bar stock was formed, and the drawing and part numbers. SPC also uses a video measuring device to perform in-process inspections to ensure that the parts meet the drawing and specification requirements. QCP EMF-P26,9061, "Smart Scope Operation," Revision 1, dated July 13, 1993, describes the practices used to operate Smart Scope for inprocess inspection of components by QC inspections. The machine shop supervisor has the responsibility to oversee all inprocess inspections of the end caps. The QC supervisor has the responsibility to have all inspections performed by following prescribed inspection and test plans. QC personnel used quality standard EMF-Q35,009, "Parts Machined from Zircaloy Rod or Bar Stock," Revision 11, dated August 27, 1993, to inspect the machined parts.

c. Conclusion

The team found that the MFs contained complete information for the various parts being manufactured. The prescribed inprocess inspections were completed, and the first completed part was inspected with the Smart Scope. No adverse findings were identified by the team.

3.6.8 Fuel Rod Loading

a. Inspection Scope

The team evaluated fuel rod loading, including the control of lattice fissile enrichments, control of the fuel column length, and internal void volume.

b. Observations and Findings

The procedures that apply to loading pellets into fuel rods are documented in EMF-22 and the process specifications are documented in EMF-268. A specification sheet for the rods being loaded identifies the pellets to be loaded into each zone, as well as the length of each zone. The weights are recorded by automatic entry into the computer system. Loading is verified by use of a passive scanner on the completed rod after welding the upper-end cap.

The team observed the loading of UO_2 and NAF pellets into clad tubing.

c. Conclusions:

The team concluded that SPC adequately controls the fuel rod loading process through the use of procedures and inprocess inspection of the loaded clad tubing. No adverse findings were identified by the team.

3.6.9 Quality Assurance Records

a. Inspection Scope

The team reviewed QA records to ascertain if the records were readily retrievable and contained adequate information to verify the quality of the manufactured component, the bar stock from which it was manufactured, and the quality of the ingot from which the bar stock was produced.

b. Observations and Findings

The team examined the following three releases for manufacturing fuel rod components:

- Release 51298 pertained to 528 lower-end caps intended for ComEd's Quad Cities Plant
- Release 51225 pertained to 3861 upper-end caps intended for Quad Cities Plant
- Release 52560 pertained to 2559 lower-end caps for CP&L's Shearon Harris Plant

All three releases contained the QC acceptance certificates, as well as the following information:

- drawing and part numbers to which the part was manufactured
- visual standard to which the part was inspected
- bar stock from which the part was manufactured
- results of QC inspections to verify the attributes specified in ANF-S35,009, "Parts Machined from Zircaloy Rod or Bar Stock," Revision 9, dated May 2, 1995
- results of the tests conducted on the parts for residual contaminants
- inprocess inspection reports documenting the results of the inspections performed during the various manufacturing steps
- dimensions of the parts measured during manufacture

Release 51053 pertained to the release of zirc-4 bar stock. Specification EMF-S35,007, "Zircaloy Bar and Rod Stock," Revision 8, dated July 17, 1995, specified the requirements for the bar stock, and required documentation on the attributes. The bars were to be in minimum lengths of 3 feet and were required to be marked in indelible ink with the type of material and the SPC lot number. The records indicated that the specification requirements were met. During a tour of the warehouse, the team verified that there were markings on all bar stock lengths to identify the type of material and the SPC lot number.

The team reviewed two SPC POs (R-073638, and R-073870) issued to Teldyne Wah Chang for the supply of various lengths of zirc-2 and zirc-4 from which components for Quad Cities and Shearon Harris had been fabricated. The objective of this review was to determine whether the certifications on the ingots were acceptable. The POs required the ingots to be triple arc melted. Teldyne Wah Chang, the manufacturer of the ingot, from which the bar stock was made, certified that the ingot was triple melted, and provided results of the chemical and physical tests.

c. <u>Conclusion</u>

The team concluded that the documentation package contained adequate information to ensure that the material met the specification requirements. No adverse findings were identified by the team.

4 ENTRANCE AND EXIT MEETINGS

During the entrance meeting on February 9, 1997, the NRC team met with members of SPC management and staff, and discussed the scope of the inspection. The team also reviewed its responsibilities for handling proprietary information, as well as those of SPC. In addition, the team established contact persons within the management and staff of the applicable SPC organizations.

During its exit meeting with SPC management and staff on May 13, 1997, in Rockville, Maryland at NRC headquarters, the team discussed its findings and concerns, as well as SPC's strengths and weaknesses.

APPENDIX A

PARTIAL LIST OF PERSONS CONTACTED

D. A. Adkisson	Manager, Engineering Quality Performance
B. F. Bentley	Manager, Plant Operations
O. C. Brown	Manager, BWR Neutronics
R. E. Collingham	Manager, Safety Analysis Methods
H. D. Curet	Manager, Product Licensing
L. J. Federico	Manager, Nuclear Engineering
B. N. Femreite	Vice President, Engineering & Manufacturing
R. L. Feuerbacher	Manager, Materials & Scheduling
M. E. Garrett	Manager, BWR Safety Analysis
R. C. Gottula	Manager, PWR Safety Analysis
L. E. Hansen	Manager, Customer Projects
B. G. Haugen	General Supervisor, Cage & Assembly Operations
M. A. Hendrickson	Supervisor, Rod Operations
D. J. Hill	Manager, Quality Engineering
T. H. Keheley	Steady State Methods
D. C. Killian	Manager, Process Engineering
W. D. Krebs	Director, Research & Technology
M. A. Law	Manager, Analytical Services
R. B. Logsdon	General Supervisor, Ceramic Operations
D. G. McAlees	Senior Vice President & General Manager
L. A. Nielsen	Manager, PWR Neutronics
J. H. Nordahl	Vice President, Sales & Projects
L. D. O'Dell	Manager, Engineering Automation & Code Maintenance
J. H. Phillips	General Supervisor, Chemical Operations
C. M. Powers	Manager, Quality
R. S. Reynolds	Manager, Methods & Codes
A. Repáraz	Manager, Product Mechanical Engineering
J. A. Shurts	Manager, Inspection Services
J. R. Tandy	Team Leader, Product Engineering
G. N. Ward	Manager, Manufacturing Engineering

