CERTIFIED

DATE ISSUED: 7/12/79

ACRS SUBCOMMITTEE ON REACTOR FUEL

MAY 8, 1979

WASHINGTON, D.C.

ACRS-1636

On May 8, 1979, the ACRS Reactor Fuel Subcommittee met in Washington, D.C., to discuss various items concerning NRC actions on fuel-rated issues. Notice of this meeting appeared on the Federal Register, on Monday, April 23, 1979. There were no requests for oral or written statements from members of the public and none were made at the meeting.

Attachment A is a copy of the meeting agenda. The attendees list is Attachment B. Attachment C is the tentative schedule for the meeting. Selected slides and handouts are Attachment D to these minutes. A complete set of slides and handouts is attached to the office copy of these minutes.

OPEN SESSION (8:30 a.m. - 4:35 p.m.) INTRODUCTION

Dr. Shewmon, Subcommittee Chairman, called the meeting to order at 8:30 a.m. The Chairman explained the purpose of the meeting, and rules and procedures of conducting the meeting, pointing out that Dr. Thomas G. McCreless was the Designated Federal Employee in attendance.

Dr. Shewmon said that concerning the discussions of extended fuel burnup, he was interested in focusing on the NRC regulatory requirements and whether or not the extended burnup programs will supply the information required by NRC.

Dr. Shewmon introduced Dr. Ralph Meyer of the Reactor Fuel Section of the NRC Division of Systems Safety and he opened the day's presentations.

NRC FUEL LICENSING CRITERIA IN THE STANDARD REVIEW PLAN - R. MEYER (NRC-DSS)

Dr. Meyer described the new criteria applied to fuel designs, noting that these same criteria would be applied in the extended burnup range (>30,000 MWD/MTU).

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1372 196

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The objectives of the fuel system safety review are to assure that: (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences; (2) fuel system damage is never so severe as to prevent control rod insertion when required; (3) the number of fuel rod failures is not underestimated for postulated accidents; and (4) coolability is always maintained. Dr. Meyer discussed the origin of these four objectives (Figures D1-4).

Dr. Meyer defined fuel rod failures as a cladding defect that allows the escape of fission gas. Dr. Shewmon asked if given a Three Mile Island type transient, would high burnup cladding be more amenable to ballooning? Dr. Meyer replied that radiation effects saturate out around 5,000 MWD/MTU and no change in ballooning behavior is expected at high burnup. Dr. Bement (ACRS consultant) noted that rod ballooning also depends on the amount of fission gas in the rod, and it is possible that blowout-type failures may be seen because of the high internal pressures due to enhanced fission gas release. Dr. Meyer replied that fission gas release at high burnup is being studied by NRC.

Dr. Meyer reviewed the various damage mechanisms (Figures D5-7).

In response to a question from Dr. Bement, Dr. Meyer emphasized the need for transient testing of fuel rods exposed to high burnup in order to establish safety margins.

Dr. Meyer solicited the Subcommittee's comments on a proposal to recommend the vendors include segmented test rods in lead test assemblies for use by NRC in a test reactor (e.g. PBF) for tests on high burnup fuel.

Dr. Shewmon noted that fuel failures have been seen recently in the LaCrosse BWR and Yankee Rowe PWR. As a result of further discussion, it was decided that these two topics would be discussed in later presentations.

DEPARTMENT OF ENERGY (DOE) EXTENDED FUEL UTILIZATION PROGRAM - P. LANG (DOE)

Dr. Lang said that he would discuss the objectives of the DOE Fuel Utilization Program as well as detail the various projects, either underway or planned. Normally DOE uses the utilities as prime contractors in these projects, with the vendors acting as subcontractors.

Dr. Lang said that the objectives of the DOE Program are to increase uranium utilization, increase power plant productivity, and decrease radiation doses to workers. The principle focus of the program, however, is high burnup, since this item offers the greatest potential for uranium saving (10-20%) in the near term (by 1988). It was noted however, that there are other improvements that can be made to increase uranium utilization, such as lattice changes, spectrum shift, and end of cycle stretch-out, among others (Figure D-8).

Dr. Lang said preliminary estimates indicate that the optimum PWR burnup is in the range of 45 to 50,000 MWD/MTU; optimum BWR burnups appear to be in the range of 40 to 45,000 MWD/MTU.

Dr. Shewmon asked if the burnup were increased by approximately 50%, would the enrichment have to be increased by a similar amount. Dr. Lang dicated the enrichment increase would be slightly less than 50% in this case (Figures D9-10).

Technical issues identified by DOE that must be addressed for high burnup fuel include: pellet-clad interaction (PCI) fission gas release and fuel rod pressure, corrosion and hydriding, dimensional and structural changes, and in-core fuel management design.

Current DOE projects identified by Dr. Lang include a Consumers Power/Exxon/Battelle PNL project to demonstrate PCI remedies (annular pellets, annular pellets with graphite coated clad, packed particle fuel), a Commonwealth Edison Project designed to demonstrate PCI barrier and/or clad liner design, a Duke Power/Arkansas Power and Light Project to demonstrate 45,000 MWD/MTU PWR fuel, and a TVA/GE Project, still under negotiation, to improve uranium utilization (see Figures D11-14). Dr. Lang noted that it is DOE's intent to have five programs that parallel the Duke Power/Arkansas Power and Light/B&W Program that would involve the remaining four fuel vendors.

EXXON NUCLEAR PRESENTATION - K. WOODS

Before beginning his presentation, Dr. Woods responded to Dr. Shewmon's question concerning the fuel failures seen at Yankee Row. Dr. Woods noted that during a fuel outage, about two years ago, a plenum spring was found in the reactor. Subsequent examination did not reveal any damage to the fuel, and the reactor ran for another cycle. During a fuel outage last October, the fuel was again examined, following a modest increase in coolant activity. This examination revealed three assemblies suspected of damage. Two assemblies were subsequently identified as damaged, the damage being confined to the upper corner rods. The damage was a result of fretting. The cause of the fretting and failure is not clear at this time. Speculation centers on possible damage during loading of the fuel, or a cross-flow of coolant resulting from a leak in the core shroud that impinged coolant on the damaged rods. The fuel will be examined during the next outage to see if this problem is still occurring.

Exxon reviewed their fuel performance as of March 1, 1979 (Figure D-15). Over 50% of the Exxon fuel had exceeded 10,000 MWD/MT burnup.

Dr. Woods reviewed the special design features that Exxon believes will allow extention of fuel burnup. These design features include strict moisture control, short dished pellets, and thicker cladding. Exxon has lead test assemblies (LTA's) in Big Rock Point, Oyster Creek, and H.B. Robinson. Exxon is also participating in PCI-related development and test programs, which include programs sponsored by EPRI at Oyster Creek, as well as power ramp tests at the Studsvick reactor in Sweden (Figure D-16). Dr. Wood said that Exxon is not planning a core reload of high burnup fuel before 1980, at the earliest.

