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SUMMARY OF NRC LWR SAFETY RESEARCH PROGRAMS ON FUEL BEHAVIOR, METALLURGY/MATERIALS AND OPERATIONAL SAFETY

G. L. Bennett

Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission

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Gary L. Bennett, Chief
Research Support Branch
U.S. Nuclear Regulatory Commission

Executive Summary

This paper summarizes the results of NRC-sponsored research into fuel behavior, metallurgy and materials, and operational safety. The NRC light-water reactor safety research program is part of the NRC regulatory program for ensuring the safety of nuclear power plants which are owned and operated by electric utility organizations. The NRC LWR safety research program focuses principally on the major subsystems or functions of a commercial LWR: thermalhydraulics, fuel behavior, metallurgy and materials, and operational safety. Computer codes are developed to analyze these various subsystems or functions. The thermalhydraulics research and associated computer codes are not discussed in this paper.

FUEL BEHAVIOR

The purpose of the NRC fuel behavior research program is to provide a detailed understanding of the response of nuclear fuel assemblies to postulated off-normal or accident conditions. This information is then used to develop physical models which are incorporated into fuel analysis codes. The fuel codes are tested against the results of integrated in-pile tests. The understanding of the release and transport of fission products from damaged fuel rods and the study of the behavior of molten fuel are included in this research program.

The Power Burst Facility (PBF) tests at the Idaho National Engineering Laboratory are sponsored by NRC. To date, nuclear LOCA blowdown tests, power-cooling mismatch tests and reactivity-initiated accident tests have been run in PBF.

Since the cladding is the first barrier to the release of fission products, NRC has funded a number of studies of this important component which have shown that the cladding will contribute less heat, less hydrogen and more strength than predicted by conservative LOCA licensing models.

Fission product release measurements made at ORNL indicate that the amount of cesium and iodine escaping from a defected PWR fuel rod during a LOCA with successful ECCS operations will be one to two orders of magnitude less than the gap release assumptions for these species used in licensing evaluations.

In the area of fuel code development, the fourth version of the transient code, FRAP-T4, has been released by INEL and undergone independent code assessment. The new steady-state code, FRAPCON-1, which is based on models from FRAP-S at INEL and GAPCON-THERMAL at PNL, has been completed and is now undergoing independent assessment.

METALLURGY AND MATERIALS

The objective of the NRC research into metallurgy and materials is to provide independent confirmation of the safe design of reactor vessels and piping and, if required, to establish ways for reducing the failure probabilities. The NRC research activites in this area are divided into the following groups: (1) fracture mechanics; (2) irradiation embrittlement, stress corrosion, and crack growth; and (3) nondestructive examination.

Some of the principal NRC confirmatory tests on fracture mechanics have been conducted on both small- and intermediate-scale pressure vessels at ORNL. These tests have shown that for flaws less than one-half of the wall thickness in depth, a pressure of nearly three times the design value must be applied to initiate rapid fracture.

Since operational effects such as irradiation embrittlement, stress corrosion and crack growth may degrade the load-bearing capability of structural components, NRC sponsors a number of research programs to quantify these effects. The results to date show that irradiation embrittlement can be "annealed" out with heat treating.

In the area of nondestructive examination, NRC-sponsored research has upgraded the ultrasonic testing techniques which form the basis for in-service inspections of the primary system.

OPERATIONAL SAFETY

NRC sponsors a broad category of research termed "reactor operational safety" which is research aimed at providing direct assistance to NRC officials concerned with the operational and operational-safety aspects of nuclear power plants. The topics currently addressed include qualification testing evaluation, fire protection, human factors, and noise diagnostics.

The NRC qualification-testing evaluation program is focused on obtaining the data needed to answer certain questions about the testing of safety-class equipment to assess the performance during and after postulated accident conditions. The specific questions considered are given in this paper. A new facility is being developed at Sandia Laboratories for more sophisticated testing.

The NRC fire protection research program emphasizes the collection of confirmatory data needed in support of current design standards and regulatory guides for fire protection and control in LWR nuclear power plants. Both small-scale and full-scale tests have been run at Sandia Laboratories and Underwriters Laboratories to study fire-retardant coatings, shields, sprinklers, and fire propagation.

The NRC human factors research program is concerned with assessing the role of human errors in reactor operational safety. It includes specific studies in support of human error investigations and the development of associated training programs, the study of safety-related operator actions, and a continuing review of the application of ergonomics in the design of nuclear power plants.

The NRC noise-diagnostics research program at ORNL has supported licensing activites by the use of noise diagnostics techniques in independent assessments of core-barrel motion in operating PWRs and in-core instrument tube vibrations in operating BWRs of the BWR-4 type. Recently, ORNL researchers have been analyzing the power oscillations in the Fort St. Vrain power plant as well as assisting in the assessment of the status of Three Mile Island Unit 2.

CONCLUSION

In general, the NRC LWR safety-research program has greatly expanded the data base on such areas as fracture mechanics, fuel behavior and operational safety. This information is being used by the NRC licensing staff in support of decisions on reactor safety.

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INTRODUCTION

The NRC's confirmatory safety research program on light-water reactors (LWRs) is structured to provide additional and/or independent information to gain confidence that the margins of safety identified in the licensing review are well defined and quantified.[1] The principal areas of NRC research in the field of LWR safety are structured approximately along the classic lines of defense-in-depth, which in this paper will include:

Fuel Behavior - Fuel-rod behavior in postulated accidents and associated failure limits;

Metallurgy/Materials - Safety design and protection of the integrity of the reactor pressure vessel and piping;

Operational Safety - Operational safety aspects of nuclear power-plant operation.

This paper complements a companion paper[2] on the NRC LWR safety research programs related to thermal-hydraulics and computer code development. In this summary paper only a brief overview of some very involved subjects can be given. A more detailed presentation is given in Reference 1.

FUEL BEHAVIOR RESEARCH

The NRC research into the behavior of LWR fuel assemblies is directed at providing a detailed understanding of the response of these fuel assemblies to abnormal or accident conditions. The importance of this research is clear since the first barrier to the release of radioactivity is the fuel cladding. The understanding of fuel behavior ultimately is expressed in terms of confirmed analytical models which are incorporated into publicly available computer codes.

NRC sponsors research covering the fuel, gap conductance, the cladding, integral fuel rods and rod-to-rod interactions. These research studies encompass both out-of-reactor and in-reactor experiments, the latter principally carried out in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory and in the Halden experimental boiling water reactor in Norway. These research studies span a range of environments from normal operation to abnormal or accident conditions.

The loss-of-coolant accident (LOCA) initiated by the rupture of a large primary-coolant pipe has been selected as a design-basis accident for evaluating many of the safety features of LWR power plants. Other postulated accident sequences that affect fuel-element behavior include the "power-cooling mismatch" (PCM) in which the boundary of the reactor-coolant system remains intact but there is an imbalance between the heat being generated by the fuel and the heat removed by the coolant. A PCM would result from a loss of coolant or an overpower transient.

A reactivity-initiated accident (RIA) could result from such causes as control-rod ejection or an anticipated transient without scram (ATWS).

The condition of the fuel element at the initiation of the accident could affect the course of the accident. The principal initial parameters that must be known for the analysis of a transient are the stored heat* and decay heat in the fuel, the gas pressure within the cladding, the extent of contact between the fuel and the cladding, and prior cladding strains. These parameters are interrelated and depend on a number of properties, such as thermal conductivity, thermal expansion, cracking and restructuring of the fuel, the initial fuel-to-cladding gap width, fission-gas release, and cladding creep. It is therefore necessary to understand the fuel design and to evaluate, either by analysis or experiment, the above-mentioned parameters. Trends that affect these parameters include prepressurization of the fuel rods, improved stabilization of pellet density during irradiation, and changes in fuel-rod diameters instituted by all reactor vendors.

*The results of a recent investigation to determine the current state of the art of fuel temperature, gap conductance, and stored energy calculations may be found in Stored Energy Calculation: The State of The Art by M. E. Cunningham, et al., PNL-2581 (May 1978).

The emergency core cooling systems (ECCS) are the principal safety features installed to maintain the integrity and long-term coolability of the fuel during a LOCA. The ECCS Acceptance Criteria[3] are intended to ensure the effectiveness of the ECCS if it should ever be needed in maintaining the structural integrity of the cladding. Two of the criteria supply direct guidance for planning the fuel behavior program:

> "Peak Cladding Temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200°F" (1477 K)

> "Maximum Cladding Oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. . ."

The following sections give a short summary of some of the principal achievements to date in the NRC fuel behavior research program.

Decay Heat

The acceptance criteria[3] for LWR emergency core cooling systems require that the decay heat applicable to a LOCA evaluation be calculated using the 1973 proposed American Nuclear Society Standard ANS 5.1.[4] The uncertainty in those data was judged to approach 15%, especially at cooling times of less than 100 seconds. The NRC has prescribed that a conservative value of 1.2 times the 1973 value be used in evaluating the effectiveness of ECCS. More recent calculations[5] and measurements[6,7] of decay heat in irradiated uranium-235 have demonstrated that the 1973 standard is itself conservative during the first seconds after shutdown. Furthermore, the uncertainties in the new data are nominally less than 5% at short cooling times and decrease as the cooling time increases. Figure 1 shows a comparison of these measured data and the summation calculation. On the basis of these data, a new standard has been developed and is being proposed to the American Nuclear Society Standards Steering Committee and then to the ANSI Board of Standards Review.

Zircaloy Oxidation

The oxidation of Zircaloy in steam is an important phenomenon in accident analysis since (1) hydrogen is generated by the reaction between zirconium and the steam, (2) the heat of reaction must be removed to prevent overheating the cladding, and (3) the oxygen consumed forms two brittle layers that reduce the wall thickness capable of carrying tensile stresses. The two brittle layers are zirconium oxide and an oxygen-stabilized alpha phase. The oxygen also dissolves in the remaining beta phase and causes it to be embrittled. For Zircaloy oxidation in steam, the Baker-Just rate-constant equation[8] is currently used for conservative evaluation of postulated accidents. Objections have been raised to the conservatism of this equation. It does not agree with experimental data in the temperature range of interest. The oxidation of Zircaloy has therefore been the subject of several investigations. From data reported by Cathcart,[9] (see Figure 2) the rate constant, $\delta^2/2$, at 1477K (2200°F) is only 58% of that of the Baker-Just equation, and thus only 76% of the oxidation predicted by the Baker-Just equation is actually observed. As a result, calculated peak cladding temperatures during a given postulated LOCA are estimated to be approximately 56K (100°F) lower with the new rate equation than with the more conservative Baker-Just equation.[10]

A new determination[11] of the rate of oxygen diffusion in beta-phase Zircaloy has indicated that the rate is approximately half that previously reported.[12] (See Figure 3)

Mechanical Properties of Zircaloy Containing Oxygen

The mechanical properties of Zircaloy have been determined[13] as functions of oxygen distribution and content, strain rate, biaxial stressing, microstructure, texture, and temperature over the range between 423 and 1700K (300 and 2600°F).

The strength and ductility of Zircaloy cladding at any temperature are strongly dependent on such factors and are important in producing and controlling cladding deformation during postulated LOCA and PCM events. The effects of quenching stresses on the properties of oxidized Zircaloy tubing have been determined.

Failure "maps" for fracture of the cladding by thermal shock were developed[14] relative to the maximum oxidation temperature and various time-dependent oxidation parameters, e.g., equivalent-cladding reacted to form ZrO_2, fractional thickness of transformed beta layer, fractional saturation of the beta-phase by oxygen, and thickness of beta phase with less than a specified critical oxygen content. The principal results are (1) if the cladding is cooled rapidly (about 100 K/s) through the beta-to-alpha phase transformation, the thermal shock failure boundary corresponds to about 20% of the wall thickness in equivalent oxidation for oxidation temperatures greater than about 1650 K, (2) if the cladding is cooled slowly through the phase transformation, the thermal shock failure boundary corresponds to 28% equivalent wall thickness oxidized to ZrO_2, and (3) the best correlation of thermal shock failure boundary with parameters related to the degree of oxidation of the cladding was that of thickness of the beta phase layer having less than 0.9 and 1.0 wt% oxygen for slow and fast-cooled cladding respectively. If the beta phase layer having less than this oxygen level was 0.1 mm in thickness, or more, the cladding did not fail irrespective of wall thickness, oxidation temperature, and total oxygen content. In-situ pendulum load impact tests at room temperature showed that the thermal shock boundary corresponded to approximately 0.03 J impact energy absorbed. As illustrated in Figures 4 and 5, data were plotted as time of oxidation versus reciprocal temperature and evaluated as to (1) failed on thermal shock quenching, (2) failed at 0.03 J impact, (3) survived 0.03 J but failed 0.3 J impact, and (4) survived 0.3 J impact load. These results allow a quantitive statement of energy absorbed by fracture, and a quantitative failure map to be drawn with any desired degree of conservatism. A finite-element model has been developed for crack growth in oxidized Zircaloy during thermal shock conditions, using mechanical properties measured for homogeneous specimens of

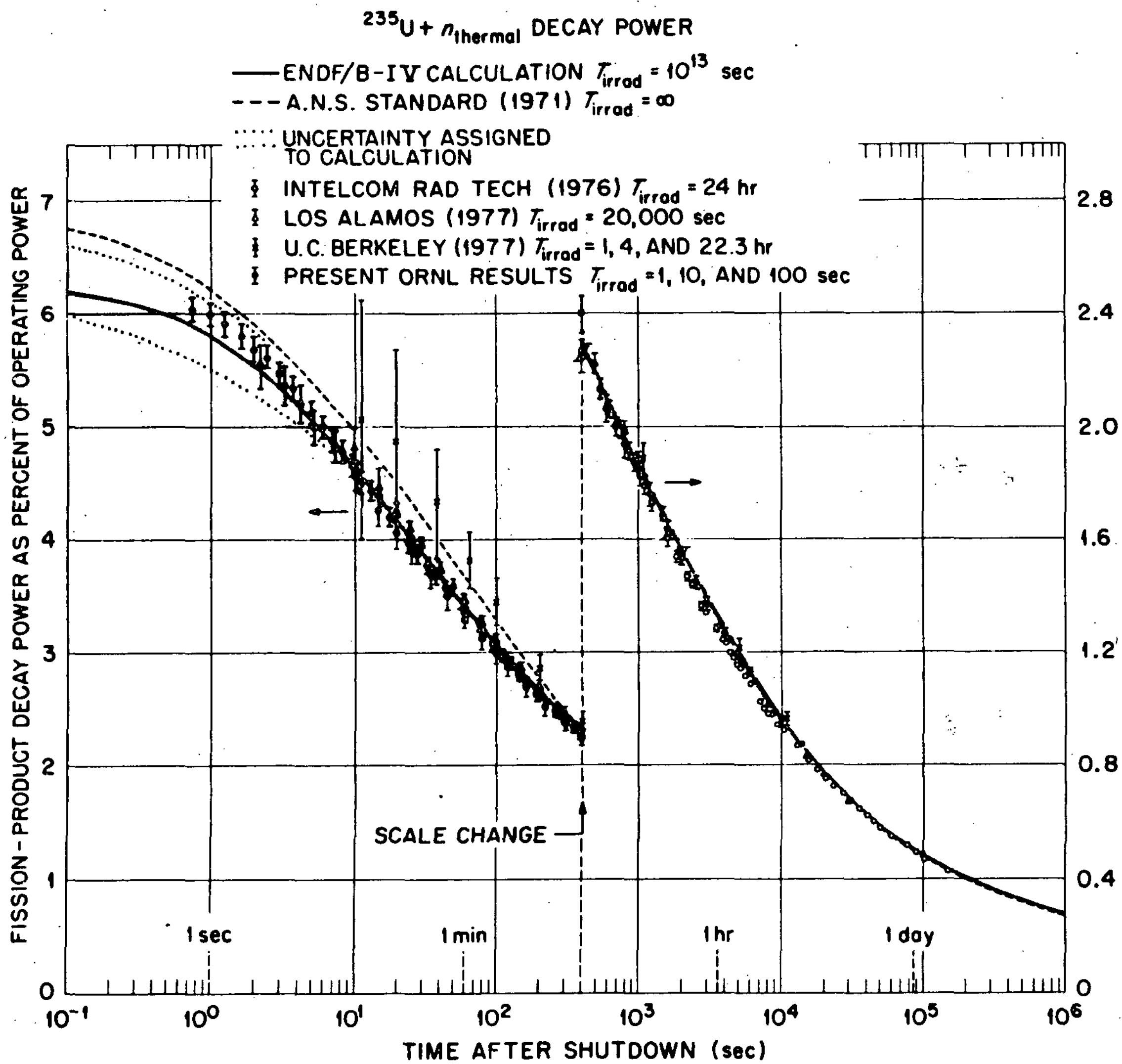
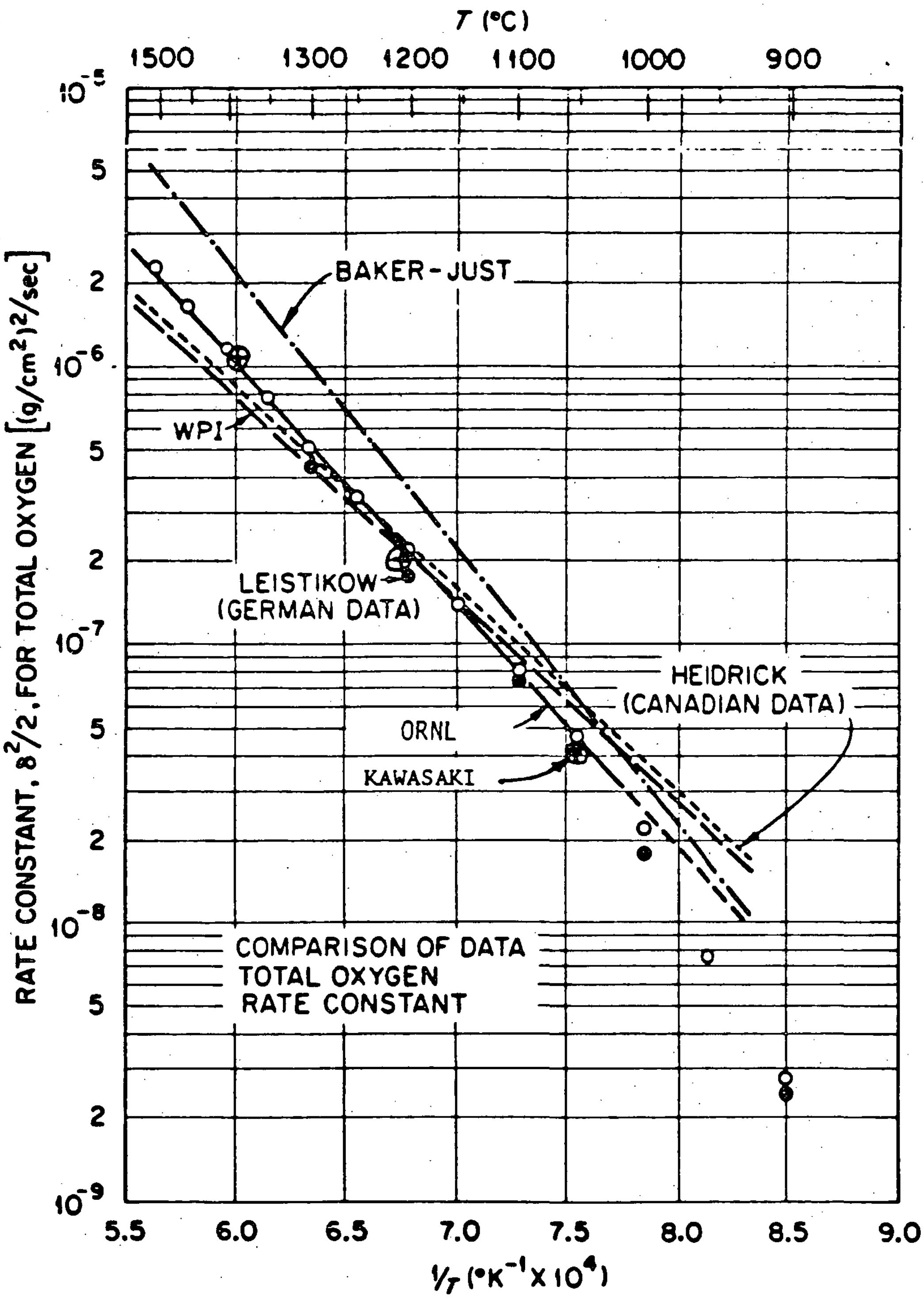


FIGURE 1. COMPARISON OF DECAY HEAT STANDARDS AND DATA

(REFS: ORNL/NUREG-14; LA-NUREG-6713; NUREG-0018-4; EPRI NP-180 & NP-616)



References:
Baker-Just: ANL-6548
WPI: EPRI 249-1
ORNL: ORNL/TM-5248
Kawasaki: JAERI-M 6181
Leistikow: private communication
Heidrick: unpublished manuscript

FIGURE 2. ARRHENIUS PLOT OF THE TOTAL OXYGEN RATE CONSTANT FOR THE STEAM: ZIRCALOY-4 REACTION

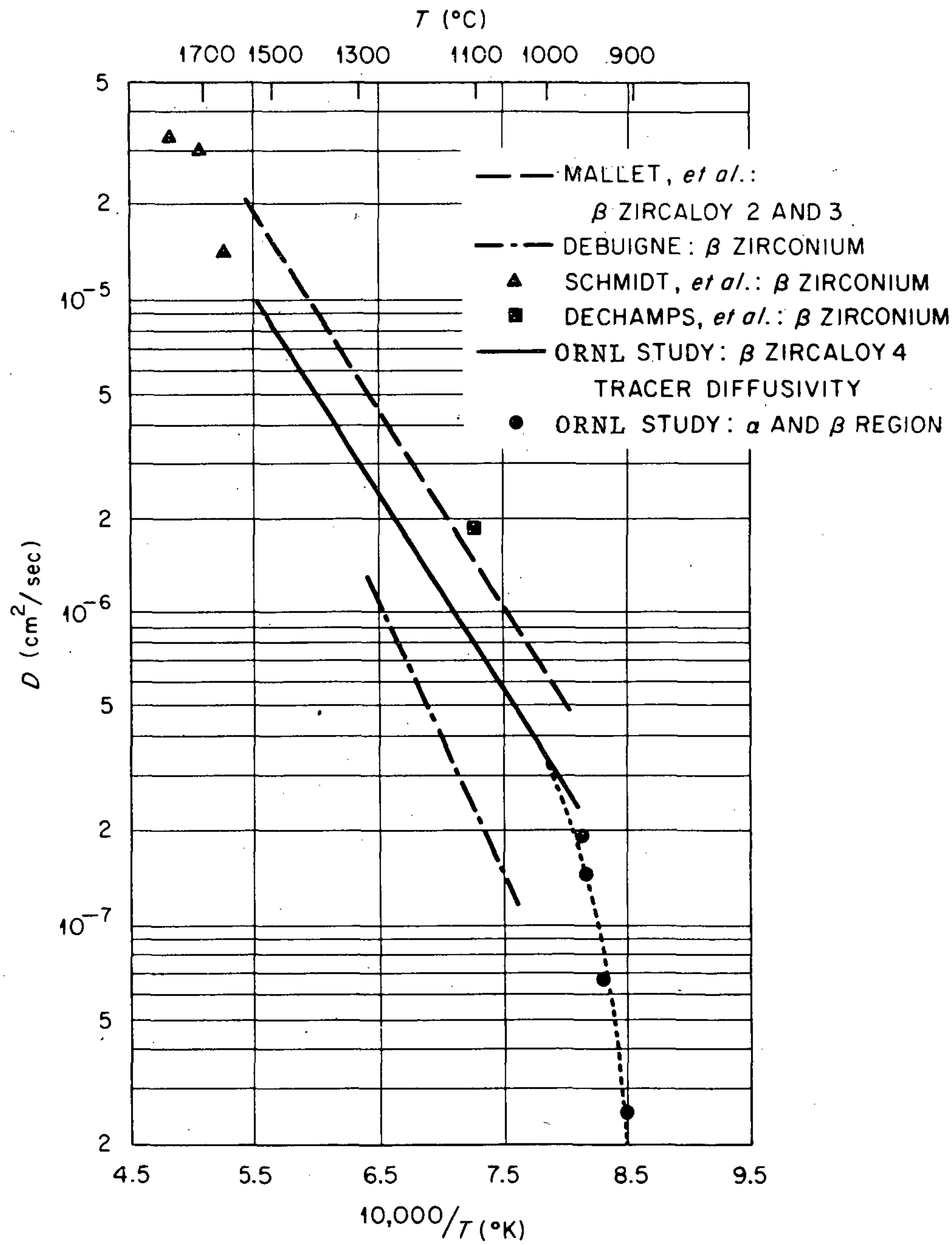


FIGURE 3. DIFFUSIVITY OF OXYGEN IN BETA-ZIRCALOY

(REFERENCE: ORNL/NUREG/TM-19)

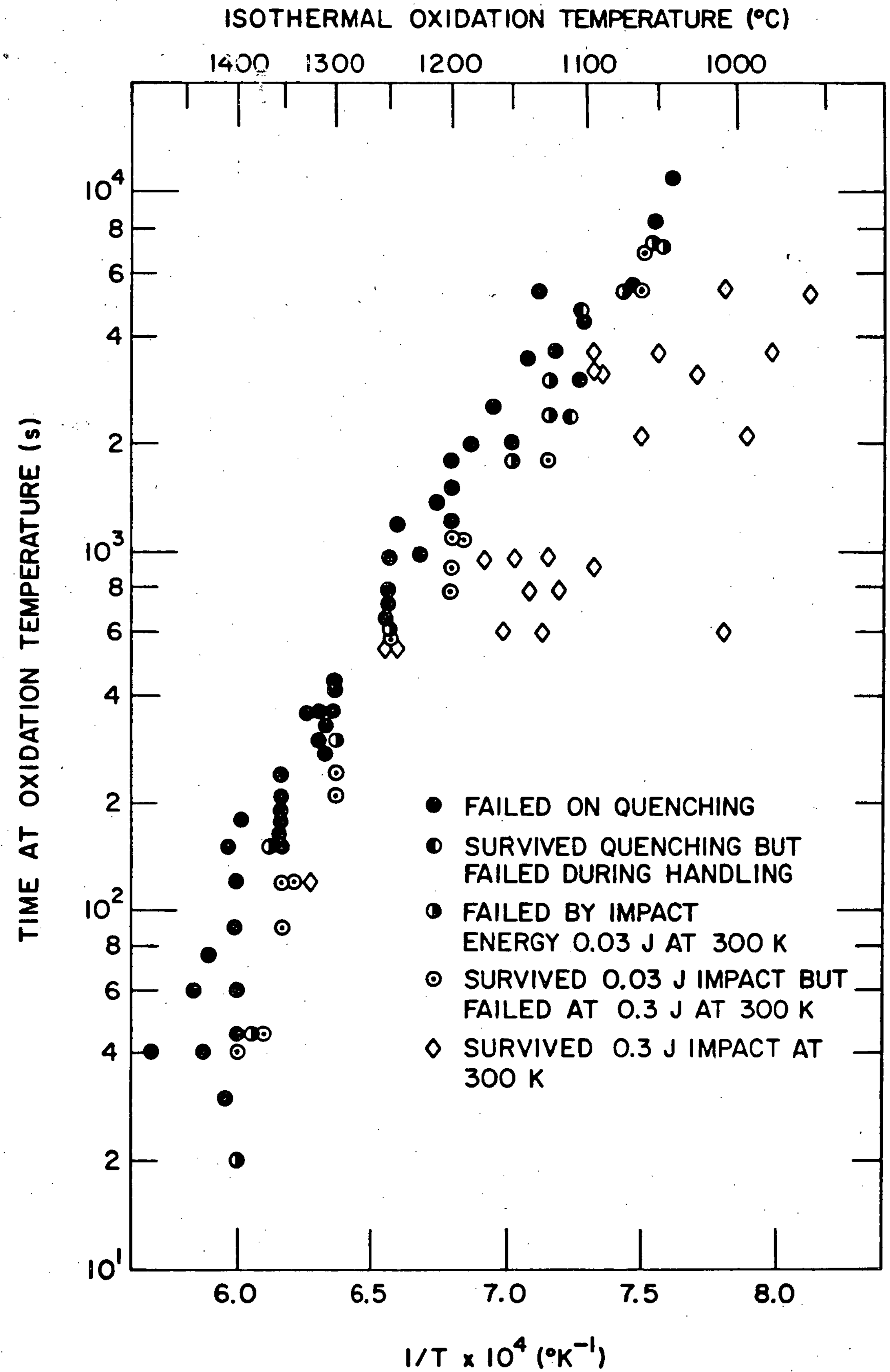


FIGURE 4. IMPACT CAPABILITY OF OXIDIZED ZIRCALOY-4 CLADDING

(REF: NUREG/CR-0201)

