UNITED STATES
NUCLEAR REGULATORY COMMISSION
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MEMORANDUM FOR: Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Robert B. Minogue, Director
Office of Standards Development

FROM: Saul Levine, Director
Office of Nuclear Regulatory Research

(?)

SUBJECT: RESEARCH INFORMATION LETTER #17 - POWER BURST FACILITY (PBF) SINGLE ROD POWER-COOLING MISMATCH (PCM) TEST RESULTS

INTRODUCTION

This memorandum transmits the results of completed research on single fuel elements exposed to power-cooling mismatch (PCM) conditions in the Power Burst Facility (PBF). It is offered for your information and use in determining possible changes in required departure-from-nucleate-boiling ratios (DNBR's) for all commercial power reactors which use zircaloy-clad uranium dioxide fuel rods.

The research results show that zircaloy fuel rod cladding normally does not fail even when prolonged film boiling occurs as a result of inadequate coolant flow rates. The cladding generally will not fail unless it becomes so heavily oxidized that it is brittle at room temperature. Such severe zircaloy oxidation would require higher cladding temperatures than are currently predicted for any light water reactor accidents which result in a PCM, whether related to a loss of coolant flow or to an increase in fuel rod power.[1]

DISCUSSION

Thirteen PCM reactor tests were performed using pressurized water reactor (PWR) type fuel pins with an active fuel length of 0.93 meters to match the PBF core length. The length of the test fuel was appreciably shorter than that of the 3.66 meter fuel typically used in commercial reactors, but fuel plenum chambers were scaled proportionately, and combinations of

[1]Memorandum of August 15, 1977, P. S. Check to T. H. Novak, "Reactor Fuels Input to ATWS Report."

test rod shroud diameter, coolant flow rate and rod power were selected to control the cladding surface heat transfer coefficients, the local coolant enthalpies and the extent of rod surfaces involved in departure from nucleate boiling (DNB) within appropriate test ranges.

Coolant flow rates in the vicinity of each rod were monitored with turbine flowmeters, and most test fuel rods were also fitted with fuel centerline thermocouples, cladding surface thermocouples, plenum pressure transducers and fuel rod length sensors. The fuel rods were tested in the reactor either singly or in groups of four, in order to permit a quick, direct comparison of the effects of programmed differences in test rod parameters. The initial part of each PCM test consisted of a nominal 24-hour fuel preconditioning period during which typical commercial fuel rod power levels were maintained, to develop typical fuel pellet crack structures and fuel-to-cladding gaps, so that appropriate stored energy profiles in the fuel could be developed during the PCM portions of the test.

In the final phase of each test, fuel rod power levels were increased to levels equal to or greater than those predicted for very severe "anticipated-transient-without-scram" (ATWS) type accidents. The test rods were then taken into DNB either by further increasing the fuel rod power or by reducing the coolant flow rate or both. Peak axial fuel rod power values were 55KW/meter to 76KW/meter.

Two series of tests were performed. The first series used only previously unirradiated (fresh) fuel rods and these were repeatedly cycled into DNB. Resultant transition boiling or film boiling at the cladding surface was terminated either by increasing the coolant flow rate, by reducing the reactor power, or both. In this series of seven tests, the number of DNB cycles varied from four to nine and the total time in film boiling reached 660 seconds for one test fuel rod.

The test rods used in the second series of tests included both previously-irradiated rods and fresh rods. They were taken into DNB only once, by reducing the coolant flow rate sufficiently to cause film boiling. The rods were then held in film boiling for from 60 seconds to 210 seconds and the test was terminated by scramming the reactor and restoring the coolant flow rate to its peak pre-DNB level.

The conditions for the two test series were chosen to provide a wide range of peak cladding temperatures and a wide range of times in transition boiling and in film boiling. The tests, therefore, produced a very wide range of cladding oxidation, from almost none to very severe, e.g., greater than 30% of the theoretical maximum in local regions.

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The resultant extremes of peak cladding temperature, time at temperature and cladding oxidation were much more severe than any which are currently postulated for commercial light water power reactors as shown in the table below.