GENERAL ELECTRIC PRESENTATION - J. CHARNLEY

Ms. Charnley noted that General Electric is taking a cautious approach to extending fuel burnup. GE's prime consideration is to assure the fuel reliability (integrity) is maintained at high burnup.

1372 199

GE's extended exposure program encompasses three projects. These projects include: (1) a high burnup demonstration project with lead test assemblies at Monticello and Peach Bottom reactors; (2) a uranium utilization study which includes a DOE/TVA program now under negotiation, and (3) a barrier fuel program which consist of copper and zirconium liner designs to mitigate PCI. Four LTA's are under irradiation in Quad Cities Unit 1 to test various barrier designs. In response to questions from Drs. Shewmon and Bement concerning reconstitution/inversion of fuel bundles, Dr. Lang (DOE) indicated that inversion of BWR fuel is a long-range (approximately 20 years) uranium utilization option.

Ms. Charnley reviewed the details of the Monticello high burnup demonstration program (Figure D-17). The Monticello program will eventually take two assemblies to a peak pellet burnup of approximately 50,000 MWD/T. There will be both non-destructive and destructive examination of selected fuel rods during, and after the program's lifetime.

Major engineering considerations that GE has under study for high burnup include: fission gas release, zircaloy corrosion, PCI, assembly dimensional changes, and lack of data concerning the effect of these considerations at high burnups. Dr. Shewmon asked if stress corrosion cracking is more likely at high burnups. Dr. Meyer (NRC) noted that PCI failures, in general, show a burnup dependence, especially early in life.

WESTINGHOUSE PRESENTATION - D. BURNMAN

Mr. Burnman noted that Westinghouse is compiling an extensive data base on fuel with high burnup. W has exposed about 20,000 fuel rods to burnups in excess of 36,000 MWD/MTU (Figure D-19). Mr. Burnman said that no failures have been seen that could be attributed to burnup. In response to a question from Dr. Shewmon, Mr. Burnman said that most of the failures that have occurred were the result of hydride moisture failures, and collapsed cladding resulting from the use of low-density fuel. Mr. Burnman noted that the objective of the W high burnup fuel performance program is to provide sufficient and necessary fuel performance data to assure that no intrinsic materials performance limitations exist at high burnups (greater than 50,000 MWD/MTU-peak pellet). Westinghouse also wants to assure sufficient information exists, to satisfy requirements of design

1372 200

margins, technical reliability, and licensing concerns. Mr. Burnman noted that presently Westinghouse high burnup experience is greater for fuel rods than fuel assemblies. Key performance areas that require assessment include fission gas release and clad corrosion (as they affect fuel rods) zircaloy component integrity, and grid relaxation (as it effects fuel assemblies).

W has lead test assemblies in the Zion and Trojan reactors that are designed to address the potential performance problems noted above (Figure D-20). Dr. Shewmon asked if Westinghouse plans to subject any high burnup rods to power ramping. Mr. Burnman replied that Westinghouse has subjected rods with around 23,000 MWD/MTU burnup to power ramps and has under consideration a program to subject higher burnup rods to power ramps. Mr. Burnman also said that he does not believe PCI is subject to a burnup dependence.

BABCOCK AND WILCOX PRESENTATION - T. COLEMAN, J.WILLSE

B&W described their extended burnup program. This program consists of two subprograms; the first subprogram is a DOE/Duke Power Company/B&W program with the objective of qualifying the current design Mark-B assembly for a burnup of 40,000 MWD/MTU. Results from the first program will be fed into a second DOE/Arkansas Power and Light/B&W subprogram with the objective of developing a extended burnup fuel design for a burnup of 50,000 MTD/MTU. The second subprogram will also begin irradiation of lead test assemblies of the advanced burnup design. Figure D-21 shows a diagram of the program schedules. Both of the programs at the Oconee and Arkansas Nuclear 1 reactors will include post irradiation examination involving non-destructive and destructive tests in the B&W hot cell facility (Figures D22-23).

The series of fuel utilization studies is being conducted as the first task in the DOE/Arkansas Power and Light improved fuel design program. B&W showed an example of the expected benefits of an 18 month cycle with a batch average burnup of 45,800 MWD/MTU (Figure D-24). B&W is also conducting a parametic study to determine the sensitivity of beginning-of-life design variables on end-of-life parameters (Figure D-25).

1372 201

Mr. Wilise discussed the impact on high burnup fuel assembly design of the licensing requirements in Section 4.2 of the Standard Review Plan. B&W believes that their post irradiation examination program will provide the necessary data to answer licensing concerns regarding such parameters as: fission gas release, fuel assembly and fuel rod bow, densification, PCI, and others (Figures D-26). B&W will also rely on the use of their TACO-II fuel code for predictions of such parameters as fuel temperatures and pin pressures.

COMBUSTION ENGINEERING PRESENTATION - M. ANDREWS (CE)

Mr. Andrews stated that additional data is needed for the understanding of fuel behavior. Programs underway at DOE, NRC, industry organizations such as EPRI, and the lead test assembly programs in power reactors all contribute to this goal.

Mr. Andrews noted that fuel reliability has been continually improving, i.e. less fuel failures are being seen. It was noted that a true-life limit for zircaloy-clad UO_2 fuel rods has not yet been identified. CE believes that more data is required to prove that 45-50,000 MWD/MTU burnups are feasible.

Mr. Andrews discussed key technical areas that need to be addressed concerning high burnup. These items include PCI/SCC, fission gas release, dimensional stability of rods and assemblies, external clad corrosion, and burnable poison requirements. Commenting on the above topics, Mr. Andrews noted that KWU data show a very low PCI defect rate below the 12.5 kw/ft. power level. He also noted that fission gas release data show no enhancement with burnup out to 30,000 MWD/MTU. Mr. Andrews did note that there is a need to judiciously separate the effects of burnup and temperature on fission gas release rates. In response to a question from Mr. Mathis, Mr. Andrews noted that there are three burnable poison systems in use today. They are: (1) ceramic pellets of aluminum oxide containing B_4C, (2) borosilicate glass, and (3) gadolinia in UO_2.

1372 202

NRC-DSS PLANS FOR REVIEW OF HIGH-BURNUP FUEL - R. MEYER

Dr. Meyer said that DSS has begun to ask for additional information from applicants in the areas of on-line monitoring for fuel failures, and improved post-irradiation surveillance in the reporting of fuel inspections. Dr. Meyer suggested that NRC support routine fuel surveillance at all plants with a mandated "floor" on the required examination hardware available at the plant site.

The NRC review of fuel behavior analysis during postulated accidents was detailed (Figure D-27). Items of particular importance included fission gas release at high burnup, PCI as a fuel failure mechanism, RIA fuel damage limits, and fuel rod bowing. NRC has taken, or is taking, licensing action concerning fission gas release and rod bow (Figures D28-29). The RIA and PCI issues are under close scrutiny by the Staff.

