DUKE NON PROPRIETARY

DPC-NE-2007A

DUKE POWER COMPANY
FUEL RECONSTITUTION
ANALYSIS METHODOLOGY

Submitted: September 1993

Approved: October 1995

Duke Power Company
Nuclear Generation Department
Charlotte, North Carolina

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 27, 1995

Mr. M. S. Tuckman
Senior Vice President
Nuclear Generation
Duke Power Company
P. O. Box 1006
Charlotte, North Carolina 28201-1006

SUBJECT: SAFETY EVALUATION REGARDING TOPICAL REPORT DPC-NE-2007P, "FUEL RECONSTITUTION ANALYSIS METHODOLOGY" (TAC NO. M88082)

Dear Mr. Tuckman:

By letter dated September 23, 1993, Duke Power Company (DPC) submitted Topical Report DPC-NE-2007P, "Fuel Reconstitution Analysis Methodology." This report presented DPC's methodology for performing analyses to support fuel assembly reconstitution, and was prepared in response to the guidance in Generic Letter 90-02, Supplement 1. The topical report is intended to be applicable to Oconee, Units 1, 2, and 3; McGuire, Units 1 and 2; and Catawba, Units 1 and 2. A supplemental response dated August 8, 1994, was submitted in response to our request for additional information dated April 26, 1994.

The NRC staff and its contractor, Pacific Northwest Laboratory (PNL), have completed their review of the topical report and the supplemental submittal. The staff's Safety Evaluation (SE) and PNL's Technical Evaluation Report are enclosed. The staff concluded that DPC-NE-2007P is acceptable for reload licensing applications for the use of solid replacement rods within the limits specified in the conclusion section of the SE. The staff does not approve of the use of vacancies in place of fuel rods because the fuel assemblies with vacancies do not conform to NRC-approved thermal-hydraulic, seismic, and loss-of-coolant accident (LOCA) analyses.

The staff does not intend to repeat its review of the matters described in the report and found acceptable when the report is referenced in a license application, except to ensure that the material presented is applicable to the specific plant involved. Staff acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, DPC must publish accepted proprietary and non-proprietary versions of this report. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an "-A" (designating accepted) following the report identification symbol.

Should NRC criteria or regulations change so that staff conclusions regarding the acceptability of the report are invalidated, DPC will be expected to revise and resubmit their documentation, or to submit justification for continued effective applicability of the topical report without revision of their documentation.

If you have questions regarding this matter, please contact Len Wiens at (301) 415-1495.

Sincerely,

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, 50-287,
50-369, 50-370,
50-413, and 50-414

Enclosure:
Safety Evaluation w/attached
Technical Evaluation Report

cc w/enclosure:
See next page

cc:
Dr. John M. Barry
Mecklenburg County
Department of Environmental
Protection
700 N Tryon Street
Charlotte, North Carolina 29202

County Manager of Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Mr. J. E. Snyder
Regulatory Compliance Manager
Duke Power Company
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Senior Resident Inspector
c/o U.S. Nuclear Regulatory
Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Mr. Dayne H. Brown, Director
Department of Environmental
Health and Natural Resources
Division of Radiation Protection
P. O. Box 27687
Raleigh, North Carolina 27611

County Supervisor of Oconee County
Walhalla, South Carolina 27621

Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
Seneca, South Carolina 29679

Mr. Ed Burchfield
Compliance
Duke Power Company
Oconee Nuclear Site
P. O. Box 1439
Seneca, South Carolina 29679

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Division
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852

Manager, LIS
NUS Corporation
2650 McCormick Drive, 3rd Floor
Clearwater, Florida 34619-1035

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

Mr. W. R. McCollum
Site Vice President
Catawba Nuclear Station
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Mr. T. C. McMeekin
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

Duke Power Company

Catawba Nuclear Station
McGuire Nuclear Station
Oconee Nuclear Station

cc:

Mr. Z. L. Taylor
Regulatory Compliance Manager
Duke Power Company
4800 Concord Road
York, South Carolina 29745

A. V. Carr, Esquire
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

North Carolina Municipal Power Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626

Mr. Peter R. Harden, IV
Account Sales Manager
Westinghouse Electric Corporation
Power Systems Field Sales
P. O. Box 7288
Charlotte, North Carolina 28241

County Manager of York County
York County Courthouse
York, South Carolina 29745

Richard P. Wilson, Esquire
Assistant Attorney General
S.C. Attorney General's Office
P. O. Box 11549
Columbia, South Carolina 29211

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

North Carolina Electric Membership Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Saluda River Electric Cooperative, Inc.
P. O. Box 929
Laurens, South Carolina 29360

Senior Resident Inspector
4830 Concord Road
York, South Carolina 29745

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW Suite 2900
Atlanta, Georgia 30323

Max Batavia, Chief
Bureau of Radiological Health
South Carolina Department of Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. G. A. Copp
Licensing - EC050
Duke Power Company
526 South Church Street
Charlotte, North Carolina 28242-0001

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT DPC-NE-2007P
DUKE POWER COMPANY FUEL RECONSTITUTION ANALYSIS METHODOLOGY

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-269, 50-270, 50-287, 50-369, 50-370, 50-413, AND 50-414

1.0 INTRODUCTION

By letter dated September 23, 1993, Duke Power Company (licensee or DPC), submitted Topical Report DPC-NE-2007P, "Duke Power Company Fuel Reconstitution Analysis Methodology," for NRC review.

Topical Report DPC-NE-2007P describes a methodology of using filler rods (Zircaloy-4 or stainless steel) or vacancies to replace failed or damaged fuel rods during reconstitution of fuel assemblies for core reloads. The filler rods require mechanical, neutronic, and thermal-hydraulic analyses to demonstrate that the inclusion of the filler rods in fuel assemblies with the specific configurations and core locations chosen for a specific fuel cycle is acceptable with respect to the overall fuel performance and safety conclusions. However, the staff does not approve the use of vacancies for reconstitution because the vacancies do not conform to the thermal-hydraulic seismic and loss-of-coolant accident analyses.

The NRC staff was supported in this review by its consultant, Pacific Northwest Laboratory (PNL). The consultant's Technical Evaluation Report (TER), which is attached, provides technical findings relative to its review.

2.0 EVALUATION

The staff has reviewed the attached TER, and concludes that the TER provides an adequate technical basis to approve Topical Report DPC-NE-2007P. Therefore, the staff agrees with PNL's conclusion that the proposed fuel reconstitution methodology is conservative and is acceptable for filler rods. Based on our review, the staff adopts the findings in the attached TER.

3.0 CONCLUSIONS

The staff has reviewed DPC's fuel assembly reconstitution methodology described in Topical Report DPC-NE-2007P, and finds it acceptable for reload licensing applications.

However, the licensee's use of solid replacement rods for fuel reconstitution in accordance with this document should conform to certain limitations as described in TER Section 6.0: (1) no more than 10 solid rods per assembly, (2) no more than 3 solid rods in a row, and (3) cycle-specific reload analyses must include the exact configuration and core location of reconstituted rods and assemblies. The staff does not approve the use of vacancies because the vacancies do not conform to thermal-hydraulic, seismic and LOCA analyses.

Attachment:
Technical Evaluation Report

Principal Contributor: S. Wu

Date: October 27, 1995

TECHNICAL EVALUATION REPORT OF TOPICAL REPORT DPC-NE-2007 (DUKE POWER COMPANY FUEL RECONSTITUTION ANALYSIS METHODOLOGY)

C. E. Beyer
J. M. Cuta

October 1995

Prepared for
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
under Contract DE-AC06-76RLO 1830
NRC FIN I2009

Pacific Northwest Laboratory
Richland, Washington 99352

LIST OF ACRONYMS

AOO	-	Anticipated Operational Occurrence
ASME	-	American Society of Mechanical Engineers
BWFC	-	Babcock & Wilcox Fuel Company
CHF	-	Critical Heat Flux
DNB	-	Departure from Nucleate Boiling
DNBR	-	Departure from Nucleate Boiling Ratio
Duke	-	Duke Power Company
ECCS	-	Emergency Core Cooling System
GDC	-	General Design Criterion
LOCA	-	Loss of Coolant Accident
NRC	-	U.S. Nuclear Regulatory Commission
PCI	-	Pellet Cladding Interaction
PCT	-	Peak Cladding Temperature
PNL	-	Pacific Northwest Laboratory
RIA	-	Reactivity Insertion Accident
SAFDL	-	Specified Acceptable Fuel Design Limit
SRP	-	Standard Review Plan
SSE	-	Safe-Shutdown Earthquake
TER	-	Technical Evaluation Report

CONTENTS

1.0 INTRODUCTION

The routine operation and handling of fuel assemblies can damage individual fuel rods and assembly structures. The commercial nuclear industry has determined the reconstitution of these damaged fuel assemblies to be economical. In order to support customer needs and remain competitive, the fuel vendors have began to reconstitute and recage individual fuel assemblies. Recently, the U.S. Nuclear Regulatory Commission (NRC) has raised several issues with the fuel vendors concerning the application of reconstituted assemblies because the number of reconstituted assemblies has been increasing in the last few years (Reference 1).

In response to the NRC concerns, Duke Power Company (Duke) has submitted to the NRC, a topical report entitled "*Duke Power Company Fuel Reconstitution Analysis Methodology*", DPC-NE-2007 (Reference 2), for review and approval. The remainder of this report will refer to Duke Power Company as only Duke. Presented in Reference 2 is the information required to support the licensing basis for the implementation of reconstituted fuel assemblies in Duke plant reloads. This Technical Evaluation Report (TER) will only review the Duke proposed criteria and evaluation methodology for reconstituted assemblies using up to ten type 304 stainless steel or zircaloy solid replacement rods or one water hole (vacancy) per assembly quadrant, i.e., four per assembly. It is assumed throughout this review that the actual act of removing fuel rods and replacing them with solid rods in a fuel assembly does not stress the regular fuel assembly components beyond the assembly design tolerance limits. The submitted topical report compliments previously approved Duke reload design methodology, including DPC-NE-2001-A (Reference 3), DPC-NE-1002-A (Reference 4), and NFS-1001 (Reference 5). Duke is also proposing to replace damaged fuel rods with fuel rods containing natural UO_2 for reconstituted assemblies. The design and analysis methodology for fuel rods with natural UO_2 are covered under References 3, 4, and 5 for reload fuel. Therefore, this review has concentrated only on the use of solid replacement rods and vacancies in reconstituted assemblies.

Pacific Northwest Laboratory (PNL) has acted as a consultant to the NRC in this review. As a result of the NRC's staff and their PNL consultant's review of the topical report, a list of questions were sent by the NRC to Duke, requesting clarification of specific evaluation methodology and licensing analyses (Reference 6). Duke responded to those questions in Reference 7. Duke was further questioned in a May 9, 1995 conference call on two unresolved issues on the use of vacancies due to grid cell damage in their reconstituted assemblies. Specifically, the two issues were 1) the potential for fretting wear of rods adjacent to the damaged spacer cell/grid and 2) the impact on seismic-LOCA loading.

This review was based on those licensing requirements identified in Section 4.2 of the Standard Review Plan (SRP) (Reference 8). The objectives of this fuel system safety review, as described in Section 4.2 of the SRP, are to provide assurance that: 1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), 2) fuel system damage is never so severe as to prevent control rod insertion when it is required, 3) the number of fuel rod failures is not underestimated for postulated accidents, and 4) coolability is always maintained. A "not damaged" fuel system is defined as fuel rods that do not fail, fuel system dimensions that remain within operational tolerances, and functional capabilities that are not reduced below those assumed in the safety analysis. Objective 1, above, is consistent with General Design Criterion (GDC) 10 [10 Code of Federal Regulations (CFR) 50, Appendix A] (Reference 9), and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks, and that the first fission product barrier (the cladding) has, consequently, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR 100 (Reference 10) for postulated accidents. "Coolable geometry" means in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels. This permits removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident (LOCA) are given in 10 CFR 50, Section 50.46.

In order to assure that the above stated objectives are met, and follow the format of Section 4.2 of the SRP, this review covers the following three major categories: 1) Fuel System Damage Mechanisms, which are most applicable to normal operation and AOOs, 2) Fuel Rod Failure Mechanisms, which apply to normal operation, AOOs, and postulated accidents, and 3) Fuel Coolability, which are applied to postulated accidents. Specific fuel damage or failure criteria are identified under each of these categories in Section 4.2 of the SRP. The Duke reconstitution analysis methodologies for solid replacement rods and vacancies are discussed in this TER under each fuel damage or failure mechanism listed in the SRP.

The purpose of the design bases and/or criteria is to provide limiting values that prevent fuel damage or failure with respect to each mechanism. Reviewed in this TER is the applicability of the design criteria, submitted in DPC-NE-2007 for employing reconstituted fuel assemblies, with up to ten solid stainless steel replacement rods in reactor reload designs. Reload design criteria have not changed from References 3, 4, and 5. The design criteria, along with certain definitions for fuel failure, constitute the SAFDLs required by GDC 10.

The Duke analysis methods assure that the design limits and, consequently, SAFDLs are met for a particular design application. Reviewed in this report is whether the NRC-approved design limits are met for the reconstituted fuel assemblies, using previously approved Duke criteria and analysis methods remains applicable to these assemblies. A description of a typical fuel assembly is briefly discussed in the following section (Section 2.0). The fuel damage and failure mechanisms are addressed in Sections 3.0 and 4.0, respectively, while fuel coolability is addressed in Section 5.0.

2.0 FUEL SYSTEM DESIGN

The fuel assembly consists of a square array of fuel rods, guide thimble tubes, and an instrumentation tube. These components are mechanically fastened together by grid assemblies, and top and bottom nozzles. The top nozzle is designed to allow for fuel assembly reconstitution. Fuel rods are supported at intervals along their length by grid springs and dimples contained within the grid assembly to maintain rod-to-rod spacing. The grid assembly consists of an egg-crate arrangement of interlocking straps that contain springs and dimples. Attached to the top nozzle are holddown springs, and spring clamps that keep the fuel assembly firmly seated on the lower core plate during normal plant operation.

The fuel rods consist of UO_2 pellets clad in Zircaloy tubing. The rod assembly is pressurized with helium, then plugged and seal welded at the ends to encapsulate the fuel. The solid replacement rod is a solid cylinder of type 304 stainless steel or zircaloy material. Both fuel and replacement rods are supported by the grid spring and dimples by frictional forces. The rod dimensions are set so that the replacement rod engages all spacer grid stops under all conditions. A clearance is maintained between the replacement rod and the top grillage under all conditions.

3.0 FUEL SYSTEM DAMAGE

The design criteria presented in this section should not be exceeded during normal operation including AOOs. The evaluation portion of each damage mechanism evaluates the analysis methods, and analyses used by Duke to demonstrate that the design criteria are not exceeded during normal operation including AOOs for the reconstituted fuel assembly design.

3.1 STRESS

Bases/Criteria - In keeping with the GDC 10 SAFDLs, fuel damage criteria for cladding stress should ensure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The Duke design basis for fuel rod cladding stresses is that the fuel system will be functional and will not be damaged due to excessive stresses. These criteria are based on guidelines established in Section III of the American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code (Reference 11). These criteria are consistent with the acceptance criteria established in Section 4.2 of the SRP and remain acceptable for reconstituted assemblies.

Evaluation - Fuel assembly reconstitution does not directly impact the cladding stress generated in the remaining fuel rods of the fuel assembly. However, as fuel rods are replaced by solid rods or vacancies, there is a small increase in the core average linear heat generation rate (LHGR).[a] In addition, there may be a small increase in the LHGR of fuel rods directly adjacent to solid replacement rods or vacancies. As part of the Duke reload design methods discussed in References 3, 4, and 5 cycle specific fuel rod design evaluations are performed using NRC approved fuel performance models. The impact of using solid replacement rods or vacancies in reconstituted assemblies is evaluated as part of this cycle specific design process. The cladding stresses generated in fuel rods adjacent to the solid replacement rods or vacancies will be evaluated on a case-by-case basis based on the analysis of LHGRs. Consequently, PNL concludes that fuel assembly reconstitution with solid replacements or vacancies is acceptable in regards to the analysis methodology for determining cladding stress.

3.2 STRAIN

Bases/Criteria - The Duke design criterion for fuel rod cladding strain specifies that maximum uniform hoop strain (elastic plus plastic) shall not exceed 1%. This criterion is intended to preclude excessive cladding deformation from normal operation and AOOs. This is the same criterion for cladding

---

[a] To maintain total core heat generation upon replacement of fuel rods by replacement rods, the remaining fuel rods must operate at a higher linear heat generation rate.

strain that is used in Section 4.2 of the SRP and, therefore, remains acceptable for reconstituted assemblies.

Evaluation - As fuel rods are replaced by solid rods or vacancies, there is a small increase in the core average LHGR. In addition, there may be a small increase in the LHGR of fuel rods directly adjacent to solid replacement rods or vacancies. As part of the Duke reload design methods discussed in References 3, 4, and 5 cycle specific fuel rod design evaluations are performed using NRC approved fuel performance models. The impact of using solid replacement rods or vacancies in reconstituted assemblies is evaluated as part of this cycle specific design process. The cladding strains in fuel rods adjacent to the solid replacement rods or vacancies will be evaluated based on the increase in LHGRs. Therefore, introduction of solid replacement rods or vacancies will be evaluated on a case-by-case basis to determine their impact on cladding strains in a fuel assembly. Consequently, PNL concludes that fuel assembly reconstitution with solid replacements or vacancies is acceptable in regards to the analysis methodology for determining cladding strain.