APPENDIX B

ITEMS OPENED AND CLOSED

Opened:

Nonconformance 99900081/97-01-01:

(1) SPC failed to verify its application of the ANFB correlation to the ATRIUM™-10 fuel design that resulted in the local peaking factor and a nonconservative flow-bias that were outside NRC-approved methodology. (Section 3.3.5.1)

(2) SPC failed to verify its application of the ANFB correlation to the ATRIUM™-9 fuel design that resulted in an inadequate number of test points and range of conditions to support the design. (Section 3.3.5.2)

Nonconformance 99900081/97-01-02:

(1) SPC failed to perform adequate V&V on codes bought from sources outside SPC. (Section 3.3.2.b.1)

(2) SPC failed to document input errors or their effect on the analysis. (Section 3.3.2.b.4)

(3) SPC failed to document and confirm assumed causal factors and consider variables. (Section 3.3.2.b.5)

(4) SPC failed to perform adequate V&V on major modifications of the RELAX and FLEX codes. (Section 3.3.2.b.6)

Nonconformance 99900081/97-01-03:

SPC failed to comply with 10 CFR 50.46. (Section 3.3.2.b.3)

Nonconformance 99900081/97-01-04:

(1) SPC failed to establish adequate procedures for V&V of code development and modifications. (Section 3.3.2.b.2)

Nonconformance 99900081/97-01-04 continued:

(2) SPC failed to establish adequate measures to control code errors. (Section 3.3.1)

(3) SPC failed to establish adequate measures for the selection of the appropriate entrainment fraction. (Section 3.3.3.2)

Nonconformance 99900081/97-01-05:

SPC failed to establish an adequate training program. (Section 3.2.1.b.3)

Unresolved Item 99900081/97-01-06:

Additional information is needed to resolve questions about the use of Semiscale pump data versus relevant pump data as required by Appendix K to 10 CFR Part 50. (Section 3.3.4.1.b.1)

Unresolved Item 99900081/97-01-07:

Additional information is needed to demonstrate that the FCTF reflood heat transfer correlation produces conservative results for other than 17 x 17 rod arrayed fuel designs. (Section 3.3.4.2.b.1)

Unresolved Item 99900081/97-01-08:

Additional information is needed to demonstrate that the FCTF reflood heat transfer correlation produces conservative results by either providing documentation that the 17 x 17 rod array data was analyzed using a valid data analysis code or perform a complete reanalysis of the data. (Section 3.3.4.2.b.2)

Unresolved Item 99900081/97-01-09:

Additional information is needed to demonstrate the adequacy of SPC's underlying assumptions about the different coefficients for each fuel design to which the ANFB correlation is applied. (Section 3.3.5.2)

ITEMS OPENED and CLOSED Continued

Unresolved Item 99900081/97-01-10:

Additional information is needed to demonstrate that the analysis performed for the St. Lucie Unit 1 Cycle 14 reload was consistent with NRC-approved methodology. (Section 3.5.2.b.2)

Unresolved Item 99900081/97-01-11:

Additional information is needed to describe how SPC's topical report XN-NF-81-58(P)(A) and the associated SER address the use of the average $\Delta\rho_{sntr}$ densification value instead of the largest mean $\Delta\rho_{sntr}$ densification value when applying the 95% UCL, and to address the confusing statements in that topical report and associated SER. (Section 3.5.3)

<u>**Closed:**</u>

Nonconformance 99900081/94-01-01:

See Section 2.1.

Nonconformance 99900081/94-01-02:

See Section 2.2.

Nonconformance 99900081/94-01-03:

See Section 2.3.

APPENDIX C

INSPECTION TEAM MEMBERS

NRC

Mr. Steven M. Matthews	Team Leader, NRR/DISP/PSIB
Dr. Steven A. Arndt	AEOD/TTD
Dr. Anthony C. Attard	NRR/DSSA/SRXB
Mr. Geoffrey R. Golub	NRR/DSSA/SRXB
Dr. Tai L. Huang	NRR/DSSA/SRXB
Mr. Edward D. Kendrick	NRR/DSSA/SRXB
Dr. Ralph R. Landry	NRR/DSSA/SRXB
Dr. Alan E. Levin	NRR/DSSA/SRXB
Mr. Kamalakar R. Naidu	NRR/DISP/PSIB
Mr. Joseph J. Petrosino	NRR/DISP/PSIB
Mr. Laurence E. Phillips	Chief, NRR/DSSA/SRXBA
Dr. Joseph L. Staudenmeier	NRR/DSSA/SASG
Mr. Anthony P. Ulses	NRR/DSSA/SRXB
Dr. Shih-Liang Wu	NRR/DSSA/SRXB

Pacific Northwest National Laboratory

Mr. Carl E. Beyer
Ms. Judith M. Cuta

Brookhaven National Laboratory

Dr. John F. Carew
Dr. Carl J. Czajkowski
Dr. Jae H. Jo
Dr. Upendra S. Rohatgi

Parámeter, Inc.

Mr. Rodney L. Grow
Mr. Patrick S. Lacy
Mr. Alexander C. Schafer