Mr. Dean Houston (NRC-DSS) briefly discussed limits in the regulations, primarily environmental, that will have to be addressed before high-burnup fuel can be shipped, transported, stored, etc.

NRC-DOR LEAD TEST ASSEMBLY RELOAD SCHEDULE FOR HIGH-BURNUP AND RELATED DOR REVIEW PROBLEMS - F. COFFMAN (DOR)

Mr. Coffman discussed DOR's review of LTA's and high burnup reload applications. DOR has taken a tolerant attitude toward LTA insertion in operating reactors since, among other reasons, there are usually few assemblies involved and they are put in low-power regions of the core. Mr. Coffman noted that for transient analysis, consideration must be given to the fact that high-burnup assemblies are more sensitive to failures of reactivity control components at beginning-of-cycle. At end-of-cycle, the concern for high-burnup assemblies is that operating margins (e.g., CPR, DNBR) are reduced.

FUEL FAILURES AT VERMONT YANKEE AND CONNECTICUT YANKEE - J. CHARNLEY (GE), R. LOBELLL (NRC-DOR)

Ms. Charnley discussed the recent fuel failures seen at Vermont Yankee. This is the first case of widescale failure of the GE 8 x 8 fuel. A total of 29 bundles have been identified as containing failed fuel. Figure D-30 details a chronology of the problem. The failure mechanism appears to be spalling of

1372 203

excessive exterior clad oxide. Most of the failures occurred on gadolina bearing rods. Preliminary investigations have not uncovered any obvious failure mechanism(s), nor has this failure method been observed elsewhere. Hot cell examinations of selected fuel assemblies, both non-destructive and destructive, are planned. Vermont Yankee is currently back in operation, with no indication of additional fuel failures.

Mr. Lobel (NRC-DOR) discussed the fuel failures observed at Connecticut Yankee (W PWR-1925 MW(t)-stainless steel clad). High offgas activity lead the vendor to suspect failure of high-burnup fuel. During a recent refueling, 36 of 48 Batch 8 assemblies, which were scheduled to be discharged, were found to be leaking. The vendor also sipped selected assemblies of the remaining batches in the core and all were found to be sound. Figure D-31 shows the location of the failed assemblies in the core.

Investigations by the licensee showed that Batch 8 fuel was unique in two respects: (1) it was the only fuel in the core fabricated by British Nuclear Fuel Limited; (2) it was the only batch to be subjected to a two-step process during fuel rod fabrication (Figure D-32). The applicant has characterized the failures as stress rupture type failures with no evidence of corrosion. Mr. Lobel noted that, as of this date, the utility has no plans for hot cell examination of the failed fuel. Dr. Bement questioned the utility's conclusion that failure was by stress rupture. Mr. Crocker felt that hot cell examination of the failed fuel was necessary for an accurate determination of the failure mechanism. Dr. Mark was of the opinion that the fuel itself was suspect.

NEW (PROPOSED) NRC EXCEPTANCE CRITERIA FOR FUEL ASSEMBLY STRUCTURAL RESPONSE TO SEISMIC AND LOCA LOADS - R. MEYER (DSS)

Mr. Meyer began by reviewing the history of the bases for combining seismic and LOCA loads. He noted that the precedent was set with North Anna, upon the discovery of the asymmetric blowdown load problem. Criteria developed at North Anna was applied to Diablo Canyon with some modification (Figure D-33).

Dr. Meyer noted, however, that with the San Onofre plant review, currently in progress, NRC, in demanding a collapsed grid assumption, would provide an incentive to use an inferior grid, rather than the strong grid now scheduled for use in San Onofre.

Dr. Meyer reviewed the proposed method for analysis of SSE and LOCA loads (Figure D-34). In response to question from Dr. Mark, Mr. Kniel (NRC-DSS) stated that the analyses for the seismic and LOCA loads are done separately.

DSS is proposing not to combine SSE and LOCA loads. DSS's reasoning for not combining these loads centers on the belief that structural failure of fuel during a SSE cannot lead to a LOCA, so there is no need to combine these loads (SSE and LOCA) for the fuel. Dr. Mark and Mr. Etherington pointed out that an SSE may induce a LOCA however, and these load may have to be dealt with in tandem. Mr. Kniel noted that there has never been a concise statement from the NRC that an earthquake will cause a LOCA. The NRC position has been to assure that a LOCA will not occur, given an earthquake. Dr. Mark noted that the Japanese do not combine SSE and LOCA loads. Dr. Meyer noted that he was informed of this fact upon a recent visit to Japan.

Dr. Meyer summarized the proposed criteria as follows: (1) use standard conservative methods, (2) add margin only for identified shortcomings, (3) use no safety factor, (4) do not combine loads, (5) use average strength values for grids, and (6) use conservative bounding strength values for other components.

Concerning the schedule of the proposed revision to the Standard Review Plan, Dr. Meyer said that a draft of the revision, and a value/impact study, is scheduled for completion on May 1. A 60 day public comment period would begin around August 1 of this year. Since R^3C is not currently scheduling Standard Review Plan actions, issuance of the revision is uncertain. In response to a question from Dr. McCreless (ACRS Staff), Mr. Meyer noted that it is a new procedure for a revision to the Standard Review Plan to be subjected to public comment.

The meeting was adjourned at 4:35 p.m.

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NOTE: A copy of a transcript of this meeting is available in the NRC Public Document Room at 1717 H St., N.W., or can be obtained from Ace-Federal Reporters, 444 North Capital St., N.W., Washington, D.C.

1372 206

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee on Reactor Fuel; Meeting

The ACRS Subcommittee on Reactor Fuel will hold an open meeting on May 8, 1979, in Room 1046, 1717 H Street, N.W., Washington, D.C. 20555 to discuss various items concerning NRC actions on fuel-related issues. Notice of this meeting was published March 23, 1979 [44 FR 17837].

In accordance with the procedures outlined in the **Federal Register** on October 4, 1978, (43 FR 45926), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: *Tuesday, May 8, 1979, 8:30 a.m. until the conclusion of business.*

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting and to formulate a report and recommendations to the full Committee.

At the conclusion of the Executive Session, the Subcommittee will hold discussions with representatives of the NRC Staff, and their consultants, pertinent to this review. The Subcommittee may then caucus to determine whether the matters identified in the initial session have been adequately covered and whether the subject is ready for review by the full Committee.

Further information regarding topcis to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the Designated Federal Employee for this meeting, Dr. Thomas G. McCreless (telephone 202/634-3267), between 8:15 a.m. and 5:00 p.m., EST.

ATTACHMENT A

1372 207

TENTATIVE SCHEDULE OF PRESENTATIONS
ACRS REACTOR FUEL SUBCOMMITTEE MEETING
MAY 8, 1979
WASHINGTON, D.C.