3.3 STRAIN FATIGUE

Bases/Criteria - The Duke design criterion for cladding strain fatigue specifies that the cumulative fatigue factor be less than 0.9, when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles (which ever is the most conservative), is imposed as per the O'Donnell and Langer design curve (Reference 12) for fatigue usage. This criterion is conservative in relation to that described in Section 4.2 of the SRP and remains acceptable for reconstituted assemblies.

Evaluation - Fuel assembly reconstitution does not significantly impact the cladding fatigue generated in the remaining fuel rods of the fuel assembly. However, as fuel rods are replaced by solid rods or vacancies, there is a small increase in the core average LHGR. In addition, there may be a small increase in the LHGR of fuel rods directly adjacent to solid replacement rods or vacancies. As part of the Duke reload design methods discussed in References 3, 4, and 5 cycle specific fuel rod design evaluations are performed using NRC approved fuel performance models. The impact of using solid replacement rods or vacancies in reconstituted assemblies is evaluated as part of this cycle specific design process, taking into account changes in the linear heat generation rate of the adjacent fuel rods. Consequently, introduction of solid replacement rods or vacancies will be evaluated on a case-by-case basis. Therefore, PNL concludes that fuel assembly reconstitution is acceptable in regards to the analysis methodology for determining cladding strain fatigue.

3.4 FRETTING WEAR

Bases/Criteria - Fretting wear is a concern for fuel, burnable poison rods, and guide tubes. Fretting or wear may occur on the fuel and/or burnable poison cladding surfaces in contact with the spacer grids, if there is a reduction in grid spacing loads in combination with flow induced vibratory forces. The Duke design criterion for fretting wear specifies that the assembly design shall provide sufficient support to limit rod vibration and fretting wear. This criterion is consistent with Section 4.2 of the SRP and remains acceptable for reconstituted assemblies.

Evaluation - Operation of a reconstituted fuel assembly does not directly impact the rod fretting wear as long as the reconstituted assembly does not have significant spacer damage. Duke was questioned about when vacancies would be left in an assembly for reconstitution (Reference 6). Duke responded (Reference 7) that a vacancy would only be left in a reconstituted assembly, if the spacer grid cell were damaged enough such that placing a solid replacement rod could possibly lead to subsequent failures in adjacent fuel rods. However, there remains the following question or issue. If the spacer grid cell is damaged enough that placing a solid rod in this cell could lead to additional failures, how can it be determined that leaving the cell vacant will not also lead to additional failures? The ability to predict fretting behavior of a damaged grid is very difficult at best for two reasons. First, spacer grid fretting is predicted from prototypic spacer designed flow tests, and there are no fretting tests for a similarly damaged grid to demonstrate acceptable behavior. Second, the extent of damage in a previously irradiated assembly spacer cell is difficult to assess from remote viewing devices. Therefore, PNL recommends that the use of vacancies in reconstituted fuel assemblies should not be approved on a generic basis due to grid spacer fretting concerns. If in the future Duke has a specific application where they feel a vacancy is warranted it may be submitted to NRC for approval on a case-by-case basis along with justification that design criteria will be met including the fretting wear criterion. This justification should include information on the extent of spacer damage in the vacant grid cell and adjacent cells along with an evaluation of the impact on grid fretting.

Another possible concern is the slight decrease in the weight of a reconstituted assembly when solid replacement rods are used because they are slightly lighter than fuel rods. This is not expected to affect fretting wear. This is because the replacement rods do not affect the assembly pressure drops nor flow distribution (see Section 4.4), and have dimensions similar to a fuel rod. Thus, PNL concludes that fuel assembly reconstitution with solid replacement rods is acceptable in regards to the analysis methodology for determining fretting wear. As noted above, generic approval is not recommended for the use of vacancies in reconstituted assemblies, due to grid spacer fretting concerns from damaged grids.

3.5 OXIDATION AND CRUD BUILDUP

Bases/Criteria - Section 4.2 of the SRP identifies cladding oxidation and crud buildup as potential fuel system damage mechanisms. The SRP does not establish specific limits on cladding oxidation and crud buildup but does specify that their effects be accounted for in the thermal and mechanical analyses performed for the fuel. Cladding oxidation and crud are not affected by the use of solid replacement rods. Consequently, these criteria remain acceptable for reconstituted assemblies.

Evaluation - Fuel assembly reconstitution does not directly impact the cladding oxidation and crud buildup generated on the remaining fuel rods of the fuel assembly. As fuel rods are replaced by solid rods, there is a small increase in the core average linear heat generation rate. In addition, there may be a small increase in the linear heat generation rate of fuel rods directly adjacent to solid replacement rods or vacancies. As part of the Duke reload design methods discussed in References 3, 4, and 5 cycle specific fuel rod design evaluations are performed using NRC approved fuel performance models. The impact of using solid replacement rods or vacancies in reconstituted assemblies on cladding oxidation and crud buildup, is evaluated as part of this cycle specific design process. The impact of cladding oxidation and crud buildup in fuel thermal and mechanical performance analyses are evaluated by Duke when solid replacement rods are introduced in reconstituted assemblies. Consequently, PNL concludes that fuel assembly reconstitution with solid replacements is acceptable in regards to the analysis methodology for cladding oxidation and crud buildup.

3.6 ROD BOWING

Bases/Criteria - Fuel and burnable poison rod bowing is a phenomenon that alters the design-pitch dimensions between adjacent rods. Fuel rod bowing affects local nuclear power peaking and the local heat transfer to the coolant. Rather than place design limits on the amount of bowing that is permitted, the effects of bowing are included in the departure for nucleate boiling ratio (DNBR) analysis by a DNBR penalty, when rod bow is greater than a predetermined amount. This approach is consistent with Section 4.2 of the SRP and remains acceptable for reconstituted assemblies.

Evaluation - Rod bowing has been found to be dependent on the distance between grid spacers, the rod moment of inertia, and flux distribution. The design characteristics will not change with the addition of solid replacement rods or vacancies, but the flux distribution will change in the immediate vicinity of the replacement rods or vacancies. This flux distribution is not expected to be significantly different than what currently exists for a fuel rod adjacent to a guide tube. Reconstituted fuel assemblies will have rod bow characteristics that are similar to normal fuel assemblies. PNL concludes that fuel assembly reconstitution with solid replacements rods is acceptable in regards to the analysis methodology used to determine rod bow.

3.7 AXIAL GROWTH

Bases/Criteria - The Duke design basis for axial growth specifies that adequate clearance be maintained between the rod ends, and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly. For assembly growth, Duke has a design basis that specifies axial clearance between core plates and the bottom and top assembly nozzles, that should allow sufficient margin for fuel assembly irradiation growth during the assembly lifetime. These criteria are consistent with Section 4.2 of the SRP and remain acceptable for reconstituted assemblies.

Evaluation - Duke provides an initial fuel rod-to-nozzle growth gap in their fuel assembly plant applications to allow for differential irradiation growth, and thermal expansion between the fuel rod cladding and the fuel assembly guide thimble tubes. The minimum gap required to allow for the irradiation growth and thermal expansion to preclude interference during operation is based on the assumption of worst case (maximum) fuel rod, and fuel assembly growth combined with worst case (minimum) fabrication tolerances. For solid stainless steel rods the concern is not the closing up of this gap (the concern with fuel rods and solid Zircaloy rods), but that the rods will remain engaged with the upper and lower tie plates with increasing burnup. The Zircaloy guide tubes and fuel rods grow at a significantly greater rate than the solid stainless steel rods that experience little or no growth with irradiation. Since stainless steel has a greater thermal expansion coefficient than Zircaloy, the disengagement of the solid rods becomes greatest at end-of-life (EOL) during cold conditions.

For reconstituted assemblies with solid stainless steel rods, Duke utilizes the upper tolerance growth curves for fuel assembly growth, worst case tolerances, and assumes no solid stainless steel rod growth in order to demonstrate that these solid rods remain engaged at cold conditions at their burnup limit. PNL concludes that the Duke analysis methodology for fuel assembly reconstitution with solid replacement rods is conservative and, therefore, acceptable in regards to axial growth.

3.8 ROD INTERNAL PRESSURE

Bases/Criteria - The Duke design basis for rod internal pressure specifies that the fuel system will remain below nominal system pressure during normal operation and AOOs. This design basis may change based on the recent NRC approval for the Babcock and Wilcox Fuel Company (BWFC) fuel designs to exceed nominal system pressure (Reference 13). The Duke application of reconstituted assemblies does not impact either rod pressure criterion.

Either criterion is applicable to reconstituted assemblies, as long as the methods in Reference 13 are found to be applicable to Duke applications.

Evaluation - As fuel rods are replaced by solid rods or vacancies, there is a small increase in the core average linear heat generation rate. In addition, there may be a small increase in the linear heat generation rate of fuel rods directly adjacent to solid replacement rods or vacancies. As part of the Duke reload design methods discussed in References 3, 4, and 5 cycle specific fuel rod design evaluations are performed using NRC approved fuel performance models. The impact of using solid replacement rods in reconstituted assemblies on rod internal pressure is evaluated as part of this cycle specific design process. The solid replacement rods in a reconstituted assembly are not affected by rod internal pressure. PNL concludes that fuel assembly reconstitution with solid replacement rods is acceptable in regards to the analysis methodology for determining rod internal pressure.

3.9 ASSEMBLY LIFTOFF

Bases/Criteria - Section 4.2 of the SRP calls for the fuel assembly holddown capability (wet weight and spring forces) to exceed worst case hydraulic loads for normal operation and AOOs. The Duke design criterion for assembly liftoff specifies that the holddown spring system shall be capable of maintaining fuel assembly contact with the lower support plate during Condition I and II events. PNL concludes that this is consistent with the SRP guidelines and, therefore, remains acceptable for use with reconstituted fuel assemblies.

Evaluation - The fuel assembly liftoff forces are a function of primary coolant flow, spring forces, and assembly dimensional changes. The only change incurred by the addition of solid replacement rods is in assembly weight because there are no changes in assembly dimensions. Replacement rods have the same outside diameters and design configurations as a fuel rod. These rods also displace essentially the same volume of coolant. However, the weight of a replacement rod is less than a fuel rod by a small amount. The lower weight of a replacement rod results in a small increase in the net upward force on the fuel assembly.

The use of vacancies in an assembly makes a larger impact on assembly liftoff because of the reduced weight in the assembly. There is a small increase in flow around the vacancy, but the overall change in assembly flow should be small. This will most likely be more than compensated for by the friction losses due to the vacant cell. Hence, Dukes assumption that the change in lift force is only due to a reduction in fuel assembly weight is valid.

The total net upward force due to lift and buoyancy forces for a typical fuel assembly is designed to be within the minimum available holddown spring force at hot full power and cold zero power conditions. The Duke liftoff analysis methodology for reconstituted assemblies takes into account the weight changes due to solid replacement rods. Consequently, PNL concludes that the Duke analysis methodology for fuel assembly liftoff fuel is acceptable.

4.0 FUEL ROD FAILURE

In the following paragraphs, fuel rod failure thresholds and analysis methods for the failure mechanisms listed in the SRP will be reviewed. When the failure thresholds are applied to normal operation including AOOs, they are used as limits (and hence SAFDLs) since fuel failure under those conditions should not occur according to the traditional conservative interpretation of the GDC 10. When these thresholds are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose assessments required by 10 CFR 100. The basis or reason for establishing these failure thresholds is thus established by GDC 10 and Part 100. Only the threshold values and the analysis methods used to assure that they are met are reviewed below.

4.1 HYDRIDING

Bases/Criteria - Internal hydriding as a cladding failure mechanism is precluded by controlling the level of hydrogen impurities in the fuel during fabrication. External hydriding can occur due to waterside corrosion. The use of solid replacement rods does not impact internal or external hydriding criteria. Therefore, the existing criteria for fuel rods remains valid.

Evaluation - As fuel rods are replaced by inert rods, there is a small increase in the core average linear heat generation rate of the remaining fuel rods. In addition, there may be a slight increase in the linear heat generation rate of fuel rods directly adjacent to solid replacement rods, or vacancies which could increase cladding waterside corrosion by a small amount. As part of the Duke reload design methods discussed in References 3, 4, and 5 cycle specific fuel rod design evaluations are performed using NRC approved fuel performance models. The impact of using solid replacement rods or vacancies in reconstituted assemblies on rod corrosion is evaluated as part of this cycle specific design process. Because the solid replacement rods are not significantly affected by hydriding, and because external hydriding due to waterside corrosion is included in Duke analysis methodology, PNL concludes that fuel assembly reconstitution is acceptable with regard to hydriding.

4.2 CLADDING COLLAPSE

Bases/Criteria - If axial gaps in the fuel pellet column were to occur due to fuel densification, the potential would exist for the cladding to collapse into a gap. Because of the large local strains that would result from collapse, the cladding is then assumed to fail. The Duke design criterion specifies that cladding collapse is precluded during the fuel rod design lifetime. This design basis is the same as that in Section 4.2 of the SRP and remains acceptable for reconstituted assemblies.

Evaluation - The cladding collapse criterion is not applicable to the solid replacement rods. As fuel rods are replaced by solid rods, there is a small increase in the core average linear heat generation rate. In addition, there may be a slight increase in the linear heat generation rate of fuel rods directly adjacent to solid replacement rods or vacancies. This could result in a small decrease in the time to creep collapse. As part of the Duke reload design methods discussed in References 3, 4, and 5 cycle specific fuel rod design evaluations are performed using NRC approved fuel performance models. The impact of using solid replacement rods in reconstituted assemblies on cladding collapse is evaluated as part of the cycle specific design process, taking into account changes in the linear heat generation rate of the fuel rods. PNL concludes that fuel assembly reconstitution with solid replacement rods is acceptable with regard to the analysis methodology used to determine cladding collapse.

4.3 OVERHEATING OF CLADDING

Bases/Criteria - The Duke design limit for the prevention of fuel failures due to overheating specifies at least a 95% probability at a 95% confidence level and that departure from nucleate boiling (DNB) will not occur on a fuel rod having the minimum DNBR during normal operation and AOOs. This design limit is consistent with the thermal margin criterion of Section 4.2 of the SRP, and thus remains acceptable for application to reconstituted fuel assemblies.

Evaluation - Fuel rod reconstitution using solid replacement rods and vacancies affects predictions of DNB in hot channels due to changes in local power, and resultant changes in the distribution of enthalpy and coolant flow in the fuel assembly. Duke uses two critical heat flux (CHF) correlations BWC and BWCMV for their Mark B and BW designs, respectively, to calculate critical power and minimum DNBR. The major issue is whether these correlations are applicable to the reconstituted assembly applications proposed by Duke. Duke has presented CHF data for their Mark BW fuel assembly design. This data demonstrates that their CHF correlation BWCMV provides a reasonably conservative prediction of critical power for test bundles with one cold rod, three cold rods, and one vacancy per test bundle. The test bundles simulate a quarter of the assembly. Examination of the CHF data does show that the critical bundle power is lowered for those bundles with either three cold rods in a row or one vacancy per test bundle. However, it appears that the BWCMV correlation provides a satisfactory prediction of critical power in these cases. It should be noted that the prediction of one vacancy rod was slightly less conservative than for the other test cases but, it is acceptable because it is small and within the error of the data.

Duke also presents calculational results using their BWC and BWCMV correlations for their Mark B and BW designs, respectively, for the cases of a regular assembly, one cold rod, three cold rods, and one vacancy. These results demonstrate that the BWC correlation predicts similar behavior as the BWCMV correlation for the regular assembly, one cold rod, three cold rods, and one vacancy. Duke has shown that for these particular cold rod, vacancy geometries, and assumed radial power distributions that the MDNBR is either similar to or equal to the MDNBR of the regular assembly. However, these calculations do not cover all possible geometries and radial power distributions.

Therefore, both the BWCMV and BWC correlations are found to be acceptable for application to reconstituted assemblies with the limitations on geometry presented in Reference 2. However, the above calculations with these CHF correlations do not cover all possible geometries and radial power distributions. Thus, PNL recommends that Duke perform calculations of MDNBR using the appropriate CHF correlation for each reconstituted assembly application using actual geometries and power distributions.

4.4 OVERHEATING OF FUEL PELLETS

Bases/Criteria - As a second method of avoiding cladding failure due to overheating, Duke precludes centerline pellet melting during normal operation and AOOs. This design limit is the same as given in the SRP and has been approved for application in Duke designs. Duke has placed a temperature limit on fuel melting at extended fuel burnups that is considered to be conservative. Therefore, PNL concludes that BWFC's design limit for fuel melting remains acceptable for application to reconstituted fuel assemblies.

Evaluation - As fuel rods are replaced by inert rods, there is a small increase in the core average linear heat generation rate which could slightly increase fuel pellet temperatures. As part of the Duke reload design methods discussed in References 3, 4, and 5 cycle specific fuel rod design evaluations are performed using NRC approved Duke fuel performance models. The impact of using solid replacement rods in reconstituted assemblies on fuel pellet temperature is evaluated as part of this cycle specific design process. Since the analysis methods used by Duke to determine fuel temperatures remain applicable to reconstituted assemblies, PNL concludes that they are acceptable for application in regards to overheating of fuel pellets.