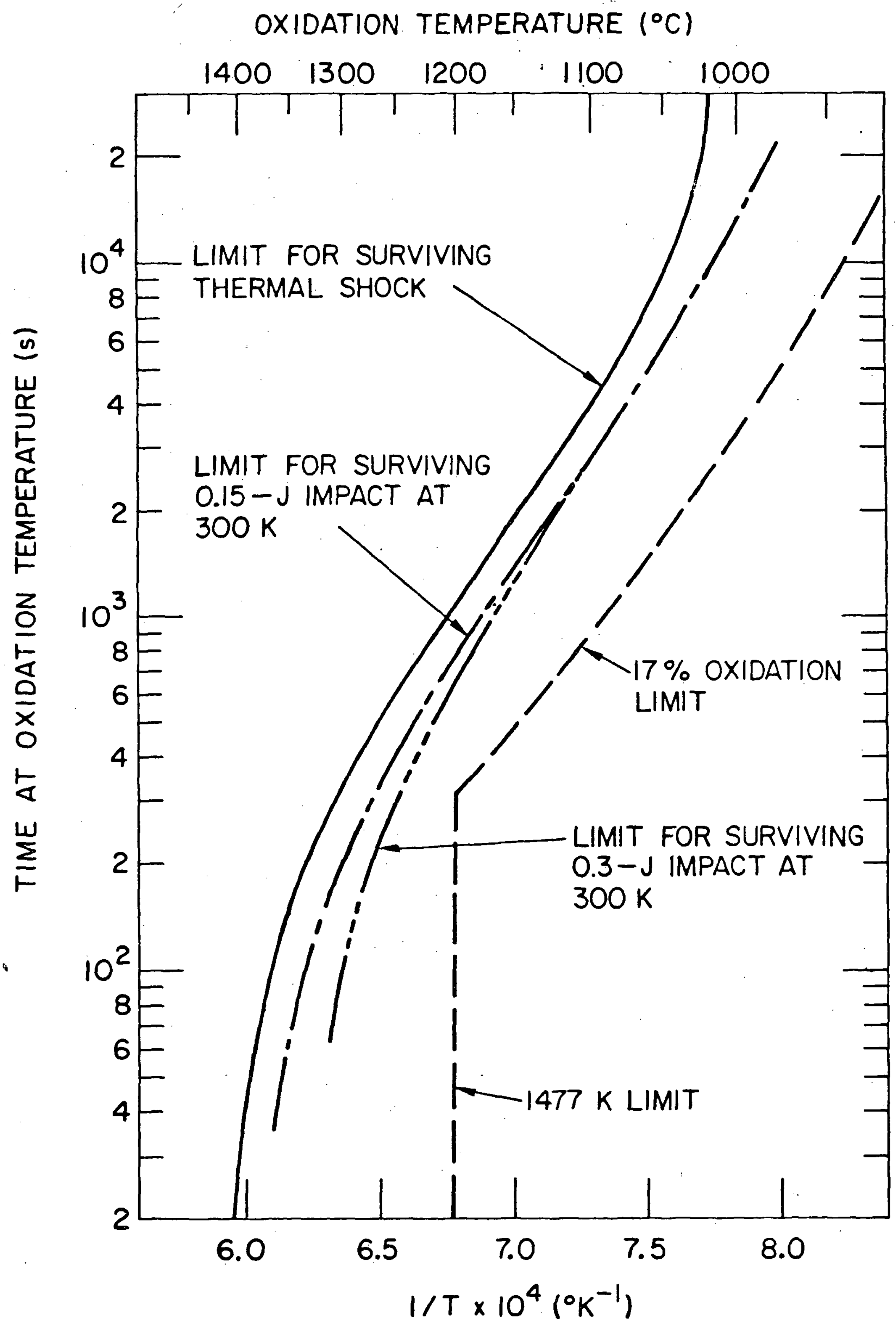


FIGURE 5. IMPACT CAPABILITY OF OXIDIZED ZIRCALOY-4 CLADDING (REF: NUREG/CR-0201)

varied oxygen contents. The data suggest that the present ECCS embrittlement criteria are conservative both to failure by thermal shock and failure by impact loads likely to be encountered in disassembly of the core.

Mechanical Properties of Irradiated Zircaloy

A satisfactory data base has been developed for the important mechanical properties of fuel cladding, e.g., yield stress, ultimate stress, uniform and total elongations, reduction in area and thickness at fracture, burst stresses, burst strains, stress rupture, collapse pressure, creep rates (both internal and external pressurization), and low-cycle fatigue over a range of environmental conditions. With respect to irradiation effects, information obtained from in-reactor tests and from post-irradiation tests is available for most of the properties of interest. For irradiated materials the scatter is generally greater (see, for example, Fig. 6), and since tests are more expensive, there are fewer data points for statistical evaluation. Until recently, most information was obtained at or below the operating temperatures of BWRs.

Irradiation results in a reduction in ductility, a reduction in impact strength, and an increase in strength at and below normal reactor operating temperatures. The yield strength of annealed Zircaloy-2 doubles and remains essentially constant above a neutron fluence of approximately 5 x 10^{20} n/cm^2. The yield strength and ultimate strength are effectively identical.[15] The consensus is that strength changes tend to saturate between 10^{21} and 10^{22} n/cm^2 (>1 MeV) but that the effect on ductility saturates between 3 x 10^{19} and 10^{20} n/cm^2 (>1 MeV).[16] The strain at which plastic instability sets in is reduced with increasing irradiation.

A significant body of data on the as-irradiated mechanical properties and on the kinetics of irradiation-damage annealing has been obtained[17] on one lot of Zircaloy tubing removed from spent commercial PWR fuel with a burnup greater than 30,000 MWd/MT UO_2. The properties examined were uniaxial tensile strengths and elongations in tubing at temperatures from 300 to 975K (80 to 1290°F) and burst tests at 645K (700°F). Specimens were tested in the as received condition, after isothermal annealing at temperatures of up to 975K (1290°F) and after annealing by transient heating to temperatures of up to 1275K (1830°F) at rates from 0.5 to 25K/sec. Similar tests are now underway on a second lot of cladding from spent commerical fuel with a burnup of less than 20,000 MWd/MT UO_2. Burst tests have been conducted during transient heating to temperatures of up to about 1275K (1830°F) at several heating rates. The kinetics of irradiation-damage annealing appear to vary with the evaluation method. Yield, tensile, and burst strengths can be fully recovered at some temperatures, while elongation decreases significantly below that observed in the as-received condition, with all tests conducted at 645K (700°F). Thus there is a "strain-aging" or "aging" phenomenon that affects elongation but not strength properties. More data are needed for cladding at lower burnup, so that the "saturation exposure" for the several properties can be determined.

Transient heating burst tests have shown that the properties at burst temperatures of 980K (1300°F) and higher are essentially the same in both irradiated and unirradiated Zircaloy tubing (See Figure 6).

Cladding Deformation During Burst Testing

One of the key items in analyzing a postulated LOCA is knowing the condition of the core, i.e., what is the deformation and extent of flow blockage of the coolant channels of a fuel assembly. Experiments are being performed with electrically heated rods to give flattened temperature gradients comparable to those in PWRs and BWRs, with internal pressures from 100 to 1800 psi and heating rates of up to 28 C/sec (50 F/sec). Single rods and clusters of 16 and 64 rods with typical PWR grid spacings have been or will be studied. The rods are approximately 2 meters (6.5 feet) long with a heated length of 0.92 meter (3 feet), and the grid spacings are about 0.61 meter (2 feet). The data from the single-rod tests[18] (over 40 have been completed-see Table 1) will be compared with the data from the cluster tests, with the goal of having a preliminary correlation of rod-to-rod interactions, scaling factors, flow blockage, heating rate, initial and burst pressures, and burst strains by 1980 (See Fig. 7). The correlation should allow the prediction of multirod performance from single-rod tests and should greatly decrease the cost of evaluating various cluster configurations and cladding modifications.

To confirm the burst test results from the electrically heated 0.9-m (3-ft) length rods experiments are planned on full length rods (3.7 m or 12 ft.) heated by nuclear power in the NRU test reactor in Chalk River, Ontario. These tests will also be of interest in comparison to the full length electrically heated bundles in the REBEKA test series being conducted by KfK in the Federal Republic of Germany. In addition to determining the burst behavior of fuel rods, a series of experiments on the thermal-hydraulic reflood behavior is also a part of the proposed NRU program. The experiments will use a bundle of 32 rods of commercial length and enrichment and will provide well-characterized data on the thermal-hydraulic and deformation behavior in a nuclear environment representative of the heatings and reflood of a LOCA in an LWR.

Gap Conductance

The thermal behavior of an LWR fuel rod is complex. Mechanisms postulated to influence the thermal behavior of an LWR fuel rod include:

- Changes in the dimensions of the fuel-to-cladding gap from pellet cracking and pellet relocation, fuel densification and swelling, thermal expansion, and cladding creepdown.
- Changes in the fuel thermal conductivity from pellet cracking (nonradial cracks) and restructuring.
- Changes in the composition of the gas in the gap or in fuel cracks from impurity-gas release, fission-gas release, and fill-gas absorption.

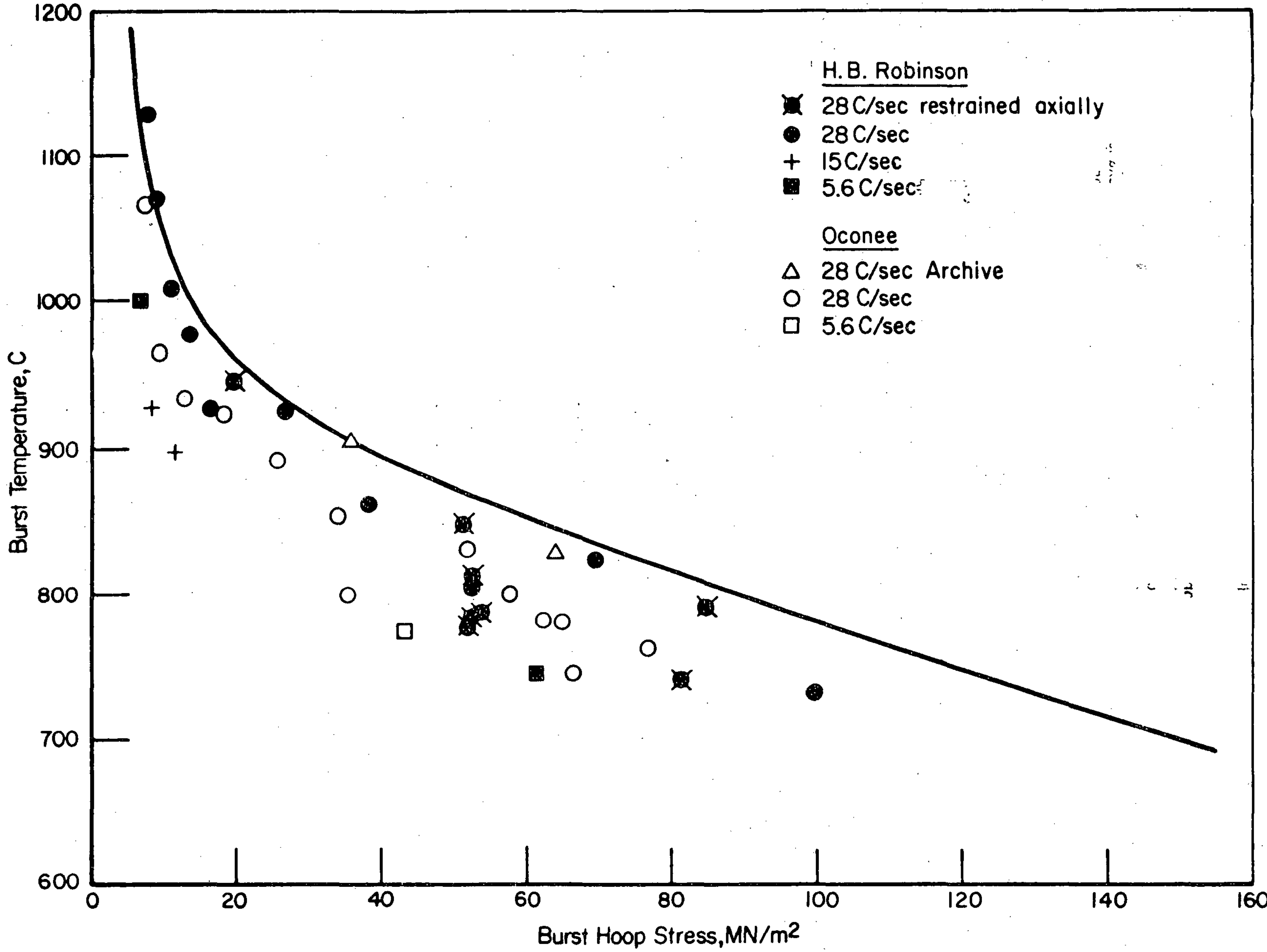


FIGURE 6. TRANSIENT BURST-TEST DATA FOR ARCHIVE AND SPENT-FUEL CLADDING INCLUDING ORNL CURVE FOR UNIRRADIATED TUBING

(REF: "MECHANICAL PROPERTIES OF SPENT FUEL CLADDING" BY L. M. LOWRY, ET AL, SIXTH NRC WRSR INFORMATION MEETING, 1978, NRC PDR)

TABLE 1A. TEST CONDITIONS AND RESULTS FOR SINGLE ROD BURST TESTS IN STEAM*

Test No.	Fuel simulator No.	FPS gas volume[a] (cm^3)	Initial conditions: Temperature (°C)	Initial conditions: Pressure (kPa)	Maximum pressure (kPa)	Burst conditions: Pressure (kPa)	Burst conditions: Temperature[b] (°C)	Burst conditions: TC No.	Burst conditions: TC position relative to tube burst	Burst strain (%)	Tube heated length change (%)	Time to burst (sec)
PS-1	PROTO 2	26.3	351	6,450	7,040	6,360	893	92	At burst, 20° around	18	NA[c]	20.0
PS-3	PROTO 1	30.0	334	6,520	6,860	5,580	873	213	27 cm below, 70° around	29	NA	20.0
PS-4	PROTO 1	29.3	343	6,440	6,780	5,860	871	201	2.5 cm below, 70° around	21	NA	21.0
PS-5	PROTO 1	29.5	343	6,410	6,760	5,720	882	201	At burst, 60° around	26	NA	21.5
PS-8	2828005	30.6	349	6,470	6,810	6,000	843	213	14 cm below, 90° around	20	NA	21.5
PS-9	2828005	35.6	346	6,480	6,890	5,650	866	201	4.5 cm above, 60° around	25	NA	23.0
PS-10	2828005	35.9	352	6,440	6,830	6,000	901	83	6 cm above, 0° around	20	-0.69	21.2
PS-12	2828006	41.0	340	6,520	6,900	6,140	898	49	At burst, 30° around	18	NA	21.75
PS-14	2828006	41.5	337	6,450	6,830	5,820	883	83	6.4 cm above, 30° around	25	-0.69	22.65
PS-15	2828006	46.9	352	6,490	6,780	6,160	885	84	18.4 cm above, 70° around	17	-0.61	20.95
PS-17	2828005	37.5	340	13,270	13,880	12,130	778	86	40.3 cm above, 140° around	25	-0.43	16.1
PS-18	2828007	45.0	350	800	862	772	1171	93	11.4 cm above, 60° around	24	0.39	42.0
PS-19	2828005	27.5	348	2,590	2,820	2,590	959	88	45.2 cm above, 50° around	28	0.0	27.45
SR-1	2828013A	46.6	347	850	910	800	1166	91	38.9 cm above, 30° around	26	1.04	31.6
SR-2	2828017	48.5	344	1,130	1,220	1,010	1082	91	17.1 cm above, 20° around	44	0.61	25.7
SR-3	2828027	46.1	346	1,770	1,900	1,720	1011	90	5.5 cm above, 0° around	43	0.48	22.4
SR-4	2828026	44.5	337	4,400	4,700	4,480	921	93	14.6 cm above, 40° around	17	0.18	20.65
SR-5	2828028A	45.9	345	10,120	10,480	9,520	810	91	8.3 cm, 100° around	26	-1.13	18.0
SR-7	2828014A	45.9	338	15,110	15,530	14,440	736	86	20.3 cm below, 130° around	20	-0.39	15.55
SR-8	2828010	49.3	336	1,420	1,520	1,230	1020	90	15.2 cm below, 70° around	43	0.61	25.15
SR-13	3838016	47.3	325	1,310	1,430	1,070	1079	89	9 cm below, 10° around	79	0.65	24.65
SR-15	2828005	38.8	342	20,350	21,280	19,150	714	91	At burst, 10° around	14	0.35	14.5
SR-17	2828010	44.9	344	1,310	1,410	1,060	1049	90	21 cm below, 40° around	53	0.78	25.25
SR-19	2828031	35.2	335	19,970	20,830	19,040	668	86	27.5 cm below, 90° around	16	0.09	14.6
SR-20	2828031	33.4	332	1,290	1,410	1,060	1049	93	30 cm above, 110° around	55	0.61	25.1
SR-21	2828005	36.7	340	1,310	1,430	1,120	1023	83	32.3 cm below, 140° around	48	0.56	24.5
SR-22	2828031	33.8	332	1,130	1,230	890	1081	90	At burst, 4° around	50	0.69	27.2
SR-23	3838005	35.4	336	1,120	1,230	960	1077	86	25.7 cm below, 130° around	35	0.74	25.7
SR-24	2828031	35.5	332	1,200	1,300	990	1057	91	34.4 cm above, 120° around	67	0.87	26.9
SR-25	2828036	34.6	345	1,130	1,240	960	1092	83	2.5 cm below, 135° around	78	0.87	26.5
SR-26	2828031	43.1	340	1,000	1,060	830	1130	87	20.4 cm below, 100° around	34	1.13	29.9
SR-27	2828036	42.3	340	1,130	1,190	920	1084	86	At burst, 0° around	41	0.87	26.9
SR-28	2828021A	48.0	335	8,930	9,400	8,400	835	89	6.2 cm below, 45° around	27	-0.69	19.3
SR-29	282028A	42.0	340	8,680	9,050	8,040	843	94	26 cm above, 90° around	27	-0.87	20.0

[a]Fuel pin simulator volume measured at room temperature; includes pressure transducer and connecting tube.

[b]Maximum measured by any thermocouple at time of burst; thermocouple number and location indicating burst are listed.

[c]Not measured.

*"Significant Results from Single-Rod and Multirod Burst Tests in Steam With Transient Heating," by R. H. Chapman, Fifth NRC Water Reactor Safety Research Information Meeting, Nov. 7-10, 1977, available in NRC PDR.

TABLE 1B. TEST CONDITIONS AND RESULTS FOR SPECIAL SINGLE ROD BURST TESTS

Test No.	Fuel simulator No.	FPS gas volume[a] (cm³)	Initial conditions: Temperature (°C)	Initial conditions: Pressure (kPa)	Maximum pressure (kPa)	Burst conditions: Pressure (kPa)	Burst conditions: Temperature[b] (°C)	Burst conditions: TC No.	Burst conditions: TC position relative to tube burst	Burst strain (%)	Tube heated length change (%)	Time to burst (sec)
						Tests conducted in an argon environment						
SR-9	2828005	38.2	335	6480	6890	6260	880	92	18 cm above, 170° around	18	-0.52	22.1
SR-11	2828010	45.9	330	1400	1530	1270	1015	91	5 cm below, 0° around	98	0.18	24.4
SR-14	2928005	38.4	334	1740	1900	1740	1004	93	14 cm above, 0° around	24	0.18	22.8
						Large-volume tests conducted in a steam environment						
SR-16	2828005	159.2	345	6500	6580	6420	880	93	22 cm above, 135° around	15	-0.61	22.15
SR-18	2828005	154.6	344	2590	2630	2590	968	89	4 cm above, 15° around	22	0.30	22.95

[a]Fuel pin simulator volume measured at room temperature; included pressure transducer and connecting tube.

[b]Maximum measured by any thermocouple at time of burst; thermocouple number and location indicating burst temperature are listed.

TABLE 1C. RESULTS OF CREEP RUPTURE AND LOW HEATING RATE BURST TESTS OF FUEL PIN SIMULATORS WITH INTERNAL HEATERS*

Test No.	Fuel simulator No.	FPS gas[a] volume (cm³)	Initial conditions: Temperature (°C)	Initial conditions: Pressure (kPa)	Maximum pressure (kPa)	Heating rate (°C/sec)	Control temperature[b] (°C)	Burst conditions: Pressure (kPa)	Burst conditions: Temperature[c] (°C)	Burst strain (%)	Tube heated length change (%)	Time to burst (sec)
					Creep rupture tests in steam							
SR-33	2828036	51.0	370	6285	6515	13[d]	762	5690	762	23.4	-1.04	103[e]
SR-34	2828031	51.6	336	6260	6540	12[d]	762	5820	766	31.6	-0.87	49[e]
SR-35	2828031	51.5	350	4830	5050	13[d]	761	4470	775	29.0	-0.26	250[e]
SR-36	2828036	51.2	330	4805	5040	11[d]	761	4555	821	28.8	-0.52	162[e]
					Transient burst tests in steam							
SR-37	2828031	50.3	305	14410	14965	28		13560	760	23.1	-1.13	17.4
SR-38	2828036	51.1	340	14660	15265	29		13775	770	20.0	-0.69	15.5
SR-41	2828031	50.0	340	10510	10915	9		9765	757	27.4	-1.04	46.9
SR-42	2828036	49.6	344	10495	10900	10		9465	761	28.4	-1.39	47.1
SR-43	2828031	48.7	340	8465	8800	4		7620	773	29.0	-1.39	89.1
SR-44	2828036	49.7	338	7935	8250	5		7310	777	30.0	-0.78	82.5

[a]Fuel pin simulator volume measured at room temperature; includes pressure transducer and connecting tube.

[b]Temperature maintained constant at control thermocouple during creep time by feedback control on power input.

[c]Maximum measured by any thermocouple at time of burst.

[d]From initial temperature of creep temperature.

[e]Hole time for creep at quasi-steady-state temperature level.

*Taken from "Effect of Creep Time and Heating Rate on Deformation of Zircaloy-4 Tubes Tested in Steam with Internal Heatups" by R. H. Chapman, et al., USNRC Report NUREG-CR-0343

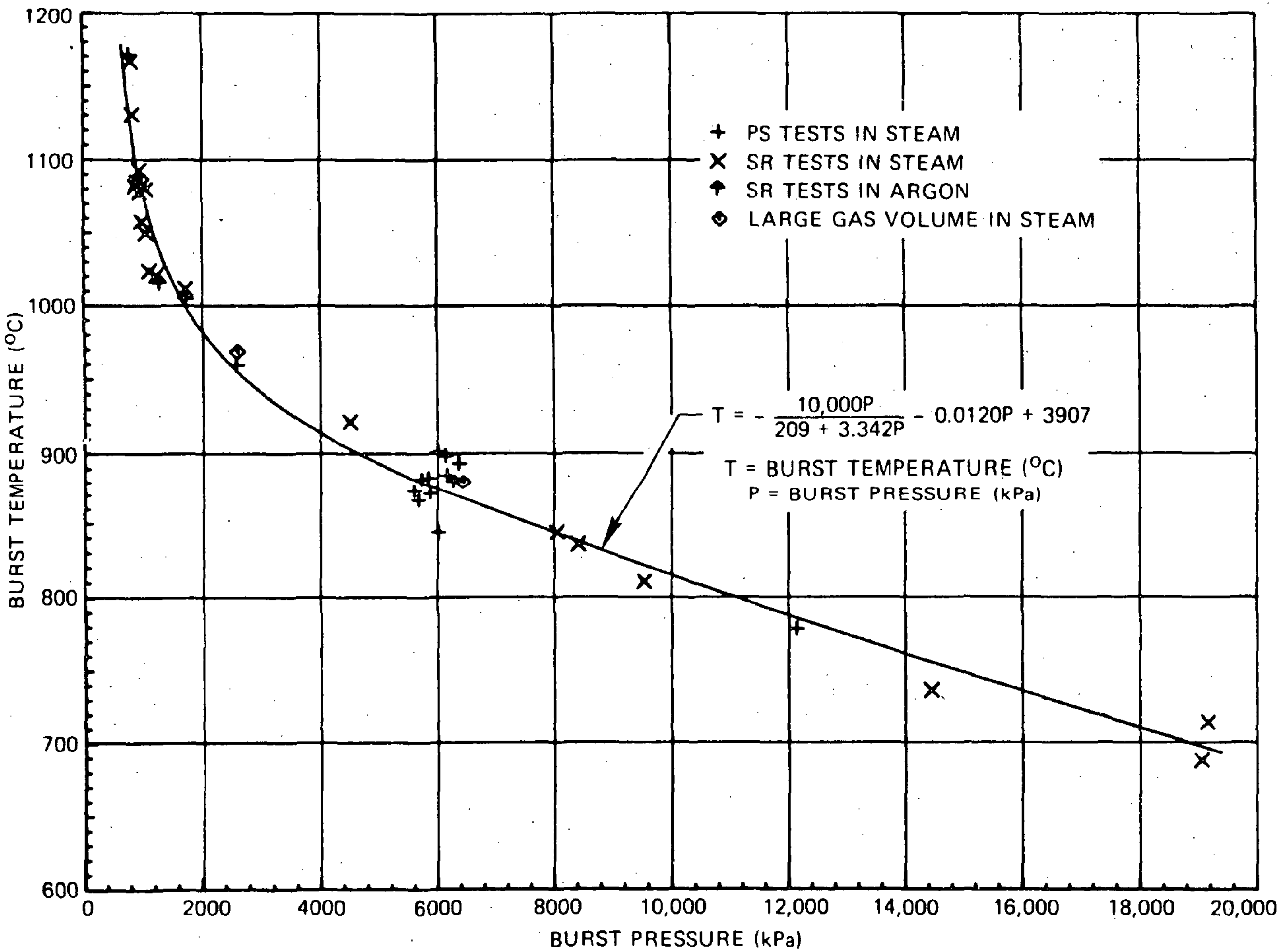


FIGURE 7. BURST TEMPERATURE AS A FUNCTION OF BURST PRESSURE.

(REF: ORNL/NUREG/TM-135)

With pressed and sintered pellets there is usually an appreciable resistance to heat transfer between the pellet surface and the cladding. The interfacial resistance may be the result of a gas-filled gap or uranium dioxide in actual contact with the cladding. Data on fuel centerline temperature and rod internal pressure tend to support the contention that the thermal response of an LWR fuel rod is strongly influenced by stochastic pellet cracking and pellet fragment relocation mechanisms. As fuel burnup progresses, pellet cracking and relocation, pellet swelling, thermal expansion, and cladding creepdown combine to close the gap. The rate of gap closure has been shown to depend on such operating variables as the rate of power increase, number of power cycles, and power level.

Several in-reactor experiments to deduce values for gap conductance have been made, and a number of investigators have attempted to infer gap conductance from the examination of fuel rods that were irradiated for other purposes. Gap conductance is not measured directly but is derived from measurements of fuel and/or cladding temperatures. Most of the reliable experiments utilized small ($\leq$ 200 μm) diametral gaps. There is very little well-characterized data for thermal reactor fuel with larger diametral gaps, especially in the 33- to 50-kW/m (10- to 15-kW/ft) operating power range. Experiments involving instrumented fuel assemblies (IFAs) and sponsored by NRC have been reported in References 19 and 20.* These experiments have been run in the Norwegian Halden Boiling Water Reactor under the technical management of EG&G Idaho and Battelle-Pacific Northwest Laboratory (PNL).

Reference 19 presents test data from the EG&G Idaho-Halden experiment IFA-429. The IFA-429 is an 18-rod test assembly designed to study fission-gas release and fill-gas absorption in prepressurized (2.58-MPa helium) PWR-type fuel rods. This data report presents assembly power history and individual fuel-rod power, temperature, pressure, and burnup data from June 1975 through June 1978. Reported fuel-rod heat ratings cover a range from 17 to 30 kW/m with measured fuel centerline temperatures of 1375 to 1475 K (2015 to 2195°F) for the highest power. Measured fuel-rod pressures showed no appreciable change during the period covered.