COMPARISON OF PBF PCM TEST EXTREME CONDITIONS WITH PROJECTED WORST CASE COMMERCIAL LWR PCM ACCIDENT CONDITIONS

	Peak Fuel Rod Power (KW/M)	Peak Clad Temperature (K)	Peak Time in Film Boil (sec)	Equivalent Oxidation (gms O/cm^2)
Projected Worst Case Commercial LWR PCM Accident Conditions (See Footnote 1)	61	1090	200*	0.0026
PBF/PCM Test (Extreme Conditions)	76	1770	660	0.1241
PBF Test Conditions At or Below Which No Failures Were Seen	--	1425	210	0.0177
	--	1600	70	0.0221

*Total time in firm boiling or transition boiling (less severe oxidation than for film boiling only)

Appendix A presents a tabulated summary of the behavior of the 37 rods exposed to film boiling conditions in the 13 Power-Cooling Mismatch (PCM) and Irradiation Effects single rod tests performed in the Power Burst Facility to date. Additional information on the procedures for determining effective clad temperature is given in Appendix B. Test data reports issued to date are listed in Appendix C. Appendix D, TREE-NUREG-1196, is a summary of the PBF-Single Rod PCM test data presented November 9, 1977, at the Fifth Annual Water Reactor Safety Research Information Meeting at Gaithersburg, Maryland.

Topical reports are scheduled for completion in July 1978; they will review and collectively evaluate the observed cladding surface heat transfer phenomena, the extent of zircaloy-water reaction, the fuel element thermal and mechanical response, and the effects of pre-irradiation on the behavior of single fuel elements under power-cooling mismatch conditions.

RESULTS

1. Thirty of the 37 fresh and pre-irradiated PWR type fuel elements which were subjected to film boiling did not fail, either during or after the test, despite elapsed times in film boiling of at least 60 seconds at effective peak cladding temperatures ranging from 1250K to 1430K, e.g., no fuel rod failure after exposure to film boiling for 210 seconds at 1425K peak clad temperature.

   Cladding for nine of these thirty intact rods had been pre-irradiated to 13 - 17 GWD/MTM equivalent and four had been pre-irradiated to 6 - 8 GWD/MTM equivalent.

2. One test rod with pre-irradiated cladding was prepressurized to a level greater than should ever occur during power operation to a burnup of 30 GWD/MTM. This rod failed while in film boiling, at a time when its internal pressure was 6MPa greater than the loop pressure. The failed rod had a maximum circumferential elongation (in the rupture area) of about 25%, and this maximum elongation occurred only for a short distance along the length of the rod (less than 1 cm). There was no significant change in coolant flow rate either during the clad ballooning or after the cladding had ruptured.

3. The remaining six fuel rods which failed did not fail while in film boiling despite total film boiling times of from one to eleven minutes and effective cladding temperatures as high as 1770K. Five of these fuel rods failed within one to three minutes after reactor scram and one did not fail until it was being examined in the hot cell several weeks after the test. Pre-irradiation levels corresponding to an equivalent burnup of 17 GWD/MTM had no significant effect on the occurrence of rod failure or on the failure mechanisms of the two failed rods with pre-irradiated cladding.

4. The cladding of the six fuel rods which failed after reactor scram had absorbed more than enough oxygen to be embrittled at room temperature, according to critical oxidation criteria proposed by Pawel.[2] In the three examinations completed to date, the film boiling times

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[2] R. E. Pawel, "Oxygen Diffusion in Beta Zircaloy During Steam Oxidation," Journal of Nuclear Materials, Vol. 50, No. 3 (April 1974) pgs. 247-258.

exceeded Pawel's critical oxidation times for the effective clad temperatures by factors of from 3 to 20. These large time excesses may explain why these three rods failed at loop ambient temperature of 494K to 600K after reactor scram. The amounts of oxygen in the zircaloy cladding of the unfailed rods sampled to date were consistent with Pawel's critical oxidation criteria, implying that the cladding of none of the 30 unfailed rods had absorbed enough oxygen to be embrittled at room temperature.

Internal clad oxidation from contact with the fuel occurred under PCM conditions where the external pressure was greater than the internal pressure and the clad collapsed onto the fuel, forming an effective diffusion couple. This source of oxygen must be considered in defining allowable film boiling times and clad temperatures.

5. There is no evidence that pellet-cladding interaction (PCI) type failures will necessarily accompany PCM events and there is some evidence that they will not. In accordance with the basic test plan (see pages 2 & 3), the second series of tests on both fresh and pre-irradiated rods were performed in a manner which would introduce PCI stresses by fairly rapid ramping of the test rods at the end of the preconditioning period to power levels approximately double the highest power levels they had been exposed to during any previous operation or preconditioning. These PCI stresses were maintained--except for relaxation effects--during the final hour of preconditioning. None of the test rods failed during this hour of critical preconditioning, although it was performed at peak axial powers up to 66KW/M.