4.5 PELLET/CLADDING INTERACTION

Bases/Criteria - As indicated in Section 4.2 of the SRP, there are no generally applicable criteria for pellet cladding interaction (PCI) failure. However, two acceptance criteria of limited application are presented in the SRP for PCI: 1) less than 1% transient induced cladding strain, and 2) no centerline fuel melting. Both of these limits have been adopted by Duke for use in evaluating their fuel designs and have been approved by the NRC. PNL concludes that these remain acceptable for application to reconstituted fuel assemblies.

Evaluation - There is no chance for PCI to occur in solid replacement rods, and the use of solid replacement rods will not cause PCI to significantly increase in the remaining fuel rods of the assembly. Also, since the issues of cladding strain and fuel pellet melting were satisfactorily addressed in Sections 3.2 and 4.4, respectively, PNL concludes that fuel assembly reconstitution with solid replacement rods is acceptable in regards to the analysis methodology used to determine pellet/cladding interaction.

4.6 CLADDING RUPTURE

Bases/Criteria - There are no specific design limits associated with cladding rupture other than the 10 CFR 50, Appendix K (Reference 14) requirements that the incidence of rupture not be underestimated. Duke uses a rupture temperature correlation consistent with NUREG-0630 guidance (Reference 15). PNL concludes that Duke has adequately addressed the criteria for cladding rupture in reconstituted assemblies.

Evaluation - Cladding rupture will not occur in solid replacement rods. The model used for cladding rupture remains applicable to the remaining fuel rods in the assembly. PNL concludes that fuel assembly reconstitution with solid replacement rods is acceptable in regards to the analysis methodology used to determine cladding rupture.

4.7 FUEL ROD MECHANICAL FRACTURING

Bases/Criteria - The term "mechanical fracture" refers to a fuel rod defect that is caused by externally applied forces such as hydraulic loads or loads derived from core-plate motion. These loads are bounded by the loads of a Safe-Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA), and the mechanical fracturing analysis is usually done as a part of the SSE-LOCA loads analysis (see Section 5.4 of this TER).

Evaluation - The discussion of the SSE-LOCA loading analysis is given in Section 5.4 of this TER.

5.0 FUEL COOLABILITY

For postulated accidents in which severe fuel damage might occur, core coolability must be maintained as required by several GDCs (e.g., GDC 27 and 35). In the following paragraphs, limits and methods used to assure that coolability is maintained are discussed for the severe damage mechanisms listed in the SRP.

5.1 FRAGMENTATION OF EMBRITTLED CLADDING

Bases/Criteria - The most severe occurrence of cladding oxidation and possible fragmentation during a postulated accident is the result of a LOCA. In order to reduce the effects of cladding oxidation during LOCA, Duke uses a limiting criteria of 2200°F on peak cladding temperature (PCT), and a limit of 17% on maximum cladding oxidation as prescribed in 10 CFR 50.46. These criteria are consistent with the SRP criteria. PNL concludes that these criteria are also applicable to the reconstituted assemblies.

Evaluation - Duke will evaluate the impact of fuel reconstitution on LOCA in a similar way to that discussed for Cladding Overheating in Section 4.3. BWFC will take into consideration the exact configuration of reconstituted rods and assemblies, to determine their impact on local power distributions near the reconstituted rods and assemblies, and also the core wide power changes. These local and core power distributions will be used to assess their impact on flow, enthalpy, and stored energy in the LOCA analysis on a cycle specific basis.

5.2 VIOLENT EXPULSION OF FUEL

Bases/Criteria - In a severe Reactivity Insertion A.accident (RIA), such as a control rod ejection accident, large and rapid deposition of energy in the fuel could result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal might be sufficient to destroy the fuel cladding and rod bundle geometry and provide significant pressure pulses in the primary system. To limit the effects of an RIA event, Regulatory Guide 1.77 recommends that the radially-averaged energy deposition at the hottest axial location be restricted to less than 280 cal/g.

The Duke design criterion for this event is identical to that in Regulatory Guide 1.77, such that the peak fuel enthalpy for the hottest axial fuel rod location shall not exceed 280 cal/g. The reconstitution of fuel assemblies does not impact the fuel enthalpy limit. Therefore, PNL concludes that Duke design limits for fuel dispersal are acceptable for reconstituted fuel assemblies.

Evaluation - The Duke reload analysis methods for RIA events takes into consideration the changes due to reconstitution. PNL concludes that the analysis methodology remains acceptable for application to reconstituted fuel assemblies with solid replacement rods.

5.3 CLADDING BALLOONING

Bases/Criteria - Fuel cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during a LOCA. There are no specific design limits associated with cladding ballooning, other than the 10 CFR 50 Appendix K requirement, stating that the degree of swelling not be underestimated.

Evaluation - The cladding ballooning model and flow blockage model used by Duke are directly coupled to the cladding rupture temperature model for the LOCA-emergency core cooling system (ECCS) analysis. These models have previously been approved by the NRC, and are considered to be conservative for a reconstituted assembly. As part of the Duke reload design methods discussed in References 3, 4, and 5 cycle specific fuel rod design evaluations are performed using NRC approved Duke fuel performance models. The impact of using solid replacement rods in reconstituted assemblies on cladding ballooning is evaluated as part of this cycle specific design process. Therefore, PNL concludes that Duke has adequately addressed the issue of cladding ballooning, and that these models remain acceptable for application to fuel assembly reconstitution with solid replacement rods.

5.4 FUEL ASSEMBLY STRUCTURAL DAMAGE FROM EXTERNAL FORCES

Bases/Criteria - Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. Appendix A to SRP Section 4.2 states that the fuel system coolable geometry shall be maintained, and damage should not be so severe as to prevent control rod insertion during seismic and LOCA events. The Duke design basis specifies that the fuel assembly will maintain a geometry that is capable of being cooled under the worst case design accident, and that no interference between control rods and thimble tubes will occur during a safe shutdown earthquake. This is consistent with the SRP and, therefore, remains acceptable for reconstituted assemblies.

Evaluation - Duke has performed a LOCA-seismic analysis to determine how solid replacement rods change the fuel assembly structural response to LOCA-seismic loadings. Reconstituted fuel assemblies have a slightly reduced weight. The structural stiffness is increased when solid replacement rods are used. This results in a shift of the natural frequencies of the reconstituted assembly. Thus, the total energy required to excite and deflect a reconstituted assembly is different from a normal assembly. Also, solid replacement rods do not significantly affect the grid dynamic properties. Therefore, the use of solid replacement rods is acceptable for seismic-LOCA analyses.

The approved seismic and LOCA loading methodologies all assume a full fuel assembly of fuel rods and guide tubes. The approval of these methodologies does not apply to the use of vacancies in an assembly. The use of vacancies will alter the dynamic response of an assembly due to changes in the physical characteristics and material properties. Therefore, based on the approved methodologies of qualifying a fuel assembly with no vacancies, the use of vacancies in reconstituted assemblies is not acceptable for seismic and LOCA analysis for Duke Power Company.

6.0 CONCLUSIONS

PNL has reviewed the Duke mechanical and thermal hydraulic analysis criteria and methods for reconstituted fuel assemblies as presented in Sections 1.0, 2.0, 4.0, 5.0 and 6.0 of Reference 1. This review has been conducted in accordance with Section 4.2 of the SRP. PNL concludes that the reconstitution methodology as described in Reference 1 is acceptable for core reload analysis licensing applications for solid replacement rods with the following restrictions: 1) no more than ten solid rods per assembly, 2) no more than 3 solid rods in a row, and 3) cycle specific reload analyses must include the exact configuration and case location of reconstituted rods and assemblies including power distributions.

This approval does not provide generic approval for the use of vacancies in reconstituted assemblies because of concerns with fretting wear and seismic-LOCA loads as discussed in Sections 3.4 and 5.4 of this report.

7.0 REFERENCES

1. Generic Letter 90-02, Supplement 1, J. G. Partlow (NRC) to Licenses and Applicants, "Supplement 1 to Generic Letter 90-02, Alternative Requirements for Fuel Assembly in the Design Features Section of Technical Specifications," dated July 31, 1992.

2. Duke Power Company, September 1993. "Duke Power Company Fuel Reconstitution Analysis Methodology." DPC-NF-2007P, Duke Power Company, Charlotte, North Carolina.

3. Duke Power Company, July 1981. "Oconee Nuclear Station Reload Design Methodology Technical Report." NFS-1001, Duke Power Company, Charlotte, North Carolina.

4. Duke Power Company, October 1985. "Oconee Nuclear Station Reload Design Methodology II." DPC-NE-1002A, Revision 2, Duke Power Company, Charlotte, North Carolina.

5. Duke Power Company, October 1990. "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel." DPC-NF-2001A, Duke Power Company, Charlotte, North Carolina.

6. Letter, R.C. Jones (NRC) to M.S. Tuckman (Duke), "Request for Additional Information on Topical Report   ", dated April 26, 1994.

7. Letter, M.S. Tuckman (Duke) to U.S. Nuclear Regulatory Commission, Topical Report DPC-2007P, "Fuel Reconstitution Analysis Methodology" dated, August 10, 1994.

8. U.S. Nuclear Regulatory Commission. July 1981. "Section 4.2, Fuel System Design." In Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants--LWR Edition. NUREG-0800, Revision 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

9. United States Federal Register. "Appendix A, General Design Criteria for Nuclear Power Plants." 10 Code of Federal Regulations (CFR), Part 50. U.S. Printing Office, Washington, D.C.

10. United States Federal Register. "Reactor Site Criteria." 10 Code of Federal Regulations (CFR), Part 100. U.S. Printing Office, Washington, D.C.

11. American Society of Mechanical Engineers. 1983 Edition. "Section III, Nuclear Power Plant Components." ASME Code. American Society of Mechanical Engineers, New York.

12. O'Donnell, W. J., and B. F. Langer. 1964. "Fatigue Design Basis for Zircaloy Components." In Nuc. Sci. Eng. 20:1.

13. Wesley, D.A., Farnsworth, D.A., and Meyer, G.A., July, 1991. "Fuel Rod Gas Pressure Criterion." BAW-10147P, Babcock and Wilcox, Lynchburg, Virginia.

14. United States Federal Register. "Appendix K, ECCS Evaluation Models." 10 Code of Federal Regulations (CFR), Part 50. U.S. Printing Office, Washington, D.C.

15. Powers, D. A., and R. O. Meyer. 1980. "Cladding Swelling and Rupture Models for LOCA Analysis." NUREG-0630, U.S. Nuclear Regulatory Commission, Washington, D.C.

ABSTRACT

This report presents Duke Power Company's methodology for performing reconstituted fuel assembly analyses. The methodology uses the models and codes currently approved for cycle design at the Oconee, McGuire, and Catawba Nuclear Stations. Additional analyses to ensure acceptable seismic behavior with filler rods or water holes and a CHF test program to verify applicability of DNBR calculations to typical reconstitution geometries are also presented. This report states the results of these analyses and the guidelines that will be used by Duke for reconstitution of both mixing and non-mixing vane fuel assemblies.

Table of Contents

List of Tables

List of Figures

1.0 INTRODUCTION

The reconstitution of fuel assemblies is becoming a more routine occurrence during refueling outages in light water reactors. This is due to the concerted effort on the part of utilities to maintain zero fuel defects during cycle operation. This goal requires aggressive programs in two areas. First, all reasonable measures must be taken in the design and manufacturing of fuel assemblies to prevent any type of known failure mechanism. Secondly, failures that do occur during operation must be identified and the failed fuel rods removed before subsequent cycles.

Given that all reasonable efforts will be pursued to identify and remove failed fuel, the options for mitigating this impact on the established design and performance of the fuel cycle are limited. Removal of the entire assembly by replacement or re-design is rarely the best option from the standpoints of time, fuel cycle costs, analyses effort, and outage impact. This leaves replacement of only the individual failed rods (reconstitution) as the best alternative for meeting all the competing goals while providing a high degree of fuel performance.

Duke Power Company's primary replacement candidate for use in reconstitution is a fuel rod that contains pellets of naturally enriched uranium dioxide (UO_2). Aside from enrichment, this rod is the same in design and behavior as a standard fuel rod and is analyzed using standard approved methods. If local grid structural damage exists, the use of a natural UO_2 replacement rod is not a sound alternative and solid filler rods made of stainless steel or Zircaloy will be used. Additionally, if the severity of the circumstances warrant, a water hole or vacancy may be left in the assembly.

This report details the methodology and guidelines DPC will use to support fuel assembly reconstitution with filler rods or water holes. The guidelines were developed to ensure acceptable nuclear, mechanical, and thermal-hydraulic performance of reconstituted fuel assemblies. Specific results are provided for the Mark-B and Mark-BW fuel designs with currently licensed codes. The methodology will be applicable if different fuel designs or codes are licensed by Duke Power Company.

2.0 EVALUATION PROCESS FOR RECONSTITUTED FUEL

The analysis of a reconstituted fuel assembly in a reload cycle follows the same process as for a standard fuel assembly. TABLE 1 depicts the analysis flow path for the reload design process. In the routine reload design process, the specific core loading pattern and peaking are compared to pre-established limits determined by design requirements and FSAR Chapter 15 accident analyses. These bounding generic limits exist for all operating and accident conditions that must be analyzed to license the core for operation. As long as the cycle specific core design meets the acceptance criteria defined by the limits, safe operation is ensured.

The reconstituted fuel assembly is analyzed in the same manner as a standard fuel assembly. Analyses detailed in this report show that the fuel assembly with filler rods or vacancies can be analyzed with the same methods and codes approved for standard fuel assemblies. This means that the generic analyses currently in place are equally valid for reconstituted and standard fuel assemblies. This also means that as long as the nuclear peaking criteria applied to the reload core are satisfied in the reconstituted fuel assembly, the configuration will have no adverse effect on safety.

The results of this report require only two limitations on the reconstituted fuel assembly geometry beyond that for standard fuel assemblies:

1) The maximum number of filler rods (Stainless Steel or Zircaloy) allowed per fuel assembly is 10.

2) The maximum number of vacancies or water holes per assembly is four, one per quadrant.

These criteria are established, not because of any adverse physical effects, but due to the maximum number of each type analyzed from a mechanical integrity standpoint. These criteria are applicable to both mixing and non-mixing vane fuel assemblies.

3.0 NUCLEAR ANALYSIS

The reconstitution of a fuel assembly perturbs the pin power distribution within that assembly relative to its reference state. The resulting power distribution perturbation is due to locally changing neutronic properties produced by the introduction of replacement fuel rods, or water holes (empty fuel cells) in the reconstituted assembly. For fuel assemblies containing a small number of interior replacement rods or water holes, power distribution perturbations will typically be limited to the reconstituted assembly. For fuel assemblies containing a small number of replacement rods or water holes at the assembly periphery, power distribution perturbations will occur in the subject and adjacent assemblies. However, for the case where a large number of replacement rods or water holes are present in a fuel assembly, core power asymmetries may occur in addition to the local pin power distribution perturbations.

Cycle-specific physics evaluations will be performed to assess the changes in both the local and global power distributions resulting from the fuel assembly reconstitution. These evaluations will be performed by either explicitly modeling the geometry of the reconstituted assembly or based on the results from generic peaking analyses. If the evaluation results in an increase in peaking, then a review of the FSAR Chapter 15 accident analyses which are sensitive to power distribution changes will be performed to confirm the acceptability of these analyses for the core design containing a reconstituted fuel assembly or assemblies.

The code system used in performing these analyses will be one of the three computer code systems documented in References 1 through 4. The application of a particular code system to an analysis will be dependent upon the plants for which it is licensed.

Cycle specific pin power distribution analyses that are outside the scope of generic analyses will be performed using either the PDQ07 or SIMULATE-3P code. The ability of these codes to predict pin power distributions has been demonstrated by comparison of calculated versus measured pin power distributions for the B&W critical experiments. Pin power uncertainties have been calculated based on a 95% probability and 95% confidence level for each code, and were performed as part of the licensing of Duke Power Company's Reload Design Methodology. Detailed discussions of the pin uncertainty calculations and the applicable uncertainty analyses for each code system can be found in References 1 through 4.

4.0 MECHANICAL ANALYSIS

Reconstitution will affect various fuel assembly structural attributes. Evaluations have been completed which address the relevant mechanical and structural issues which are affected by fuel assembly reconstitution. Three BWFC Mark-B 15 x 15 non-mixing vane fuel assembly designs (Mark-B8, Mark-B9, and Mark-B10) and the BWFC Mark-BW 17 x 17 mixing vane fuel assembly design were each considered in the evaluations. Since the three Mark-B fuel assembly designs are similar in mechanical make-up and the evaluations produced identical results for each, the Mark-B designation will be utilized to encompass these designs throughout the remainder of Section 4.

The structural acceptability of a given reconstituted fuel assembly configuration is dependent on the number and location of filler rods or water holes within the subject assembly. Therefore, the limiting reconstituted assembly configurations addressed in Section 2 were assumed for generic structural evaluation. However, for fuel centerline melt and LOCA concerns, it will be necessary to perform cycle-specific evaluations in order to justify actual reconstituted assembly configurations. Cycle specific analyses are required since the actual reconstituted geometry can have an effect on nuclear peaking and therefore fuel rod thermal analyses results. A brief review of the individual structural evaluation results and description of the cycle-specific methodologies is provided in the following sections.