Results from PNL-Halden experiment IFA-431 were reported in Reference 20 and analyzed in subsequent reports. For one of the rods in this experiment, the average gap conductance uncertainty over the range of measurement was ± 19%. The uncertainty in the gap-conductance measurement changed as a function of linear heat rating. (See Fig. 8). The experiment includes two fill gases (pure xenon and pure helium) and three pellet-to-cladding gaps. The absolute error in determining the temperature drop across the gap is less than 100°C for any of the combinations of gap diameter and fill gas used.

Power Burst Facility

The Power Burst Facility (PBF) is a water-cooled and water-moderated reactor, contained in an open-top steel vessel and is used for in-reactor tests of fuel rods. (See Figure 9). It is operated for the U.S. Department of Energy and the NRC by EG&G Idaho, Inc.

The reactor core is designed for both steady-state and pulsed-mode operation. One to twenty-five test fuel elements with an active length no greater than 91 cm are fitted into a test train together with the necessary test instrumentation. The assembled test train is then fitted into a pressurizable thick-walled metal cylinder 15.5 cm in diameter (the inpile tube or IPT). The IPT is mounted vertically and concentric to the vertical axis of the reactor core and the containing vessel.

The IPT has six to eight openings, permitting the use of up to 100 pairs of instrumentation test leads. Typical test instrumentation includes inlet and/or exit flow meters (up to five per test); absolute- and differential-pressure transducers for monitoring fluid and fuel-element plenum pressures; surface and internal thermocouples for monitoring fuel, cladding, plenum, and coolant temperatures; ultrasonic thermometers; linear variable differential transformers (deflection indicators); radiation-flux monitor wires and foils; and self-powered neutron detectors. Suitable instrumentation, signal-conditioning equipment, and data-accumulation and data-reduction equipment and services are available.

The PBF test program includes the following areas: (1) power-cooling mismatch, with both unirradiated and pre-irradiated fuel rods (16 tests); (2) LOCA with both unirradiated and preirradiated fuel rods (10 tests); (3) flow blockage, with previously unirradiated fuel rods (3 tests); (4) reactivity-initiated accident, with both fresh and pre-irradiated fuel rods (12 tests); and (5) gap conductance (stored energy--7 tests).

Table 2 is a summary of the PBF tests conducted to date. The PBF test series may be described as follows:

Power-Cooling-Mismatch Tests. These tests study the critical-heat-flux (CHF) and post-CHF behavior of single rods (four at a time) and nine-rod clusters under a variety of power and cooling conditions, in which CHF is achieved either by increasing the fuel-rod power at a steady coolant flow, or by decreasing the coolant flow at a steady fuel-rod power, or by simultaneously decreasing the coolant flow and increasing the fuel-rod power. (To date, only the specific combinations of final fuel-rod power level and final flow rate appear to be important.) These tests also study the effects of irradiation and burnup on the thermal-mechanical properties of fuel-rod components (particularly claddings).

Coolant flow, stored energy, and test-termination temperatures and post-CHF cladding deformation are among the test variables measured.

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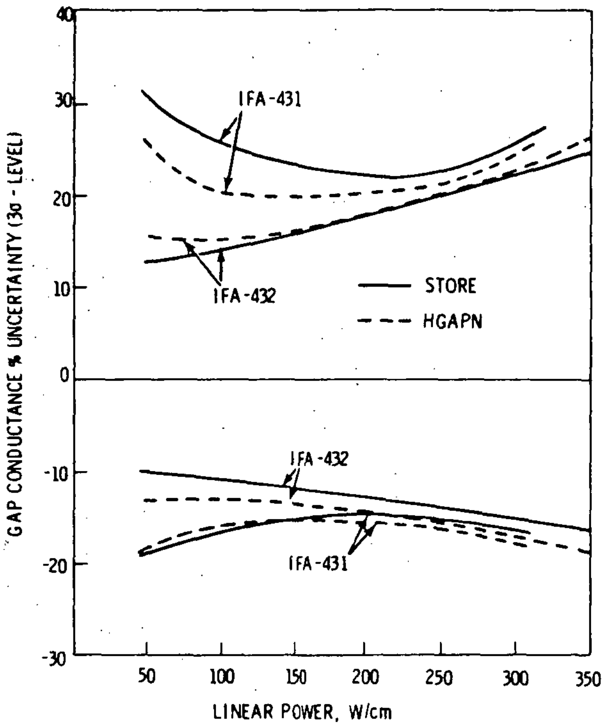
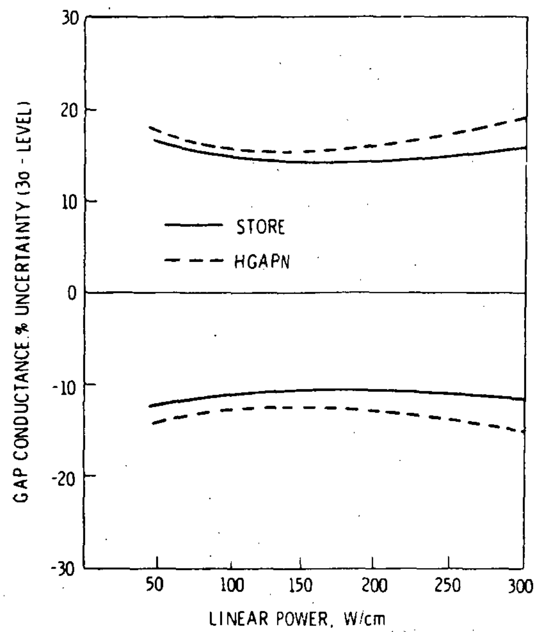
*A report has recently been published summarizing data on mixed oxide fuel rods irradiated in the Halden reactor: Fuel Rod Temperature and Pressure Response in Halden Reactor Experiment IFA-226 by P. E. MacDonald et al., NUREG/CR-0267 (August 1978).



Gap Conductance Uncertainty Bounds for Rod 1, 0.0229 cm Initial Gap



Gap Conductance Uncertainty Bounds for Rod 2, IFA-431, 0.0381 cm Initial gap

FIGURE 8. GAP CONDUCTANCE UNCERTAINTY BOUNDS

(REF: PNL-2581)

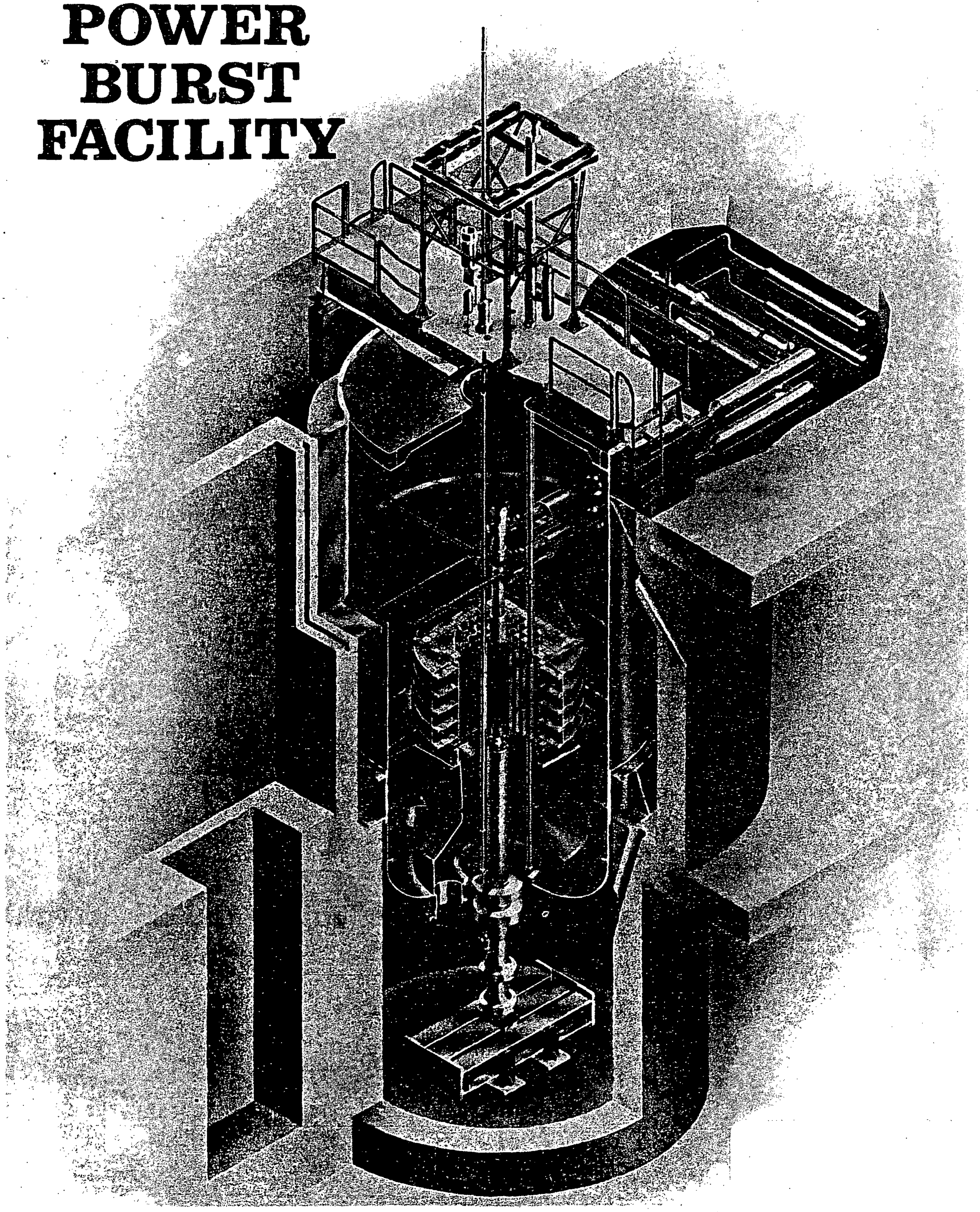


FIGURE 9. POWER BURST FACILITY (PBF)

TABLE 2

PROGRAMMATIC TESTS PERFORMED IN THE POWER BURST FACILITY

Type of Test	# of Tests	Total Rods	Test Conditions	Results
Power Cooling Mismatch	8	17	DNB at PWR pressure & temperature	Basis for failure prediction at high cladding temperatures; brittle failures predictable from ex-reactor criteria; almost complete oxidation at power to get failure
PCM-Cluster	1	9	3 x 3 rod array	Similar to single rod tests; 2 rods failed at power or during cooldown
Loss of Coolant Accident	5	4	PWR blowdown from 67 kW/m (20 kW/ft)	Good blowdown heat transfer caused low cladding temperatures
LOFT-LOCA	3	12	Simulate LOFT L2 series (from 53 kW/m)	No cladding collapse; temperatures lower than predicted
Gap Conductance	4	16	BWR rods with various gases & gaps	Steady state & transient data for code development
PCM - Irradiation Effects	6	21	PWR rods from Saxton Reactor	Similar behavior to unirradiated tests; some fission-gas-induced swelling
Reactivity Initiated Accident	6	12	Energy input up to 300 cal/gm	Failure thresholds near prediction

The objective of the power-cooling-mismatch test, PCM-1, was to simulate the worst possible PCM incident leading to fuel rod failure at power with molten fuel-coolant interaction (MFCI) when failure occurred. The critical heat flux was surpassed during a power ramp from 40 kW/m to 78.7 kW/m. The fuel rod operated approximately 8 minutes after the onset of film boiling (DNB) before it failed. After failure, the fuel rod power was maintained at 78.7 kW/m for approximately 7 minutes, at which time the reactor was scrammed. Following the scram, the central portion of the fuel rod disintegrated. No violent MFCI (vapor explosion) occurred as a result of the PCM-1 fuel rod failure even though over 50% of the UO_2 fuel at the hot-spot was molten at the time of rod failure. The oxidation of the cladding at the time of failure was considerably in excess of present licensing criteria for rod failure.

The first 9-rod cluster power-cooling-mismatch test, PCM-5, was conducted in May 1978. The test consisted of 9 PWR-type fuel rods held in a 3x3 cluster with spacing typical of a 15x15 PWR lattice. The objective of PCM-5 was to study the film boiling behavior of a central rod when surrounded by other rods also in film boiling. Cladding temperatures for the center rod during the transient were in the β-phase Zircaloy range ($T > 1245$ K). The corner rod, which was held in DNB for about 11 minutes, failed about 5.5 minutes after DNB occurrence. The center rod entered film boiling 4 minutes after the corner rod and was subjected to film boiling for approximately 4 minutes. Seven of the fuel rods achieved stabilized film boiling, and four of the nine rods failed at power or during the rapid post-test shutdown. Two of the rods did not experience DNB (see Figure 10).

Recently a review[22] has been completed of the available literature on the operation of nuclear fuel rods under film boiling or dryout conditions. The following material is taken directly from that review.

Test fuel rods from both pressurized water reactors and boiling water reactors were subjected in-reactor to a combined total of more than 170 power ramp cycles and more than 250 flow coastdown cycles in order to force the rods into film boiling or dryout. Only 13 of the 667 tests rods failed and these 13 failed in a manner consistent with an ex-reactor time-at-temperature brittle-ductile boundary curve. This relationship shows that long times at very high temperatures are required for failure.

Figure 11 is a plot of the reactor test data discussed in sections 3 and 4 and summarized in Tables 1, 2 and 3 of Reference 22. The cladding peak equivalent temperatures are plotted against the logarithm of the time-at-equivalent-temperature. The unfailed rods are shown as open circles, squares and triangles and the rods which were oxidized severely enough to fail are shown as filled circles, squares and triangles.

The studies of Chung, Garde and Kassner[23] at Argonne National Laboratory have shown that the severe shock of quenching the hot oxidized 0.6 mm thick Zircaloy cladding is roughly equivalent to a 0.03 Joule impact at 300K (See Figure 5). These studies have also shown that if oxidized cladding can withstand a 0.3 Joule impact at 300K, the cladding is still tough enough to withstand both operating stresses when at working temperature and the mechanical stresses of handling when at room temperature. This is true despite the fact that the ANL test rods have effectively the same hydrogen pickup from the hydrogen released by the zirconium-steam reaction as would be picked up after clad rupture in an accident where reduced coolant pressure occurs, e.g., a LOCA. This level of hydrogen pickup is higher than would be expected for most abnormal operating transient accidents.

Therefore, the thermal shock curve and the 0.3 Joule at 300K impact curve represent brittle-ductile boundary curves between the elevated isothermal exposure temperatures required to produce brittle behavior of cladding and those isothermal exposure temperatures which will not cause brittle behavior either during or after a given exposure time.

In Figure 11 the boundary for brittle vs. ductile behavior for shocked, oxidized Zircaloy from the out-of-pile isothermal time-at-temperature studies of Chung, et al.[23] is plotted against the reactor test data for failed and unfailed rods.

It is apparent from Figure 11 that, for any selected exposure time, the required equivalent temperature for in-reactor embrittlement of both fresh and pre-irradiated Zircaloy cladding is effectively the same as the required isothermal temperature for out-of-pile embrittlement of that same cladding. Therefore, the equivalent temperature concept for identifying probable embrittlement by the non-isothermal accident time-temperature profile seems to be valid.

Based on the good agreement of the out-of-pile data on unirradiated cladding and the in-pile data on both fresh and pre-irradiated cladding it would appear that brittle-ductile boundary curves generated out-of-pile can then be used with reasonable confidence to predict cladding oxidation embrittlement and failure which might be caused by reactor upset and accident conditions particularly where good models exist for predicting peak cladding temperatures and times-at-temperature.

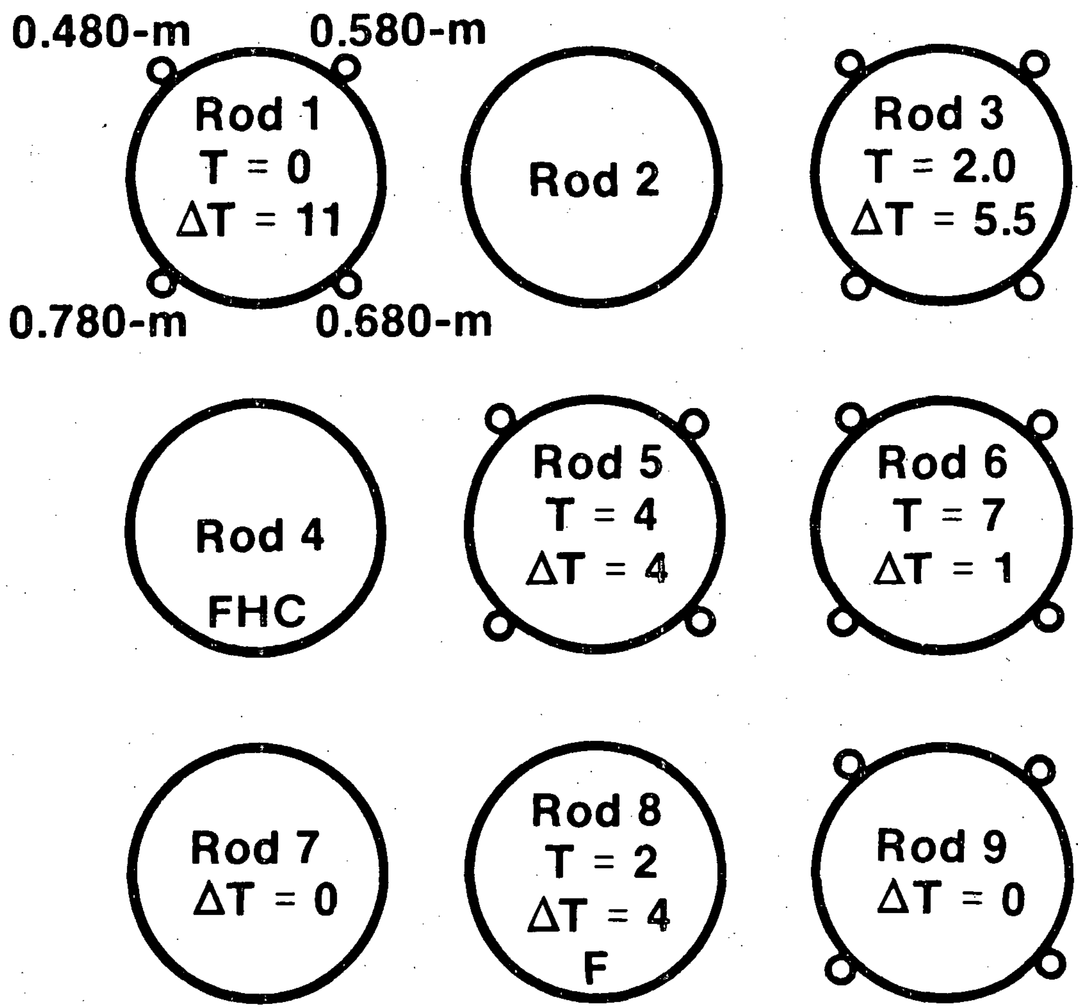
Both the in-pile and out-of-pile test data show that cladding integrity is generally maintained despite exposure to very severe accident environments.

Projected equivalent times-at-temperatures (see Figure 12) for most BWR and PWR reactor transients in commercial reactors are far below the values of time-at-temperature required for brittle failure by oxidation which were observed in the reported tests. Therefore, it would appear that fuel rod cladding should not fail by oxidation embrittlement either during or after most of the PCM-related DNB or dryout events identified to date.

LOCA Tests. These tests will study fuel-rod behavior, e.g., cladding deformation and oxidation of single-rod (four at a time) assemblies under blowdown conditions. Parameters to be varied include irradiation history and cold internal pressures. Sixteen-rod clusters will be tested under heatup conditions. Results will be correlated with those of out-of-reactor tests.

Fig. 10

Rods in Film Boiling During Test PCM-5



(REF: "RESULTS OF THERMAL FUELS BEHAVIOR PROGRAM RESEARCH DURING FY-1978" BY H. J. ZEILE, SIXTH NRC WRSR INFORMATION MEETING 1978, NRC PDR)

Comparison of In-Reactor Post-DNB Survival/Failure Data for Zircaloy Cladding with FBRB/ANL Zircaloy Ductile-Brittle Boundary Curve

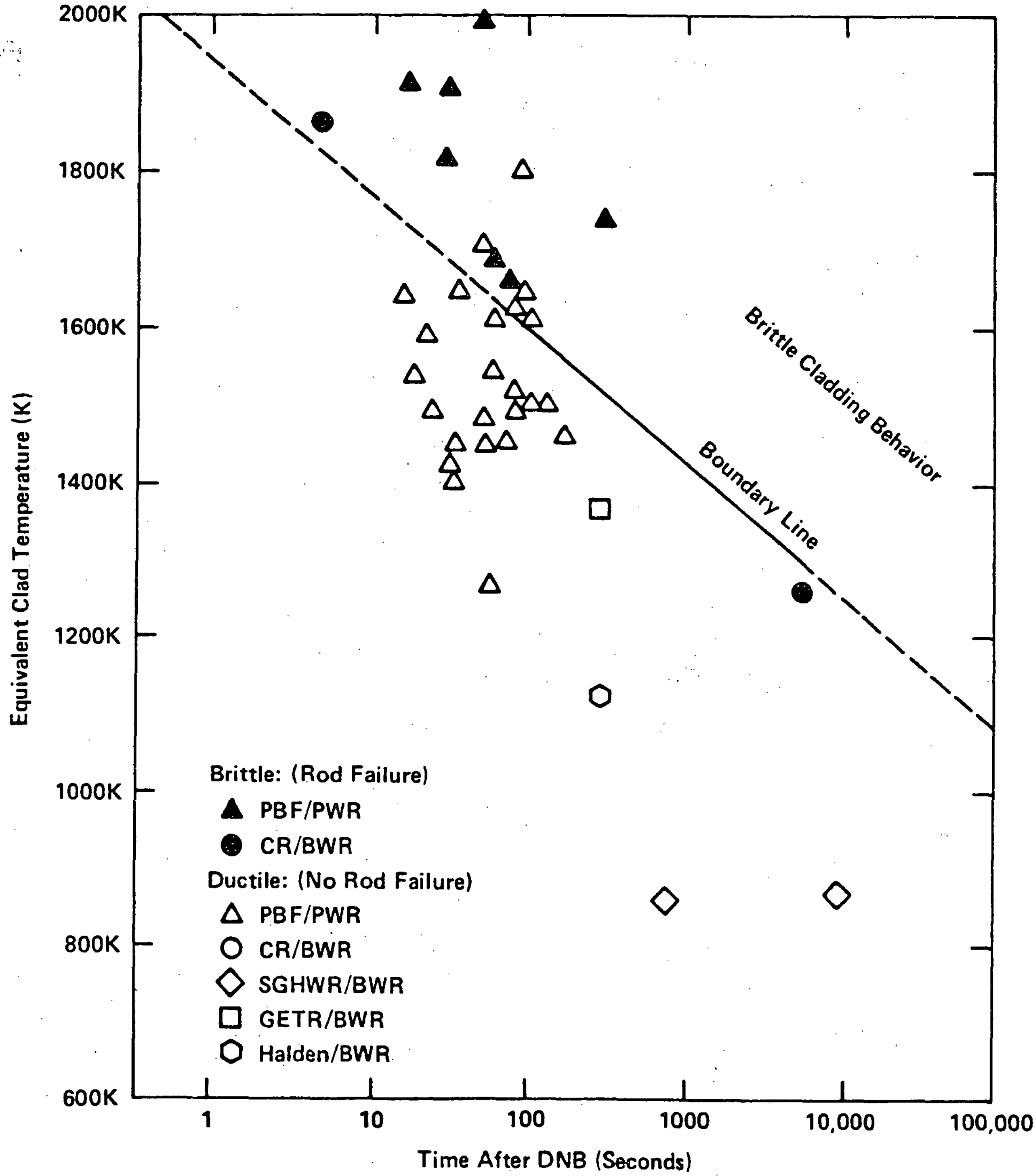


FIGURE 11

(REFERENCE: NUREG-0562)

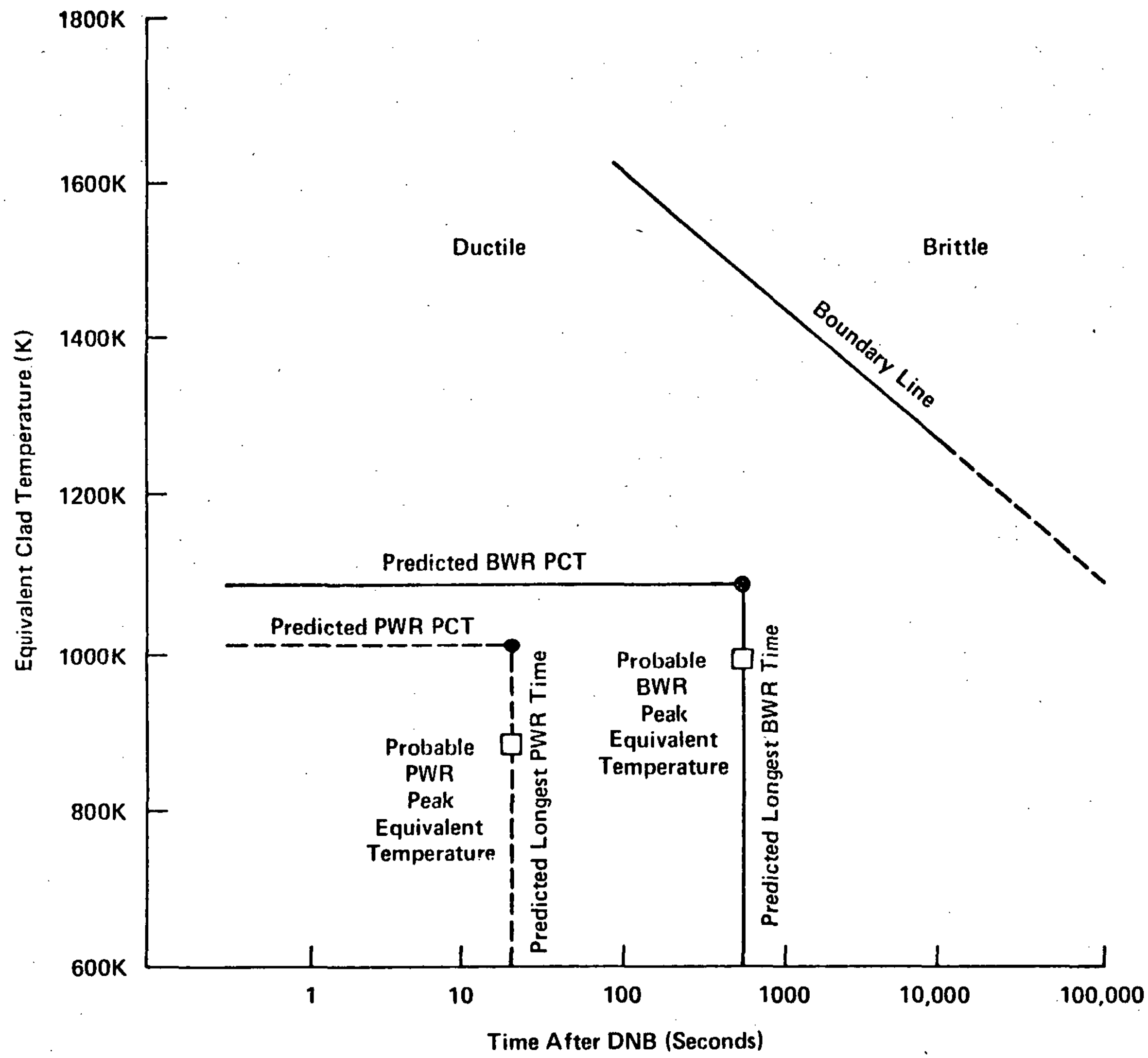


FIGURE 12. COMPARISON OF PREDICTED BWR AND PWR UPSET CONDITION PEAK CLADDING TEMPERATURE AND TIME AFTER DNB WITH NRC FUEL BEHAVIOR RESEARCH BRANCH/ ARGONNE NATIONAL LABORATORY ZIRCALOY DUCTILE-BRITTLE BOUNDARY CURVE (REFERENCE: NUREG-0562)

Fuel behavior LOCA testing in the PBF was started in January 1978 when the world's first nuclear blowdown tests from PWR initial conditions, LOC-11A, LOC-11B and LOC-11C, were conducted sequentially with four separately shrouded PWR 15x15 design fuel rods.[24] Calculations of coolant behavior generally agreed well with the measured coolant behavior, but the calculated cladding surface temperatures were slightly greater than measured (see Figure 13). None of the four test fuel rods failed. Two of the fuel rods were essentially unpressurized and the other two fuel rods were prepressurized at ambient temperature to about 2.42 and 4.81 MPa. The cladding of the unpressurized rods collapsed very slightly and the cladding of the pressurized rods ballooned very slightly. The cladding deformation observed during the LOC-11 tests is very well calculated with available high temperature Zircaloy plastic deformation models.