SIGNIFICANCE OF RESULTS

The significance of the reported results is:

1. These test results provide further evidence that cladding failure is not a necessary consequence of exceeding the critical heat flux (CHF) and incurring film boiling.

2. Use of the concept of "allowable oxidation," based on the Pawel room temperature embrittlement criteria, is a promising way of predicting clad failure and consequent probable fission product release to the coolant as a result of a PCM. This criterion should provide a more realistic estimate of coolant fission product content than the assumption that coolant flow rates below some selected DNBR value (currently 1.3) will lead to clad failure and consequent release of the total volatile fission product inventory.

3. The results to date indicate little or no likelihood of failure propagation during PCM events for fuel elements during the first three operating cycles because during these three cycles, the coolant pressure will be greater than the internal gas pressure within the fuel element and the cladding will collapse rather than balloon during a severe PCM. Based on the test results for the highly pre-pressurized fuel rod, there is also little likelihood of failure propagation during the fourth or fifth cycles because there is no hint of significant fuel rod distortion and flow blockage associated with the failure of this fuel rod. This assertion will be subjected to confirmation in future tests of clusters of fuel rods.

EVALUATION OF RESULTS

The PBF Experimental Program Review Group members have reviewed the preliminary release of test result information for single rod PCM tests and concur that there is substantial evidence that DNB caused by the power-cooling mismatch does not necessarily result in fuel rod failure. They also concur that clad failure appears to be closely associated with severe clad oxidation.

Some members of the PBF Program Review Group have expressed an interest in determining the extent that PCI type failures may be associated with power-ramp type PCM events. In the irradiation effects tests, there were no PCI-caused clad failures despite the use of moderate to high over-power ramp rates, followed by a hold at the over-power to induce PCI. These results offer limited assurance that the primary clad behavior in power-ramp type PCM's as well as in flow coastdown PCM's is little influenced by PCI-type interactions.

For flow coastdown PCM's, the clad temperature would, of course, rise simultaneously with or slightly ahead of any rise in fuel temperature, while for power ramp PCM's, the fuel temperature would rise slightly before any change in clad temperature. The PBF test data support the hypothesis that for both types of PCM, the risk of PCI-type clad failure is reduced as soon as the cladding becomes hot enough to creep rapidly under the primarily compressive creep conditions which would exist because the external pressure is greater than the internal pressure for all but extremely high-burnup, end-of-life rods.

RECOMMENDATION

These research results from tests on single fuel elements exposed to power-cooling mismatch are offered for NRR's consideration in determining possible changes in required departure-from-nucleate-boiling ratios for those commercial power reactors which use zircaloy-clad uranium dioxide fuel rods. Technical questions concerning these results may be referred to Dr. Robert Van Houten, Project Manager, or to Dr. William V. Johnston, Chief, Fuel Behavior Research Branch.

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Enclosures: as stated

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GLOSSARY

ATWS - anticipated transient without scram
CHF - critical heat flux
DNB - departure from nucleate boiling
DNBR - departure from nucleate boiling ratio

G - giga (= 10^9)
GWD - giga watt days
K - degrees Kelvin (= Degrees Celsius plus 273.1)
KW - Kilowatts

LWR - light water reactor

M - meter
MTM - metric ton (= 1000 Kg) of metal

PBF - Power Burst Facility
PCI - pellet-clad interaction
PCM - power-cooling mismatch
PWR - pressurized water reactor