4.1 FILLER ROD ENGAGEMENT \ SHOULDER GAP CONSIDERATIONS

Since irradiation growth of a stainless steel replacement rod will be negligible, provision must be made so that at end of life (EOL) when fuel assembly irradiation growth is maximum, the replacement rod is engaged by the upper spacer grid. Both the Mark-B and Mark-BW stainless steel replacement rods will remain engaged in the upper end grid of the fuel assembly at an EOL fuel assembly burnup of [ ].

At [ ], a minimum Mark-BW stainless steel replacement rod length, or engagement, of [ ] inches will be maintained above the top hard stop of the Mark-BW fuel assembly upper spacer grid. For the Mark-B replacement rod, at least [ ] inches of the replacement rod length will remain above the soft stop of the Mark-B upper spacer grid. This level of engagement is acceptable since the rod will be effectively fixed at four positions located 90° from one another, with the two lower hard stops and two soft stops of a given upper spacer grid cell.

The stainless steel replacement rods will also experience greater axial thermal expansion relative to standard Zircaloy clad fuel rods. Therefore, evaluations were performed to ensure that the shoulder gap, or distance between the top of the replacement rod to the bottom of the fuel assembly top nozzle (i.e. upper end fitting), will not be closed.

The shoulder gap analyses for both the Mark-BW and Mark-B replacement rod were conservatively performed with no consideration given to irradiation growth of the guide thimbles due to previous operation in the core. A temperature change from $70^{\circ}F$ to $650^{\circ}F$ was considered in the evaluation and the stainless steel replacement rods were assumed to be seated on, or in contact with, the bottom nozzle. Considering these assumptions, a minimum shoulder gap of [ ] inches will be maintained with the Mark-BW stainless steel replacement rod and [ ] inches for the Mark-B stainless steel replacement rod.

4.2 GRID RELAXATION

Stainless steel replacement rods will also experience greater diametral thermal expansion relative to the standard Zircaloy clad fuel rods. The additional movement of the grid soft stops should not cause interference between the grid and fuel rods of adjacent cells and sufficient grip force must be maintained on the stainless steel rod throughout operation considering the additional soft stop deflection.

The diametral expansion of both the Mark-BW and Mark-B stainless steel replacement rod is approximately two mils or approximately one mil more than a standard Zircaloy clad fuel rod. This small amount of grid stop deflection is within the elastic range of the Mark-B and Mark-BW spacer grids and will not result in interference between the grid and adjacent rods within the fuel assembly bundle.

4.3 IMPROPER ROD SEATING

The result of fuel rods not being consistently positioned in fuel assemblies, such that their active fuel stack elevations are inconsistent with the other fuel rods of the core, could possibly have a small effect on the core axial flux distribution (AFD). Unlike fuel rods, stainless steel replacement rods will be non-power producing and also few in number in comparison to the entire core. Therefore, the axial position of stainless steel replacement rods will not have an impact on AFD. However, established field reconstitution procedures and fuel assembly design characteristics ensure that both the Mark-B and Mark-BW stainless steel replacement rods are properly seated in the fuel assembly.

4.4 FRETTING

Both the Mark-B and Mark-BW stainless steel replacement rod designs weigh slightly less than standard Zircaloy clad fuel rods. However, these rods are structurally stiffer than standard rods. The frequency response of the replacement rods were evaluated to assure fretting of adjacent fuel rods or fuel assemblies due to the hydraulic flow forces will not result.

The maximum amplitude of vibration for both the Mark-B and Mark-BW stainless steel replacement rods were determined to be less than [ ] mils and insignificant in relation to the limiting rod-to-rod gaps in each of the fuel assembly designs. Therefore, no fretting is anticipated.

An evaluation has also been performed which assures that the presence of water holes will not bring about fuel rod fretting conditions in those cells located immediately adjacent to the vacancies. The evaluation utilized Mark-B and Mark-BW grip force data obtained during grid keying tests to show that preload from both Inconel end grid and Zircaloy intermediate grid soft stops will be maintained on rods located in grid cells which receive no support from rods in adjacent cells.

4.5 CENTERLINE FUEL MELT LIMITS

To ensure fuel rod integrity, the maximum local fuel temperature must remain below the melting temperature of the UO_2 fuel. A linear heat rate to melt (LHRTM) analysis is performed to determine the local linear heat rate (LHR) corresponding to centerline fuel melt (CFM). This limiting LHR is considered in the nuclear design maneuvering analysis performed each cycle to confirm Reactor Protection System setpoints.

Standard methodology (References 9 and 10) will be utilized on a cycle-specific basis to ensure the changes in power distributions resulting from the presence of stainless steel replacement rods or water holes (discussed previously in Section 3.0, Nuclear Design) do not invalidate the established CFM LHR limits. Cycle-specific analyses will be performed which compare the pin power history utilized to obtain the CFM limits against predicted pin power histories which account for reconstituted fuel pin power effects. The power history used in defining the CFM limits will be shown to envelope the resultant pin power predictions, thus justifying the limits, or a new bounding power history will be used to redefine the CFM limits.

4.6 LOCA LIMITS

Linear heat rate (LHR) limits have been set to ensure peak cladding temperatures (PCTs) are controlled below critical levels in the event of a loss-of-coolant accident (LOCA). Established Emergency Core Coolant System (ECCS) interface criteria verify the applicability of the LOCA analyses and LHR limits. Standard methods, which involve utilizing the ECCS interface criteria, will be used to determine the applicability of the LOCA analyses and LHR limits for core designs consisting of reconstituted fuel assemblies.

4.7 SEISMIC AND LOCA LOADINGS

Seismic-LOCA analyses have been performed for both Mark-B and Mark-BW reconstituted fuel assembly configurations containing ten stainless steel replacement rods or four water holes (one per each quadrant of the fuel assembly). The analyses were performed to evaluate the impact of the following fuel assembly attribute changes which result from reconstitution:

1) fuel assembly weight and lateral stiffness,

2) spacer grid strength and stiffness, and

3) fuel assembly natural frequency and seismic-LOCA loadings used in the dynamic simulation of fuel assembly for faulted conditions.

The analysis also considered the dynamic response of a mixed core of reconstituted/standard fuel during seismic-LOCA events.

Adding stainless steel replacement rods or water holes to a fuel assembly reduces the weight and affects the lateral stiffness characteristics of a fuel assembly. A weight reduction of less than 2% of the fuel assembly mass is applicable to both Mark-B and Mark-BW fuel assembly designs. A two-dimensional fuel assembly model was developed to quantify any changes to the lateral stiffness due to the presence of stainless steel rods or water holes. An analysis of the model shows a change of less than [ ] in lateral stiffness for both Mark-B and Mark-BW fuel assemblies.

The weight and lateral stiffness changes will impact the natural frequency of the fuel assembly. Considering the changes, the difference in the natural frequency between a standard fuel assembly and a reconstituted fuel assembly containing the filler rods or water holes was determined to be less than [ ]. This frequency change is negligible, therefore no factor was applied to the resultant load magnitudes.

Grid impact tests were performed with the solid stainless steel fuel rods to analyze the interaction between the fuel rods and spacer grid during a seismic-LOCA event. The test results showed that solid stainless steel rods produce about the same grid strength as Zircaloy fuel rods. Likewise, the absence of fuel rods (maximum of four) should have a minimal effect on spacer grid strength.

Data from the same test shows that stiffness increases a maximum of [ ] in the grid holding ten stainless steel replacement rods. This increase in stiffness increases the force in the spacer grid during a seismic-LOCA event by approximately [ ]. Positive loading margins have been determined for the fuel assembly spacer grids considering the increased load. These positive margins ensure that reconstituted fuel assembly geometries with ten stainless steel replacement rods will remain coolable, therefore the [ ] seismic-LOCA load increase is acceptable. The spacer grid impact loads were calculated based on assuming an all reconstituted core. These loads are conservative for a mixed-core configuration (Mark-B or Mark-BW and reconstituted Mark-B or Mark-BW fuel assemblies).

Removing four rods decreases spacer grid stiffness by [ ]. A decrease in spacer grid stiffness decreases the force in the spacer grid during a seismic-LOCA event. Consequently, acceptable spacer grid impact loads which were previously calculated based on all standard Mark-B or all standard Mark-BW fuel cores, are conservative for a reconstituted fuel assembly with four water holes.

In summary, adding ten stainless steel replacement rods or four water holes to a Mark-B or Mark-BW fuel assembly produces a minimal effect on the seismic-LOCA loads. Reconstituted fuel assemblies configured in this manner will remain coolable during a seismic-LOCA event.

5.0 THERMAL-HYDRAULIC ANALYSIS

Thermal-hydraulic analyses are performed by using a thermal-hydraulic code and Critical Heat Flux (CHF) correlation that together predict the fluid behavior of and critical heat flux values for the fuel assembly. This information is used to calculate the Departure from Nucleate Boiling Ratio (DNBR). The local condition predictions are the basis for the correlation itself since the thermal-hydraulic code is used to create the correlation from measured test data. The use of the VIPRE-01 computer code with the BWC CHF correlation (Reference 5) and the BWCMV CHF correlation (Reference 6) has been approved by the NRC for DPC analyses.

Each CHF correlation data base is comprised of test bundles that represent the unit and guide tube cell geometries for a particular fuel type or types. Non-standard geometries, such as multiple unheated rods or vacancies, are not typically present in the CHF correlation's data base. In order to determine the impact of such severe changes on DNBR predictions, DPC and B&W Fuel Company (BWFC) completed a CHF test program that included both standard and non-standard geometry configurations (Reference 7).

5.1 CHF TEST PROGRAM

A CHF test program was conducted at the Columbia Heat Transfer Test Facility during 1992. This test program consisted of five different Mark-BW fuel assembly bundle geometries. The test program included the following 5 by 5 rod geometries:

Unit Cell - 25 nominal diameter heated rods (Test Bundle BW13.1).

Guide Tube Cell - 24 nominal diameter heated rods with one larger diameter cold rod in the center to represent a Guide Tube (Test Bundles BW14.1, BW19).

Cold Unit - 24 nominal diameter heated rods with one nominal diameter cold rod in the center to represent a single unheated rod (Test Bundle BW15.1).

Cold Row - 22 nominal diameter heated rods with three nominal diameter cold rods to represent multiple unheated rods in close proximity (Test Bundle BW16).

Water Hole - 24 nominal diameter heated rods with one rod completely removed to represent a water hole or vacancy within the assembly (Test Bundle BW17).

The confirmatory test bundle from the 1987 test program (25 nominal diameter heated rods), denoted BW12, was also included.

FIGURES 1 through 5 show the layout of each test bundle geometry. All test bundles used a 1.55 chopped cosine axial power shape. The heated length of 143.4 inches, the rod pitch of 0.496 inches, and the rod diameters were all selected to represent production Mark-BW fuel assembly nominal values. The inlet condition test data matrix of power, pressure, temperature and mass flux for all bundles had the same range. These parameter ranges bound the limits of the existing BWCMV correlation and are listed in Reference 7.

The intent of the test program was to provide measured data to evaluate the response of currently approved CHF correlations and for inclusion into future correlations. The geometries were selected to envelope the most limiting reconstituted configurations that could be tested within the limits of the facility. The standard test configurations (unit and guide tube cells) provide comparison data for the test program.

The single unheated rod or cold unit represents one filler rod surrounded by fuel rods. The unheated rod in this run has the same outside diameter as a standard fuel rod (slightly smaller than a guide tube). The cold row test bundle represents up to three unheated rods in the same vicinity (connected subchannels). The water hole test bundle shows the impact of a vacancy on the local hydraulic conditions and thermal behavior.

5.2 TEST PROGRAM RESULTS

In this report, the evaluations for each test geometry were performed with VIPRE-01 and the BWCMV correlation. The results are listed in TABLE 2. The table shows the test geometry, the average Measured to Predicted (M/P) ratio of all test points, the number of data points for each test section, and standard deviation for the each test section. The bottom of the table summarizes the data in the test program data base.

The indication of how well a correlation performs against the measured test data is shown by the M/P ratio. Since the M/P ratio has the correlation's prediction in the denominator, a value greater than 1.0 indicates a conservative prediction. All the data analyzed in this analysis shows the current form of the BWCMV CHF correlation has considerable DNBR margin when used to analyze the Mark-BW fuel assembly. This margin is provided by the mixing vane grid design and a separate submittal has been made to the NRC to modify the correlation to use this margin (Reference 7).

The most important indication from the data to observe for this report is the relative behavior of one test section to another, not the absolute magnitude of the M/P ratios. This predictable response of the thermal hydraulic code/correlation confirms the applicability of existing methods to reconstituted geometries. The general behavior characteristics of the different test sections compared to each other is the major point of interest.

Table 2 shows that the thermal-hydraulic code does an excellent job of predicting local conditions even with very large changes from the normal heated geometry. This is evident by the M/P ratios of each test section being close to each other and to the average for the entire test program. The M/P values for the single and multiple unheated rod test bundles are very close to the unit and guide tube test sections. The vacancy test bundle is slightly lower [ ] than the average for all geometries. This is much smaller than one standard deviation (σ) for the data base and is well within the 5% repeatability historically assumed for CHF testing.

The main conclusion of the test program with respect to non-standard geometries is that the current DPC licensed code and correlation (VIPRE-01 and BWCMV) can predict the DNB behavior of mixing vane fuel assemblies that include multiple unheated rods or vacancies. No penalties or adjustments are necessary for DNB calculations performed for reconstituted geometries.

5.3 APPLICATION TO NON-MIXING VANE FUEL

As stated previously, the test program was based on the Mark-BW fuel assembly design which has mixing vane grids. The key point demonstrated by the results of the test program is that the prediction of the local fluid conditions by the thermal-hydraulic code consistently represents the physical conditions, even for very different fuel rod geometries.

Since the correlation predictions of critical heat flux depend solely on local conditions, the BWC correlation for non-mixing vane fuel will be just as valid in non-standard geometries as the BWCMV correlation. In order to show that this relationship is valid, several cases containing unheated rods and water holes were analyzed with the approved models for both Mark-BW/BWCMV and Mark-B/BWC and the DNB behavior compared.

The complex models (75 channel and 64 channel for Mark-BW and Mark-B respectively) from References 5 and 6 were used in this evaluation. The reference case that all runs are compared to is a high pressure, high temperature, high power Maximum Allowable Total Peaking (MATP) case with the reference radial and axial power distributions. Several cases with multiple cold rods or a vacancy in different configurations were run with each model. FIGURE 6 shows the fuel rod and channel ID's and radial power distribution values for Mark-BW runs and FIGURE 7 shows the same information for the Mark-B runs.

5.3.1 Unheated Filler Rods

To evaluate the impact of multiple zero power rods, five cases were evaluated for each fuel design. TABLE 3 shows the MDNBR values for selected channels for all the cases evaluated. Section 1 has the results for Mark-BW fuel and Section 2 for Mark-B.

In each configuration, the MDNBR for the non-standard geometry case is always higher or nearly identical than the reference case (with all heated rods). Comparisons of the channel mass flux and enthalpy at the point of MDNBR (FIGURES 8 through 11) show that both parameters are either unaffected or move in the direction that increases DNBR (increased mass flux, lower enthalpy). The location of the channel with MDNBR moves depending on the proximity of the cold rod to the highest power fuel rod. This comparison (TABLE 3) also shows that the impact of unheated rods on DNB is localized within a [ ] subchannel neighborhood.

FIGURES 8 through 11 depict the channel to channel distribution of the selected parameter (either mass flux or enthalpy). The top graph in all the figures shows the parameter gradient, calculated with the reference radial power distribution, for each fuel design. The second (bottom) graph shows the changes in parameter distribution for the non-standard geometry. All the channel ID's are consistent with the nomenclature in FIGURES 6 (Mark-BW) and 7 (Mark-B) but are rotated 180 degrees for clarity. The parameter scaling is consistent for all graphs of the same parameter and fuel design for both non-standard configurations.

FIGURES 8 and 9 show the behavior of mass flux when an unheated filler rod is modeled as Rod 3. FIGURE 8 is for the Mark-BW fuel type and FIGURE 9 for Mark-B. Channels 2 and 4 are adjacent to the unheated rod for both models. The mass flux increases [ ] for these channels and the behavior is consistent for both fuel designs.

This case is representative of the response in all cases analyzed. For all configurations, the subchannels next to the cold rods increased in mass flux. Additionally, the most limiting DNBR channel mass flux either increased (depending on proximity to the unheated rod) or remained unchanged.

FIGURES 10 and 11 show the response of enthalpy for the same unheated rod case. The decrease in enthalpy for the channels adjacent to the cold rod is [ ]. Again, the behavior is consistent for both the mixing and non-mixing vane fuel assembly designs. The response for all the cases analyzed was identical. As with the mass flux example, the limiting DNBR channel enthalpy either improved (lower enthalpy value) or was the same as in the reference case.

The conclusions of this analysis are two-fold. First, both fuel designs behave in the same manner which indicates that if CHF testing were performed on Mark-B fuel, the code and CHF correlation would correctly represent the local conditions and DNBR values. Secondly, the limiting DNBR always occurs in a non-adjacent subchannel regardless of the number and location of cold rods. This means that as long as the peaking changes caused by the addition of cold rods are evaluated by Nuclear Design (as described in Section 3.0) and determined to be acceptable, all established DNB criteria are satisfied. In other words, acceptable DNB performance in a core with stainless steel rods is ensured by compliance to the radial core peaking limit represented by the Maximum Allowable Total Peaking (MATP) Limits (References 5 and 6).