Flow-Blockage Tests. These tests will study fuel-rod behavior, e.g., cladding temperatures and geometric profiles of multiple-rod assemblies (25 rods) under flow blockages of 80 to 98%.

Reactivity-Initiated Accident Tests. These tests will study the behavior of irradiated and unirradiated fuel rods under rod-drop and rod-ejection conditions. Independent rod tests, cluster tests, and model development/ evaluation tests will be performed. The effects of irradiation, cluster size, coolant flow, and initial power level will be studied.

A series of RIA tests was begun in PBF to determine the behavior of the fuel rods under rapid power burst conditions that could be caused, for example, by a control rod ejection from the core of a power reactor. In a power burst test the PBF reactor is given a sudden increase in reactivity by the ejection of fast moving control rods from the core, which causes a power burst to be initiated. The first RIA test ever conducted at power reactor temperature, pressure and flow conditions, RIA-ST, was a scoping test performed to (1) identify the energy deposition failure threshold for BWR hot-startup conditions, (2) evaluate calorimetry techniques for RIA transient tests, and (3) determine if sizable pressure pulses would result from fuel failure in a water-filled system. The RIA-ST test was comprised of five single rod tests. The first three tests addressed the determination of the failure threshold and the evaluation of calorimetry techniques, and the final two tests were performed at high energy depositions to determine pressure pulse magnitudes.

Prior to performance of the first RIA test, forty power burst tests were performed to determine the dynamic characteristics of the PBF core for use in RIA tests and to qualify the core down to a burst period of 1.6 milliseconds with an associated peak power of 92 GW.

Figure 14 compares the calculated and measured cladding surface temperatures for one of the RIA tests. Figure 15 shows the influences of fuel burnup on failure during an RIA.

Gap Conductance (Stored Energy) Tests. These tests study the gap conductance and stored energy of irradiated and unirradiated rods. Parameters measured include irradiation history, gap size, fill-gas pressure, and pellet densities. Power oscillation (transfer function) and integral k dt gap-conductance measurement methods are being compared.[21]

In Reactor Tests at Other Facilities

The Power Burst Facility is particularly useful for the in-reactor testing of LWR fuels under abnormal operating conditions. Other reactor safety test facilities also provide useful data. Accordingly, complementary fuel testing programs are performed or planned at a number of other facilities, including:

- The LOFT Facility at the Idaho National Engineering Laboratory
- Nuclear Safety Research Reactor (NSRR) in Japan
- The FR-2 reactor in the Federal Republic of Germany
- The Halden experimental boiling water reactor in Norway
- The ESSOR/SARA test loop in Italy
- The PHEBUS reactor in France
- The NRU reactor in Canada
- The BR-2 reactor in Belgium

Fuel Behavior Computer Codes

Fuel behavior codes must analyze the thermal, mechanical, and internal gas response of fuel-rod components with the goal of predicting rod condition and integrity. Modeling of thermal behavior during normal and accident conditions must include the surface heat transfer, heat transfer across the fuel-to-cladding gap, the thermal conductivity of fuel and cladding, the power generation distribution in the fuel, and the solution of the conduction equation. These aspects of the thermal calculations are listed approximately in order of their importance.

Modeling the mechanical response of fuel rods involves consideration of fuel-cladding mechanical interactions (FCMI); cladding creep, ballooning, and failure; fuel thermal expansion, swelling, densification, and creep. The phenomena that are important in steady-state operation (creep, swelling, etc.) significantly affect behavior during transients. The transient codes must therefore in some manner consider these phenomena, either by direct calculation or by linkage to a steady-state code. These response phenomena are, of course, coupled to one another as well as to the thermal behavior factors. One very important parameter that is difficult to calculate because of this strong coupling is the fuel-to-cladding gap width.

Modeling of the internal gas response is important for determining the loading that it applies to the cladding and for determining heat transfer across the fuel-to-cladding gap. The key modeling areas associated with these effects are axial gas flow, fission-gas release, plenum gas temperature, and voids and void temperature.

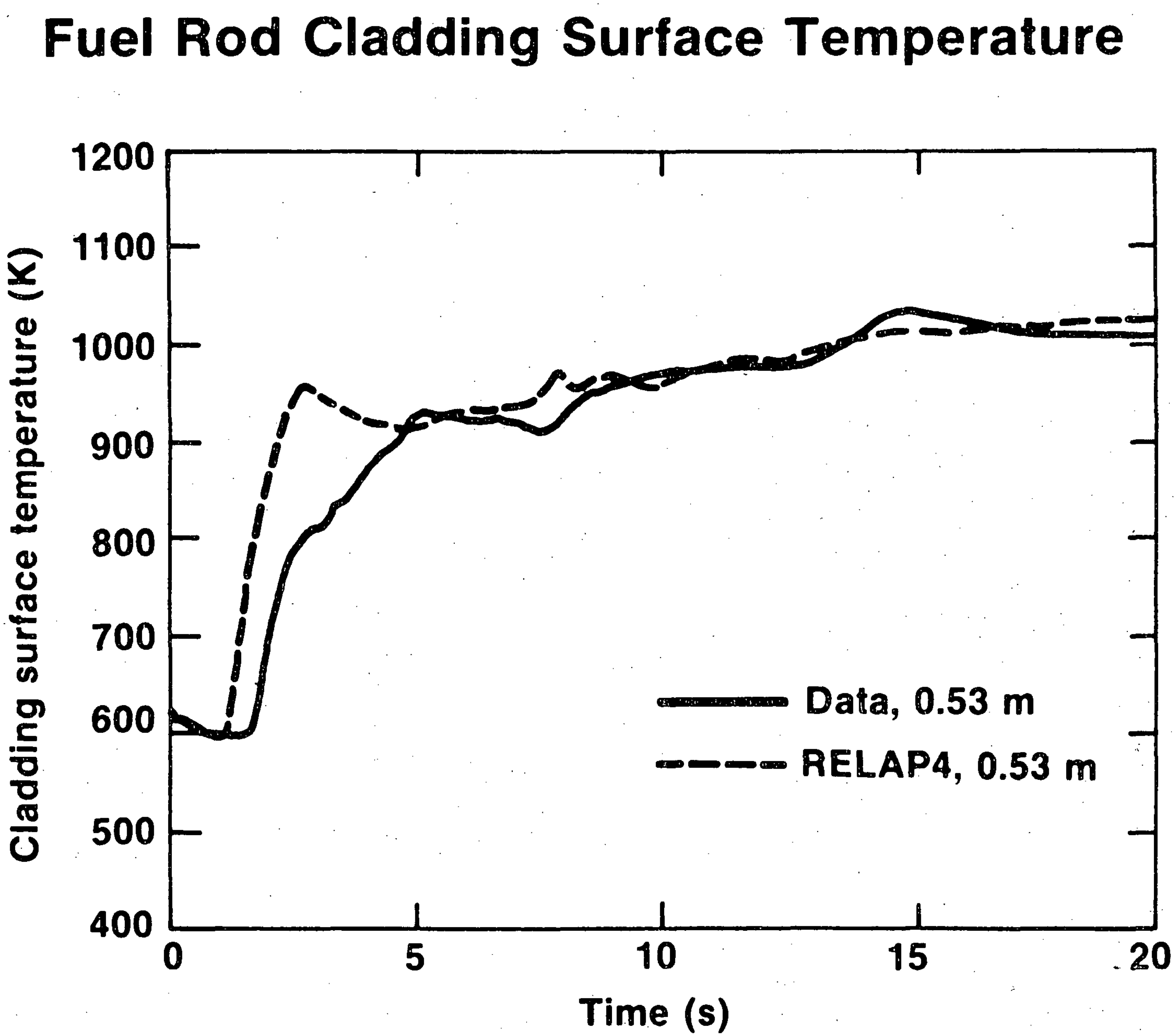


FIGURE 13. PBF-LOCA TEST LOC-11C (REFERENCE: NUREG/CR-0618)

Cladding Surface Temperature
Rod 2-1
0.46 m - 180°

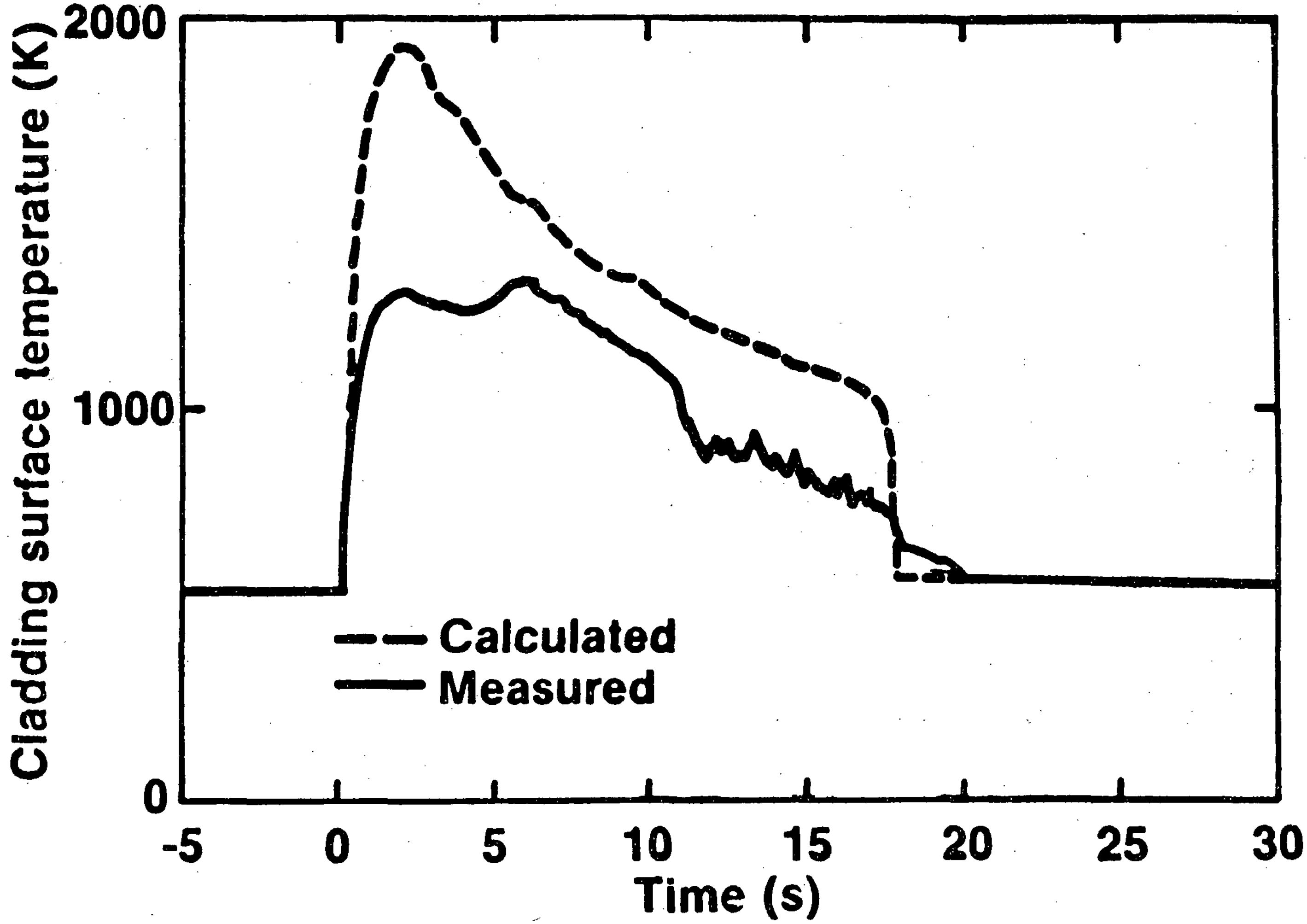


FIGURE 14. COMPARISON OF MEASUREMENT WITH PREDICTION FOR REACTIVITY INITIATED ACCIDENT TEST RIA 1-2 CONDUCTED IN THE POWER BURST FACILITY (REFERENCE: NUREG/CR-0765)

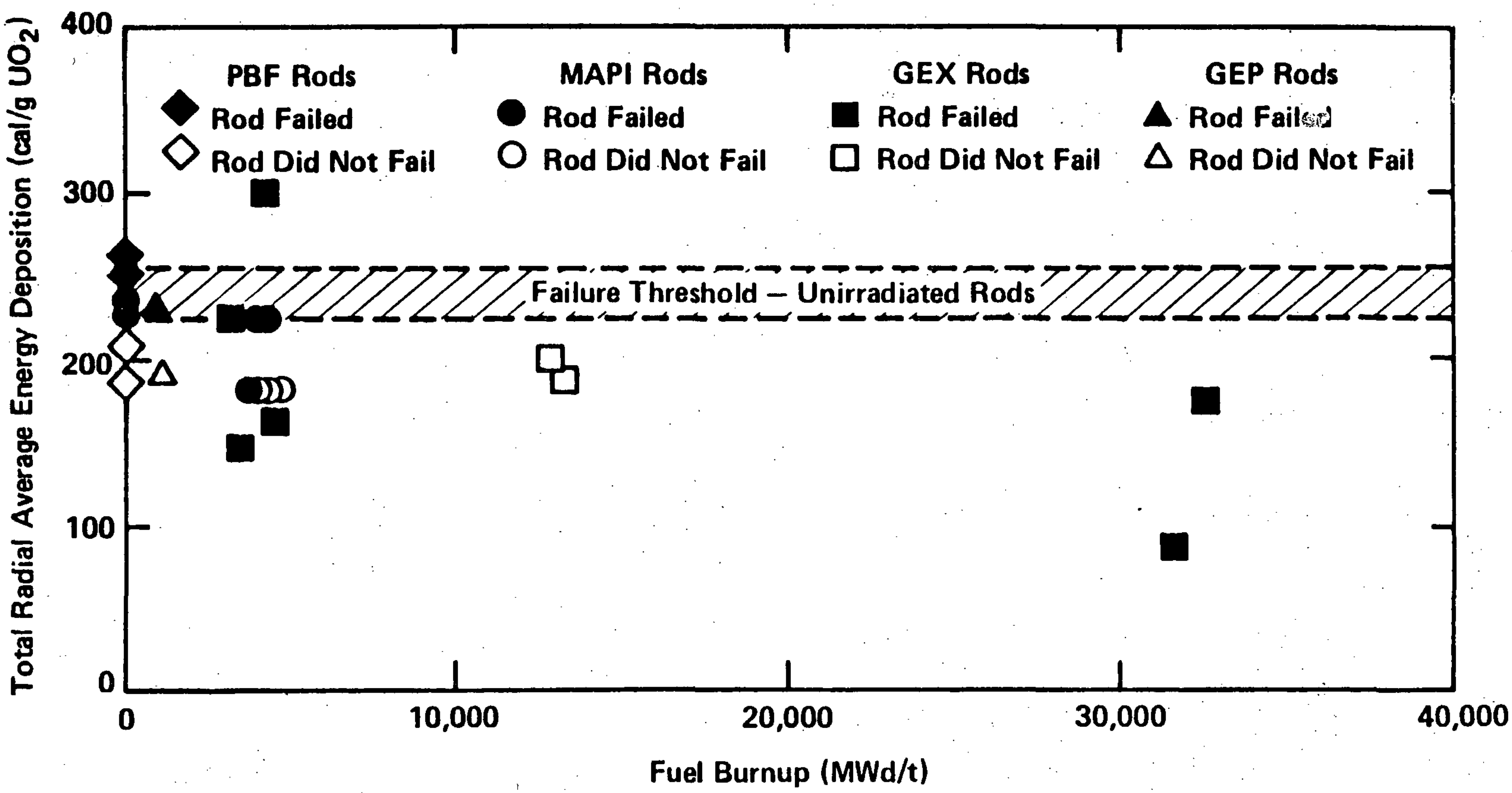


FIGURE 15. INFLUENCE OF FUEL BURNUP DURING RIA TESTS

(REFERENCE: NUREG/CR-0269)

The NRC Fuel Behavior Research Branch is sponsoring the development of two fuel behavior codes FRAP-S (Fuel Rod Analysis Program - Steady State)[25] and FRAP-T (Fuel Rod Analysis Program - Transient).[26] Recently NRC decided to combine FRAP-S with the code GAPCON-THERMAL-3[27] to produce the combined steady-state code FRAPCON. [28]

The steady-state code was developed for use as a normal-operation analysis tool and as the generator of burnup-dependent initial conditions required for the FRAP-T (transient) code. FRAP-S/FRAPCON seeks to model all of the important phenomena involved in nonaccident situations during the life of LWR fuel rods. It iteratively calculates the interrelated effects of fuel and cladding temperature, rod internal pressure, fuel and cladding elastic and plastic deformation, release of fission-product gases, fuel swelling, cladding growth resulting from irradiation, cladding corrosion, and crud deposition, all as a function of time and specific power.

FRAP-S3 included a number of features such as failure prediction models, sophisticated fuel-cladding interaction models, and a package of material properties.* Frequent and independent assessment studies are also performed and published as a guide to code users and code developers. One study[25] found that improvements incorporated since the previous version (including a new treatment of fuel pellet-relocation and related fuel thermal conductivity effects) have resulted in a more realistic description of fuel behavior under moderate operating conditions. The thermal and mechanical model development generally relates to the response regimes associated with highest power rods as opposed to core-average rods. FRAP-S3 now yields a standard fuel centerline temperature error of 198 K for unpressurized rods and 254 K for pressurized rods. (See Table 3). These discrepancies approach the present experimental uncertainties, however, and are of the same magnitude as those experienced by the other codes. Figure 16 shows a comparison of FRAP-S centerline temperature predictions to measured data obtained from a Halden fuel rod. Future research work is aimed at reducing these experimental uncertainties and to create a corresponding improvement in modeling capability.

FRAP-T is being developed to calculate the temperature increases and the accompanying time- and temperature-dependent processes expected during postulated occurrences such as LOCA's, power-cooling-mismatch accidents, reactivity-initiated accidents, and inlet flow blockage, as well as the processes occurring during normal operation. The code will eventually be general enough to treat expected asymmetries and all of the phenomena occurring up to and including fuel melt.

The models for FRAP-T have been developed primarily from fundamental formulations so that they will apply over a wide range of response conditions and will not be limited by the range of available data. FRAP-T predicts the time dependence of many coupled variables at an arbitrary number of axial positions for any transient power history. Calculated are the fuel-rod temperature distribution, gap conductance, internal pressure, cladding strain, time and location of cladding failure, cladding surface temperature (including surface heat transfer), and coolant conditions (including temperature, enthalpy, and quality). The primary input data required are the following: power history, descriptions of the fuel-rod cold state, the time-dependent conditions of the coolant surrounding the rod, the axial power profile, and code running requirements, including the mesh size, time step, and convergence criteria for pressures and temperatures. The results of either a steady-state or an earlier transient calculation may, of course, be stored on tape and read by FRAP-T to satisfy these input requirements. FRAP-S/FRAPCON or FRAP-T may be used for this purpose, and transient coolant conditions calculated[29] with RELAP-4 may be read from a tape. The output may optionally include plots of up to 20 variables as a function of time.

Both FRAP-T and FRAPCON have and will continue to undergo rigorous independent assessment to determine how well they predict experimental results. A summary of the standard model errors for FRAP-T4 is given in Table 4.

Data comparisons[30] for two TREAT LOCA simulation tests were performed using FRAP-T4 to evaluate the effect of relocation and ballon model changes on cladding temperature and rod internal pressure calculations (see Figure 17). Surface temperature response under single-phase steam cooling conditions was well represented by the model. Relocation effects on gap volume redistribution contributed to the overprediction of pressure for the relatively large gap, small plenum rod. An adequate representation of measured pressure response was obtained for the larger plenum rod, which is more typical of power reactor void volume conditions. Incorporation of strain rate dependence in the burst model improved predictions for time and duration of cladding rupture.

Versions up through FRAP-S3, FRAPCON-1 and FRAP-T4 are currently available for use from the National Energy Software Center at Argonne National Laboratory. FRAP-T5[31] is under development.

Fuel Meltdown

As a result of the Reactor Safety Study[32] analysis of a core meltdown, uncertainties in some of the physical phenomena were reexamined more closely. A review of available experimental data[33] concluded that in general a sizable body of useful data exists, but the experimental conditions are usually such that the results are not always directly applicable to the case of a reactor meltdown. The report also suggested that sensitivity studies using state-of-the-art models of meltdown should be conducted to determine the importance of physical phenomena in relation to the overall consequences of the postulated meltdown accident.

As a result, programs were instituted to examine fission-product behavior during meltdown, natural convection in molten pools, interactions between molten core materials and concrete, steam explosions, and the effects of these phenomena on meltdown probabilities and consequences. Close cooperation with a similarly oriented program

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* G. A. Reymann (ed.), MATPRO-Version 10, A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, USNRC Report TREE-NUREG-1180, February 1978.

TABLE 3

Standard Model Errors

FRAP-S3

Output parameter	Sample (rods/pts)	Standard error $[(\sum_{i=1}^{n} (P_i - M_i)^2/n-1)^{0.5}$
Fuel center temperature	33/290	254 K (pressurized)
Fuel center temperature	64/511	198 K (unpressurized)
Released fission gas	176/176	18.8% generated gas
Rod internal pressure	28/309	0.66 MPa (unpressurized)
Rod internal pressure	20/349	1.34 MPa (pressurized)
Gap closure heat rating	77/77	13.4 kW/m (local)
Fuel axial thermal expansion	19/173	0.37% active length
Fuel axial permanent expansion	100/368	0.44% active length
Cladding hoop permanent strain	170/393	0.59% cladding OD
Cladding axial permanent strain	115/161	0.47% active length
Cladding surface oxide layer	48/84	6.6 μ
Cladding hydrogen concentration	38/53	39 ppm

(REFERENCE: "INDEPENDENT FRAP-T4 ASSESSMENT" BY D. R. COLEMAN, PRESENTED AT SIXTH NRC WRSR INFORMATION MEETING, 1978, NRC PDR)

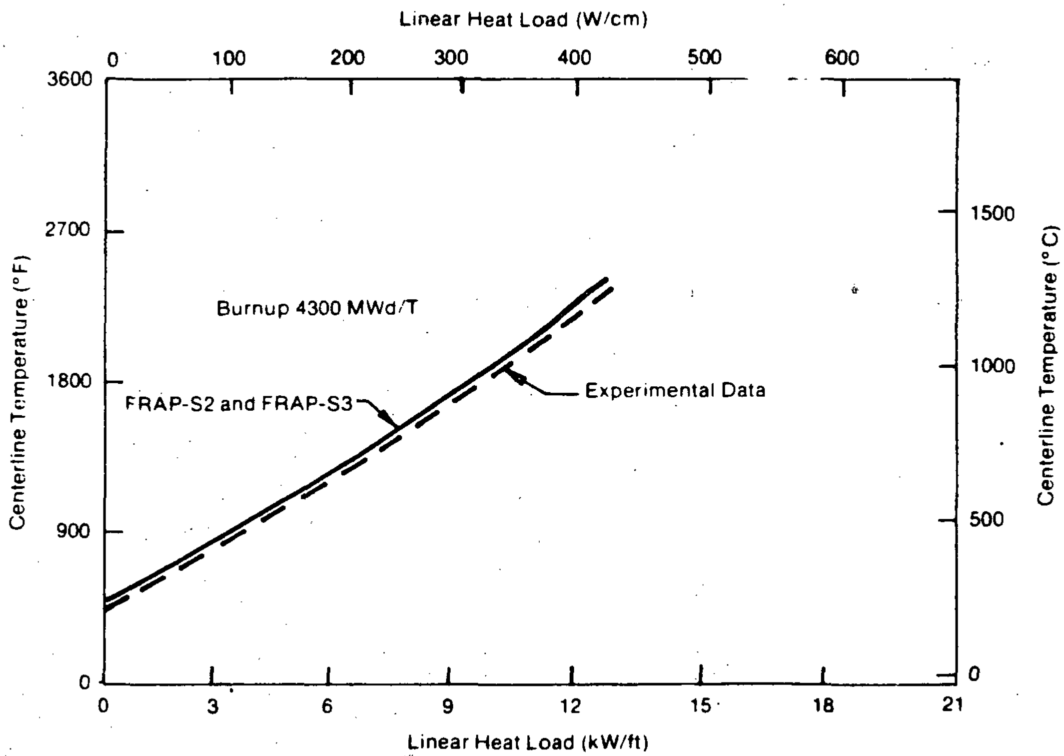

FIGURE 16. FRAP-S CENTERLINE TEMPERATURE PREDICTIONS COMPARED TO HALDEN ROD HBB END OF LIFE DATA

(REF: TFBP-TR-172, JANUARY 1977)

TABLE 4

Standard Model Errors

FRAP-T4

Output parameter	Sample (rods/pts)	Standard error $[(\sum_{i=1}^{n} (P_i - M_i)^2/n-1)^{0.5}$
CHF power at known flow	18/78	0.06 kW/CC channel
CHF flow at known power	18/78	400 kg/s-m^2
Initial fuel center temperature at SCRAM	21/32	280 K
Fuel thermal decay constant during SCRAM	21/32	4.5 s
Equilibrium fuel center temperature during SCRAM	21/32	54 K
Cladding burst temperature at known pressure	158/158	290 K
Cladding burst pressure at known temperature	64/64	3.4 MPa
Cladding permanent hoop strain	370/370	57% cladding OD

(REFERENCE: "INDEPENDENT FRAP-T4 ASSESSMENT" BY D. R. COLEMAN, PRESENTED AT SIXTH NRC WRSR INFORMATION MEETING, 1978, NRC PDR)

Internal Pressure and Cladding Surface Temperature Response for Treat Test FRF-2

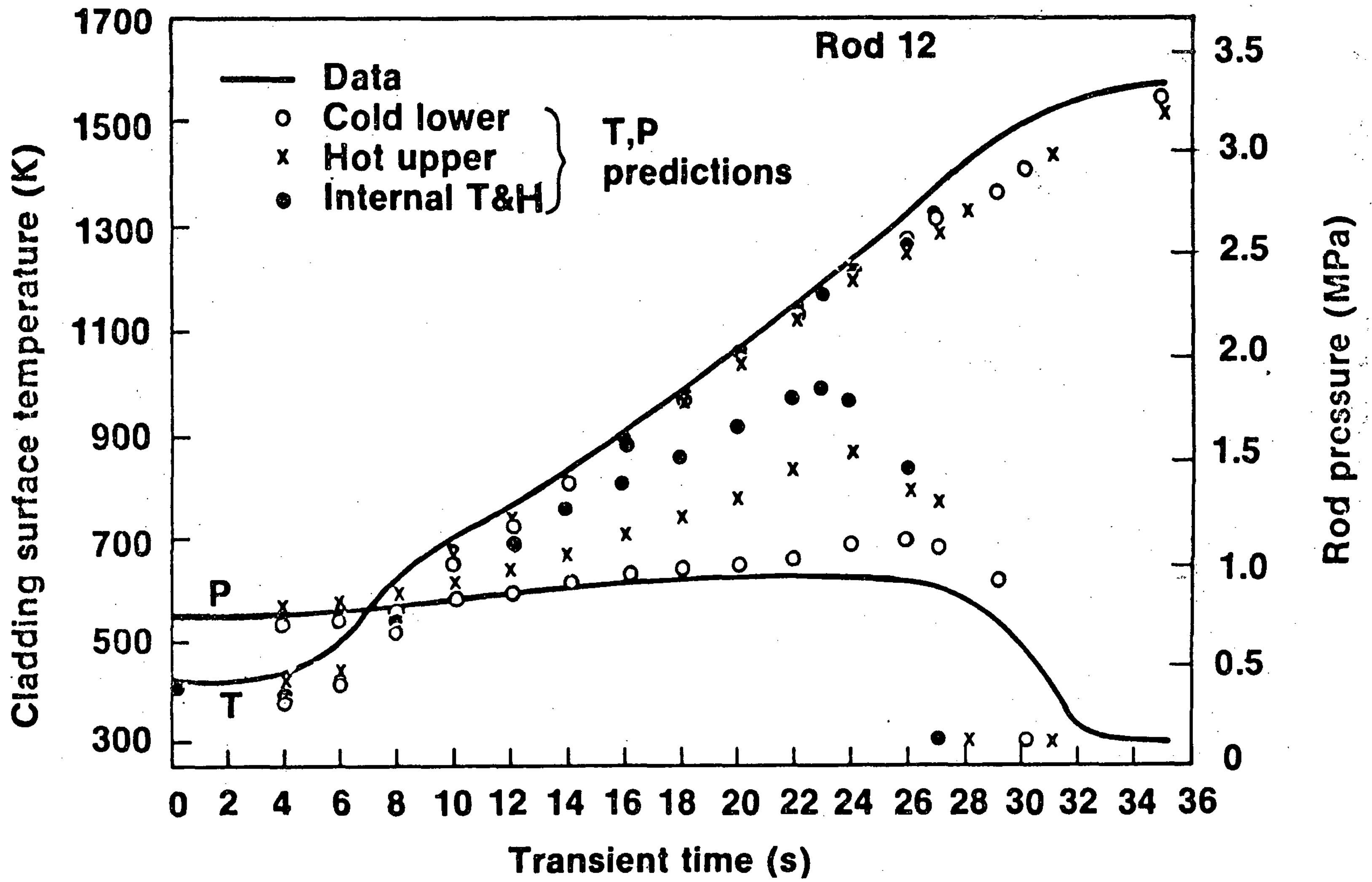

FIGURE 17

(REF: "INDEPENDENT FRAP-T4 ASSESSMENT" D. R. COLEMAN, SIXTH NRC WRSR INFORMATION MEETING) in the Federal Republic of Germany has also been instituted. Some of the key findings to date include the following:

- Three series of experiments conducted at three separate laboratories indicate that elevated system pressures are effective in reducing the probability of vapor explosion.[34]
- A relatively long time is required to establish steady-state single-phase convection in molten pools at high Rayleigh numbers.[35]
- Concretes with high and low carbonate contents behave in a qualitatively similar manner on contact with molten core materials.[36]
- Concrete penetration is thermally dominated (1 cm/min is a nominal value).[36]
- Steel in the melt will chemically reduce decomposition gases to carbon monoxide and hydrogen, which will then burn on contact with air.[36]
- The initial heat transfer is dominated by the effects of concrete decomposition gases which tend to keep the melt thermally well mixed.
- Single-phase natural convection is not a dominant heat-transfer mechanism during core-concrete interactions.[36]
- A first generation mechanistic model of core-concrete interactions has been developed[37] and an improved model is now undergoing assessment testing.