APPENDIX A

PCM & IE TEST RESULTS SUMMARY

Test Desig-nation	Test Rod Desig-nation	Test Rod Type [g]	Pre-Pressur-iza-tion (MPa)	Rod Flow Shroud ID (mm)	Pre-Condi-tion-ing Time (h)	No. of DNB Cycles	Fuel Rod Peak Power During Final DNB Cycle (kW/m)	Coolant Mass Flux for DNB Cycle (kg/s-m2)	Total Time in Film Boil-ing (s)	Max Clad Temp From TC (K at/ Elev m)	Max Clad Temp From Met Exam (K at/ Elev m)	Max Fuel Q Temp from TC (K at m)	Increase in Rod Internal Pressure after DNB (MPa)	Clad Axial Strain after DNB (mm)	Zirc Crystal. Phase at Hot-Spot	Max Thick. of ZrO_2 α-layer (μ)	Cladding Collapse Region (Elev. m)
8-1RS	4	PWR	3.79	17.9	2	4	80	1383	600 [a]	1105/ 0.635	1683/ 0.597	2564/ 0.737	0.69	2.29	β	200	0.48 - 0.76
CHF Scoping	5	PWR	3.79	19.2	1.5	7	63	1397	60	1061/ 0.533	1372/ 0.584	2422/ 0.679	0.26	2.92	β	30	0.48 - 0.79
8-1RF	6	PWR	3.82	19.4	17	6	61	407	30	993/ 0.737	1311/ 0.787	1889/ 0.737	0.21	[d]	α + β	6	0.66 - 0.86
PCM-2A	7	PWR	2.49	16.3	35	9	58	827	210	861/ 0.789	1425 0.86	1894/ 0.787	0.24	1.04	β	90	0.66 - 0.86
PCM-2	8	PWR	2.49	16.3	14	8	46	722	100	1311/ 0.635	NC[c]	2289/ 0.686	0.36	2.90	β	NC[c]	NC[c]
	9	PWR	2.49	16.3	14	8	46	768	50	908/ 0.787	NC[c]	1900/ 0.533	0.33	[d]	α	NC[c]	0.74 - 0.79
	10	PWR	2.49	16.3	14	8	46	700	50	874/ 0.787	922/ 0.762	2082/ 0.686	0.08	[d]	α	2	0.74 - 0.79
	14A	PWR	3.79	16.3	14	8	46	735	50	[b]	NC[c]	[b]	[b]	1.70	β	NC[c]	NC[c]
PCM-3	11	PWR	2.49	16.3	22	5	45	736	60	887/ 0.686	NC[c]	[d]	0.16	0.15	[c]	NC[c]	NC[c]
	13	PWR	2.49	16.3	22	5	45	765	0	649/ 0.686	NC[c]	[d]	[d]	0	α	NC[c]	NC[c]
	15A	PWR	3.79	16.3	22	5	45	780	60	[b]	1430/ 0.572	[b]	[b]	3.81	β	56	0.43 - 0.76
	21A	PWR	2.49	16.3	22	5	45	732	60[b]	1103/ 0.584	1411/ 0.572	[b]	[b]	0.40	β	0.048	0.43 - 0.76
PCM-4	14	PWR	2.49	16.3	25	4	63	1605	115	925/ 0.584	NC[c]	[d]	0.20	1.00	NC[c]	NC[c]	NC[c]
	15	PWR	2.49	16.3	25	4	63	1605	160	1050/ 0.686	NC[c]	[d]	0.40	1.55	NC[c]	NC[c]	NC[c]
	16	PWR	2.49	16.3	25	4	63	1590	130	1000/ 0.584	NC[c]	[d]	0.30	1.55	NC[c]	NC[c]	NC[c]
	17A	PWR	3.79	16.3	25	4	63	1515	120	[b]	NC[c]	[b]	[b]	1.24	NC[c]	NC[c]	NC[c]
IE ST-1	1	SL-U	2.59	19.3	19	1	67	2090	288 [a]	1310/ 0.50	1620/ 0.60	[d]	[d]	2.0	β	230	0.48 - 0.80
IE ST-2	2	SL-I	2.67	19.3	34	[f]	55	[f]	[f]	[b]	NC[c]	[b]	[f]	[f]	[f]	[f]	[f]
	4	M	2.72	19.3	34	[f]	61	[f]	[f]	[b]	NP[c]	[f]	[f]	[f]	[f]	[f]	[f]
	5	SW-I	2.66	19.3	34	1	61	1280	60	[b]	1330/ 0.81	2750/ 0.74	0.60	[d]	β	28	0.56 - 0.93
	6	SL-U	0.1	19.3	34	1	61	1414	90 [e]	1100/ 0.61	NP[c]	[b]	[b]	[d]	N [c]	NP[c]	NP[c]
IE-1	7	SL-[illegible]	1.91	16.3	33	1	68	2520	72 [a]	[b]	NC[c]	[b]	0.6	>2.8	β	NC[c]	NC[c]
	8	SL-I	1.72	16.3	33	1	63	1690	60	[b]	NC[c]	[b]	[d]	1.6	β	NC[c]	NC[c]
	9	SL-I	1.21	16.3	33	1	64	1840	60	[b]	NC[c]	[b]	0.8	0.9	NC[c]	NC[c]	NC[c]
	10	SL-I	2.49	16.3	33	1	64	2030	60	[b]	NC[c]	[b]	0.6	3.1	β	NC[c]	NC[c]
IE-2	11	M	2.6	16.3	34	1	65	<2600	90	[h]	NC[c]	1870/ 0.74	0.07	1.83	NC[c]	NC[c]	NC[c]
	12	M	2.6	16.3	34	1	65	>2390	90	[h]	NC[c]	[d]	0.32	0.08	NC[c]	NC[c]	NC[c]
	13	SL-U	2.6	16.3	34	1	62	>2390	90	1080/ 0.62	NC[c]	[d]	0.18	[b]	NC[c]	NC[c]	NC[c]
	14	SL-U	2.6	16.3	34	1	61	>2390	90	1020/ 0.62	NC[c]	[d]	1.17	0.28	NC[c]	NC[c]	NC[c]
IE-3	15	SL-I	2.48	16.3	31	1	66	2110	90 [a]	[b]	NC[c]	[b]	1.24	[d]	NC[c]	NC[c]	NC[c]
	16	SL-I	2.48	16.3	31	1	63	2160	90	[b]	NC[c]	[b]	2.06	3.75	NC[c]	NC[c]	NC[c]
	17	SL-I	2.48	16.3	31	1	67	2290	90	[d]	NC[c]	[b]	0.64	5.57	NC[c]	NC[c]	NC[c]
	18	SL-I	2.48	16.3	31	1	60	2100	90	[d]	NC[c]	[b]	0.88	2.81	NC[c]	NC[c]	NC[c]