5.3.2 Vacancies or Water Holes

The analysis method for the behavior of Mark-B fuel/BWC correlation in the case of a vacancy was evaluated in the same manner as for unheated rods. The complex models for both fuel types were run for a high pressure, high temperature, high power MATP case and with a single vacancy in different locations within the eighth core modeled. Only one vacancy was modeled due to the limit of one vacancy per fuel assembly quadrant imposed by the mechanical analysis (Section 4.7).

The DNBR results of these runs are presented in TABLE 4. Section 1 has the results for Mark-BW fuel and Section 2 for Mark-B. As with the unheated rod configurations, the MDNBR is higher or nearly identical in the reference case (with all heated rods) than in the runs modeling vacancies. Again, comparisons of the mass flux and enthalpy at the point of MDNBR shows that both parameters either remain constant or move in the direction that increases DNBR (increased mass flux, lower enthalpy). The response of both fuel types is graphically depicted in FIGURES 12 through 15. The same graph orientation previously described in Section 5.3.1 is also used in these four figures.

FIGURES 12 (Mark-BW) and 13 (Mark-B) show the behavior of mass flux when a vacancy is modeled. The vacancy was modeled in place of Rod 5 for the Mark-BW runs and in place of Rod 6 for the Mark-B runs. Again, as with the unheated rod case, the mass flux increases in the channels adjacent to the vacancy. The mass flux increase is larger than with the unheated rod [ ] as would be expected. This is due to the increase in flow area and smaller form loss coefficient for the channels with the water hole. The results for the channels adjacent to the water hole are consistent in both magnitude and direction for both fuel designs.

FIGURES 14 and 15 show the response of enthalpy for the vacancy model. The decrease in enthalpy for the channels adjacent to the water hole is [ ]. Again, the behavior is consistent for both the mixing and non-mixing vane fuel assembly designs.

The response of the limiting DNBR channel to the water hole is also the same as with unheated rods. The most limiting channel's mass flux and enthalpy either improved the DNBR (increased mass flux and lower enthalpy) or remained virtually unchanged. At a distance of more than [ ] subchannels, the local fluid parameters are unaffected by the presence of a water hole. Thus, the effect of the vacancy non-standard geometry is localized and it introduces no significant change or a benefit when compared to standard geometry analyses.

The conclusions for the analysis with vacancies present is the same as for the presence of unheated rods. Both fuel designs behave similarly and the thermal-hydraulic code predicts that the local fluid conditions important to DNBR calculations either improve or are basically unaffected when vacancies are modeled in a limiting assembly. The improvement is shown by an increase in mass flux and a decrease in enthalpy at the point of MDNBR.

6.0 CONCLUSIONS

This report documents Duke Power Company's methodology for performing reconstituted fuel assembly analyses. The methodology uses the models and codes currently approved for cycle design at the Oconee, McGuire, and Catawba Nuclear Stations. The limitations beyond the criteria for standard fuel assemblies are listed in Section 2.0.

The affects on fuel analyses in the nuclear, mechanical, and thermal hydraulic areas were evaluated. The nuclear analysis impact, changes in local and global power distributions caused by reconstituted geometries, will be analyzed with the licensed nuclear codes. Cycle specific evaluations will assess the magnitude and extent of this change. Bounding mechanical analyses were performed to demonstrate the mechanical integrity of the fuel assembly is maintained with filler rods or water holes. All mechanical design criteria are either generically bounded or will be analyzed on a cycle specific basis.

A CHF test program was performed to verify applicability of the BWCMV CHF correlation when applied to the Mark-BW fuel assembly. Several reconstituted geometries, such as multiple cold rods and a water hole were included in the test. All CHF test sections showed very similar thermal-hydraulic behavior when analyzed with VIPRE-01 and the BWCMV CHF correlation. Additionally, the applicability of the thermal hydraulic code/correlation response for non-mixing vane Mark-B fuel has also been demonstrated by comparison of the behavior characteristics of reconstituted geometries in both mixing and non-mixing vane fuel designs. Both responded in the same manner.

7.0 REFERENCES

1. Oconee Nuclear Station Reload Design Methodology Technical Report, NFS-1001, Duke Power Company, Charlotte, North Carolina, July 1981.

2. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002A, Revision 2, Duke Power Company, Charlotte, North Carolina, October 1985.

3. McGuire Nuclear Station, Catawba Nuclear Station Nuclear Physics Methodology for Reload Design, DPC-NF-2010-A, Duke Power Company, Charlotte, North Carolina, May 1985.

4. Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power Company, Charlotte, North Carolina, November 1992.

5. Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003P-A, Duke Power Company, Charlotte, North Carolina, October 1989.

6. McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2004P-A, Duke Power Company, Charlotte, North Carolina, December 1991.

7. BAW-10189P, CHF Testing and Analysis of the Mark-BW Fuel Assembly Design, D.A. Farnsworth and G.A. Meyer, BWFC, Lynchburg, Virginia, August 1993.

8. DPC-NE-2001-A, Fuel Mechanical Reload Analysis Methodology For Mark-BW Fuel, Duke Power Company, Charlotte, North Carolina, October 1990.

9. DPC-NE-1002A, Oconee Nuclear Station Reload Design Methodology II, Duke Power Company, Charlotte, North Carolina, October 1985.

TABLE 1

Reload Design & Analysis Flow Path

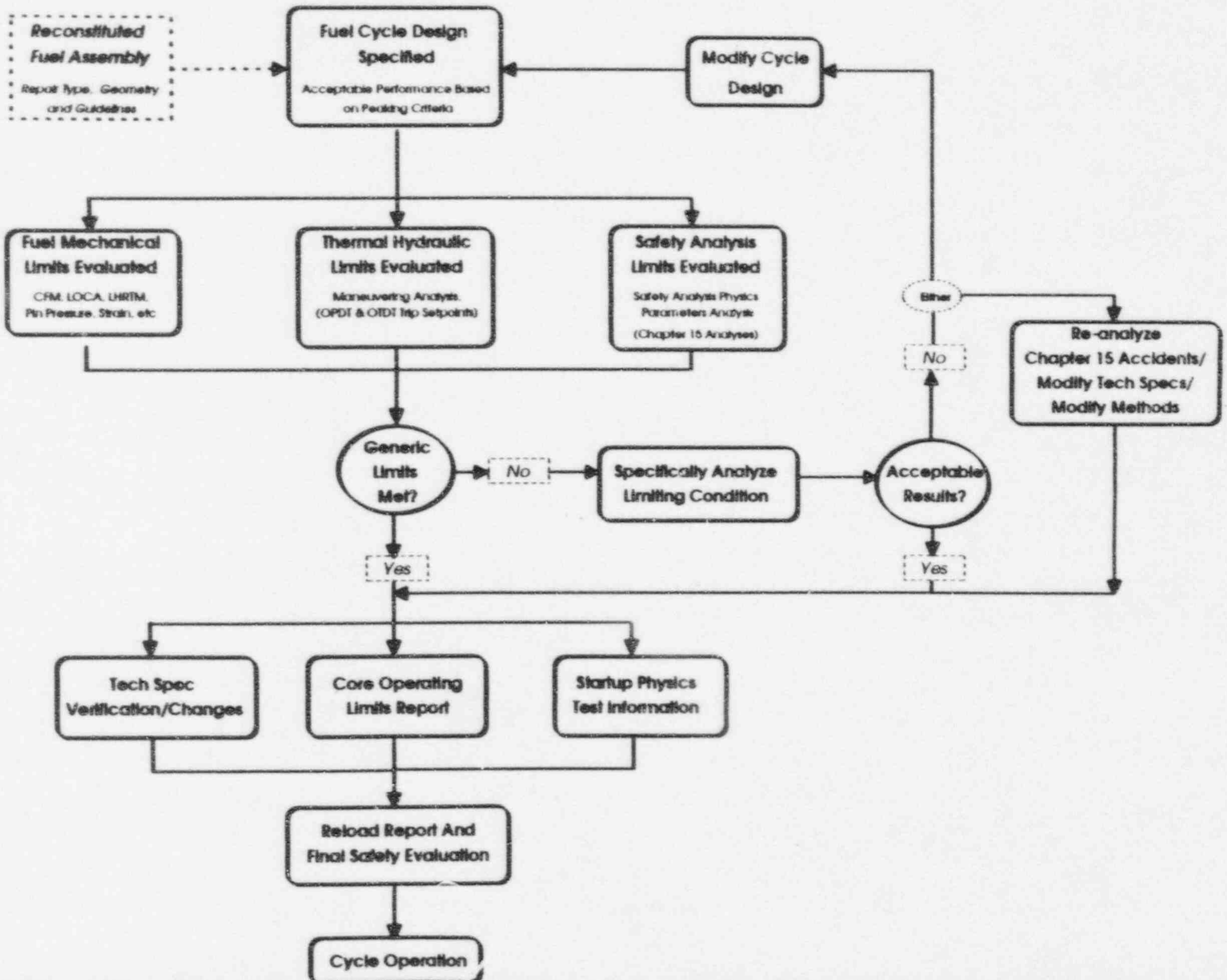


TABLE 2

Summary of Test Bundle Measured to Predicted (M/P) Critical Heat Flux Ratios

BWCMV Critical Heat Flux Correlation

Test Bundle	M/P	# Data Points	σ
12 - Unit Cell	[		]
13.1 - Unit Cell	[		]
14.1 - Guide Tube	[		]
15.1 - Cold Unit	[		]
16 - Cold Row	[		]
17 - Water Hole	[		]
19 - Guide Tube 2	[		]
Average/Totals	**[**		**]**

Table 3
Model Analyses with Unheated Rods

Section 1 - Mark-BW Runs

Case 1 - Reference Radial Power Distribution (See Figure 6)
Case 2 - Rod 3 Unheated
Case 3 - Rods 3 and 4 Unheated
Case 4 - Rods 3, 4, and 5 Unheated
Case 5 - Rods 3 and 5 Unheated
Case 6 - Rods 13, 14, and 18 Unheated

MDNBR (Minimum for Case boxed in bold)

Case #	Ch 1	Ch 2	Ch 3	Ch 4	Ch 5	Ch 12	Ch 14	Ch 19
1								
2								
3								
4								
5								
6								

Section 2 - Mark-B Runs

Case 1 - Reference Radial Power Distribution (See Figure 7)
Case 2 - Rod 2 Unheated
Case 3 - Rod 3 Unheated
Case 4 - Rods 3 and 4 Unheated
Case 5 - Rods 6 and 7 Unheated
Case 6 - Rods 11, 12, and 15 Unheated

MDNBR (Minimum for Case boxed in bold)

Case #	Ch 1	Ch 2	Ch 3	Ch 4	Ch 5	Ch 7	Ch 11	Ch 12
1								
2								
3								
4								
5								
6								

Table 4
Model Analysis With Vacancies

Section 1 - Mark-BW Runs

Case 1 - Reference Radial Power Distribution (See Figure 6)
Case 2 - Rod 2 Removed
Case 3 - Rod 5 Removed
Case 4 - Rod 18 Removed

MDNBR (Minimum for Case boxed in bold)

Case #	Ch 1	Ch 2	Ch 3	Ch 4	Ch 5	Ch 12	Ch 14	Ch 19
1								
2								
3								
4								

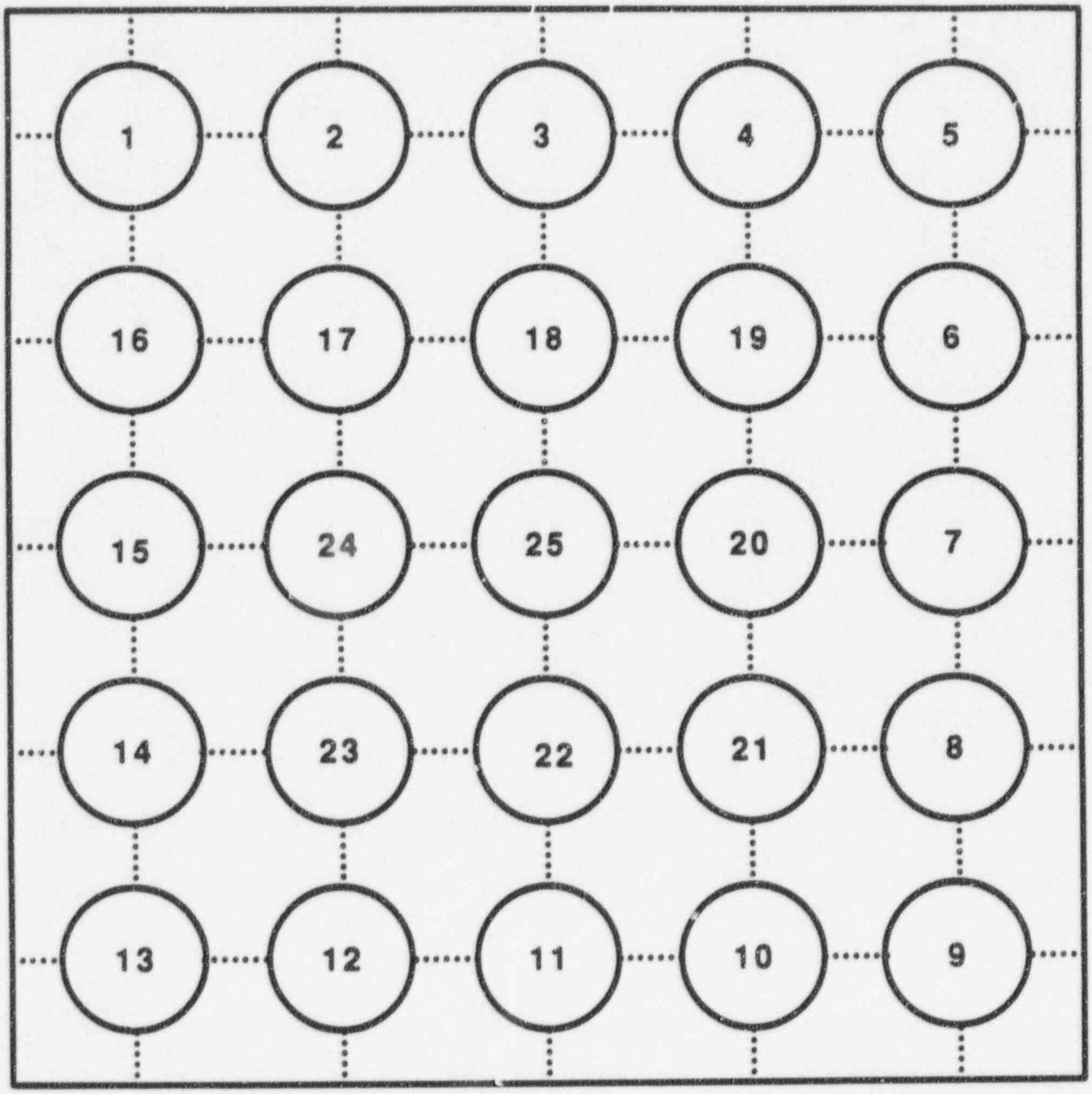
Section 2 - Mark-B Runs

Case 1 - Reference Radial Power Distribution (See Figure 7)
Case 2 - Rod 3 Removed
Case 3 - Rod 6 Removed
Case 4 - Rod 11 Removed

MDNBR (Minimum for Case boxed in bold)

Case #	Ch 1	Ch 2	Ch 3	Ch 4	Ch 5	Ch 11	Ch 20
1							
2							
3							
4							

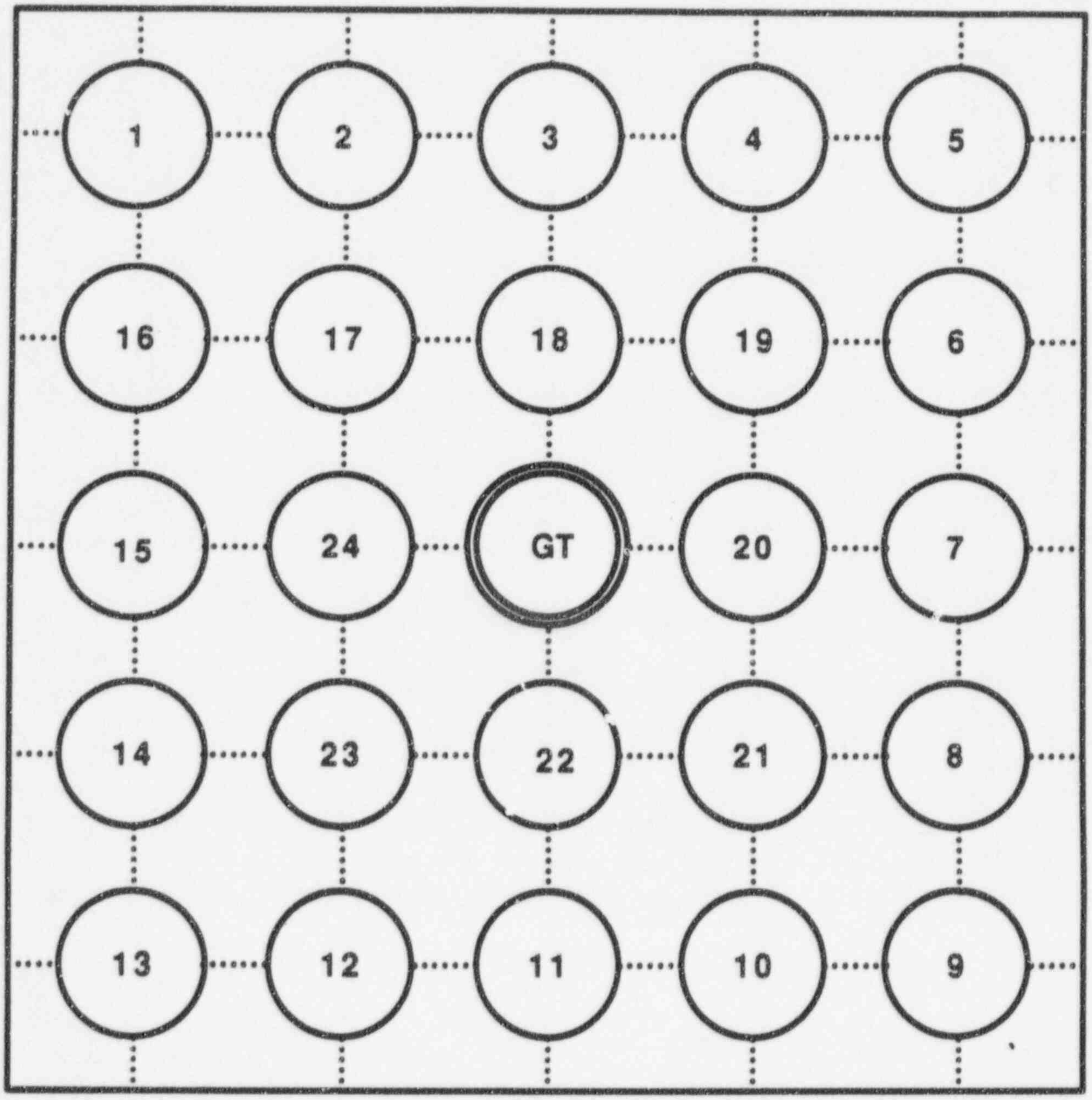
Figure 1
Unit Cell Test Bundle



Rod OD = 0.374 inches

Rod Powers
1-16 = [ ]
17-25 = [ ]