		PRESENTATION TIME	ACTUAL TIME
I.	INTRODUCTION	10 min	8:30 am
	P. SHEWMON, CHAIRMAN		
II.	NRC FUEL LICENSING CRITERIA IN THE STANDARD REVIEW PLAN	30 min	8:45 am
	R. MEYER, NRC-DSS		
III.	EXTENDED FUEL BURNUP PRESENTATIONS		
	A. DOE PRESENTATION	45 min	9:30 am
	P. LANG		
	- BREAK -	10 min	10:50 am
	B. FUEL VENDORS PRESENTATIONS		
	1. EXXON PRESENTATION	20 min	11:00 am
	J. OWSLEY AND K. WOODS		
	2. GE PRESENTATION	20 min	11:30 am
	J. CHARNLEY		
	3. WESTINGHOUSE PRESENTATION	20 min	12:00 noon
	D. BURMAN		
	4. BABCOCK AND WILCOX PRESENTATION	20 min	12:30 pm
	T. COLEMAN		
	5. COMBUSTION ENGINEERING PRESENTATION	30 min	1:00 pm
	M. ANDREWS		
	- LUNCH -		1:45 pm

1372 208

ATTACHMENT C

- CONTINUATION OF -

TENTATIVE SCHEDULE OF PRESENTATIONS
ACRS REACTOR FUEL SUBCOMMITTEE MEETING
MAY 8, 1979
WASHINGTON, D.C.

	PRESENTATION TIME	ACTUAL TIME
C. NRC PRESENTATIONS		
1. DSS PLANS FOR REVIEW OF HIGH-BURNUP FUEL R. Meyer - DSS	15min	2:45 pm
2. DOR - LEAD TEST ASSEMBLY RELOAD SCHEDULE FOR HIGH-BURNUP AND RELATED DOR REVIEW PROBLEMS F. COFFMAN - DOR	15 min	3:10 pm
IV. NRC DOR AND DSS LICENSING ACTIVITIES		
A. FUEL FAILURES AT CONN. YANKEE AND VERMONT YANNKEE R. LABEL - DOR	30 min	3:30 pm
B. NEW (PROPOSED) ACCEPTANCE CRITERIA FOR FUEL ASSEMBLY STRUCTURAL RESPONSE TO SEISMIC AND LOCA LOADS R. MEYER - DSS	30 min	4:15 pm
V. ADJOURN		5:00 pm

ACRS SUBCOMMITTEE ON REACTOR FUEL
MAY 8, 1979
WASHINGTON, D.C.

ATTENDEES LIST

ACRS

P. Shewmon, Chairman
H. Etherington, Member
C. Mark, Member
W. Mathis, Member
J. Crocker, Consultant
A. Bement, Consultant
T. McCreless, Staff*
P. Boehnert, Staff
D. Bessette, Fellow

*Designated Federal Employee

GENERAL ELECTRIC

J. Charnley
C. Richard

EXXON NUCLEAR

G. Owsley
K. Woods

NUSCO

M. Pitek

B&W

T. Coleman
J. Willse
J. Tulenko
E. Coppola

DUKE POWER

R. Snipes
G. Swindlehurst

VEPCO

N. P. Wolfhope

EMBASSY OF JAPAN

A. Yuki
A. Morishima

NRC

M. Tokar
D. Powers
J. Voglewede
R. Meyer
M. Houston
S. Kim
K. Kniel
S. Sands
R. Lobel

WESTINGHOUSE

G. Antaki
D. Burman
J. McInerney

INST. OF RADIATION PROTECTION

L. Willberg

THE TOKYO ELECTRIC POWER CO

H. Hamada

YANKEE ATOMIC ELECTRIC CO

W. Metevia

CE

M. Andrews
M. Marugg
C. Brinkman

EG&G

J. Crocker

NUCLEAR ASSOCIATES INTERNL.

D. Coleman

RAINBOW FAMILY OF LIVING LIGH

M. Williamson

1372 210

ATTACHMENT B

1. The Fuel System is Not Damaged as a Result of Normal Operation and Anticipated Operational Occurrences

10 CFR 50 Appendix A

II. Protection of Multiple Fission Product Barriers

Criterion 10 -- Reactor Design

. . . Assure That Specified Acceptable Fuel Design Limits are Not Exceeded . . .

FIGURE D-1

1372 211

2. Fuel System Damage is Never so Severe as to Prevent Control Rod Insertion When It is Required

10 CFR 50 Appendix A

III. Protection and Reactivity Control Systems

Criterion 26 – Reactivity Control Systems Redundancy and Capability

. . . Reliably Controlling Reactivity Changes . . . Under Conditions of Normal Operation Including Anticipated Operational Occurrences

Criterion 27 – Combined Reactivity Control Systems Capability

. . . Reliably Controlling Reactivity Changes . . . Under Postulated Accident Conditions . . .

1372 212

D-2

3. The Number of Fuel Rod Failures is Not Underestimated for Postulated Accidents

10 CFR 50 Part 100.11

The Fission Product Release . . . Should . . . Result in Potential Hazards Not Exceeded by Those From Any Accident Considered Credible.

1372 213

0-3

4. Coolability is Always Maintained

10 CFR 50 Part 46

Paragraph (b) (4) Coolable Geometry

Calculated Changes in Core Geometry Shall be Such That the Core Remains Amenable to Cooling.

1. FUEL SYSTEM DAMAGE

a. Stress, Strain and Loading Limits

b. Strain Fatigue

c. Fretting

d. Oxidation and Hydriding

e. Rod Bowing and Irradiation Growth

f. Rod Pressure

g. Hydraulic Loading

h. Control Rod Reactivity

1372 215

O-5

2. FUEL ROD FAILURE

a. Overheating
b. Pellet/Cladding Interaction (PCI)
c. Hydriding
d. Cladding Collapse
e. Bursting
f. Mechanical Fracturing
g. Fretting

1372 216

0-6

3. FUEL COOLABILITY

a. Cladding Embrittlement
b. Expulsion of Fuel
c. Cladding Melting
d. Structural Deformation
e. Fuel Rod Ballooning

SUMMARY OF LWR IMPROVEMENTS

	POTENTIAL FOR URANIUM SAVING	TIMELINESS OF INTRODUCTION	REQUIREMENTS FOR BACK FIT
INCREASED BURNUP	10–20%	NEAR	NONE/FUEL REDESIGN
LATTICE CHANGES	$\sim$4%	NEAR/MEDIUM	FUEL REDEDSIGN
SPECTRUM SHIFT	MEDIUM	MEDIUM/LONG	FUEL AND/OR PLANT CHANGES
SPATIAL VARIATION OF ENRICHMENT	LOW/MEDIUM	NEAR/MEDIUM	NONE/FUEL REDESIGN
FULL USE OF STARTUP CORE	$\sim$1%	NEAR	NONE
IMPROVED FUEL MANAGEMENT AND CONTROL DESIGNS	MEDIUM	NEAR/MEDIUM	NONE/FUEL REDESIGN
END OF CYCLE STRETCHOUT	LOW	NEAR	NONE
RECONSTITUTION/ INVERSION OF BWR FUEL	$\sim$4%	NEAR/MEDIUM	NONE/FUEL REDESIGN