Significant progress has been made in identifying and characterizing these phenomena, but several key issues remain to be resolved. These include:

- Directional partitioning of heat flux during interactions between concrete and molten core materials.
- Probabilities associated with steam explosions.
- Necessary scale of steam explosion experiments.
- Prediction of containment failure modes.
- Leakage of radioactivity from containment.

Fission-Product Release and Transport

Accurate estimates of fission-product inventory in a reactor core can be made from a knowledge of the operating history. However, the amount and form of radionuclides released during a postulated accident can only be inferred by extrapolation of limited experimental data and application of simplified analytical models. Experimental investigations have been conducted primarily out of reactor on small samples (1 to 100 g) of relatively low burnup fuel (trace to 4 GWd/MTM) with rare instances of higher burnup (up 20 GWD/MTM). Current practice is to make what are judged to be conservative assumptions[38,39] regarding fission-product release from the fuel and transport through the various barriers without detailed consideration of the mechanisms of release and transport. Greater emphasis has recently been placed on instrumenting commercial and test reactors so that releases of activity from the fuel can be monitored with greater reliability and accuracy. Out-of-reactor experiments[40,41] using irradiated fuel under more controlled conditions are elucidating the mechanisms of release and transport. In-reactor experiments monitoring releases of activity during normal operating conditions as functions of power and of defect size for rods intentionally made defective are under way in Japan and France. The development of mechanistic models to trace the path of fission products once they leave the fuel is also in progress.[42]

Research to date on the behavior of airborne fission products has involved the use of scaled facilities having up to about 1% of the volume of the containment in a typical presssurized water reactor. This research has examined atmospheric conditions, airborne concentrations, physical and chemical states of various species, mechanisms of fission-product removal from the airborne state, and thermal effects. Particular attention has been paid to iodine, and one outcome of these investigations has been the establishment of Regulatory Guides[38] specifying the partition of available iodine in the containment into discrete physical forms. Elemental vapor and chemically active particulate forms of iodine are readily removed from the air by chemical sprays and filtering systems. Methyl iodine, other organic iodides, and possibly hypoiodous acid have been identified as persistent airborne species but are conservatively believed to make up no more than 4% of the total iodine released into the containment. Fission-product and fuel aerosols are effectively removed from the containment space by agglomeration and gravitational settling as well as by engineered safety features. Analytical models have been developed to predict the airborne concentration of fission products as a function of time for a given set of input data, including fission-product concentration, particle-size distribution, containment atmospheric conditions, and geometry.[43]

ORNL researchers have developed preliminary empirical models for the release of cesium and iodine in steam under LOCA temperature conditions (cladding temperature less than 1477 K, 2200F.)[44] The models assume that the release is the sum of two components: burst release (that carried out with escaping plenum gas when the rod ruptures), and diffusional release (that diffusing from the fuel rod after the plenum gas has vented).

A measure of the precision of the model was obtained by comparing the model predictions with the experimental data. This is done graphically in Figure 18 for both cesium and iodine.

Comparisons[44] of these best-estimate predictions for total cesium and iodine release with those used in the Reactor Safety Study[32] show that the best estimate values are lower by a factor of 200 and 60, respectively than those used in the Reactor Safety Study.

More recent experiments[45] conducted at ORNL at higher temperatures (1200°C to 1600°C) than would be expected under terminated LOCA conditions indicate a greatly accelerated release of cesium, iodine, and noble gases occurs at temperatures greater than approximately 1300 to 1400°C. The results of these tests suggest that some mechanism other than diffusion from either the pretest gap or the UO_2 matrix is controlling fission product release at these higher temperatures. Table 5 compares the observed fission product release for four tests conducted at temperatures ranging from 1200°C to 1610°C.

Table 5. Comparison of Results of Fission Product Release Tests

Test No.	Temperature (°C)	Test period (min)	Percent of total inventory released[a] ^{85}Kr	^{134}Cs	^{129}I
HBU-11[b]	1200	10	1.3[c]	0.012	0.018
HT-1	~1300	10	1.07 + ~0.5[d]	0.112	0.165
HT-2	~1445	7	5.0 + ~1.0[d]	4.82	2.35
HT-3	~1610	3	8.3 + ~1.0[d]	8.27	e

[a]Percentage release values are based on total inventory in the 12-in. (30.5-cm) segment length. Percentage releases based on the heated length of 6 to 6.5 in. (15.2 to 16.5 cm) will be approximately double the listed values.

[b]Total test time was 27 min; release values were adjusted for a 10-min period.

[c]Includes ^{85}Kr released when this segment was used in a previous test at 900°C, test HBU-7.

[d]Estimated release during cladding expansion.

[e]Data acquisition and processing not yet complete.

METALLURGY AND MATERIALS RESEARCH

The NRC metallurgy and materials research program is concerned with the integrity of the primary-system pressure boundary in light-water reactors (LWRs). It is an experimental and analytical program designed to upgrade the bases for design, fabrication, operation, and inspection criteria, as well as for the analytical procedures required to evaluate performance under normal, upset, accident, and faulted conditions. Thus, a primary goal is to improve the definition of failure probabilities and failure modes, and to establish ways by which the failure probabilities can be reduced if this is considered necessary.

The primary system integrity research program consists of three major areas of research: (1) fracture and structural mechanics, (2) operational effects, and (3) flaw detection and evaluation.

The fracture and structural mechanics work encompasses (1) reactor vessel and piping-system performance under pressure and thermal loading; (2) crack initiation, propagation, and arrest (including static and dynamic studies and the use of irradiated specimens); and (3) response to operational and postulated conditions. In particular, the work on vessel response to postulated conditions includes thermal shock and steam-line-break accident conditions to assess the effects of abnormal pressures and thermal shock following the injection of cold emergency core cooling (ECC) water after various postulated loss-of-coolant accidents (LOCA's) or steam-line breaks.

The operational effects work is directed at obtaining data on (1) irradiation embrittlement, (2) annealing and re-irradiation, (3) residual element effects, (4) cycle crack growth, (5) steam-generator tube integrity, (6) intergranular sensitization and stress-corrosion cracking and (7) neutron dosimetry.

The flaw detection and evaluation work covers (1) improved ultrasonic characterization of flaws, (2) acoustic emission studies of flaw growth in piping and pressure vessels and of flaws produced during welding, (3) improved eddy-current inspection of steam-generator tubing, and (4) advanced nondestructive examination techniques.

Fracture and Structural Mechanics

Vessel/Piping Performance and Response

Fracture toughness and crack arrest in LWR vessel and piping materials have been studied extensively by many organizations over the years. Much of the data was obtained with relatively small laboratory specimens, 1 inch

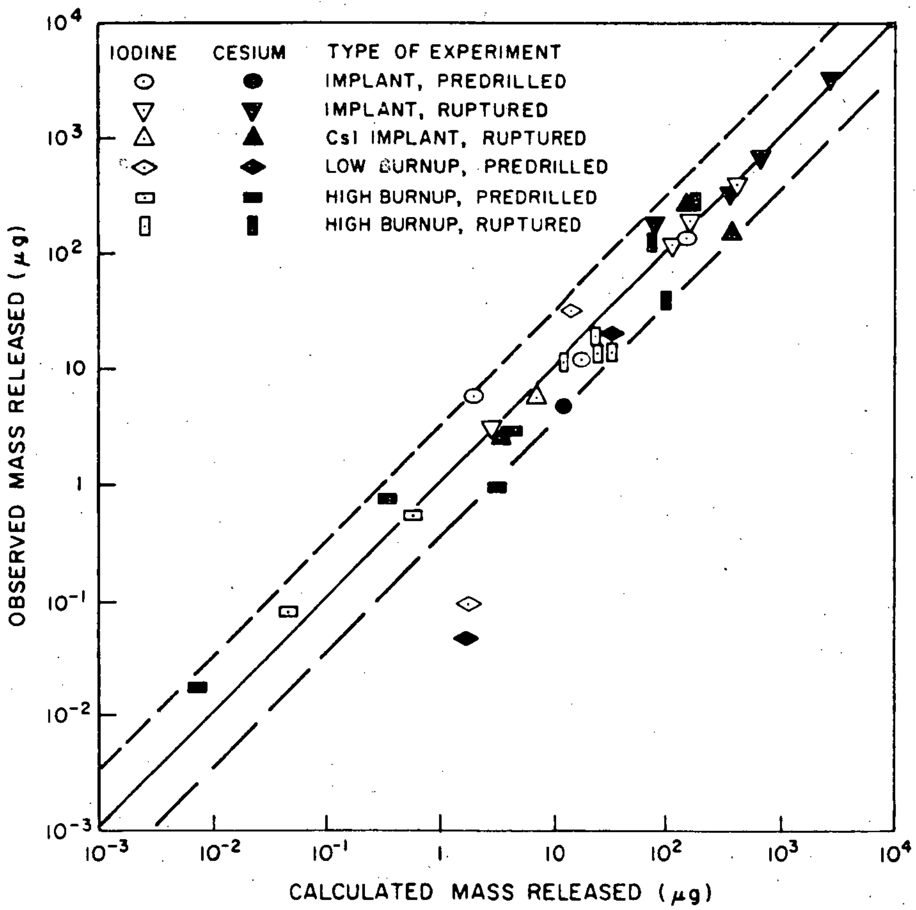

FIGURE 18 COMPARISON OF CALCULATED AND OBSERVED CESIUM AND IODINE RELEASE FROM DEFECTED LWR FUEL RODS INTO STEAM

(REFERENCE: NUREG/CR-0091)

or less in thickness.[46-50] However, some very useful correlations have been made between small-specimen data and data obtained with 8- to 12-inch-thick compact specimens and 12-inch-thick dynamic tear specimens.[51,52] Pipe-rupture studies have been conducted in the ductile regime, using piping generally less than 2 inches thick.

Research in this area has resulted in the development and validation of linear elastic fracture mechanics as a method for analyzing pressure vessels and piping with cracks having sufficient constraint, under elastic stresses, and at or below the transition temperature.[53,54] The criterion based on this method is embodied in ASME Boiler and Pressure Vessel Code Section III, Appendix G, and has been validated for slow-load fracture toughness with up to 12-inch specimens (steel plate 02 from the Heavy Section Steel Technology Program) and for rapid-load fracture toughness with up to 8-inch specimens. Criteria for fracture-safe operations under elastic-plastic and fully plastic conditions are now of interest. For example, the arrest of a crack initiating in a local brittle region may be dependent on the increased toughness of the surrounding material. To quantify the transition from brittleness to increasing toughness, additional information is needed on material responses (e.g., possible crack initiation, propagation, and arrest) under appropriate test conditions of temperature, stress, and radiation-induced changes, and their gradients. Research is under way at several laboratories and universities to develop an elastic-plastic fracture-analysis criterion based on the J-R Curve[55] and on tearing instability concepts,[56] as well as to predict the elastic-plastic stress state at the tip of a crack by means of a three-dimensional finite-element computer code. These techniques are being studied carefully from a theoretical standpoint for both pressure vessels and piping. The results will then be validated by experiments on a spectrum of test configurations, ranging from small to large specimens and to vessels. Unirradiated materials must be used to validate the criterion, but it must subsequently be validated with irradiated or simulated-irradiated materials as well.

Pressure tests of 6-inch-thick pressure vessels have been carried out with carefully sized flaws placed in walls and in nozzle regions, under a series of temperature and stress conditions.[57] (See Table 6). The flaw regions were formed either from brittle electron-beam welds to produce a "natural" crack as a result of a dynamic crack pop-in or were fatigue precracked. The tests were designed to produce very long (~ 12 inches) and deep (one-half the thickness) cracks for evaluating elastic and elastic-plastic loading criteria and methodologies. The results will aid in establishing the relationship between the fracture-toughness and crack-stability criteria and the details of the ultimate fracture, and hence will aid in establishing the design margin of safety against failure.

Four thermal shock tests have been performed (see Table 7) on cracked steel cylinders with degraded toughness properties somewhat simulating irradiation embrittlement. The results confirmed predictions of crack noninitiation under severe thermal shock loadings as well as crack initiation and estimated crack-arrest positions. It has been shown that linear elastic fracture mechanics does characterize the thermal shock methodology and that operational pressure-temperature transients can be accurately analyzed.[58] The beneficial effects of warm prestressing are currently being studied in more detail because this type of treatment is expected to severely limit crack initiation, and thereby penetration, in the thickness direction.[59]

Sustained-load testing of a vessel has been conducted to investigate the extent of fracture that would occur if a reactor vessel were to fail in a fully pressurized mode. Specifically, the investigation considered the progress and extent of crack propagation that could result from the stored energy available in a high-temperature system (essentially loading resulting from system blowdown). The test showed that even though a through-the-wall crack occurred at ductile shelf toughness temperatures, no further propagation occurred as a result of the sustained load.[60]

Crack Arrest

Crack initiation is governed by slow-load fracture toughness or rapid-load fracture toughness, whereas crack arrest is presently defined by the limits of the K_{IR} curve (See Figure 19). Crack arrest has been actively studied in recent years.[61] The methodology for predicting when, where, and under what conditions a running crack will stop has now been established for specimen geometries,[61] and experimental validation is under way.

Standard test methods and specimen geometries were submitted to ASTM Committee E-24 in March 1977 to be considered as a tentative standard. An international cooperative testing program has been organized with 29 participants. The objective was to test the applicability of the proposed methods for measuring crack-arrest toughness in the range of practical interest and to define the clarifications and refinements that are found to be necessary. Two-dimensional, dynamic, finite-element analyses are developed for cylinder geometries that will be applicable to test cylinders as well as reactor vessels. They have been used to analyze the thermal shock experiments and will be used to analyze reactor vessels subjected to ECCS operation following a LOCA. Thick specimens will be tested to determine the relationship between crack-arrest toughness measurement capacity and specimen thickness. An irradiation has been completed and specimen testing is ready to begin.

Theory and methodology are being developed for the analysis of crack arrest in reactor pressure vessels for situations such as thermal shock (resulting from ECCS operation) and embrittled region crack pop-in. The analysis procedure will permit prediction not only of whether or not a crack will be arrested but also of the conditions for crack arrest (crack size in a given wall thickness, etc.) in terms of the nature of the event, the initial operating conditions, the structural configuration and the initial crack configuration, and the appropriate toughness criterion. The relation between rapid-load fracture toughness and change in crack length as a function of temperature and irradiation is being used as input to the structural analysis of crack arrest. The analysis considers the influence of the kinetic energy of the structure (during crack extension) on crack

TABLE 6

SUMMARY OF TEST RESULTS FROM EIGHT 6-IN.-THICK INTERMEDIATE TEST VESSELS

(REFERENCE: ORNL/TM-5090)

VESSEL NO.	TEST TEMPERATURE (°F)	FLAW DIMENSIONS DEPTH (in.)	FLAW DIMENSIONS LENGTH (in.)	FLAW LOCATION	FRACTURE PRESSURE (ksi)	NOMINAL FRACTURE STRAIN (%)[a]
V-1	130	2.56	8.25	BASE METAL (o)[b]	28.8	0.92
V-2	32	2.53	8.30	BASE METAL (o)	27.9	0.19
V-3	130	2.11	8.50	WELD METAL (o)	31.0	1.47
V-4[c]	75	3.00	8.25	WELD METAL (i)[d]	26.5	0.17
	75	3.10	8.10	BASE METAL (o)	26.5	0.17
V-6[e]	190	1.87	5.25	WELD METAL (o)[d]	31.9	2.0
	190	1.34	5.20	BASE METAL (i)	31.9	2.0
	190	1.94	5.30	WELD METAL (i)	31.9	2.0
V-5	190	1.20	3.75	BASE METAL (i)[f]	26.6[g]	0.25
V-7	196	5.30	18.0	BASE METAL (o)	21.4[g]	0.12
V-9	75	1.20	3.75	BASE METAL (i)[f]	26.9	1.05

[a] OUTSIDE CIRCUMFERENTIAL STRAIN ON CENTER LINE OF VESSEL REMOTE FROM FLAW.

[b] (o): OUTSIDE SURFACE, (i): INSIDE SURFACE.

[c] CONTAINED TWO FLAWS

[d] FLAW WHERE FRACTURE OCCURRED.

[e] CONTAINED THREE FLAWS.

[f] NOZZLE CORNER FLAW.

[g] LEAK-BEFORE-BREAK.

TABLE 7. TEST CONDITIONS FOR TSE-1, TSE-2, TSE-3, AND TSE-4

Test Conditions	TSE-1	TSE-2	TSE-3	TSE-4
Test specimen	TSV-1	TSV-2	TSV-1	TSV-2
Test specimen dimensions, m (in.)				
OD	0.53 (21)	0.53 (21)	0.53 (21)	0.53 (21)
ID	0.24 (9.5)	0.24 (9.5)	0.24 (9.5)	0.24 (9.5)
Length	0.91 (36)	0.91 (36)	0.91 (36)	0.91 (36)
Test specimen material	A508, class 2	A508, class 2	A508, class 2	A508, class 2
Heat treatment	Quench only from 871°C (1600°F)	Quench only from 871°C (1600°F)	Quench only from 871°C (1600°F)	Quency only from 871°C (1600°F)
Flaw	Long axial crack, a = 11 mm (0.42 in.)	Semicircular axial crack, a = 19 mm (0.75 in.)	Long axial crack, a = 11 mm (0.42 in.)	Long axial crack, a = 11 mm (0.42 in.)
Temperatures, °C (°F)				
Wall (initial)	288 (550)	289 (552)	291 (555)	291 (555)
Sink (initial)	4 (40)	-23 (-10)	-23 (-10)	-25 (-13)
Sink (final)	7 (45)	-15 (4.5)	-15 (4.5)	-19 (-2)
Coolant	Water	40 wt % methyl alcohol, 60 wt % water	40 wt % methyl alcohol, 60 wt % water	40 wt % methyl alcohol, 60 wt % water
Coolant flow rate, m^3/hr (gpm)	59 (260)	114 (500)	$\sim$114 (500)	$\sim$114 (500)
Coolant pressure in test section, kPa (psi)	1520 (220)	917 (133)	965 (140)	1000 (145)
Back-pressure orifice diameter, mm (in.)	25.43 (1.001)	43.18 (1.700)	43.18 (1.700)	43.18 (1.700)
Heat transfer coefficient, W m^{-2} K^{-1} (Btu hr^{-1} ft^{-2} $°F^{-1}$)	$\sim$2800 ($\sim$500)	$\sim$5700 ($\sim 10^3$)	$\sim$5700 ($\sim 10^3$)	$\sim$5700 ($\sim 10^3$)
$(K_I/K_{Ic})_{max}$	0.74	1.33 (θ = 75°)	1.13	1.29
Time of occurrence[a] of (K_I/K_{Ic}) = 1, min		b	$\sim$3	1.7
Time of occurrence[a] of $(K_I/K_{Ic})_{max}$, min	$\sim$8	$\sim$4	$\sim$6	5
Duration of experiment, min	30	30	30	30

[a] Calculated times based on measured temperature distributions. For TSE-3, the effect of core holes is not included.

[b] Varies along crack front.

REFERENCE: NUREG/CR-0107, October 1978

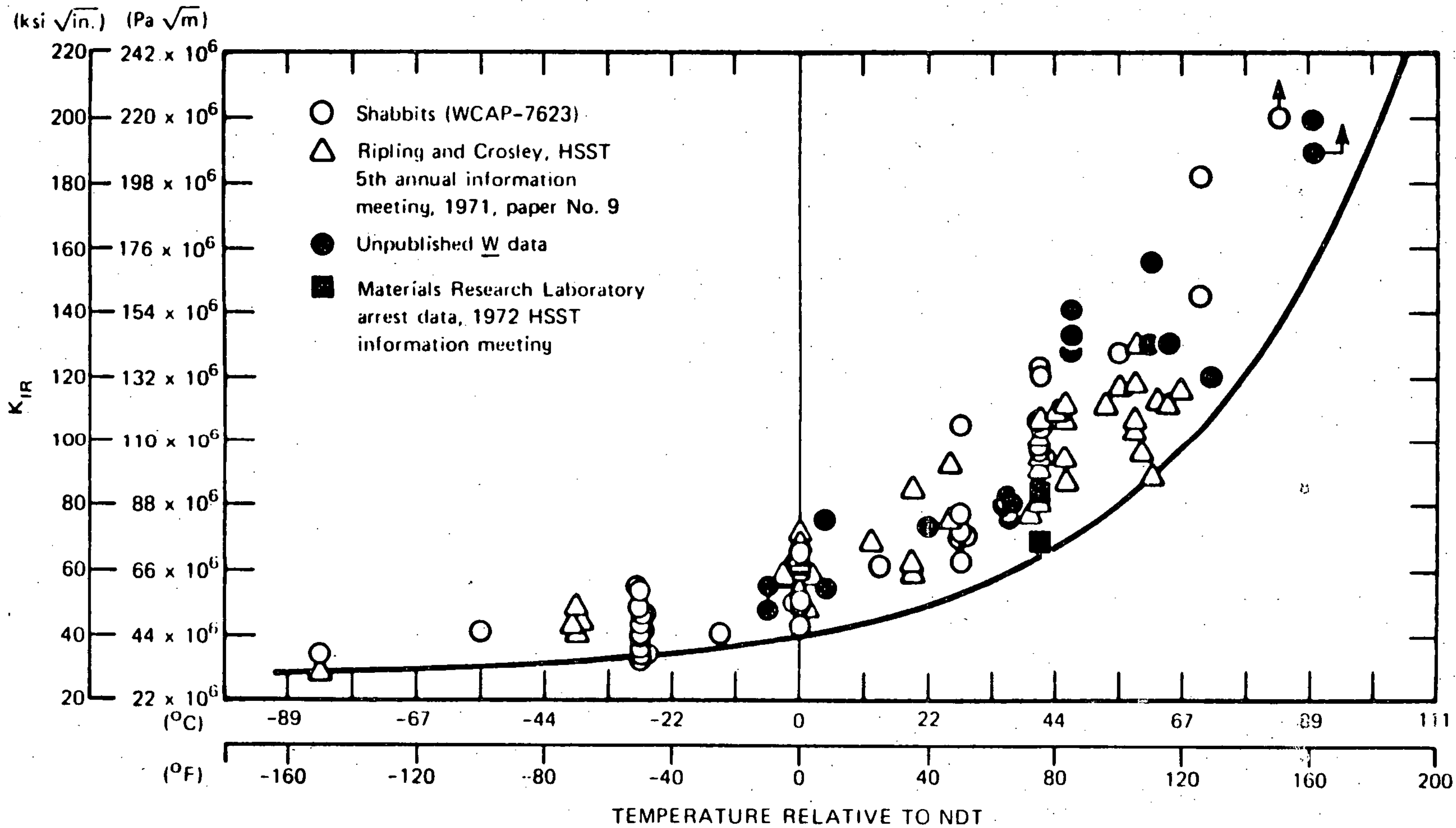

FIGURE 19. REFERENCE CURVE FOR MINIMUM TOUGHNESS

(REFERENCE: WRC BULLETIN 175 AND NUREG-0234) arrest by evaluating the "path dependency" of the process, including multiple events of onset of crack extension and arrest. The initial analysis will be based on linear elastic material behavior. The need for developing an elastic-plastic analysis will be evaluated from (and based on) the elastic analysis results. The crack arrest specimens and test method and analysis must lead to the adoption of standards by such organizations as ASTM and ASME. The specimens and test methods must be evaluated for adequacy in reactor surveillance. Validation is required for the initiation of a running crack in simulated-embrittled material in a pressure vessel configuration. The objective is to confirm the capability of the crack-arrest methodology to predict the arrest of a running crack before it reaches a critical size.

Operational Effects

Irradiation Effects

The effects of neutron bombardment on reactor vessel beltline structural materials include an upward shift in the reference nil-ductility transition temperature by several hundred degrees Fahrenheit, a reduction in ductile shelf-level energy absorption strength, and a reduction in tensile ductility. All of these factors combine to reduce fracture toughness and the potential for crack arrest. Because of the problems inherent in research on irradiated materials--including space limitations in reactors, shipping-cask sizes, and radiation limits for hot cells--most research on irradiation effects on vessel materials has been conducted with test specimens 1 inch or less in thickness. However, initial correlations have been made between results (see Fig. 20) from small specimens and those from 2- and 4-inch thick compact specimens tested in 1975.[62] Systematic studies of post-irradiation heat treatment at temperatures above the operating temperature of irradiated steel have shown that a significant portion of the pre-irradiation toughness can be recovered in this way to extend the useful life of reactor vessels with renewed fracture-toughness capability.[46] (See Figure 21)

Neutron-induced embrittlement in ferritic pressure vessel steels has been studied extensively.[46,63] It has been shown that embrittlement can be significantly reduced simply by complying with the draft ASTM recommendation for upper limits of 0.10 wt% copper and 0.012 wt% phosphorus in the chemical composition of the steel. Furthermore, a mechanism by which copper affects neutron embrittlement in steels has been proposed.[64] (See Figure 22)

Cyclic Crack Growth Rate

It is recognized that small flaws, material defects, and inhomogeneities will always exist to some extent in materials to be used in nuclear service. Although such irregularities will initially be below the established limits that would require repairs, they can grow as a result of cyclic loading during normal operation. The potential for cyclic crack growth in reactor structural materials should be experimentally assessed to gain confidence that flaws cannot grow to a "critical" size. Therefore, NRC is sponsoring research to extend the data base and to improve the understanding of cyclic crack growth in reactor structural materials, especially for the environment and cyclic loading rates that represent realistic reactor operating service.

Some crack growth rate data have been established for irradiated materials under reactor service. Much of the work on unirradiated steels was conducted at relatively rapid cyclic frequencies. More recently, however, slower cyclic rates (of 1 cpm and less) have shown significant increases in crack growth rates from cycling in a water environment.[65] Because this slow cycling more nearly approaches the realistic service performance of an operating reactor, and using the results of an extensive study to determine the most realistic test parameters, current research is emphasizing the use of a 1 cpm loading time for R ratios of 0.2 and 0.7 at pressure and temperature in water duplicating the chemistry of PWR water in the development of data on cyclic crack growth rates.

Steam-Generator Tube Integrity

Tubing for PWR steam generators is subject to wastage, cracking, and denting at support-plate locations. Denting is particularly insidious because it precludes meaningful eddy-current inspection. The large accumulated strain in the dented region causes primary-side intergranular stress-corrosion cracking, which can remain undetected until leakage. Large safety margins are established for steam-generator tubing, so that large in-service degration (40 to 60%) of the tube wall can be tolerated.