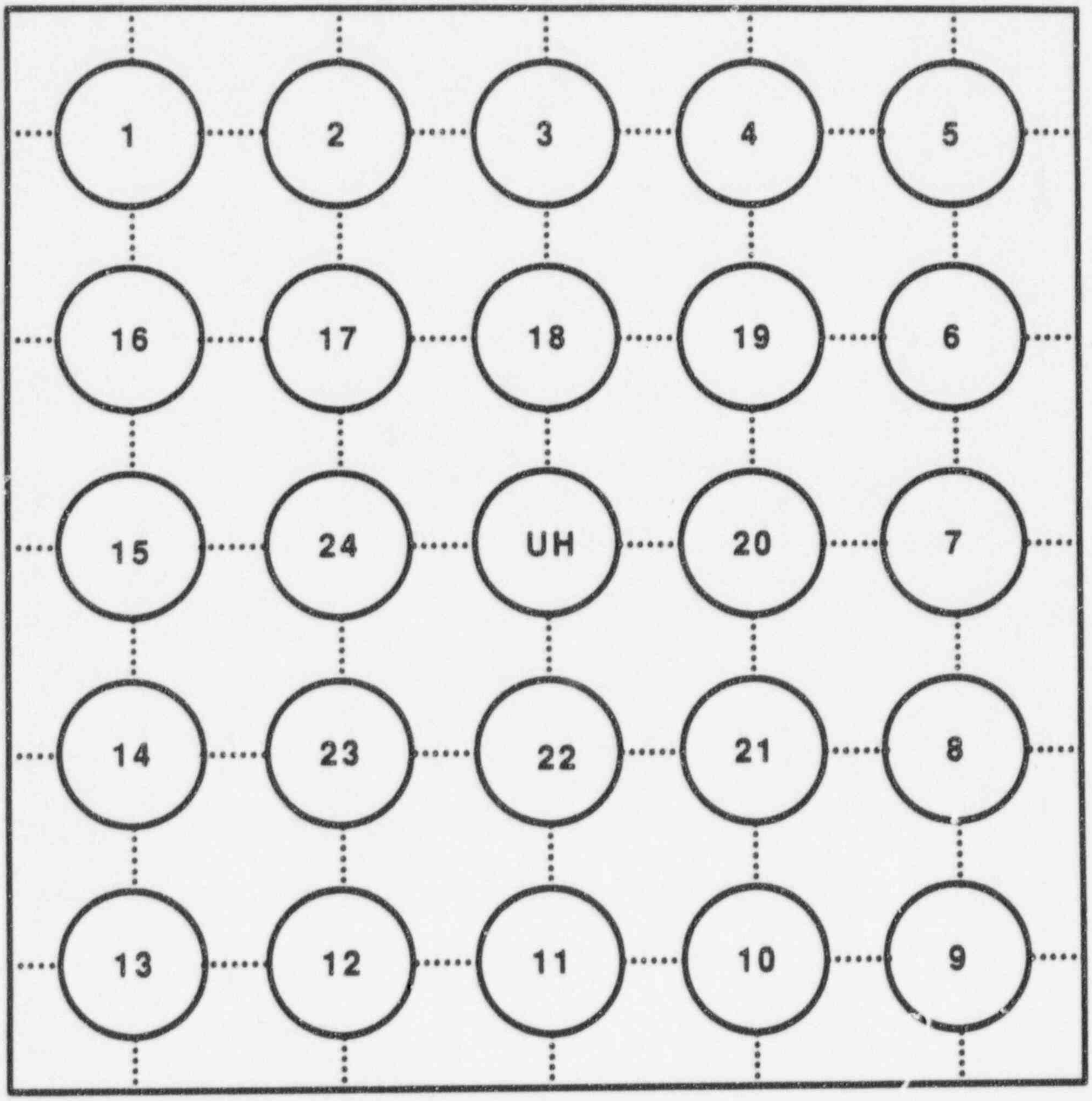
Figure 2
Guide Tube Test Bundle



Rod OD = 0.374 inches
Guide Tube (GT) OD = 0.482 inches

<u>Rod Powers</u>
1-16 = [ ]
17-24 = [ ]

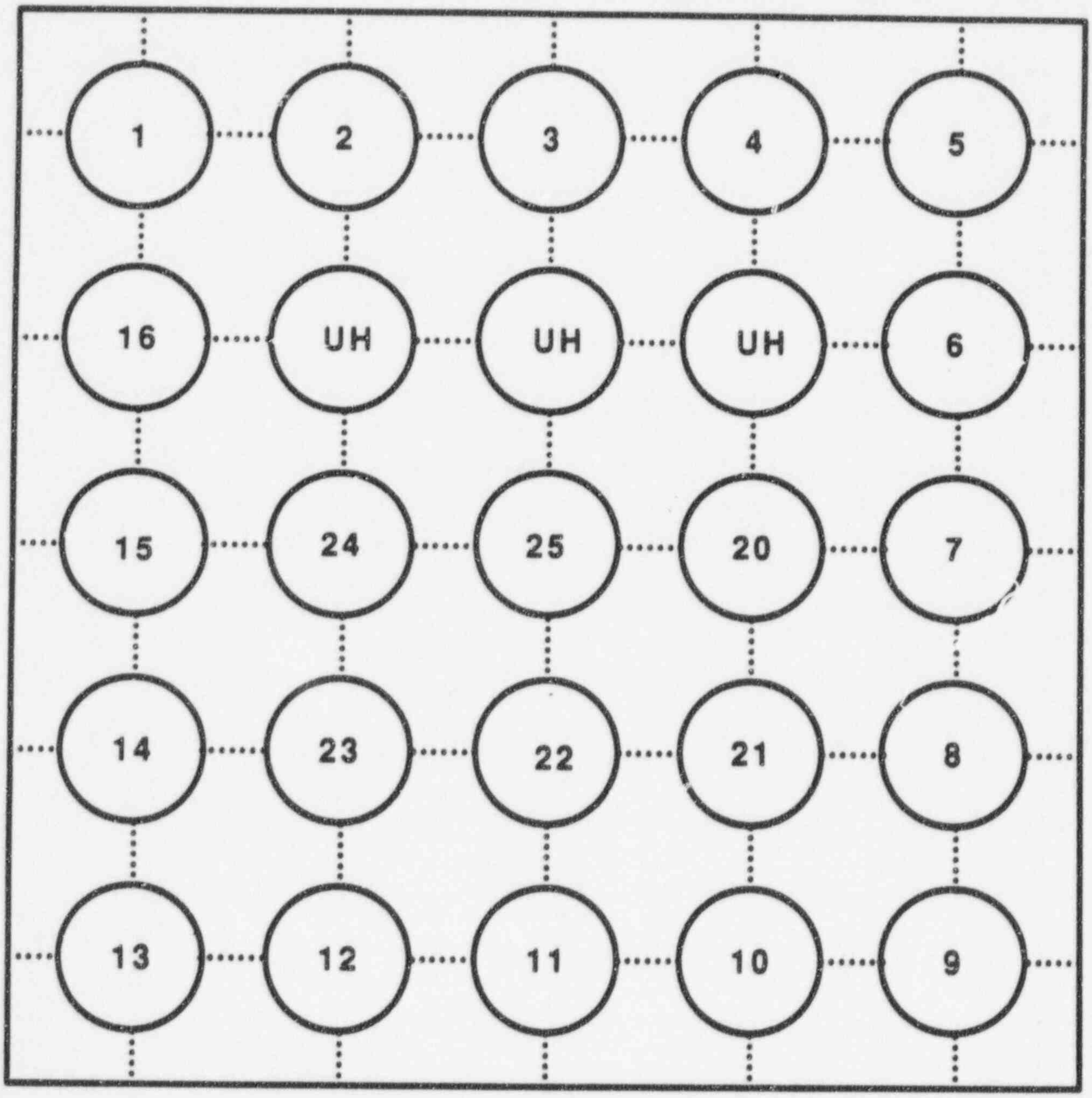
Figure 3
Cold Unit Test Bundle



Rod OD = 0.374 inches
Unheated (UH) Rod OD = 0.374 inches

Rod Powers
1-16 = [ ]
17-24 = [ ]

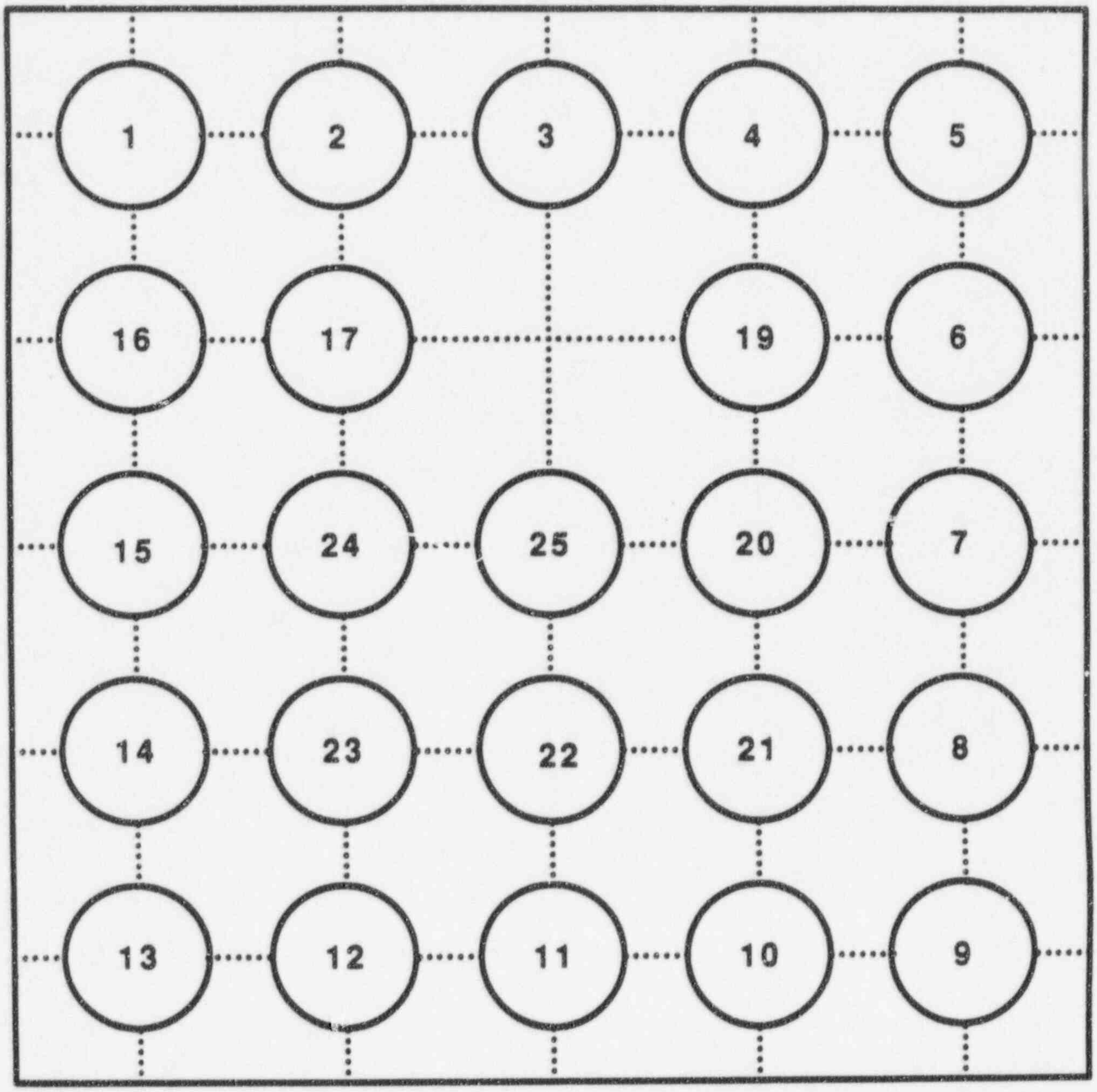
Figure 4
Cold Row Test Bundle



Rod OD = 0.374 inches
Unheated (UH) Rod OD = 0.374 inches

Rod Powers
1-16 = [ ]
20-25 = [ ]

Figure 5
Water Hole Test Bundle



Rod OD = 0.374 inches

<u>Rod Powers</u>
1-16 = [ ]
17,19-25 = [ ]

FIGURE 6

Mark-BW 75 Channel Model

Limiting Assembly Subchannel and Rod Geometry

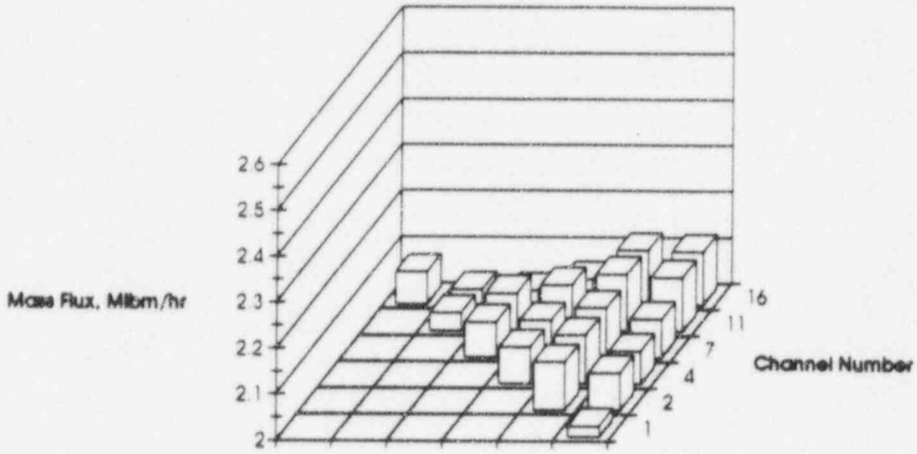
FIGURE 7

Mark-B 64 Channel Model

Limiting Assembly Subchannel and Rod Geometry

FIGURE 8
Mark-BW Mass Flux Profile
Filler Rod

Mark-BW Reference Case



Mark-BW Stainless Steel Rod In 3

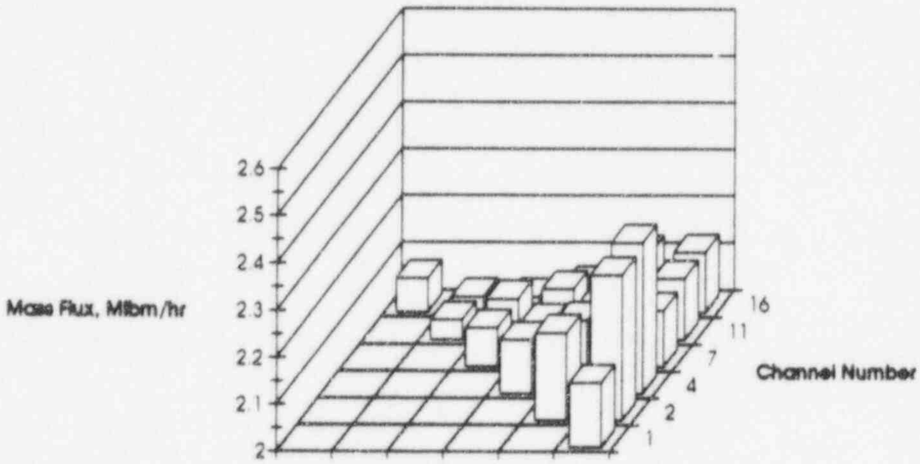
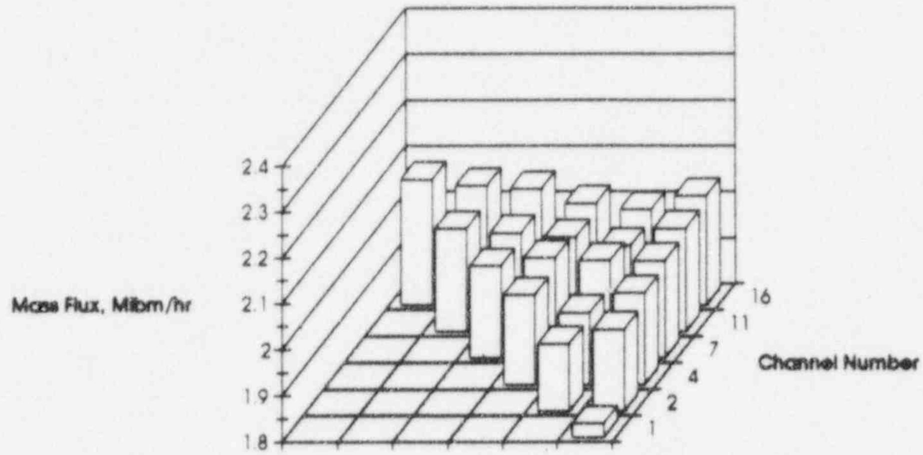