LOW: 0–2%
MEDIUM: 2–10%

NEAR: BY 1988
MEDIUM: 1988–2000
LONG: AFTER 2000

1372 218

0-8

HIGH BURNUP FUEL BENEFITS—BWR
12 MONTH CYCLE
1270 MWe AT 75% CAPACITY

	STANDARD	IMPROVED	PERCENT CHANGE
BURNUP (MWD/MT)	28,400	47,000	+65
ANNUAL RELOAD BATCH	212 ASSEMBLIES 1/4 OF CORE	122 ASSEMBLIES 1/7 OF CORE	-42
FEED ENRICHMENT	2.77%	3.81%	-
ANNUAL SWU REQUIREMENTS	141,000	138,400	-2
ANNUAL U_3O_8 REQUIREMENTS (MT)	219	188	-14
ANNUAL TOTAL PLUTONIUM CONTENT IN SPENT FUEL (KG)	312	241	-23

25 → 30

0-9

HIGH BURNUP FUEL BENEFITS—PWR
12 MONTH CYCLE
1270 MWe AT 75% CAPACITY

	STANDARD	IMPROVED	PERCENT CHANGE
BURNUP (MWD/MT)	30,400	50,600	+66
ANNUAL RELOAD BATCH	80 ASSEMBLIES 1/3 OF CORE	48 ASSEMBLIES 1/5 OF CORE	-40
FEED ENRICHMENT	3.0%	4.3%	—
ANNUAL SWU REQUIREMENTS	150,000	148,500	-1
ANNUAL U_3O_8 REQUIREMENTS (MT)	227	197	-13
ANNUAL TOTAL PLUTONIUM CONTENT IN SPENT FUEL (KG)	311	234	-25

1372 220

0-10

CONSUMERS POWER PROJECT

SCOPE

- TO DEVELOP AND DEMONSTRATE 3 CANDIDATE REMEDIES FOR PCI, TO FACILITATE HIGHER BURNUP
  - ANNULAR PELLETS
  - ANNULAR PELLETS WITH GRAPHITE COATED CLAD
  - PACKED PARTICLE FUEL
- REACTOR IRRADIATION IN BIG ROCK POINT
- HALDEN REACTOR SCREENING AND RAMP TESTS

PARTICIPANTS

- CONSUMERS POWER COMPANY (PRIME CONTRACTOR)
- EXXON NUCLEAR CO. (SUBCONTRACTOR)
- BATTELLE NORTHWEST LABS (SUBCONTRACTOR)

COMMONWEALTH EDISON PROJECT

SCOPE

- TO PLAN AND CONDUCT A LARGE SCALE DEMONSTRATION OF BARRIER AND/OR LINER FUEL DESIGNS, AS REMEDY FOR PCI, TO FACILITATE HIGHER BURNUP
- THREE PRINCIPAL TASKS:
  - DESIGN OF DEMONSTRATION (132 ASSEMBLIES)
  - SUPPORTING TEST REACTOR AND LABORATORY INVESTIGATIONS
  - FOUR LEAD TEST ASSEMBLIES:
    - 1 ASSEMBLY OF CU BARRIER FUEL
    - 1 ASSEMBLY OF CU BARRIER FUEL WITH OXIDE COATING
    - 1 ASSEMBLY OF PURE ZIRCONIUM LINER FUEL
    - 1 ASSEMBLY OF SPONGE ZIRCONIUM LINER FUEL
- DEMONSTRATION REACTORS–QUAD CITIES 1 AND 2

PARTICIPANTS

COMMONWEALTH RESEARCH CORP. (PRIME CONTRACTOR)

GENERAL ELECTRIC COMPANY (SUBCONTRACTOR)

DUKE POWER/ARKANSAS POWER & LIGHT PROJECTS

SCOPE

- TO DEVELOP AND DEMONSTRATE 45,000 MWD/T PWR FUEL
- IRRADIATION OF CURRENT FUEL TO 41,000 MWD/T IN OCONEE–1, WITH EXAMINATION AFTER EACH CYCLE
- DESIGN AND LEAD TEST ASSEMBLY IRRADIATION OF 45,000 MWD/T FUEL IN ANO–1, WITH EXAMINATIONS
- FULL RELOAD BATCH DEMONSTRATION OF 45,000 MWD/T FUEL IN ANO–1

PARTICIPANTS

- DUKE POWER COMPANY (PRIME CONTRACTOR–OCONEE)
- ARKANSAS POWER AND LIGHT (PRIME CONTRACTOR–ANO–1)
- BABCOCK AND WILCOX (SUBCONTRACTOR–BOTH)

1372 223

TVA-GE PROJECT TO IMPROVE BWR URANIUM UTILIZATION

POTENTIAL DESIGN IMPROVEMENT PHASE I PRELIMINARY ALTERNATIVES	EARLIEST IMPLEMENTATION DATE
FOUR BUNDLE ENRICHMENT INITIAL CORE	1983
REDUCED GD RESIDUAL IN RELOADS (AXIAL SHAPING)	1981
BURNUP OPTIMIZATION (INCREASED BURNUP)	1982
REFUELING PATTERN OPTIMIZATION (INCLUDING REINSERT)	1981
CONTROL ROD PATTERN OPTIMIZATION	1981
AXIAL BLANKET OPTIMIZATION	1981
AXIAL ENRICHMENT ZONING	1982
FUEL BUNDLE RECONSTITUTION, REINSERTION	1981

EXXON NUCLEAR FUEL PERFORMANCE HIGHLIGHTS

IRRADIATION PERFORMANCE SUMMARY AS OF MARCH 1, 1979

	Total Assemblies	Maximum Assembly Exposure MWD/MT
IN-CORE FUEL		
BWR Assemblies	1,139	30,400
Big Rock Point	64	30,400
Oyster Creek	560	23,000
KRB	183	22,388
Humboldt Bay	122	15,900
Kahl	18	12,000
Dresden-1	66	5,200
Oskarshamn-1	94	11,500
LaCrosse	32	5,000
PWR Assemblies	596	31,700
H.B. Robinson	157	31,700
Yankee Rowe	76	18,350
Palisades	136	25,490
D.C. Cook	129	21,200
Biblis A	57	13,700
R.E. Ginna	32	8,265
LOFT	19	130
DISCHARGED FUEL		
BWR Assemblies	134	27,900
Big Rock Point	4	21,514
Humboldt Bay	2	22,700
Oyster Creek	128	27,900
PWR Assemblies	110	31,520
R.E. Ginna	2	25,400
Palisades	68	14,212
Yankee Rowe	40	31,520
TOTAL IRRADIATED FUEL ASSEMBLIES	1,979	