[a] Rod broke after test during cooldown.
[b] Fuel rod not instrumented for this measurement.
[c] Metallurgical examination not completed (NC) or not planned (NP).
[d] Instrumentation not functioning properly.
[e] Rod broke during posttest handling in hot cell.
[f] Rod did not go into film boiling operation.
[g] SL-U - Saxton Load Follow Rod - unirradiated.
SL-I - Saxton Load Follow Rod - irradiated.
SW-I - Saxton water tubes irradiated - fresh fuel.
M - MAPI irradiated cladding with fresh fuel.

[h] No indication of change

Appendix B
Effective Clad Temperature

Effective Clad temperature is the temperature needed to produce the same oxygen pickup in zircaloy clad in an isothermal test in steam for the same time as the total DNB time. Effective clad temperatures are established by metallographically determining the local thickness of oxide and oxygen stablized alpha phases. Time in film boiling can be established accurately by up to four independent test measurements (clad temperature, centerline temperature, fuel rod elongation and fuel rod plenum pressure).

Effective clad temperatures agree well with peak temperatures which could be determined by direct or indirect evidence of the presence or absence of rapidly formed high temperature phases.

Appendix C

List of Relevant Reports (in Alpha-Numerical Order)

A. S. MEHNER et al., Postirradiation Examination Results for the Irradiation Effects Scoping Test 1, ANCR-NUREG-1336 (September 1976).

G. W. CAWOOD et al., Power-Cooling-Mismatch Test Series, Test PCM-2A Test Results Report, ANCR-NUREG-1347 (September 1976).

A. S. MEHNER, Postirradiation Examination Results for the Irradiation Effects Scoping Test 2, TREE-NUREG-1022 (January 1977).

S. L. SEIFFERT and G. R. Smolik, Postirradiation Examination Results for the Power-Cooling-Mismatch Test-2A, TREE-NUREG-1029 (February 1977).

Z. R. MARTINSON et al., Power-Cooling-Mismatch Test Series, Test PCM-2 Test Results Report, TREE-NUREG-1038, (February 1977).

W. J. QUAPP et al., Irradiation Effects Test Series, Scoping Test 2 Test Results Report, TREE-NUREG-1044 (September 1977).

W. J. QUAPP et al., Irradiation Effects Test Series, Test IE-1 Test Results Report, TREE-NUREG-1046 (March 1977).

W. J. QUAPP et al., Irradiation Effects Test Series Scoping Test 1 Test Results Report, TREE-NUREG-1066 (September 1977).

S. L. SEIFFERT, Power-Cooling-Mismatch Test Series PCM-2 Postirradiation Examination, TREE-NUREG-1069 (March 1977).

L. C. FARRAR et al., Irradiation Effects Test Series Test IE-3 Test Results Report, TREE-NUREG-1106 (October 1977).

D. W. CROUCHER et al., Irradiation Effects Test Series Test IE-5 Test Results Report, TREE-NUREG-1130 (January 1978).

A. W. CRONENBERG and M. S. El-Genk, An Assessment of Oxygen Diffusion During UO_2-Zircaloy Interaction, TREE-NUREG-1192 (January 1978).