FIGURE 9
Mark-B Mass Flux Profile
Filler Rod

Mark-B Reference Case



Mark-B Stainless Steel In Rod 3

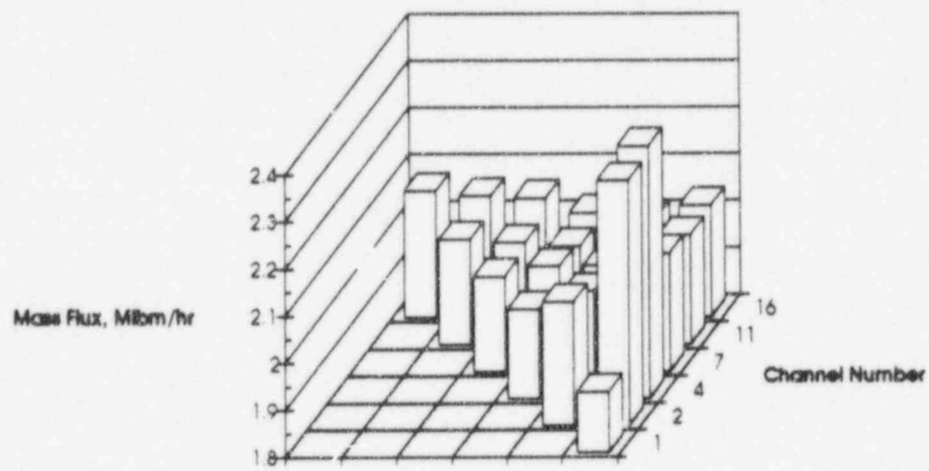
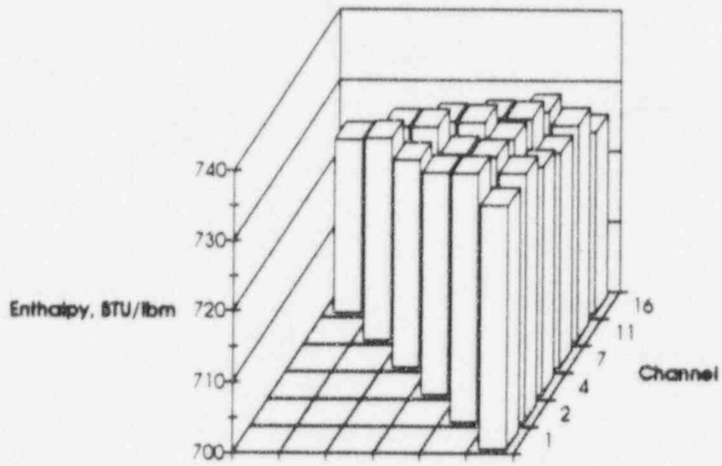


FIGURE 10
Mark-BW Enthalpy Profile
Filler Rod

Mark-BW Reference Case



Mark-BW Stainless Steel Rod In 3

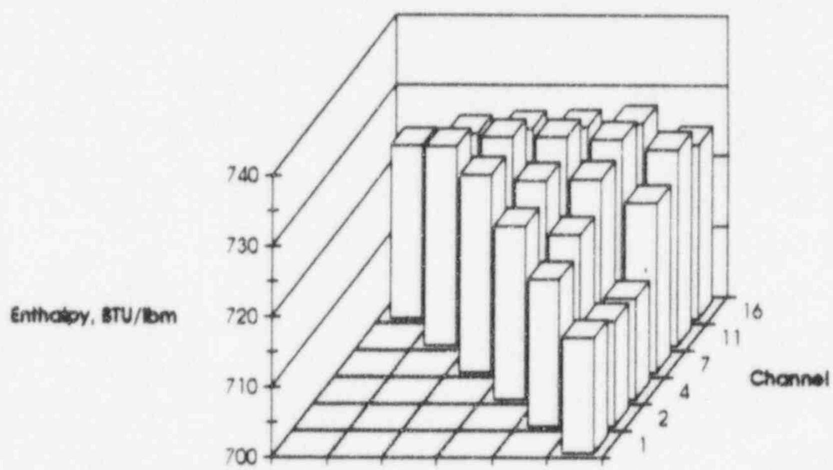
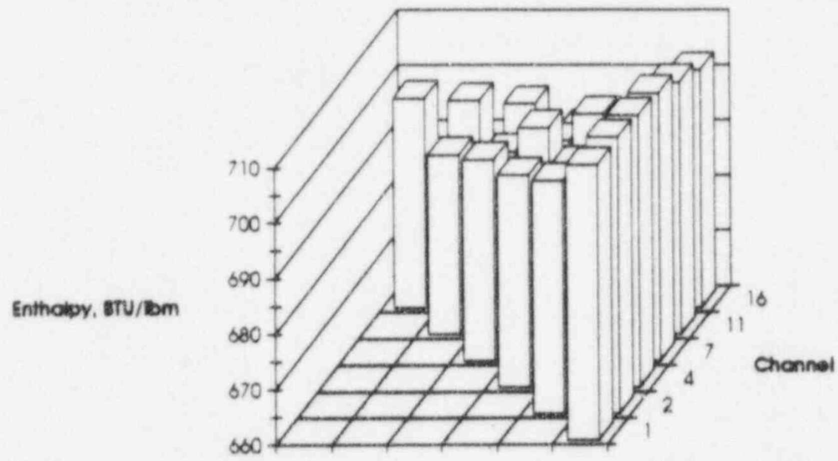


FIGURE 11
Mark-B Enthalpy Profile
Filler Rod

Mark-B Reference Case



Mark-B Stainless Steel Rod in 3

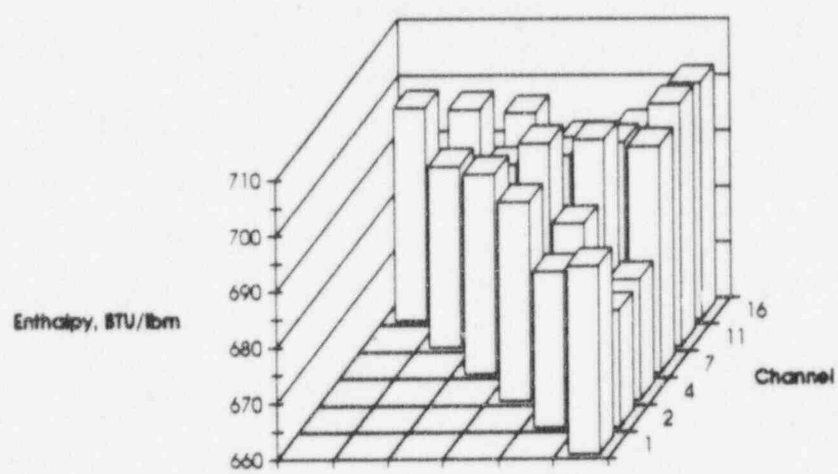
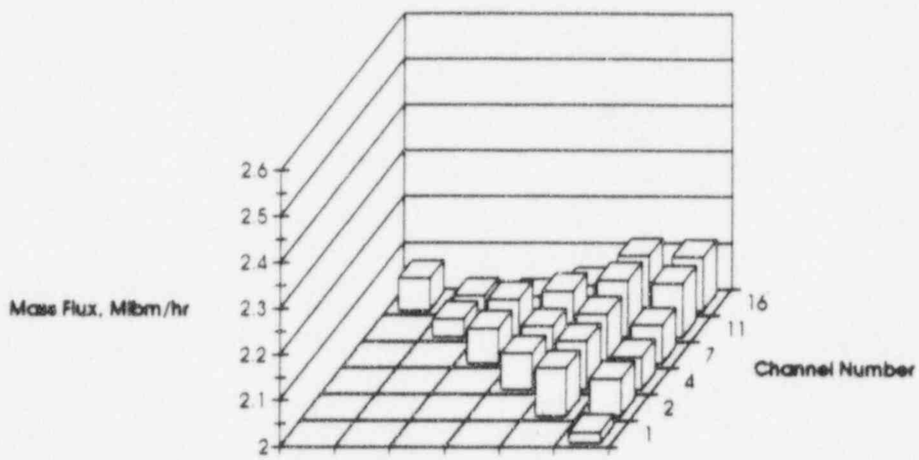


FIGURE 12
Mark-BW Mass Flux Profile
Vacancy

Mark-BW Reference Case



Mark-BW - Water Hole in Rod 5

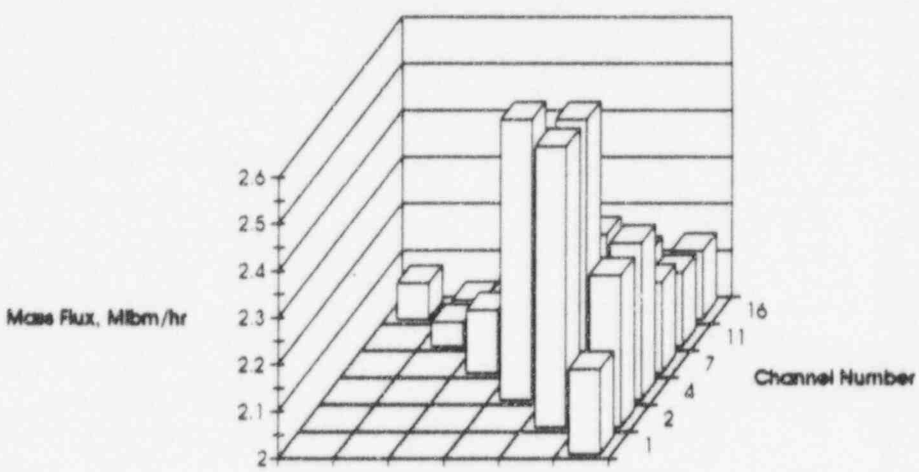
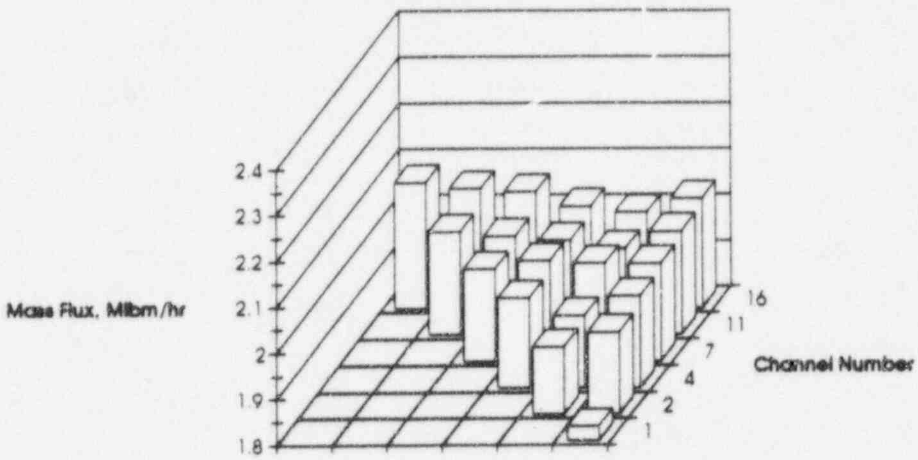


FIGURE 13
Mark-B Mass Flux Profile
Vacancy

Mark-B Reference Case



Mark-B Water Hole in Rod 6

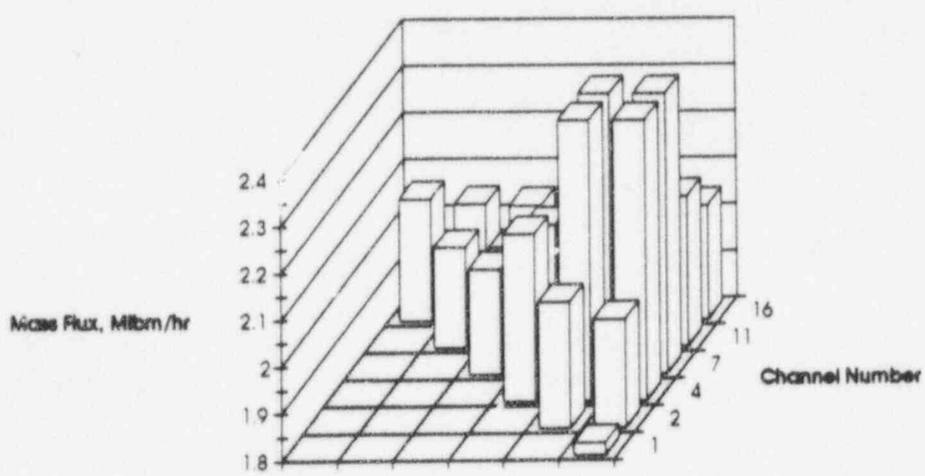
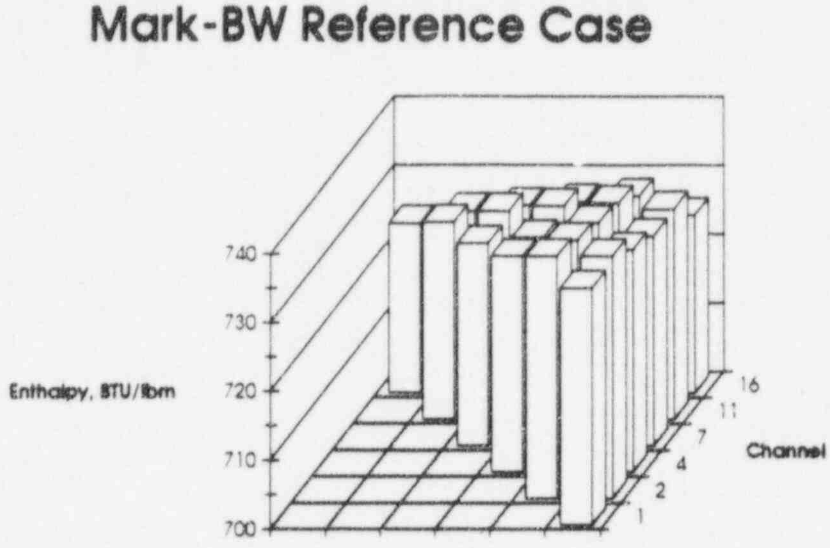


FIGURE 14
Mark-BW Enthalpy Profile
Vacancy



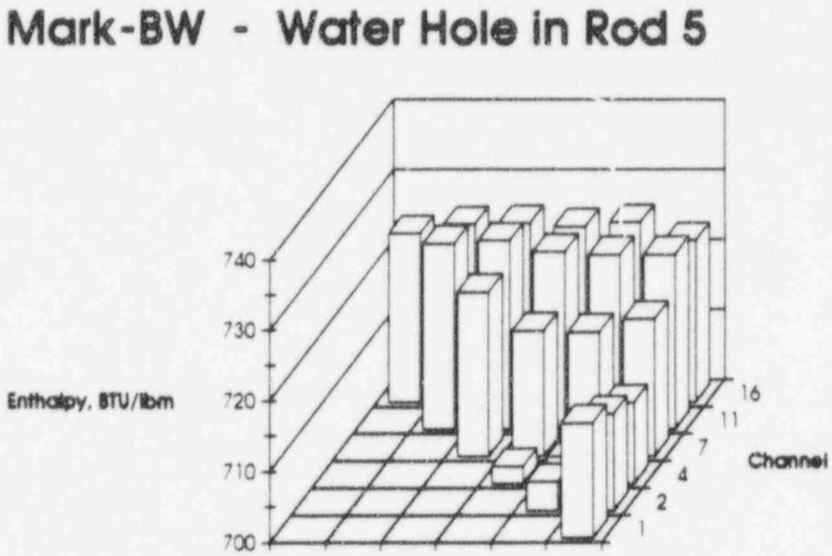
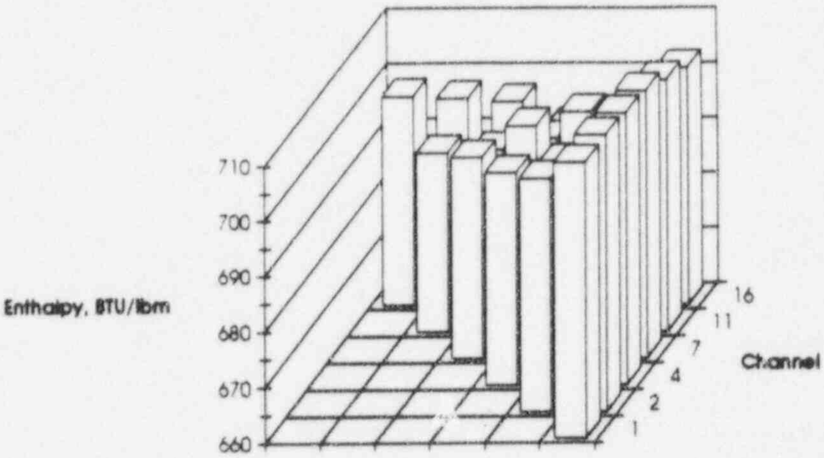
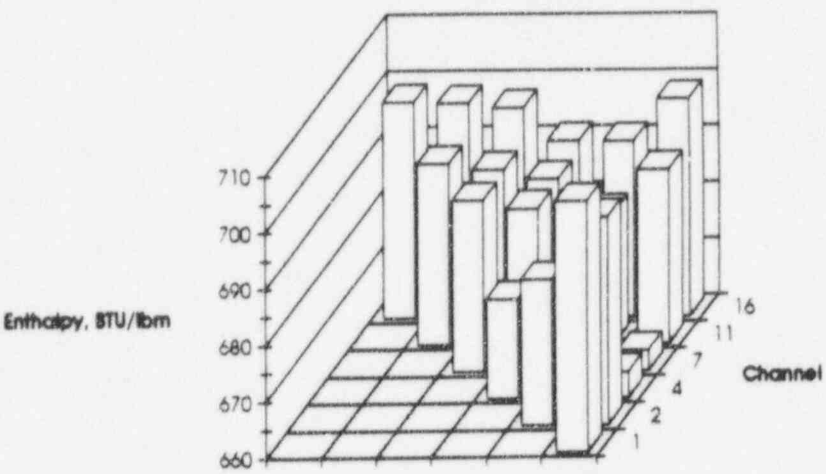