Tubing typical of several major designs of PWR steam generators has been tested in both the burst and the collapse modes. In the first phase of the study, electric-discharge-machined (simulated) cracks with depths ranging from about 25% of the wall thickness to through-wall and areas of wastage ranging in depth from 25 to 90% of wall thickness were burst and collapse tested under simulated reactor conditions to determine their effect on both burst and collapse pressure. Leak rates from various size flaws were also determined. The second phase of the study will involve similar testing of tubing containing defects induced by chemical corrosive means. Flawed tubing removed from operating steam generators will be tested in the future when available. Margins of safety against burst and collapse were established from the test results, and an empirical predictive equation for steam-generator tube burst pressure as a function of tube and flaw dimensions was developed.[66]

Stress-Corrosion Cracking

Stress-assisted intergranular corrosion cracking in the BWR coolant environment continues to occur in seamless small- and intermediate-diameter austenitic steel piping. The primary factors causing this phenomenon are known. They include oxygen in the coolant, high stress, and sensitization of the stainless steel. The exact combination of factors that actually produces cracking has not yet been conclusively established.

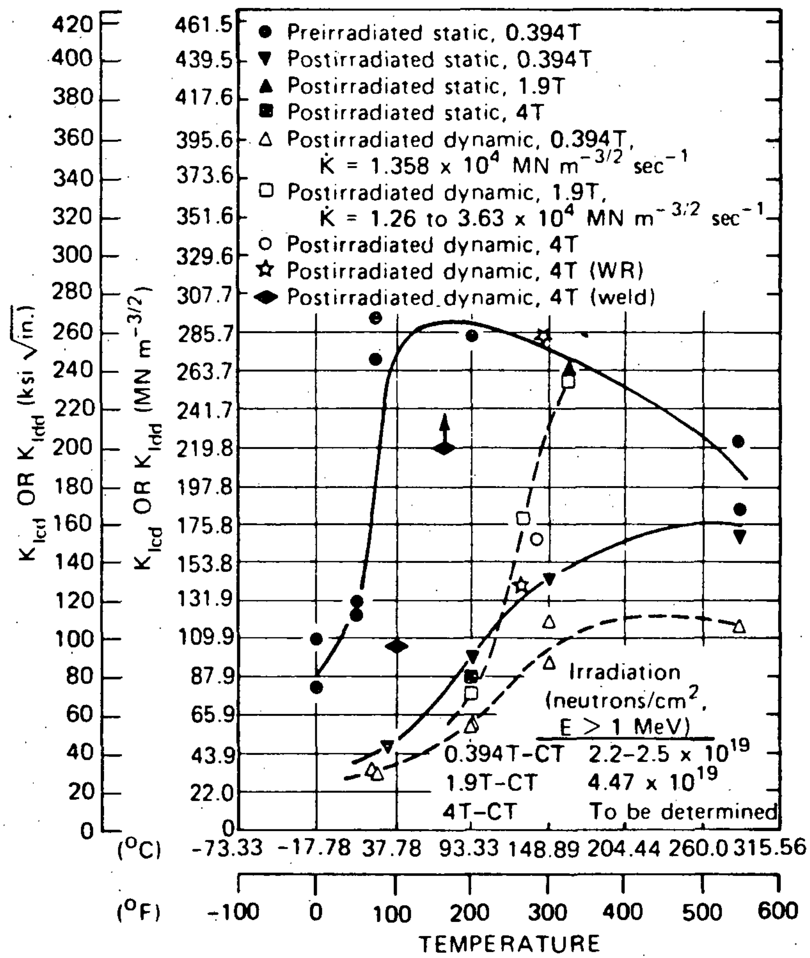

FIGURE 20. COMPACT TENSION SPECIMEN TEST RESULTS FOR ASTM A533, GRADE B C 1 STEEL

(REF: NUREG-0234)

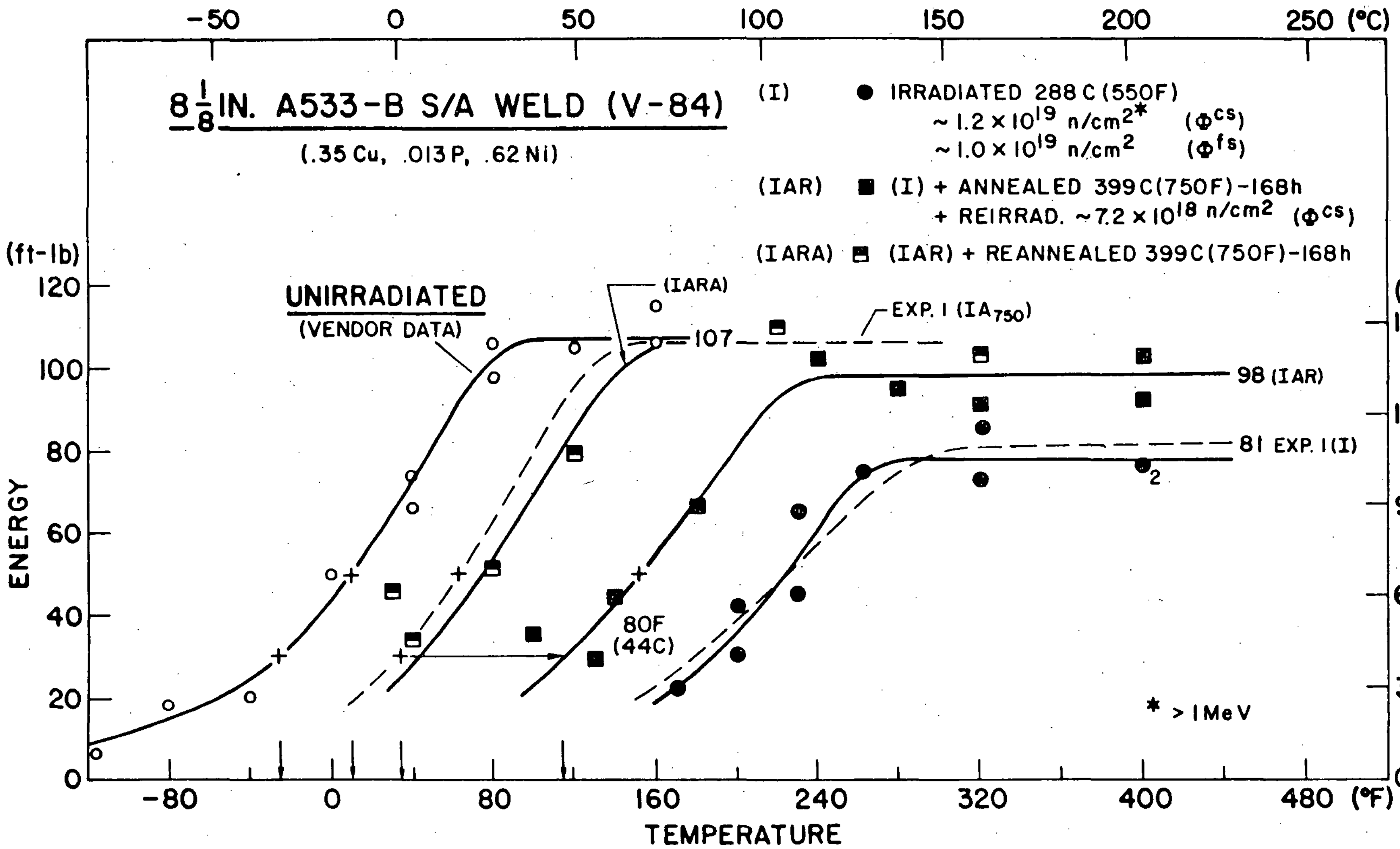

FIGURE 21. EFFECTS OF ANNEALING ON IRRADIATED REACTOR PRESSURE VESSEL STEEL

(REFERENCE: NUREG/CR-0486)

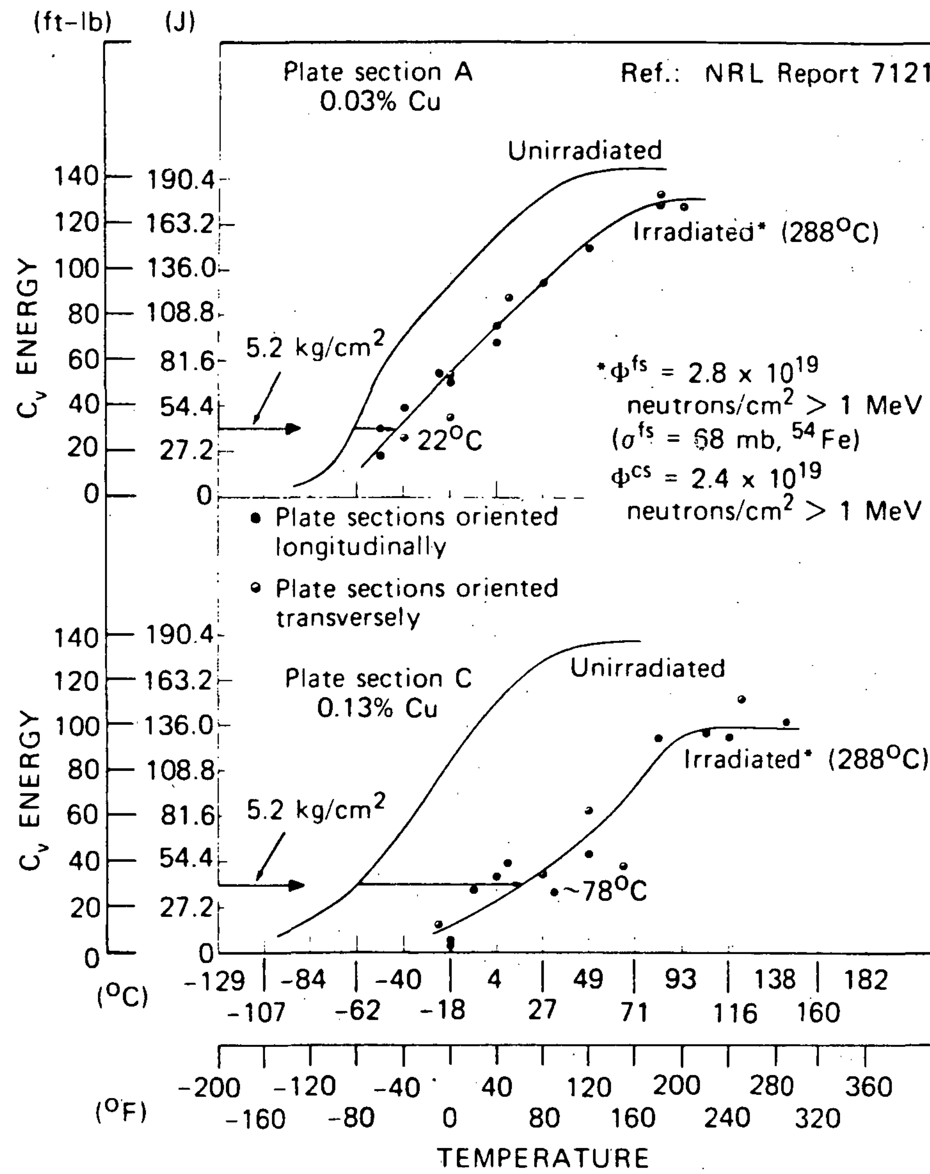

FIGURE 22. EFFECTS OF RESIDUAL ELEMENTS IN IRRADIATED STEEL

(REFERENCE: NUREG-0234)

Research is under way in three primary areas of interest for intergranular stress-corrosion cracking (SCC). A field test is nearly complete for detecting and measuring the degree of sensitization and susceptibility to SCC of welded stainless steel piping.[67] Laboratory studies are being extended to field confirmation. Adoption of the test as a standard acceptance and inspection procedure is anticipated.

The residual stress level resulting from welding in piping has been studied to develop an analytical method for predicting such residual stress levels on the basis of fabrication parameters.[68] Evaluation of the analytical procedure was conducted on a variety of sizes and types of nuclear grade weldments.

The SCC characteristics of steam-generator tube material (Inconel 600) are being investigated as a function of stress, strain, temperature, metallurgical condition, water chemistry, and other important parameters. The aim is to establish a predictive capability for stress-corrosion cracking in steam-generator tubing during the service life of a steam generator.

Neutron Dosimetry

Neutron dosimetry--the measurement of the neutrons causing embrittlement of pressure vessel steel--is needed to accurately correlate steel embrittlement from surveillance irradiations with embrittlement in the power reactor vessel wall to assess safe reactor life. At present, mechanical property measurements are more accurate than neutron dosimetry. Laboratories and LWR vendors use ASTM procedures for neutron dosimetry in test and power reactor surveillance, but each uses its own set of experiments to verify that calculations and extrapolations are consistent with measurements. Furthermore, an energy threshold of $E > 1$ MeV is currently being used as the criterion for neutron flux and fluence, even though research data[69] show that neutrons with energies between 0.1 and 1 MeV can contribute as much as 40% of the embrittlement attributed to neutrons with energies higher than 1 MeV.

The neutron flux and the spectrum of neutrons by energy level are being both calculated and measured in a wide variety of test and power reactors. The objective is to confirm procedures for calculating and extrapolating the neutron flux in reactor surveillance irradiations. For the experimental irradiations, as many as 20 different flux-monitor materials will be included to cover the entire neutron energy spectrum. In selected instances, spectrometry will be used to establish the spectrum in the important energy range between 0.1 and 1.0 MeV. Confirmation procedures will use a simulated pressure vessel wall wherein neutron-flux monitors and mechanical property specimens can be irradiated for comparison with the pretest calculations. Primary mechanical property characterizations will be in fracture-mechanics terms.

Flaw Detection and Evaluation

In-Service Ultrasonic Inspection

Inspection of nuclear reactor components by ultrasonic techniques is required both prior to service and during shutdown for periodic in-service inspections. Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for In-Service Inspection of Nuclear Power Plant Components," defines inspection criteria and allowable flaw sizes, based on linear elastic fracture mechanics, for various locations within reactor components. In the present inspection procedure, the pulse-echo amplitude and search-unit position are evaluated as a basis for flaw detection and sizing. Although ultrasonic testing is the presently accepted and most useful volumetric inspection technique, its reliability for flaw detection and sizing (using the Code procedure) is questionable and often inadequate. Significant advances have been made recently in the signal processing of pulse-echo data to form a synthetic aperture focused image of high resolution for greatly enhanced flaw characterization.[70-73]

The upgrading of ultrasonic inspection is focusing on developing more aspects of the information resulting from a pulse-echo test, including phase, frequency, amplitude, and search-unit position. Sensitivity of the results to the specific operator, a specific calibration test, or a specific transducer is also being reduced or eliminated. Greater detection sensitivity is being developed, and flaws are being characterized with much improved resolution. Means for the storage and ready retrieval of the information for meaningful reevaluation are also being developed. The importance of ultrasonic inspection records is expected to increase, with reference being made to past records for comparison with current information. For such comparisons to be most accurate, flaw and sensor locations must be accurately determined and recorded, and the inspection results must be made independent of changing transducers and electronics properties. The difficulty of ultrasonically inspecting austenitic stainless steel base metal, weld deposits, and the interface is being minimized by processing the data to greatly decrease electronic and grain-boundary scattering noise and to obtain focused images independent of signal amplitude above a detection level. These procedures greatly increase the sensitivity and resolution of ultrasonic inspection and flaw evaluation in stainless steel. The laboratory test procedures are being validated on realistic plate and piping samples, in addition to being adapted for typical in-service inspection procedures.

Flaw Detection By Acoustic Emission

Continuous on-line surveillance represents a goal because feasibility is yet to be demonstrated. The technique to be employed is acoustic emission, and while important advances in instrumentation have been recently realized, acoustic emission data analysis and extrapolation to real structures still require development and final proof testing in operating nuclear systems.[74] Furthermore, it is noted that acoustic emission data on crack growth may need validation by ultrasonic testing during a shutdown-period inspection.

Continuous acoustic emmission inspection during welding, for the detection of microcracking during weld solidification and cooling, has been established in nonnuclear applications[75] and has been carried forward in the nuclear application to demonstrate and characterize detection of cracks and other types of rejectable flaws.[76]

Acoustic emission is being developed for on-line flaw monitoring. Acoustic emission probes eventually will be placed on vessels and piping to monitor signals emitted during operation; improved nondestructive examination methods would then be used during shutdown periods for further characterization of the flaws detected. At present, relationships are being drawn between acoustic emission signals and mechanical property effects obtained during the testing of fracture and fatigue-type specimens of both plate and weldments. Acoustic emission signals from other sources that may be present during reactor monitoring are also being evaluated. A laboratory program will then be conducted on fully characterized natural flaws to validate both the detection and the quantification abilities of the techniques developed (See Figure 23). The techniques will then be extended to the full range of conditions prevalent in reactor operation, and procedures will be established for distinguishing among acoustic emission signals from various sources such as flaws, strained regions, reactor operations, etc. With such a baseline library available, it will be possible to begin actual structure and reactor vessel monitoring so that signal detection and quantification will be much more meaningful.

Improved Eddy-Current Inspection for Steam-Generator Tubing

The presently used ASME Code eddy-current inspection techniques are fast, but they can produce unreliable inspection results because of the many independent variables that affect the signals. For example, the detection of flaws in a tube dented region surrounded by corrosion products and the steam-generator support plate is extremely difficult. Existing mathematical models will be used to develop computer programs for designing optimum probes, instrumentation, and techniques for improved eddy-current inspection of steam-generator tubing.

Improved eddy-current techniques will be developed to separate the effects of diameter variations, probe wobble, tube supports, and conductivity variations from defect-size, defect-depth, and wall-thickness variations. Mathematical models and computer codes for eddy-current tests will be used to computer-design optimized probes, instrumentation, and techniques for multi-frequency, multiproperty examinations. The program will develop at least two optimized designs: one for the general inspection of steam-generator tubing and another for special conditions such as denting. Optimized probes and instrumentation will subsequently be built, evaluated in the laboratory, and finally validated by in-service inspections of steam-generator tubing.

Advanced Techniques

Numerous new techniques for nondestructive examination are being developed. Such techniques are continually reviewed and, if seen to be especially promising, are funded in carefully controlled assessment studies. A program was recently initiated to study the feasibility of internal friction monitoring techniques for the prediction of incipient intergranular stress-corrosion cracking in welded stainless steel BWR piping.

OPERATIONAL SAFETY RESEARCH

NRC sponsors a category of research termed "reactor operational safety," that is, research aimed at providing direct assistance to NRC officials concerned with the operational and operational-safety aspects of nuclear power plants.

The NRC requires a defense-in-depth philosophy to ensure the safety of nuclear power plants. Essentially, this means that three levels of safety are incorporated: (1) the plant is designed and fabricated for maximum safety, (2) protective systems are provided to monitor and correct abnormal conditions, and (3) engineered safety features are installed to mitigate the consequences of accidents.*

Criteria for the defense-in-depth concept are presented in Chapter I of Title 10 ("Energy") of the U.S Code of Federal Regulations, in particular, Appendix A ("General Design Criteria for Nuclear Power Plants") and Appendix B ("Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"), of Part 50 of Chapter I. The Operational Safety Research programs are related primarily to research in support of NRC implementation of the criteria contained in these two appendices and the related NRC standards, guides, and branch technical positions. The topics currently addressed include qualification-testing evalution, fire protection, human factors, and noise diagnostics.

A bibliography of NRC-sponsored reports on operational safety research is included at the end of this report.

Qualification Testing Evaluation

The qualification-testing evaluation program is focused on obtaining the data needed to answer certain questions about the testing of Class IE safety related** equipment to assess performance during and after postulated

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*An excellent short discussion of the defense-in-depth concept is contained in U.S. Nuclear Regulatory Commission Annual Report 1976.[77]

**Safety classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal or are otherwise essential in preventing significant releases of radioactive material to the environment.

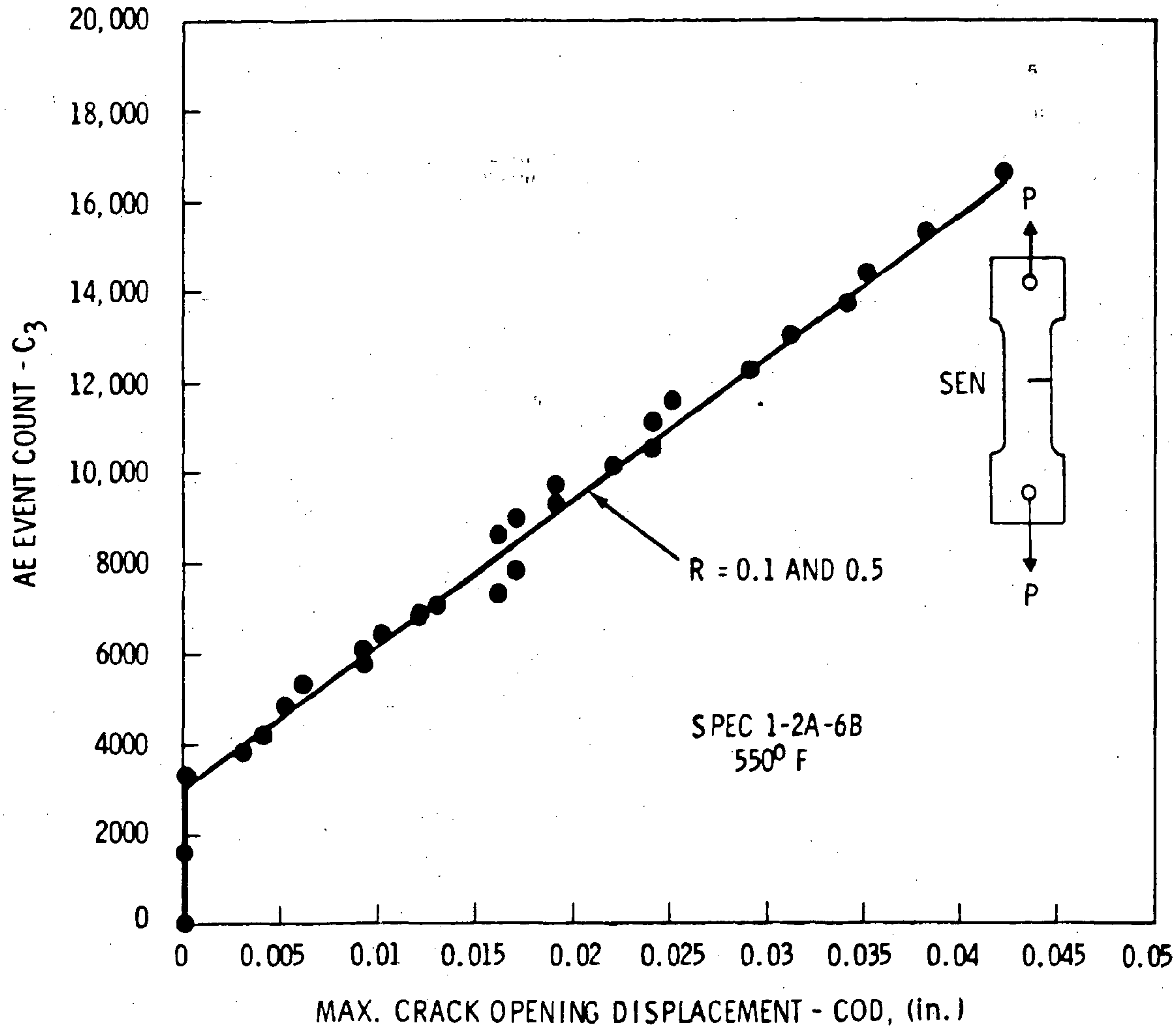


FIGURE 23. SUMMATION AE EVENT COUNT VS. COD, SPECIMEN 1-2A-6B, 550°F

(REFERENCE: "ACOUSTIC EMISSION-MATERIAL PROPERTY RELATIONSHIPS FOR CONTINUOUS MONITORING OF REACTORS" BY P. H. HUTTON, ET AL, SIXTH NRC WRSR INFORMATION MEETING, 1978, NRC PDR) accident conditions. The near-term qualification tests program is being conducted to answer questions about assessing conformance with IEEE Std 323-1974,[78] "Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations," and NRC Regulatory Guide 1.89,[79] "Qualification of Class IE Equipment for Nuclear Power Plants," which endorses IEEE Std 323-1974 with certain changes. The end products are data, criteria, and procedures that will enable the applicant and the NRC to ascertain that an acceptable qualification-testing program has been conducted.

The specific questions considered include aging; nuclear source term definition; synergisms; the performance indicators that must be monitored; failure definition; allowable thermal and nuclear-radiation-flux gradients; test sample preparation, quality control, mounting, and connections; chemical and steam flow rates; degree of mixing required; degree of impingement; and vibration. The first three will be discussed further in the following sections.

Aging

Considerations of aging in the qualification test program are important because of the potential for some aging mechanism, not detected through routine periodic testing, to create a weakened condition in a safety-related component. Such a weakened condition could result in common-mode failures in redundant safety-related equipment subjected to overstress conditions resulting from an accident condition such as a LOCA. Qualification testing is difficult because the weakened conditions must be simulated by exposing the component to low levels of stress over long periods of time. Any practical aging qualification test must therefore be based on an accelerated-aging methodology. At present, thermal aging is simulated by using the Arrhenius equation as a basis for performing accelerated aging. The Arrhenius equation, which is used to extrapolate from high thermal stress applied for a short time to lower stress applied for a longer time, is based on the assumptions that the chemical reaction rate is dependent only on temperature and that over the range of consideration it is constant.

For many safety-related materials in use today, there is some doubt as to the validity of using this assumption to extrapolate from times of less than 1 year to times up to 40 years. The NRC research program is centered around this issue, and its goal is to develop and prove techniques that will adequately simulate long periods of aging at low levels of stress for currently used materials.

The current research effort on aging consists of six tasks as described in the succeeding sections:

Task 1 - Single-Environment Aging Tests

Single-environment aging tests are being conducted to obtain data on the separate effects of radiation, temperature, and humidity.[80] The tests at present are limited to polymeric electrical cable materials utilized with safety-related systems but will be expanded to include other safety-related materials at a later date. The testing is based on the assumption that the important failure mode of electrical cable will be mechanical damage of the insulating or jacket material caused by embrittlement. Data obtained as part of the state-of-the-art report and scoping tests have confirmed this assumption. The parameter used as a measure of this damage is relative elongation of the insulating and jacket material with the conductor removed. From these tests, accelerated damage at the higher stress levels (acceleration functions) in a single environment have been obtained, using a test period of about 1 year. Elongation is being used as a relative damage indicator to evaluate the aging-simulation techniques and for comparison with naturally aged cable samples.

Task 2 - Combined-Environment Aging Tests

Combined-environment aging tests will be conducted to obtain data on the synergistic aging effect of temperature and radiation. Synergism with other aging parameters will be evaluated later in the program. As in the case of the single-environment aging tests, relatively low stress levels and long test cyles will be used. Temperatures from 90 to 150°C and radiation dose rates from 10^3 to 10^5 rad/hr are planned for this testing.

Preliminary indicators show there is a synergistic effect with some materials when radiation and temperature stress are applied. The test methodology for accelerated aging testing for combined stress has been developed and is currently being extended and verified with other materials.

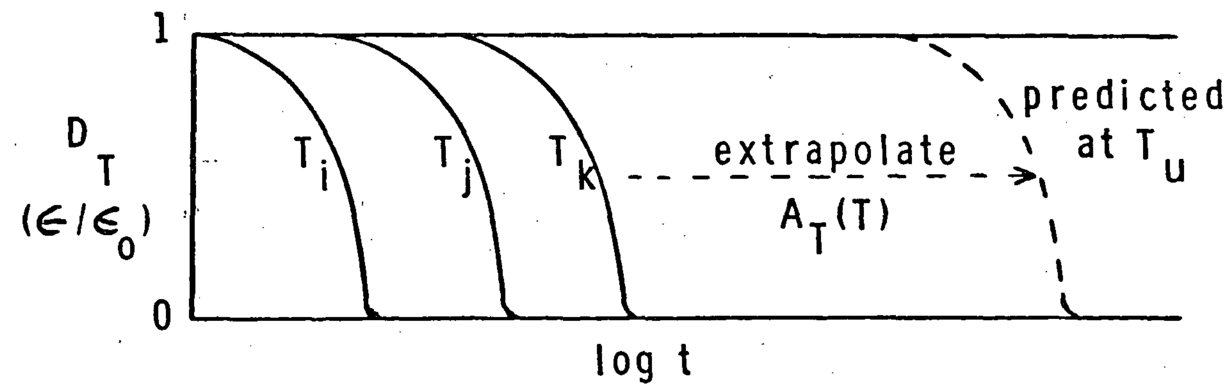
Figure 24 summarizes the method by which single environment aging is normally carried out. Generally, the experimenter overstresses the environment to accelerate the aging. For combined environments, Gillen[80-82] proposed accelerating matched sets of environments (e.g., temperature and radiation as shown in Figure 25) to see if the same predicted result is obtained. Figure 26 shows aging data for chloroprene jacketing material including the separate aging effects of radiation and temperature plus the combined aging effect. Using these data, Figure 27 shows how the single and combined environment studies can be analyzed according to the formalism of the proposed method. In effect, one is building on actual data, including synergistic effects, to predict the damage to the material under various radiation and temperature conditions. In addition, the matched set approach can be used to accelerate by a chosen factor any ambient aging conditions.