PCI RELATED DEVELOPMENT AND TEST PROGRAMS

- Power shape monitoring at Oyster Creek — sponsored by EPRI
  - — Quantitative on line monitoring of ramp rate expected in all fuel types
  - — Correlation of any fuel failure (or lack of it) with experienced ramp rates and characterized physical data
- Ramp rates to failure at Studsvik — sponsored by consortium including ENC
- Fuel performance improvement program ramp
  - — Tests to failure at Halden
  - — Test rods at Big Rock aimed at increasing resistance to power cycling

Consumers power prime contractor to Department of Energy ENC and Battelle Northwest Laboratories subcontractors

10-76

PROPOSED
MONTICELLO PROGRAM

FOUR LEAD BURNUP BUNDLES EXTENDED ONE CYCLE

- o BEGINING-OF-CYCLE 7 BURNUP
  - 23600 MWD/STU BUNDLE AVERAGE
  - 31300 MWD/STU PEAK PELLET
- o INCLUDES ONE 8X8 SURVEILLANCE BUNDLE

DISCHARGE ONE BUNDLE (~40 GWD/STU) AT END-OF-CYCLE 7

- o EXTEND OTHER THREE WITH FIVE LOWER BURNUP BUNDLES
- o VISUAL EXAM OF EXTENDED RODS
- o NONDESTRUCTIVE AND DESTRUCTIVE BUNDLE EXAMINATIONS
- o OTHER ASSEMBLY COMPONENT EXAMINATIONS
- o MONITOR AND STORE OPERATING HISTORY DATA

DISCHARGE ONE LEAD BUNDLE (~45 BWD/STU) AT END-OF-CYCLE 8

- o DISCHARGE THREE FOLLOW-ON BUNDLES
- o EXTEND TWO LEAD AND TWO FOLLOW-ON BUNDLES
- o VISUAL EXAMS AND GAMMA SCAN OF EXTENDED FUEL
- o MONITOR/STORE OPERATING HISTORY

DISCHARGE REMAINING BUNDLES AT END-OF-CYCLE 9

- o PEAK PELLET EXPOSURE ~50 GWD/STU
- o NONDESTRUCTIVE AND DESTRUCTIVE EXAMINATIONS
- o EXAMINE OTHER ASSEMBLY COMPONENTS

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5/8/79

1372 227

D-17

MAJOR ENGINEERING CONSIDERATIONS

FISSION GAS RELEASE

- EXTEND DATA BASE/VERIFY PREDICTIONS
- POTENTIAL DESIGN IMPACT

ZIRCALOY CORROSION

- EXTEND OPERATING DATA
- POTENTIAL MATERIAL/OPERATION FEEDBACK

PELLET-CLAD INTERACTION

- KNOWN ACTIVE FAILURE MECHANISM
- EXTEND EXPERIENCE
- REFLECT BARRIER PROGRAM RESULTS

ASSEMBLY DIMENSIONAL CHANGES

- INCONEL RELAXATION
- CHANNEL LIFETIME COMPATIBILITY

NUCLEAR CHARACTERISTICS

- VERIFY MODEL PREDICTIONS
- GADOLINIA ENRICHMENT/RESIDUAL CONSIDERATIONS

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5/8/79

1372 228 D-18

HIGH BURNUP FUEL DISCHARGED

Plant	Reg.	Cycles of Exposure	Discharge Cycle-Date	No. of F/A	Average Burnup	Max. Rod Burnup	I-131 (μCi/cm)
Point Beach 2	2	2	2-2/76	33	31,900	35,200	0.02
	3	3	3-2/77	37	36,140	42,800	0.004
Zorita	5	4	6-3/77	3	34,000	37,000	0.01
Beznau 1	4	4	5-5/76	4	34,500	37,000	0.03
	4A	4	6-5/77	1	35,000	36,600	0.02
	4U2	4	6-5/77	4	33,200	36,400	0.02
	5	4	6-5/77	8	31,200	33,500	0.02
Point Beach 1	3U2	4	5-10/77	2	36,400	40,000	0.04
	4	3	5-10/77	27	33,180	40,000	0.04
	5	6	6-10/78	8	32,600	37,300	0.025
	6	6	6-10/78	13	37,500	40,400	0.025
Ginna	5	4	6-4/77	20	31,000	35,000	0.20
Turkey Point 3	3	3	3-11/76	52	29,600	34,600	0.03
Turkey Point 4	3	3	3-11/77	48	28,900	34,300	0.02
	2	4	4-8/78	16	30,080	35,279	0.008
Prairie Island 1	2	3	3-4/78	1	36,500	40,900	0.0001
	3	3	3-4/78	40	34,380	40,000	0.0001
Kewaunee	3	3	3-5/78	40	33,500	40,400	0.002
	3	3	3-9/78	64	36,000	42,000	0.03
Zion 1	2	3	3-9/78	5	36,400	37,500	0.03
Zion 2	3	3	2-15-79	64	36,900	42,700	0.01