FIGURE 15
Mark-B Enthalpy Profile
Vacancy

Mark-B Reference Case



Mark-B Water Hole in Rod 6



Duke Power Company
P.O. Box 1006
Charlotte, NC 28201-1006

M. S. Tuckman
Senior Vice President
Nuclear Generation
(704)382-2200 Office
(704)382-4360 Fax

DUKE POWER

August 10, 1994

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Duke Power Company
Catawba Nuclear Station
Docket Numbers 50-413 and -414
McGuire Nuclear Station
Docket Numbers 50-369 and -370
Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Topical Report DPC-2007P, "Fuel Reconstitution Analysis Methodology"

By letter dated September 23, 1993, the subject topical report was submitted for NRC staff review. The staff has requested, by letter dated April 26, 1994, additional information. Attachment I contains responses to the questions contained in that letter.

In accordance with 10CFR 2.790, Duke Power Company requests that this report be considered proprietary. Information supporting this request is included in the attached affidavit. Attachment II contains a non-proprietary version of the responses.

If you have any questions, or need more information, please call Scott Gewehr at (704) 382-7581.

M. S. Tuckman

M. S. Tuckman

cc: Mr. L. A. Wiens, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop 14H25, OWFN
Washington, D. C. 20555

Mr. S. D. Ebneter, Regional Administrator
U.S. Nuclear Regulatory Commission - Region II
101 Marietta Street, NW - Suite 2900
Atlanta, Georgia 30323

bxc: G. A. Copp
K. R. Epperson
File

AFFIDAVIT OF M. S. TUCKMAN

1. I am Senior Vice President, Nuclear Generation Department, Duke Power Company ("Duke"), and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing, and am authorized to apply for its withholding on behalf of Duke.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission ("NRC") and in conjunction with Duke's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that providee a competitive advantage to Duke.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, it is to be received in confidence by the NRC.

   (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

   (v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the report DPC-NE-2007, "Fuel Reconstitution Analysis Methodology" and supporting documentation, and omitted from the non-proprietary versions.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN (Page 2)

This information enables Duke to:

(a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

(b) Describe the method by which damaged fuel assemblies are repaired.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

M. S. Tuckman

M. S. Tuckman

Sworn to and subscribed before me this 11th day of August, 1994.
Witness my hand and official seal.

Mary P. Nims

Notary Public

My commission expires Jan 22, 1996.

ATTACHMENT I
RESPONSES TO QUESTIONS

Responses to Questions on DPC-NE-2007P, "Duke Power Company Fuel Reconstitution Analysis Methodology"

1. *What effect will the replacement of up to 10 fuel rods with inert filler rods have on the fuel assembly lift-off analysis for hot and cold (zero power) conditions?*

This question will be answered first for Mark-BW (17x17) fuel then for Mark-B (15x15) fuel.

MARK-BW - The weight difference between the filler rod and a standard fuel rod is [ ] pounds so the maximum fuel assembly weight decrease is [ ] pounds. Replacement filler rods have no hydraulic effect on fuel assembly lift characteristics so the only change is the weight difference. The holddown force margin (difference between assembly weight and lift force) at the mechanical design flow limit (420,000 gpm) is [ ] pounds cold and [ ] pounds hot. This gives more than adequate margin against fuel assembly lift for an assembly with 10 replacement filler rods.

MARK-B - The weight difference between the filler rod and a standard fuel rod is [ ] pounds so the maximum fuel assembly weight decrease with 10 filler rods is [ ] pounds. Replacement filler rods have no hydraulic effect on fuel assembly lift characteristics so the only change is the weight decrease. As with Mark-BW fuel, the limiting lift scenario is at cold conditions. The holddown force margin (difference between assembly weight and lift force) is calculated for each core configuration to determine the minimum RCS temperature at which the fourth reactor coolant pump can be started. This change in fuel assembly weight will be generically included in these calculations.

2. *Are replacement rod engagements based on hot or cold (zero power) conditions? If they are based on cold conditions, what are the expected radial and axial hot dimensions of the solid stainless-steel filler rods for the Mark-B and Mark-BW fuel assemblies? Also, please provide a description of how assembly growth is determined and the growth model used in the engagement analysis.*

The replacement rod/upper end grid engagements reported are based on end-of-life cold conditions. This philosophy was utilized because the coefficient of thermal expansion for stainless steel is greater than that for Zircaloy. Also, since axial irradiation growth experienced by the Zircaloy guide thimbles will be greater than that of a stainless steel replacement rod, the worst case engagement between the upper spacer grid and stainless steel replacement rod will occur at end-of-life cold conditions.

Note that the thermal expansion difference between the Zircaloy and stainless steel was considered in Sections 4.1 and 4.2 to address shoulder gap and grid relaxation issues with regard to the stainless steel replacement rod. The following radial and axial hot dimensions of the solid stainless steel filler rods are included in order to fully respond to this question:

Mark-BW hot dimensions (650 degrees F)

RADIAL: [ ] inches (for maximum specified as-fabricated diameter of [ ] inches)

AXIAL: [ ] inches (for maximum specified as-fabricated length of [ ] inches)

Mark-B hot dimensions (650 degrees F)

RADIAL: [ ] inches (for maximum specified as-fabricated diameter of [ ] inches)

AXIAL: [ ] inches (for maximum specified as-fabricated length of [ ] inches)

Fuel assembly growth was determined by conservatively calculating the amount of guide tube growth which will be experienced during incore operation. Guide tube growth was calculated with support from conservative growth models obtained from BWFC. The growth models were established from fuel assembly post irradiation examination results obtained on fuel containing Zircaloy guide tubes produced to material and manufacturing specifications representative of BWFC fuel currently in operation at Duke Power reactors. Growth data is available on this material up through an in-reactor fuel assembly burnup of [ ] MWd/mtU. Both upper and lower tolerance growth curves were supplied by BWFC to enable conservative calculation of fuel assembly growth in engagement/fit-up evaluations.

3. *The methodology described in the submittal has been proposed for use with fuel assemblies comprised of mixing-vane spacers and fuel assemblies comprised of standard spacers. Will the actual reconstitution of damaged fuel rods also include the replacement of spacer grids? If so, will spacer designs be mixed in the assembly reconstitution, i.e., will mixing-vane spacers or standard spacers used on fuel assemblies comprised of mixing-vane spacers?*

No. Fuel assembly reconstitution will remove failed fuel rod(s) and replace the failed rod(s) with either a natural Uranium replacement rod, a stainless steel filler rod, or a vacancy as described in the report. No grid configuration or structural changes are made to the fuel assembly.

4. *In order to assist in the review of the submittal document, please provide Reference 7 of the submittal, BAW-10189P, "CHF Testing and Analysis of the Mark-BW Fuel Assembly Design," so that the use of the BWC and the BWCMV CHF correlations with reconstituted assemblies can be assessed. We would also like the CHF data on magnetic media for further examination.*

A copy of the reference topical report submitted by B&W Fuel Company is attached. Also included is a diskette with the CHF test data in ASCII format. The file names correspond to the test section identification number. The data includes the individual test point ID, the pressure, inlet temperature, inlet mass flux, bundle power, average surface heat flux, and CHF location (rod, channel, and axial location).

5. *What criteria has been established to decide whether a damaged fuel rod will be replaced by a depleted uranium rod, solid inert rod, or left vacant? Of particular concern are the reasons for leaving a vacancy rather than using a replacement rod. Please provide all possible scenarios for why a grid position would be left vacant in a reconstituted fuel assembly.*

As discussed in Section 1.0 of the report (pg. 1 and 2), the primary replacement rod candidate is a natural uranium fuel rod. This rod is the same design as a standard fuel rod with non-enriched U^{235} pellets. If the failure mechanism of the removed rod is such that the replacement rod would likely fail in subsequent cycles, a filler rod will be used. The most likely failure mechanism in this scenario would be grid-to-rod fretting at one axial location.

The use of a vacancy will be a last resort. If the fuel assembly is deemed repairable but there is doubt that a filler rod will perform as designed (i.e. could cause additional failures in adjacent cells), a vacancy would be used. An example of this is extensive grid damage to one cell location. Another example is grid-to-rod fretting at multiple axial elevations such that the probability is high that the filler rod would not stay in place. For these conditions, a vacancy would be the best alternative. The use of a vacancy will be allowed only if the change in nuclear peaking is acceptable. In such severe damage cases, a core redesign to remove the damaged fuel assembly will also be considered.

The following questions are related to the use of vacant fuel rod positions in a reconstituted fuel assembly.

6. *The reconstitution methodology presented in the submittal is supported by analyses performed for either up to 10 inert filler rods or 4 water holes. Are combinations of inert filler rods and water holes allowed in a reconstituted assembly? If so, in what combinations? Please provide the appropriate mechanical analyses because no analyses have been presented in the submittal.*

Combinations of up to ten inert filler rods and four water holes (one per fuel assembly quadrant) is considered allowable in a reconstituted fuel assembly. All mechanical analysis results and methods provided in Section 4 of DPC-NE-2007P, with the exception of seismic/LOCA loadings, are independent of the location and number of replacement rods or water holes. Furthermore, the results and methods detailed in Section 4 would not be adversely impacted due to a combination of stainless steel replacement rods and water holes in a single reconstituted assembly.

Spacer grid impact test data indicates that removing four rods (one per fuel assembly quadrant) decreases spacer grid stiffness by [ ]% without having a significant overall effect on reducing spacer grid impact strength and mode of failure. Since a decrease in spacer grid stiffness decreases the force in the spacer grid during a seismic/LOCA event, the addition of water holes in a reconstituted fuel assembly would serve to increase seismic/LOCA loading margins. Consequently, seismic/LOCA analyses applicable to a reconstituted fuel assembly containing ten stainless steel replacement rods envelope the combined condition of four water holes/ten stainless steel replacement rods.

The thermal hydraulic and nuclear peaking effects of a combination of replacement rods and water holes will be evaluated in the geometry that exists, on a case by case basis. The methodology used will be that described in the report.

7. *Please demonstrate that a vacancy will not increase the potential for fretting wear in the surrounding fuel rod locations. This demonstration should include any contributing factors such as grid damage and fretting test data verifying any analyses.*

The hydraulic effect of vacancies was evaluated by analyzing the change in subchannel lateral cross flow velocities. The vacancy geometry cross flow velocity profile was compared to the predictions in standard geometry. The cross flow velocities increased by approximately [ ] ft/sec but are still well within the hydraulic limits used to ensure that flow induced vibration does not occur with the fuel assembly.

No fretting tests were utilized to illustrate that the existence of fuel rod vacancies will not increase the potential for wear in surrounding fuel rod locations. Rather, Mark-B and Mark-BW spacer grid spring rate tests were consulted to obtain the necessary data to show that adjacent rod fretting is not a concern with the water hole condition.

The spring rate tests were performed to obtain load/deflection characteristics of the various Mark-B and Mark-BW grid cell configurations. Spacer grid spring load/deflection data was obtained for cladding segments in adjacent grid cell locations and with vacancies in surrounding cells. Load/deflection data was obtained after initially opening the grid cells to dimensions larger than the replacement rod diametral dimensions at hot in-core conditions. This simulated a manufacturing procedure known as spacer grid keying, which BWFC uses in their normal fabrication process to load rods into the fuel assembly bundle while minimizing degradation to the fuel rod cladding due to interface with the grid springs.

Results from the two test procedures indicate that the adjacent cell configurations have only a minimal effect on the adjacent rod preload imparted by the grid springs. More significantly, the tests indicate that preload is maintained on fuel rods fabricated to current Mark-B and Mark-BW diameter specifications in the event of adjacent vacancies. Therefore, the existence of water holes will not negatively influence grid/fuel rod fretting in adjacent spacer grid cell locations. Note that this conclusion may not be valid in the event of damage or degradation to the spacer grid. The increased potential for grid/fuel rod fretting due to spacer grid damage must be addressed on a case-by-case basis since the potential cell configurations resulting from grid damage is large.

8. *Please demonstrate that differential nuclear peaking resulting from a vacant fuel rod position would not lead to abnormal asymmetric growth of the adjacent grids and fuel rods.*

Fuel assembly growth is primarily a function of the fast flux distribution, specifically, neutrons with energies greater than 1 Mev. The fast flux distribution with vacancies present is spatially flat and can therefore be characterized by a smooth function which varies slowly. Because this function varies very little spatially, asymmetric fuel assembly growth is not a concern for vacant fuel rod locations within a fuel assembly.

9. *What effect will the replacement of up to 4 fuel rods with vacancies have on the fuel assembly lift-off analysis for hot and cold (zero power) conditions?*

This question will be answered first for Mark-BW fuel then for Mark-B fuel.

MARK-BW - The weight of a standard fuel rod is [ ] pounds so the maximum fuel assembly weight decrease with four vacancies is [ ] pounds. Vacancies within the lattice will cause a flow increase in the adjacent subchannels. From a lift perspective, this flow increase is balanced by a reduction in friction loss and a lower forms loss in the vacant cells. The net effect is no increase in the fuel assembly lift force. Therefore, only the weight change needs to be accounted for. As stated previously in response to Question 1, the minimum holddown force margin is [ ] pounds lift. This gives more than adequate margin against fuel assembly lift for an assembly with 4 vacancies.

MARK-B - The weight of a standard fuel rod is [ ] pounds so the maximum fuel assembly weight decrease with four vacancies [ ] pounds. Vacancies within the lattice will cause a flow increase in the adjacent subchannels. From a lift perspective, this flow increase is balanced by a reduction in friction loss and a lower forms loss in the vacant cells. The net effect is no increase in the fuel assembly lift force.

Therefore, only the weight change needs to be accounted for. As with Mark-BW fuel, the limiting lift scenario is at cold conditions. The holddown force margin is calculated for each core configuration to determine the minimum RCS temperature at which the fourth reactor coolant pump can be started. This change in fuel assembly weight will be generically included in these calculations.

10. *Please provide test data demonstrating that the absence of four fuel rods will not have an adverse effect on the spacer grid crush strength for seismic/LOCA loads.*

Grid impact tests were performed with the solid stainless steel fuel rods instead of the Zircaloy fuel rods to analyze the interaction between the fuel rods and a spacer grid during a seismic/LOCA event. The test results showed that solid stainless steel rods produce about the same grid strength as the Zircaloy fuel rod segments. These results suggest that impact failure of the grid occurs due to a column buckling type failure mechanism. This indicates that the grid strength is primarily a function of the grid component design and structure alone (i.e., strip interaction, weld nugget size, strip thickness, grid height, grid length, etc.) with little strength contribution due to fuel rods. Extrapolating from the results of the Zircaloy and stainless steel rod impact tests, the existence of vacancies (one per fuel assembly quadrant) is not anticipated to significantly affect the strength of the spacer grid.

11. *Please provide the details of analyses of the fuel assembly structural response to externally applied forces due to seismic/LOCA loading on a fuel assembly containing four vacant fuel rod positions. These analyses should consider the most conservative geometric locations for the vacant rods.*

Data from the grid impact tests discussed above was utilized to calculate a spacer grid stiffness applicable to the four vacant cell condition. As stated in the question 6 response, the data indicates that removing four rods (one per fuel assembly quadrant) decreases the spacer grid stiffness by [ ] without having a significant overall effect on reducing spacer grid impact strength and mode of failure. A decrease in spacer grid stiffness decreases the force in the spacer grid during a seismic/LOCA event. Consequently, existing calculations for the spacer grid impact loads based on an all Mark-B and Mark-BW fuel core are conservative for a reconstituted fuel assembly with four water holes.