P. E. MACDONALD et al., Response of Unirradiated and Irradiated PWR Fuel Rods Tested Under Power-Cooling-Mismatch Conditions, TREE-NUREG-1196 (January 1978)

Appendix C (continued)

Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, TREE-NUREG-1004 April-June (October 1976)

Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, TREE-NUREG-1017, July-September, 1976 (January 1977)

Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, TREE-NUREG-1070, October-December 1976, (April 1977)

Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, TREE-NUREG-1128, January-March 1977 (July 1977)

Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, TREE-NUREG-1147, April-June 1977 (September 1977)

Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, TREE-NUREG-1188, July-September 1977 (December 1977)

Reports to be issued by July 15, 1978

TREE-NUREG-1199	IE-1 PIER
TREE-NUREG-1195	IE-2 PIER
TREE-NUREG-1200	IE-3 PIER
TREE-NUREG-1201	IE-5 PIER

Supplementary Reports

R. R. Hobbins, G. R. Smolik, G. W. Gibson, "Zircaloy Cladding Behavior During Irradiation Tests Under Power-Cooling-Mismatch Conditions," Proceedings, ASTM Symposium on Zirconium in the Nuclear Industry, August 10-12, 1976.

W. J. Quapp, R. K. McCardell, "Behavior of Zircaloy Clad UO_2 Fuel Rods During Film Boiling in a PWR Environment," Specialists Meeting on the Behavior of Water Reactor Fuel Elements Under Accident Conditions, Spatind, Norway, September 1976.

R. K. McCardell, Z. R. Martinson, "Experimental Results of Post CHF Fuel Rod Behavior." Transactions of the American Nuclear Society 1975 Winter Meeting, San Francisco, California, November 16-21, 1975, Vol. 22, pp. 485-486.

W. J. Quapp, C. M. Allison, L. C. Farrar, "PWR Fuel Rod Behavior Under Film Boiling Conditions," Transactions of the American Nuclear Society 1976 Annual Meeting, Toronto, Canada, June 14-18, 1976, Vol. 23, pp 300-302.

R. R. Hobbins, A. S. Mehner, W. J. Quapp, G. W. Gibson, "Evaluation of Fuel Microstructures in Rods Operating Above Critical Heat Flux," International Conference of the American Nuclear Society, November 15-19, 1976.

W. J. Quapp, et al., "Behavior of Irradiated Fuel Rods During Film Boiling Operation," ANS Annual Meeting, New York, June 1977.

A. S. Mehner, O. Goetzmann, W. J. Quapp, "Reaction Between UO_2, and Zircaloy in a Fuel Rod Operated Above the Critical Heat Flux," American Ceramic Society Pacific Coast Regional Meeting, San Francisco, California, November 1976.

A. S. Mehner, W. J. Quapp, et al., "Performance of Unirradiated and Irradiated PWR Fuel Rods Tested Under Power-Cooling-Mismatch Conditions," Sun Valley, Idaho, July 31-August 4, 1977.

Z. R. Martinson, "Comparison of Experimental and Calculated Cladding Temperatures During Film Boiling," American Nuclear Society Topical Meeting, "Thermal Reactor Safety," Sun Valley, Idaho, July 31-August 4, 1977.

D. T. Sparks & R. W. Garner, "Results of Gap-Conductance Tests in the PBF," American Nuclear Society Topical Meeting, "Thermal Reactor Safety," Sun Valley, Idaho, July 31-August 4, 1977.

J. D. Kerrigan & J. A. Dearien, "Fuel Cladding Failure Models in the FRAP Codes," American Nuclear Society Topical Meeting, "Thermal Reactor Safety," Sun Valley, Idaho, July 31-August 4, 1977.

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RECOMMENDATION

These research results from tests on single fuel elements exposed to power-cooling mismatch are offered for NRR's consideration in determining possible changes in required departure-from-nucleate-boiling ratios for those commercial power reactors which use zircaloy-clad uranium dioxide fuel rods. Technical questions concerning these results may be referred to Dr. Robert Van Houten, Project Manager, or to Dr. William V. Johnston, Chief, Fuel Behavior Research Branch.

Original Signed by
Saul Levine

Saul Levine, Director
Office of Nuclear Regulatory Research

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F. Coffman, DOR
M. Mendonca, DOR
V. Stello, DOR
E. Brown, SD
J. Norberg, SD
J. Yore, ASLBP
J. Buck, ASLBP
R. Van Houten, RES
W. V. Johnston, RES
G. P. Marino, RES
D. A. Hoatson, RES
M. L.PPicklesimer, RES
T. McCreless, SECY ACRS

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