OPPORTUNITIES FOR EXTENDED BURNUP ON WESTINGHOUSE FUEL

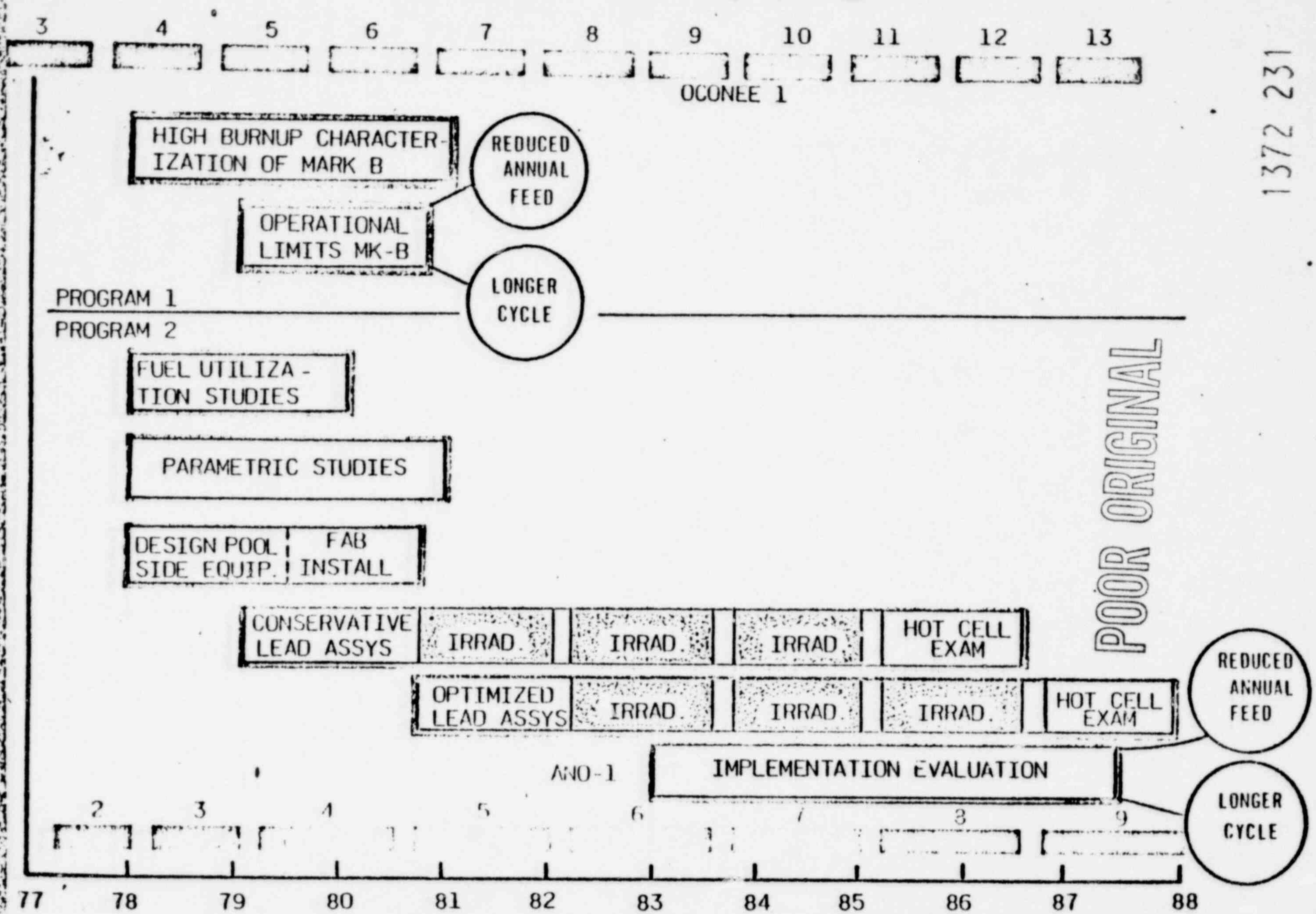
Year

Item	1977	1978	1979	1980	1981	1982	1983	1984
15x15 Assemblies			39,000					
Zion No. 1	Cycle-2	Cycle-3		47,000	55,000			
Zion Unit 2			Cycle-4	Cycle-5				
Surry No. 2	Cycle-3	Cycle-4	28,000	41,000				
17x17 assemblies			S.G. Repair	Cycle 5				
Trojan		Seismic Issue	Cycle-2	21,000 Cycle-3	31,000 Cycle-4	40,000 Cycle 5	48,000	

1372 230

0-20

Extended burnup program



POOR ORIGINAL

1372 231

2-21

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Non-destructive

- Visual examination
- Gross gamma scans
- Assembly length and grid locations
- Water channel measurements
- Line scan profilometry
- Holddown spring load-deflection characteristics
- Surface deposit characterization
- Spacer grid load-deflection characteristics

POOR ORIGINAL

1372 232

D-22

PIE/destructive test phase

- Rod lengths; pulling forces
- Clad profilometry, thickness
- Clad defect scans, gamma-scans
- Burnup, fission gas analysis
- Crud sampling and analysis
- Pellet density
- Clad mechanical properties
- Metallography

0-23

High burnup fuel benefits-PWR

	BASE CASE	IMPROVED	ANNUALIZED	PERCENT CHANGE
	12 MONTH CYCLE 80% CF OUT-IN SHUFFLE	18 MONTH CYCLE 84% CF IN-OUT SHUFFLE		
BATCH BURNUP (MWd/MTu)	27,000	45,800		+ 70
RELOAD BATCH	60 ASSY'S. ~ 1/3 OF CORE	60 ASSY'S. ~ 1/3 OF CORE	40 ASSY'S.	- 30
FEED ENRICHMENT	2.8	4.1		
SWU REQUIREMENTS	106,000	190,000	112,000	+ 5
U_3O_8 REQUIREMENTS (STU_3O_8)	183	279	164	- 11
UTILIZATION (MWY/STU_3O_8)	11.2	12.5	12.5	+ 12

1372 234

O-24

Parametric studies of design variables at high burnups

	BOL parameters		EOL parameters
Determine effects of	Cladding thickness	On	Pin pressure
	Degree of cold work		Fuel temperatures
	Pre-pressurization level		Creep collapse
	Fuel density		Stress and strain levels
	Pellet dimensions		Fatigue
	Gap size		Pellet swelling
	Plenum volume		Enrichment
	Grid		Rod bow
			Ovality
			Fretting
			Hold down springs

Identify key design constraints on extended burnup assembly design

B&W
W1L25B

Application of PIE data

Test parameter	Primary application					
	Fuel ass'y lift	Rod bow	Creep collapse	Pellet cladding int'ract'n	Ass'y bow, twist	Densi-fication
Rod ΔL/L		X		X		
Ass'y ΔL/L	(X)	X			X	
Fuel stack length			X	X		X
Holddown force	(X)				X	
Rod ΔD/D		X	X	X		X
Rod bow		X			X	
Rod to rod gap		(X)				
Crud	X		X	X		
Rod pull force		X				
Visuals	X	(X)	(X)	X	(X)	
Eddy current				X		
Rod profilometry			X	X		
Fission gas pressure		X	(X)	X		X
Pellet density			X	X		(X)
Clad tensile properties				(X)		
Rod cross sections				X		X

(X) Important data supporting licensing or design

X Data contributing to code verification, licensing, etc.

1372 236

ANALYSIS OF FUEL BEHAVIOR AT HIGH BURNUP

Background

Current Restrictions on Mixed-Oxide Utilization and Recycle Technology Have Placed Increased Emphasis on Extended Burnups.

Regulatory Interest in Fuel Behavior at High Burnup Includes:

1. Enhanced Fission Gas Release
2. Increased Potential for Fuel Failure by PCI
3. Rod Bowing
4. Material Property Changes
5. Cladding Collapse
6. Cladding Axial Growth
7. Fretting and Wear
8. Fatigue
9. Oxidation and Crud Deposition
10. Relaxation of Springs

0-27

ANALYSIS OF FUEL BEHAVIOR AT HIGH BURNUP

Status of Fission Gas Release:

1. All Vendors Asked to Revise Codes by February 1979.
2. Westinghouse, General Electric, and Babcock & Wilcox Have Developed Improved Codes.
3. The Combustion Engineering and Exxon Codes With the NRC Fission Gas Model (Nureg-0418) will be Given Additional NRC Review.
4. The ANS-5.4 Model is Being Incorporated in the NRC FRAPCON Code.
5. An NRC High-Burnup Standard Problem Set will be Available Soon.

FUEL ROD BOWING

Background

1. Bowed Fuel Rods Alter Local Heat Transfer and Neutronics.
2. Early Vendor Analytical Models Were Inconsistent and Inaccurate.
3. Conservative Licensing Models Developed by NRC in 1976 Give Excessive Heat-Transfer Penalties.