Task 3 - Rate Effects

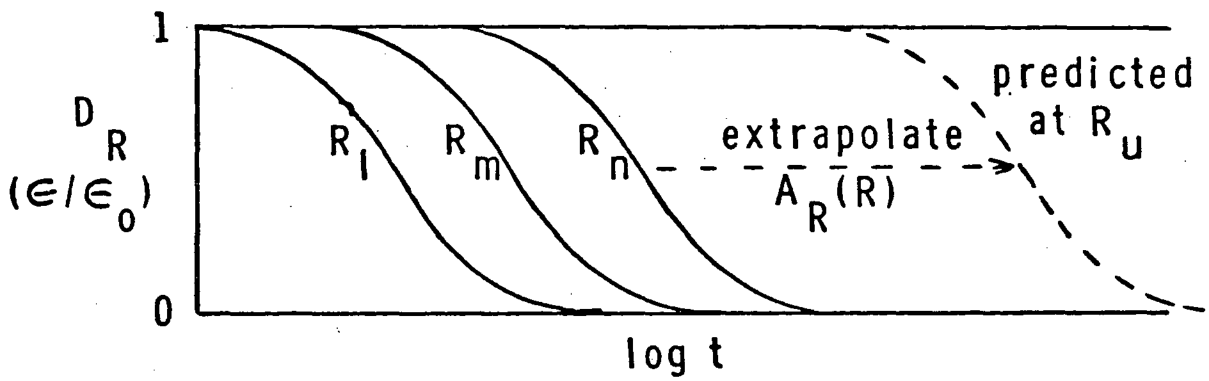
Tests to determine rate effects are underway. Of particular concern are the rate effects associated with oxygen diffusion and with radiation. These tests will determine the minimum accelerated aging period from which extrapolation to the required material life can be made.

CONVENTIONAL ACCELERATED AGING

Single environment thermal aging (const. accel.)



Single environment radiation aging
(constant acceleration)



ABOVE ASSUMES FUNCTIONAL RELATIONSHIP BETWEEN ACCELERATING STRESS AND TIME ($A_T(T)$ AND $A_R(R)$) CAN BE DETERMINED AND EXTRAPOLATED TO USE STRESS LEVEL

FIGURE 24 (REF: SAND 78-1907A)

COMBINED ENVIRONMENT AGING

Use environment: T_u, R_u

To accelerate by a factor X, use temperature T_x which accelerates thermal degradation by X simultaneous with radiation dose rate R_x which accelerates radiation degradation by a factor X. Assume any synergistic reactions also accelerated by factor X.

T_x, R_x called "matched set" to T_u, R_u

Example; do matched set experiments using accelerations of 20 times and 80 times. Predictions should agree.

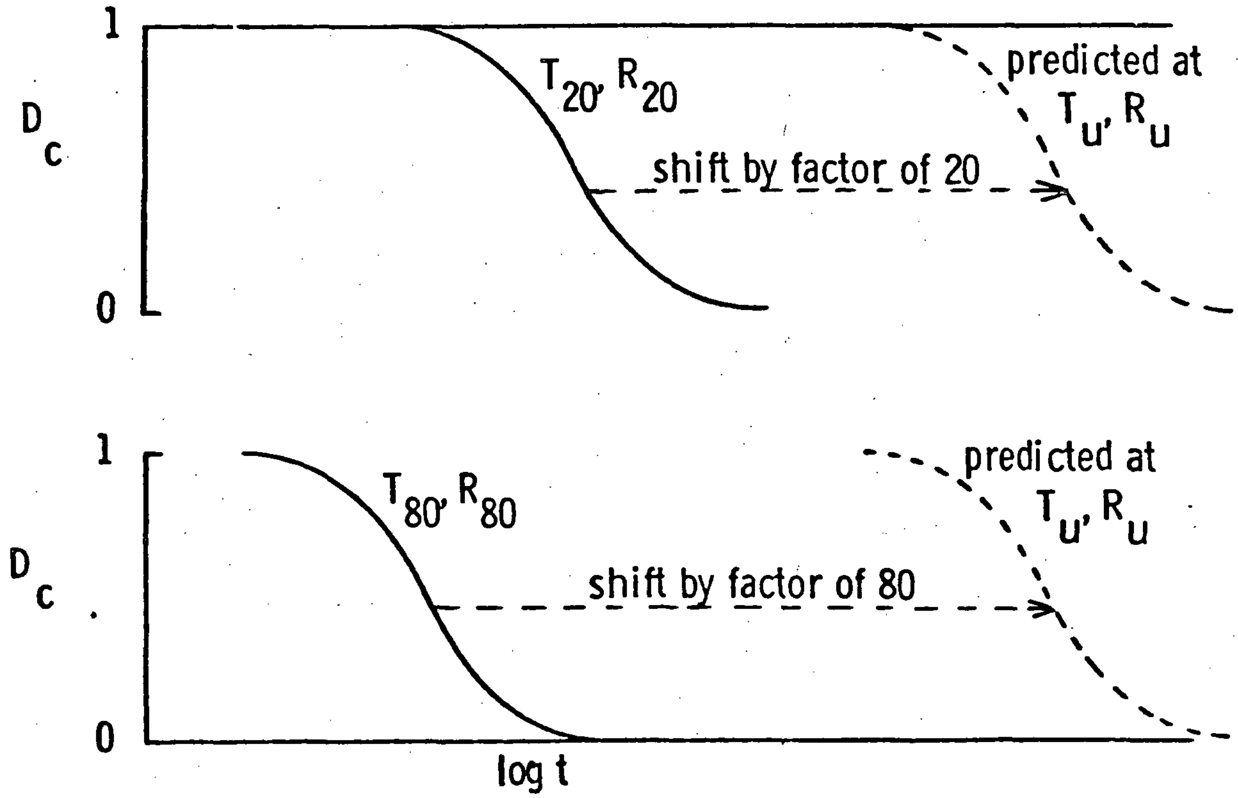


FIGURE 25

(REF: SAND 78-1907A)

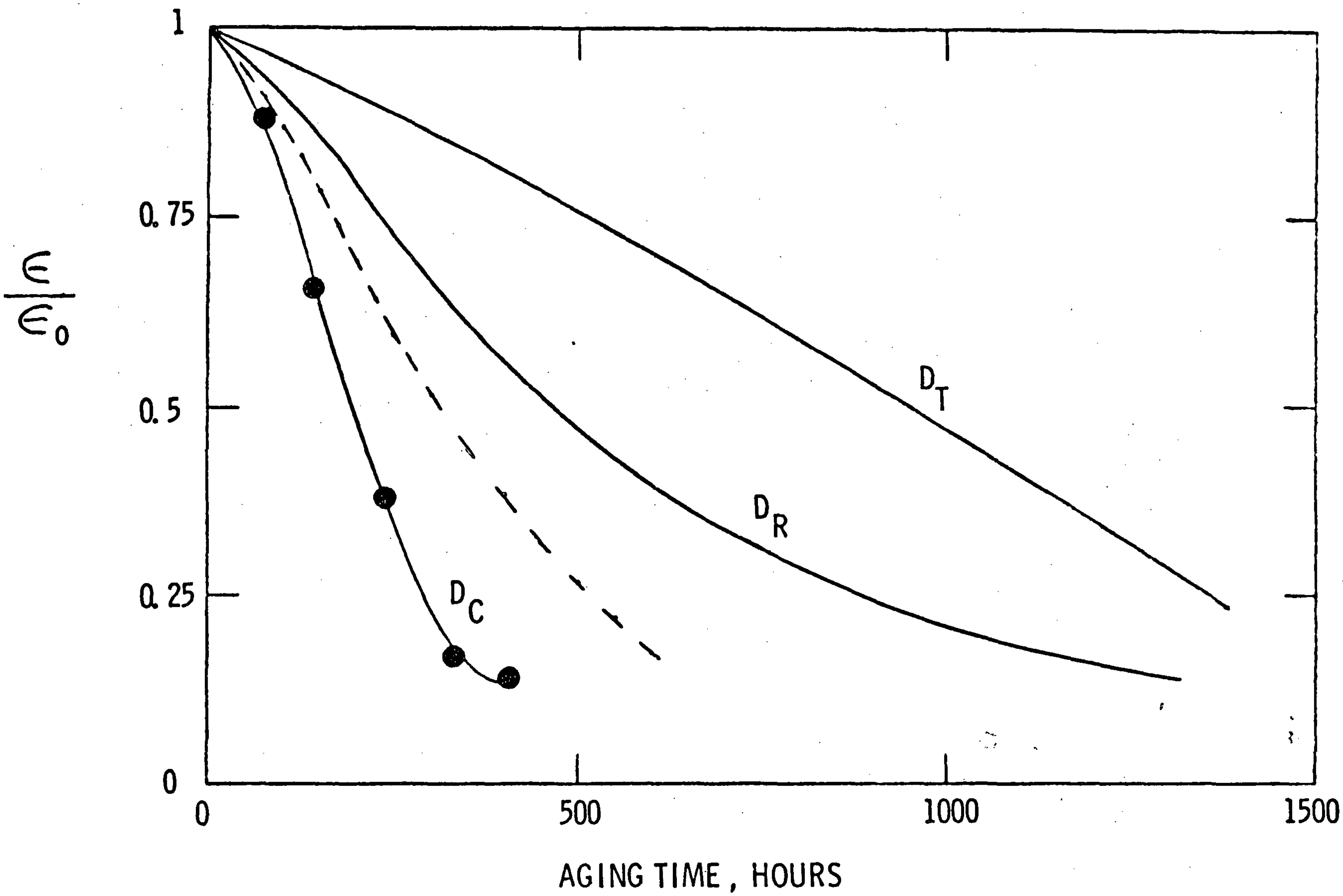


FIGURE 26. AGING OF CHLOROPRENE: D_T, 361 °K; D_R, 95 KRAD/HOUR; D_C, 361 °K COMBINED WITH 95 KRAD/HOUR; DASHED CURVE, PREDICTION WITHOUT SYNERGISM (REFERENCE: SAND 78-1907A)

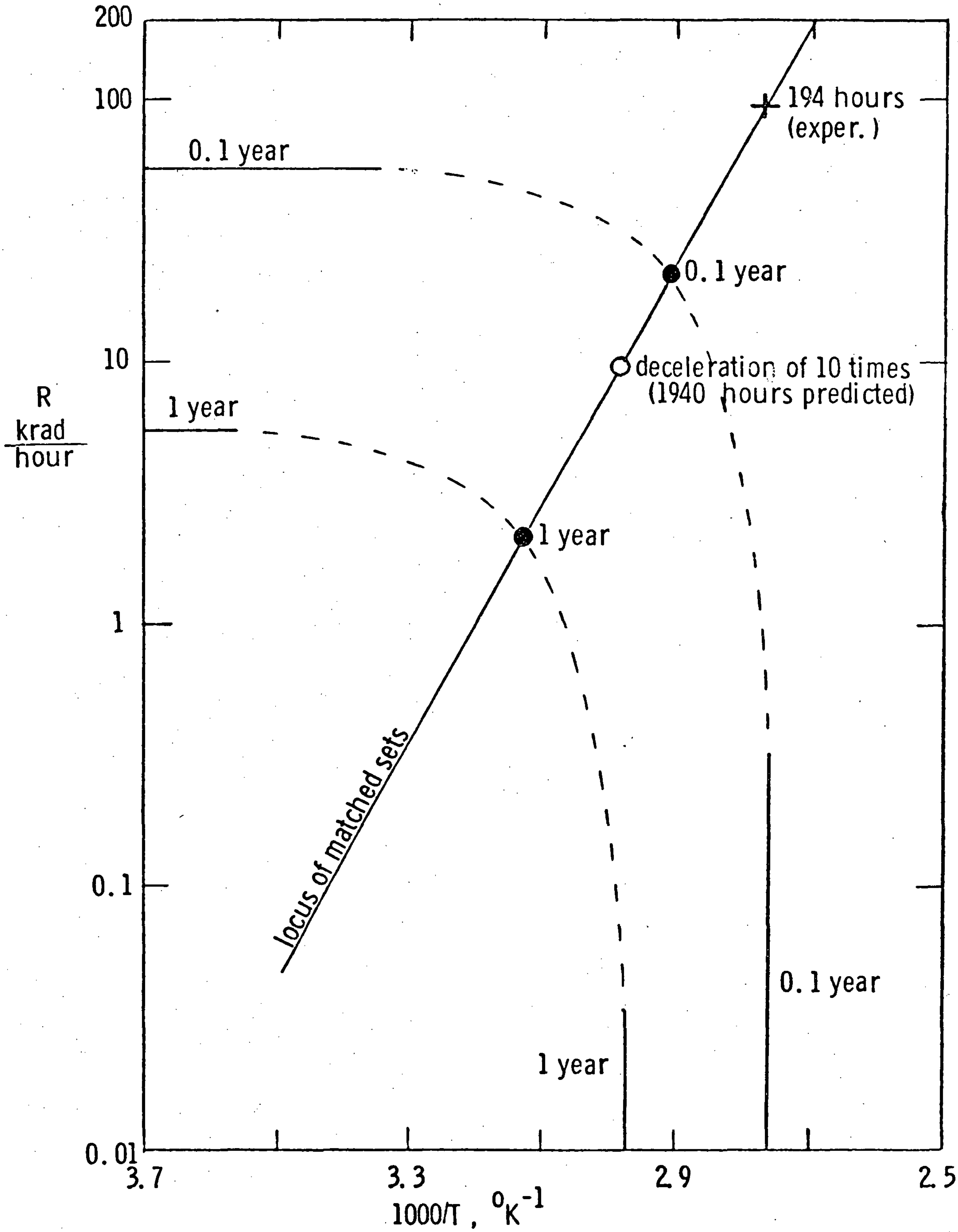


FIGURE 27. ANALYSIS OF CHLOROPRENE AGING DATA USING PROPOSED METHOD; CONTOURS OF TIME REQUIRED FOR $e/e_0 = 0.5$

(REFERENCE: SAND 78-1907A)

Task 4 - Damage Indicators

A study is in progress to identify alternative damage indicators that could be utilized in addition to the material elongation criterion currently used as the reference aging-damage indicator for electrical cable. Although this study is at present limited in scope to the selection of a relative damage indicator to confirm the aging-simulation techniques for electrical cable insulation, the general question of damage criteria will have to be addressed in conjunction with LOCA qualification testing of other safety related equipment. The current qualification-testing standards require that the component under test continue to carry out its prescribed function following the LOCA. It is usually not practical to demonstrate this in any test that can be designed and used. A more basic measure of damage will have to be developed and used for important safety-related equipment.

Task 5 - Comparison Study

Naturally aged samples are being collected so that the aging-simulation techniques being developed in tasks 1 through 4 can be checked with naturally aged material. Some comparisons have been completed showing that the accelerated aging methodology developed to date does simulate the long term natural aging.

Task 6- Requalification Tests

As a backup to the aging-simulation techniques being developed in tasks 1 through 4, an alternative method is being evaluated in which resistance to aging degradation for short periods of time would be assessed and requalification tests used to extend the acceptable lifetime in short time increments through the use of duplicate sacrificial samples.

Nuclear Source Term Definition

This work covers the calculation of the nuclear source terms for the accident assumptions made in Regulatory Guide 1.89. Progress to date has consisted of analysis to determine the time relationships following a LOCA for radiation doses, dose rates, energy spectra, and particle types. These data show that current industry practice with regard to radiation simulation testing may differ significantly in terms of dose rate, spectra, and particle type from that implied by Regulatory Guide 1.89. The ongoing work in this area is aimed at determining the importance of these differences in terms of potential damage to safety-related equipment.[83-85] The current effort consists of three tasks.

Task 1 - Source-Term Calculations

Additional source-term calculations will be made on the basis of Regulatory Guide 1.89 assumptions, taking into account new codes and test data developed in other programs. Also performed will be calculations based on the proposed revision to Regulatory Guide 1.89, which allows for reduced release assumptions for certain classes of safety-related equipment. In addition, source-term calculations will be made with best estimate LOCA release assumptions as required. Figures 28 and 29 show some of the calculations which have been made. Note that the "gap release" values (LOCA source term) are much lower than these derived from Regulatory Guide 1.89.

Task 2 - Evaluation of Radiation Simulators

An evaluation is being made of the adequacy of radiation simulators currently used to duplicate the hypothetical environment following the radioactive release postulated in Regulatory Guide 1.89. (See Figure 30 as an example). An initial assessment has been made by comparing the dose rates and energy spectra resulting from the conservative accident assumptions in Regulatory Guide 1.89 with the dose rates and energy spectra obtainable from practical simulators. The final simulator evaluation will take into account the practical significance of these differnces as evaluated in Task 3 of the LOCA testing. Cobalt-60, cesium-137, spent-fuel elements, and beta-particle machines will be included in this assessment.

Task 3 - Assessment of Radiation Effects

The damage to safety-related equipment materials will be determined as a function of the gamma and beta dose rates. A determination will also be made of how closely the dose-rate profiles resulting from the Regulatory Guide 1.89 assumptions must be simulated during qualification testing. Materials studies will be conducted using available radiation damage data, and additional experimental data will be obtained as needed. Also of concern is the damage of safety-related equipment materials by beta particles as a function of depth-dose profiles. Materials studies will be conducted and supplemented by experimentation if needed.

Assessment of Test Methodologies

This work, which was initiated in fiscal year 1975, covers the assessment of LOCA and MSLB (main steam line break) testing procedures, including synergisms among the stresses applied. Progress to date has been the development of test equipment and methods to compare sequential and simultaneous tests on identical test samples of safety-related material.[86] The current effort consists of four tasks.

Task 1 - LOCA Qualification Tests

At Sandia Laboratories, LOCA qualification tests have been conducted (1) sequentially, as recommended in IEEE Std 323-1974, with radiation exposure preceding the steam and chemical spray environment, and (2)

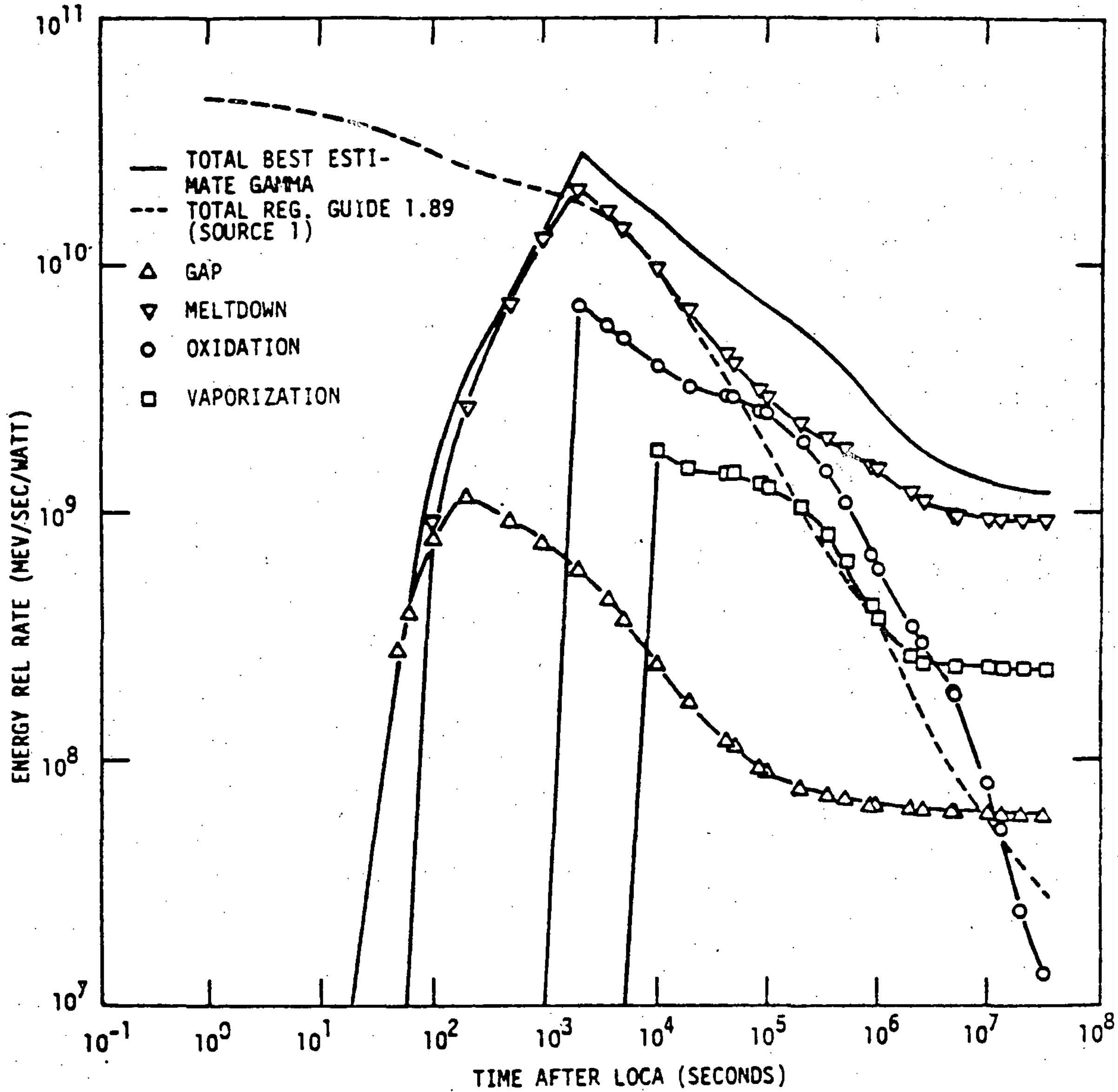


FIGURE 28. GAMMA-RAY ENERGY RELEASE RATE FOR "BEST-ESTIMATE" SOURCE SHOWING THE CONTRIBUTIONS OF THE CONSTITUENTS, AND COMPARED TO THE TOTAL REGULATORY GUIDE 1.89 SOURCE RESULT.

(REFERENCE: SAND 78-0349)

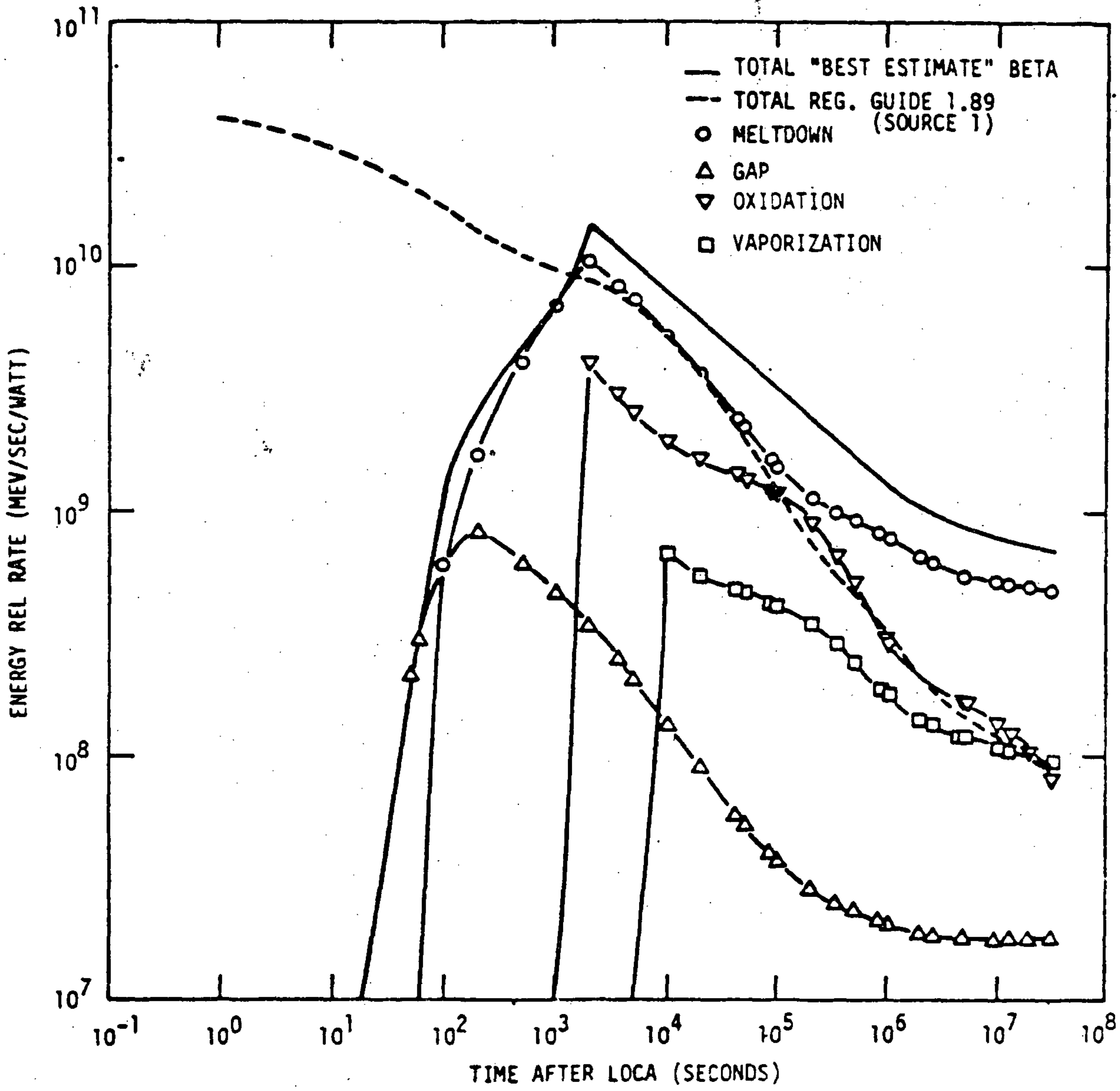


FIGURE 29. BETA ENERGY RELEASE RATE FOR "BEST-ESTIMATE" SOURCE SHOWING THE CONTRIBUTIONS OF THE CONSTITUENTS, AND COMPARED TO THE TOTAL REGULATORY GUIDE 1.89 SOURCE RESULT.