Status

1. NRC Letters to Vendors in Summer of 1978 Agreed to Accept More Realistic Models and Presented Guidelines.
2. New Rod Bow Models are Expected from Vendors During Next 2 Years.

1372 239

D-27

VERMONT YANKEE

OPERATING HISTORY

OFF-GAS ACTIVITY INCREASE	MAY 1978
SEVENTEEN LEAKING RODS FOUND	SEPT. 1978
PLANT RESTART, LOW OFF-GAS	OCT. 1978
OFF-GAS ACTIVITY INCREASED	NOV. 1978
REDUCED POWER DUE TO OFF-GAS	FEB. 1979
SHUTDOWN FOR SIPPING (24 LEAKING BUNDLES)	MAR. 1979
RESTART WITH REPLACEMENT FUEL	APRIL 1979
OFF-GAS ACTIVITY LOW AND STEADY	MAY 1979

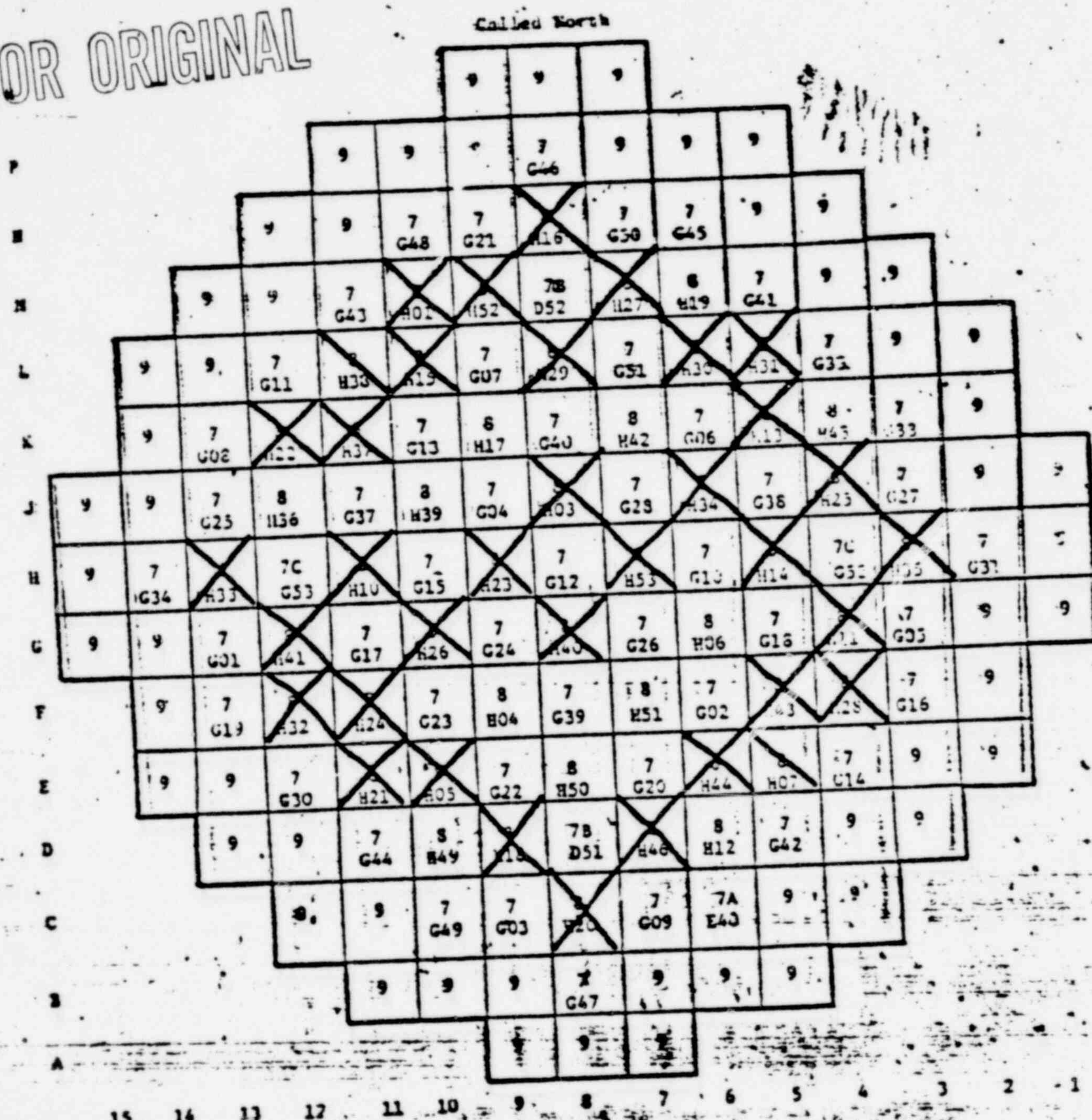
JSC:BJW/1024 5/8/79

1372 240

D-30

POOR ORIGINAL

Figure 3-1. Connecticut Yankee Cycle 7 Loading Diagram



	Batch No.	Initial enrichment, wt % ^{235}U
X — Batch No.	7, 7A	4.0
XXX — Fuel Assembly Serial No.	7B, 7C	3.67
	8, 9	4.0

0-31

1372 241

POOR ORIGINAL

CONNECTICUT YANKEE
FUEL SUPPLIER COMPARISONS

PARAMETER	RELOAD BATCH			
	10	9	8	7
FUEL PELLET SUPPLIER	B&W	B&W	BNFL	GUNF
FUEL ROD FABRICATION	B&W	B&W	GUNF/ B&W	GUNF
CLADDING SUPPLIER	SUPERIOR TUBE COMPANY			
AVG BURNUP (TO DATE) (GWD/MTU)	10	24.2	33.8	33.5

1372 242

0-32

THE DIABLO CANYON ASSUMPTION

-

1. PERIPHERAL BUNDLES COULD NOT MEET NORTH ANNA PRECEDENT.

2. ASSUMED GRIDS IN PERIPHERAL BUNDLES WERE FULLY COLLAPSED.

3. FULLY COLLAPSED GRIDS RESULTED IN ONLY 25°F INCREASE IN PCT.

4. LOW POWER IN PERIPHERAL BUNDLES REDUCED PCT BY MORE THAN 25°F THUS COMPENSATING FOR GRID DEFORMATION.

1372 243

ANALYSIS OF LOADS

A. INPUT: USE OUTPUT OF PRIMARY SYSTEM ANALYSIS THAT PRODUCES LARGEST FUEL LOADS.

B. METHODS: REVIEW INCLUDES SAMPLE PROBLEM.

C. UNCERTAINTY ALLOWANCE: ADD MARGIN FOR STEAM-FLASHING AND PRONOUNCED SENSITIVITY TO INPUT VARIATIONS.

D. AUDIT: NRC WILL AUDIT ANALYSIS OF CURRENT PLANT DESIGN WITH INDEPENDENT CODE.

E. COMBINATION OF LOADS: NONE.

proposing loads, not
be combined

1372 244