(REFERENCE: SAND 78-0349)

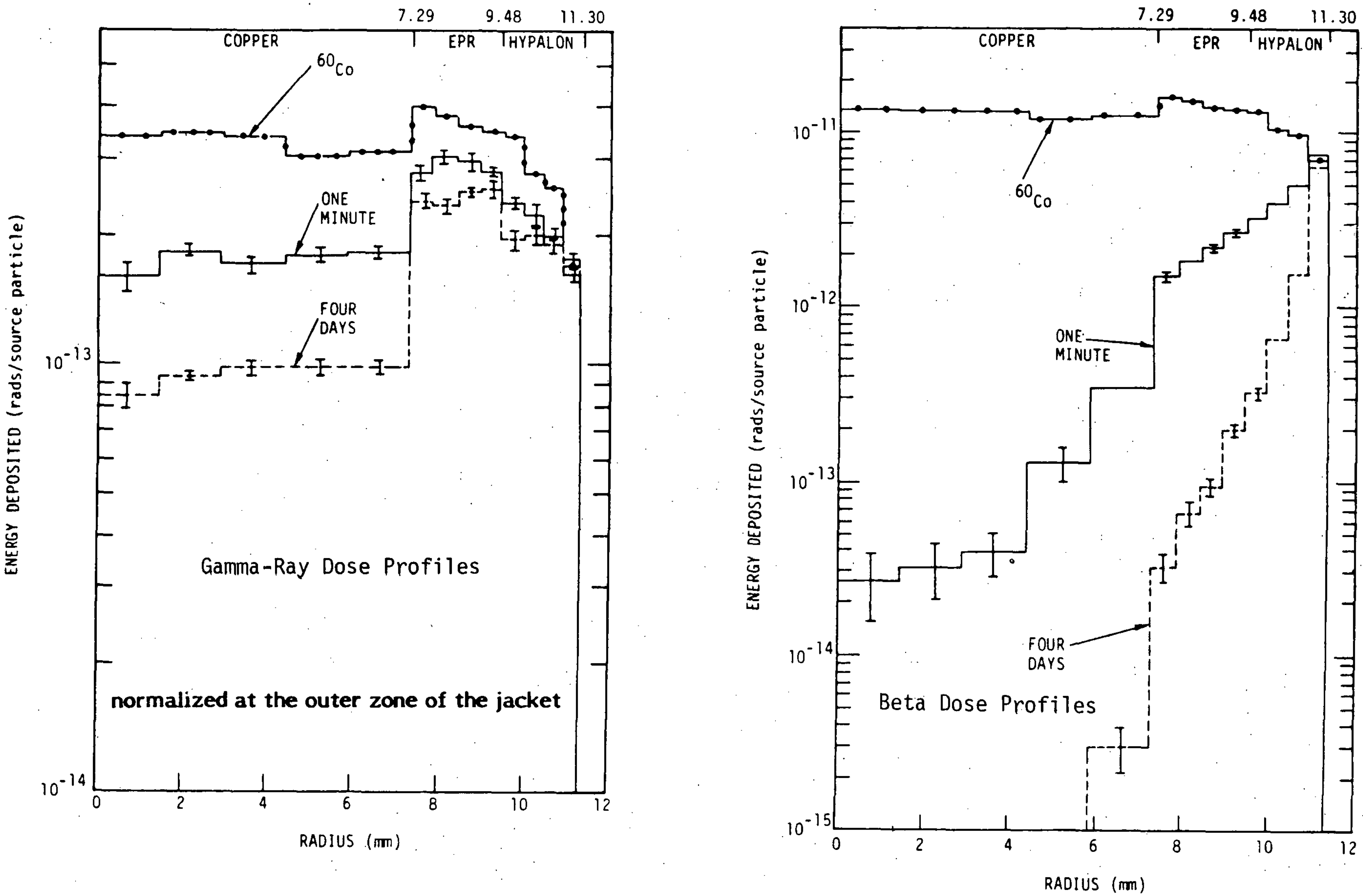


FIGURE 30. CALCULATIONS OF RELATIVE DEPTH-DOSE PROFILES IN A TYPICAL ELECTRICAL POWER CABLE FOR COBALT-60 COMPARED TO THE BEST ESTIMATE BETA AND GAMMA-RAY PLATE-OUT SOURCES AT COOLING TIMES OF ONE MINUTE AND FOUR DAYS AFTER A LOCA

(REFERENCE: "BEST-ESTIMATE LOCA RADIATION SIGNATURE" IRT 0056-005) simultaneously, with radiation and steam and chemical environments imposed together. The same experimental test chamber and identical test samples are used in both cases. The specimens from both tests are subjected to a qualitative comparison of performance, using on-line measurements and post-test evaluation. Tests have been conducted on electrical cables, paints, connector assemblies, cable splices, and lubricants. The radiation source utilized was cobalt-60, and the usable test chamber size was approximately 10 by 20 by 51 cm (4 by 8 by 20 in.) The test profile was a composite of the PWR/BWR profile recommended in the appendix to IEEE Std 323-1974. The current series of tests was completed in 1977, and a report on synergistic effects in LOCA qualification testing has been prepared.[86] No functional synergisms were found in these preliminary tests.

Task 2 - Generic Test Data

Generic test data have been obtained through a contract with a commerical testing laboratory. These data have been used by Sandia Laboratories in its evaluation of the synergistic effects associated with LOCA testing. The commerical testing laboratory has been used in an advisory capacity for the design of a new test facility at Sandia Laboratories and will continue to be used in formulating a long-term test program at Sandia.

Task 3 - New Test Facility

Sandia has recently completed construction of a new test facility (see Figure 31) which will be able to accommodate larger and more diverse Class IE test items. The actual test chamber is 52 cm ID by 150 cm high. The test facility is designed to allow a wide selection of radiation dose rates with better control of dose profiles. Typical coolant steam environments for postulated accidents can be applied simultaneously with the postulated accident radiation environment.

Task 4 -Test Plan

A test plan will be prepared for the new test facility by Sandia Laboratories, defining the components and materials, the types of tests to be conducted, and the performance parameters to be monitored and evaluated. The basis for this test plan is an evaluation of the LOCA-sensitivity of safety-related equipment. The design data on safety-related equipment have been obtained by subcontract with a reactor plant design organization using a pressurized-water reactor currently under design. The specific data obtained include a list of safety-related equipment along with the functional requirements for normal service and for accident conditions, plant location, and normal-service and accident environmental conditions. In addition to the items that are currently being tested (electrical cable, cable connectors, and splices) it is anticipated that the test plan will include such items as electric motor and solenoid valve operators, limit switches, electronic components, junction boxes, pressure transmitters, neutron detectors, electrical signal cable for instruments, resistance temperature detectors, and materials and components utilized in hydrogen recombiners, decay heat removal equipment, and containment penetrations. Specific tests will probably include aging, thermal radiation, LOCA, and main-steam-line break.

In order to ensure that all qualification tests are conducted on materials and components that meet current NRC requirements for quality assurance, a comprehensive review of all potential test specimens will be conducted before conducting any methodology tests. Test specimens will be chosen from generic categories identified by a survey of manufacturers supplying the material or component to be tested. A pretest review of the specific test item will be made to identify factors that could impact the methodology assessment. When test specimens have been judged to have been designed, fabricated, and procured in accordance with NRC quality assurance requirements and this judgment has been confirmed by a pretest inspection and/or test, methodology-oriented qualification tests will be conducted for the purpose of confirming existing test procedures and to provide data that can be used to modify these procedures where required.

Fire Protection Research

The fire protection research program is based on research areas identified in NUREG-0050[87] and by review of current design standards and guidelines. The program is aimed at providing confirmation data relating to the operation of Class I system under the conditions they would encounter during the appropriate design-basis fires.

The following specific program elements are included in the fire protection research program:

1. Obtain data on the effectiveness of cable-tray separation criteria in ensuring the functional integrity of redundant safety systems.

2. Obtain data on the effectiveness of conduits, fire barriers, and penetration firestops.

3. Obtain data on the effectiveness of coating materials.

4. Obtain data on the fire retardancy of aged materials.

5. Obtain data on IEEE Std 383-1974 and development of improved small-scale cable-system qualification tests.

6. Obtain data on the effectiveness of safety-related equipment (other than cable) when subjected to exposure fires.

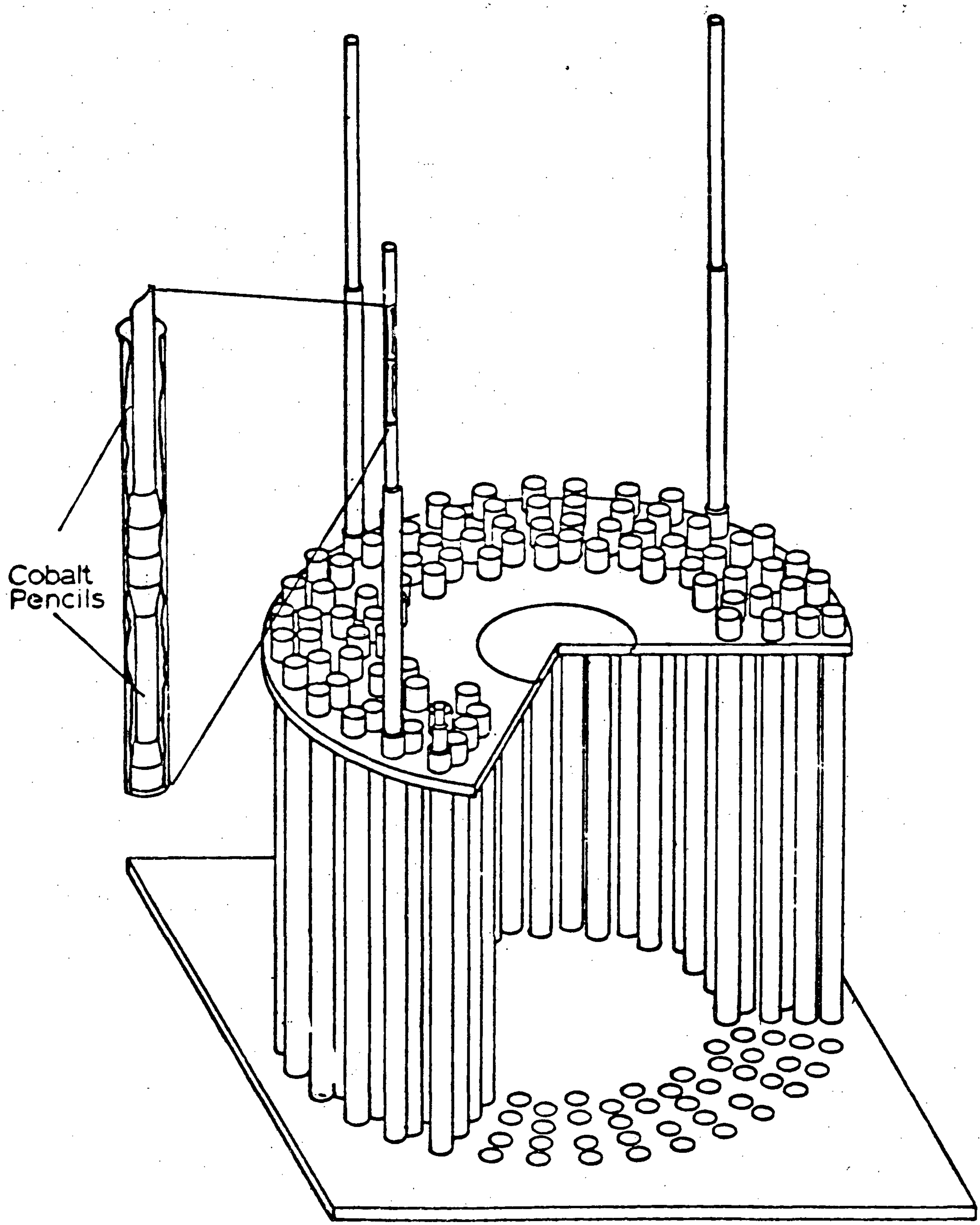


FIGURE 31. HIGH INTENSITY ADJUSTABLE COBALT ARRAY

7. Obtain data on fire detection system performance.

8. Obtain data on the effectiveness of water and other fire-extinguishing agents.

The first program element is based on work at Sandia Laboratories started in 1975 to confirm cable-tray-separation design practice. The scope of this effort was expanded, and program elements 2 through 6 added to cover the effectiveness of additional cable-tray configurations and components and an evaluation of separation criteria utilized for other safety-related equipment. Program elements 7 and 8 cover specific research areas required for licensing decisions and will be implemented in fiscal year 1979 and 1980. These elements will also be used in support of Regulatory Guide 1.120,[88] "Fire Protection Guidelines for Nuclear Power Plants."

Program Element 1 - Cable-Tray Separation

In support of some of the provisions of NRC Regulatory Guide 1.75[89] "Physical Independence of Electric Systems," tests were conducted at Sandia with varying separation distances to determine the minimum separation necessary for cables most susceptible to fire. Vertical separation distances from 152 cm (5 ft) down to 26.7 cm (10.5 in) and horizontal separation distances from 91 cm (3 ft) down to 20 cm (8 in) were tested. For electrically initiated fires in a horizontal open-space configuration, it was determined that a fire will not propagate from the ignited tray to adjacent trays. These tests were conducted with fire retardant 12-gage single-conductor and 12-gage triplex wire, utilizing both uniform and random-pattern cable packing.

Tests were also conducted with an experimental exposure (fuel) fire. The objective was to determine whether cable-tray separation alone is sufficient to prevent fire propagation between trays and between redundant safety divisions if an exposure fire resulted in a fully developed cable-tray fire.

The type and size of the worst-case exposure fire that must be considered for licensing are based on a fire-hazard analysis and will vary from plant to plant; they will also differ among different locations within the plant. Accordingly, no attempt was made to define a design-basis fire for the exposure-fire tests. Single-tray tests were conducted to find a reasonable set of conditions that would result in a fully developed cable-tray fire. The experimental exposure fire was then used in all full-scale cable-tray exposure-fire tests. Propane burners were used to start an exposure fire in one tray, with a barrier placed between it and the tray above. When a fully developed fire was obtained in the first tray, the burners were turned off and the barrier was removed. This method allows experimental study of fire propagation from tray to tray under specific conditions and without the exposure fire effecting the other cable trays.

A series of tests were conducted on arrays of cable trays, with both electrical and exposure-fire initiation. An array of 14 closely spaced cable trays was used to simulate a single safety division. Simulated redundant safety divisions were separated by the required 152-cm (5-ft) vertical and 91-cm (3-ft) horizontal distance. (See Figure 32). The results of these tests were summarized in a Research Information Letter.[90] The principal conclusion was that a fully developed fire in the bottom cable tray of a stacked array may propagate to a redundant safety division without fire suppression systems (as expected). On the other hand, electrically initiated fires do not propagate because they do not result in a fully developed cable tray fire.

In order to determine the characteristics of a cable-tray fire in cable tunnels or in areas where structural walls are close enough to the tray to influence the fire, some of the tests were repeated to include the effect of walls and ceilings. The first tests were conducted with a 91-cm (3-ft) vertical and 30-cm (1-ft) horizontal separation. Other separation distances, representative fire loadings, and closed (dead-ended) tests may be included in the testing program. The preliminary indication is that there is a greater chance of fire propagation under these conditions than with a similar configuration in an open area.

In typical plant installations, cable trays are oriented vertically at some locations and in others are oriented both vertically and horizontally. Vertical cable trays have been[91] and will be tested in both the open-space configuration and with walls and ceilings close enough to affect the fire.

Program Element 2- Effectiveness of Fire Shields

Researchers at Sandia Laboratories completed a series of tests using different fire shields:

- ceramic wool blanket over ladder tray
- solid bottom tray with no cover
- solid cover on ladder tray with no vents
- vented cover on solid bottom tray
- 2.54-cm (1-inch) fire barrier (thermal board) between trays

using single and double tray configurations as well as electrical cable which passed the IEEE Std 383-1974 flame retardancy test[92] and cable which did not pass this test.

The results[93] of the Sandia research showed that all fire shield designs offered some protection. None of the cable which passed the flame retardancy test in IEEE Std 383-1974 ignited. It is possible to ignite the cable which did not pass this flame retardancy test; however, no propagation was observed past the fire shields.

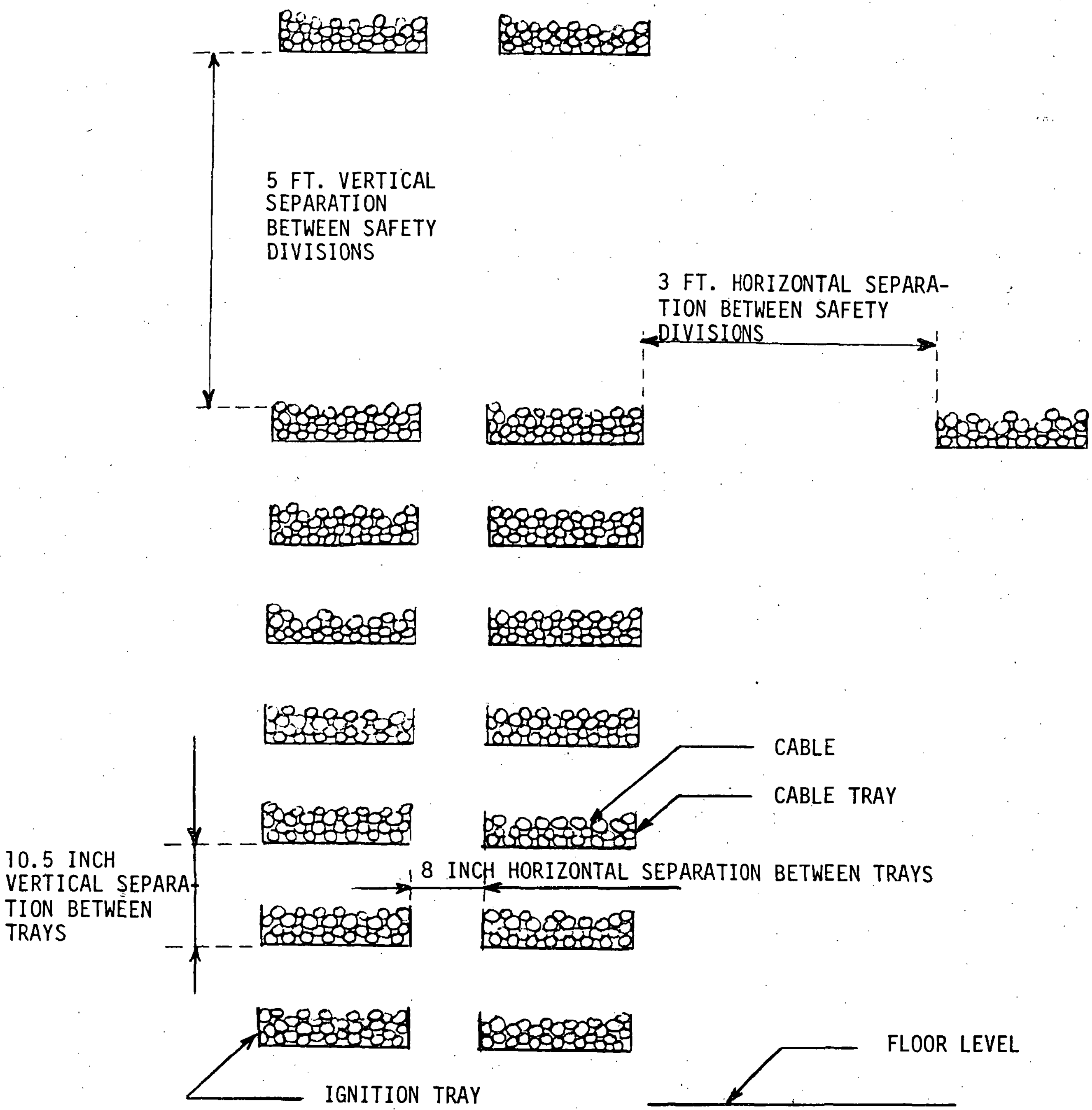


FIGURE 32. HORIZONTAL OPEN SPACE CABLE TRAY ARRANGEMENT (STACKED TRAY TESTS)

Program Element 3 - Effectiveness of Fire-Retardant Coating-Materials

The objective of this program element is to provide information on the effectiveness of fire-retardant coating materials when used in typical cable-tray installations.

A survey of coating materials available for use in cable trays was initiated in August 1976. Generic types were chosen for testing and evaluation in small- and large-scale cable systems tests. Small-scale tests on basic coating properties have been conducted by using six coatings and two cable types. Full-scale tests were conducted using both single and double trays.

While the results[93] showed that all coatings offer a measure of additional protection, there was a wide range in the relative effectiveness of the different coatings tested. No propagation to the second tray was observed in any of two-tray tests in which cable that passed the IEEE Std 383-1974 test was used. (Propagation was observed in three tests involving cable which did not pass the IEEE Std 383-1974 test). Overall, a good correlation was obtained between small-scale and large-scale tests.

Program Element 4 - Fire Retardancy of Aged Materials

The objective of program of element 4 is to provide information on the fire retardancy of aged materials. As presently planned, the program is aimed at cable and coating materials used in the first three program elements but may be expanded at a later date to include materials from other safety-related equipment.

Both accelerated and natural aging will be considered in the program. The results of an ongoing material-aging study at Sandia Laboratories will be applied. Some naturally aged cable and coating samples will be tested and the fire retardancy compared with that of cable aged by accelerated methods. The physical properties of cable material will be evaluated only to the extent that they affect the fire retardancy of the material. Cracking and spalling of protective coating material applied to a cable tray and changes in the bond between coating and cable and between coating and tray will be examined after simulated long-term aging.

The present plan calls for a survey of generic cables to identify the fire retardants in current use. A screening test will be developed for determining the figure of merit for stability as a function of age and ambient temperatures. Small-scale tests will be conducted on different generic cable insulations and fire-retardant additives to determine the worst-case combinations. Aging tests will be performed to determine aging-acceleration functions using the methods being developed in the qualification-testing evaluation program at Sandia Laboratories. Concurrently, cables will be aged by currently used methods, and full-tray electrical initiation and/or exposure-fire tests will be conducted.

Program Element 5 - Fire Tests for Cable Systems and Systems Components

This program element was set up to establish how well defined and repeatable is the cable flame-retardancy test of IEEE Std 383-1974 and to what extent total cable-system performance can be predicted from this small-scale tests.

Certain test features not specifically addressed or defined in the current standard were studied at Underwriters Laboratories to determine their significance, specifically:

- Test cell size and configuration
- Air flow requirements
- Cable-tray design and orientation to the flame source
- Flame source energy rate
- Extent to which tray is filled with cable (percentage of volume)
- Cable size and material
- Mixing cable sizes and materials
- Cable-tray orientation (vertical and/or horizontal)

As a result of this UL study[94] suggestions for revisions of IEEE Std 383-1974 were made with respect to (1) construction of cable trays, (2) test enclosure, (3) ventilation, (4) type, size and spacing of cable ties, (5) measurement of fuel and air rates, (6) flame temperature measurement, (7) initial ambient temperature, and (8) reporting of results.

Program Element 6 - Effects of Fires on Other Safety-Related Equipment

The objective of this program element is to evaluate the effects of exposure fires on safety-related equipment. Only safety-related equipment that is not totally isolated from redundant equipment will be considered.

The scope of this program element will be established by other NRC studies on the probability of redundant safety-related equipment being affected by the same exposure fire and the risk assessment for this event. The risk assessment will be made by first identifying those areas in the plant in which it is difficult to totally isolate redundant safety-related equipment (e.g., the cable spreading room and the control room) and then examining the effects of a plant-specific design-basis exposure fire on all safety-related equipment in that area.

Program Element 7- Performance of Fire-Dectection Systems

This program element is concerned with testing the performance of fire-detection systems. It will provide data that can be used as the basis for a guide or standard for the design, installation, and utilization of systems for detecting fire and products of combustion. A survey and performance evaluation of currently used fire and smoke detection systems will be conducted. The evaluation will take into account constraints and requirements imposed by typical plant installations, including air flow and local air stratification. Detection system sensitivities will be established and evaluated for adequacy under conditions prevailing in typical design-basis fires; this should help determine detector location and response-time requirements. Development of in-place test methods will be considered for detection systems.

Program Element 8 - Effectiveness of Extinguishing Agents

The objective of this program element is to test the effectiveness of water and other fire-extinguishing agents and their potential damage to safety-related equipment. An improved technical base will be provided for establishing criteria on when and where water and other fire extinguishing agents can be used. This effort is planned not as a complete study on fire-extinguishing systems, but rather as a study of technical issues identified as requiring evaluation.

Current plans call for determining the minimum concentration and soaking time required for Halon and carbon dioxide to extinguish a fully developed cable-tray fire. Later testing will study the effectiveness of water as a means of fire suppression. The testing will be conducted in a facility large enough to simulate a full-scale cable fire and typical plant enclosure. Data on smoke and fire detection and actuation systems will also be obtained.

Noise Diagnostics

Noise diagnostics refers to the study of the fluctuating portion of plant signals to gain an improved understanding of function, especially system dynamics. Typical plant signals include neutron flux, coolant pressure, temperatures, acoustic pressure, vibration (acceleration, velocity, displacement), and coolant flow. Noise diagnostic techniques are attractive for system studies because they are nonperturbative to operations.

The noise diagnostic research program at the Oak Ridge National Laboratory [conducted in conjunction with the NRC Offices of Nuclear Regulation (NRR) and Standards Development (SD)] has supported licensing activities by the use of noise diagnostics techniques in independent assessments of core-barrel motion in operating pressurized water reactors and in-core instrument-tube vibrations in operating boiling water reactors of the BWR-4 type. More recent noise diagnostic studies jointly supported by RES, NRR, and SD have been concerned with assessing the performance of existing loose-parts-monitoring systems in operating reactors. The current RES program includes methods development studies, laboratory research on loose-parts monitoring, and assessment of the use of noise diagnostics techniques to determine reactor stability. The noise diagnostics studies support the development of guides and standards such as Regulatory Guide 1.133,[95] "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," and proposed regulatory guides on core-barrel motion monitoring and pipe vibration. This research program supports NRR efforts as needed for independent evaluation of the cause and correction of various vibration-induced malfunctions at operating nuclear power plants.

Noise diagnostics has been used in the past to identify and monitor system malfunctions in a number of operating nuclear power plants; for example, it was used by NRC to analyze and quantify excessive core-barrel motion in the Palisades nuclear power plant.[96] (See Figure 33). The problem was studied with the aid of a Fourier analyzer, using onsite measurements of signals from in-core and out-of-core neutron-flux chambers, primary coolant temperature variations, and out-of-vessel accelerometers to compute the power spectral density. Technical specifications were later issued by NRC to provide guidance on monitoring for excessive core-barrel motion in operating PWRs.[97] More recently, noise analysis was used by NRC to assist in determining the cause, and monitoring the magnitude, of instrument-tube vibrations to BWR-4 cores.[98] (See Figure 34).

Noise analysis is a powerful tool for diagnosing and identifying the source of malfunctions that cause the vibration of reactor internals and flow anomalies. The ultimate use of this technique would be in conjunction with an automated surveillance and diagnostic system to provide an alarm in the control room on the detection of an impending component failure or system malfunction.

Monitoring systems for loose parts are being installed in nuclear stations currently starting operation. They use signals from accelerometers and sonic detectors mounted on primary-system piping and the reactor pressure vessel to locate and identify loose metallic parts in the primary system. Loose parts could result in primary-system degradation by blocking fuel coolant channels, impairing valve functioning, or damaging pumps. Early detection and identification of the size and location of loose parts are thus important to safe operation. The performance and use of existing systems were evaluated in 1977.[99]

Reactor kinetics and stability measurements were made in 1978 at several BWRs, using standard transfer-function techniques and analytical calculations of reactor systems kinetics. Several months are required to obtain stability results by this technique. In addition, the result is dependent on an exact knowledge of many reactor coefficients and the model employed. Noise diagnostics has been proposed by Japanese researchers as a means of measuring reactor stability directly.

Noise diagnostics has been applied to two recent operational incidents: the power oscillations at the Fort St. Vrain nuclear power plant and the transient at Three Mile Island Unit II (TMI-2). In both cases, ORNL

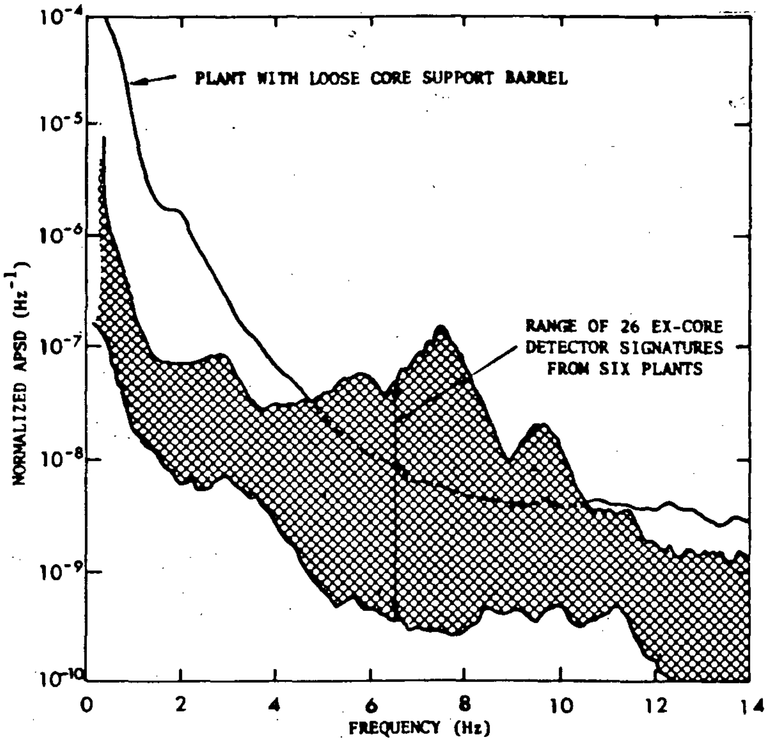


FIGURE 33. COMPARISON OF AN ABNORMAL SIGNATURE WITH A RANGE OF BASELINE SIGNATURES FOR PWRs.

(REFERENCE: NUREG/CR-0525)

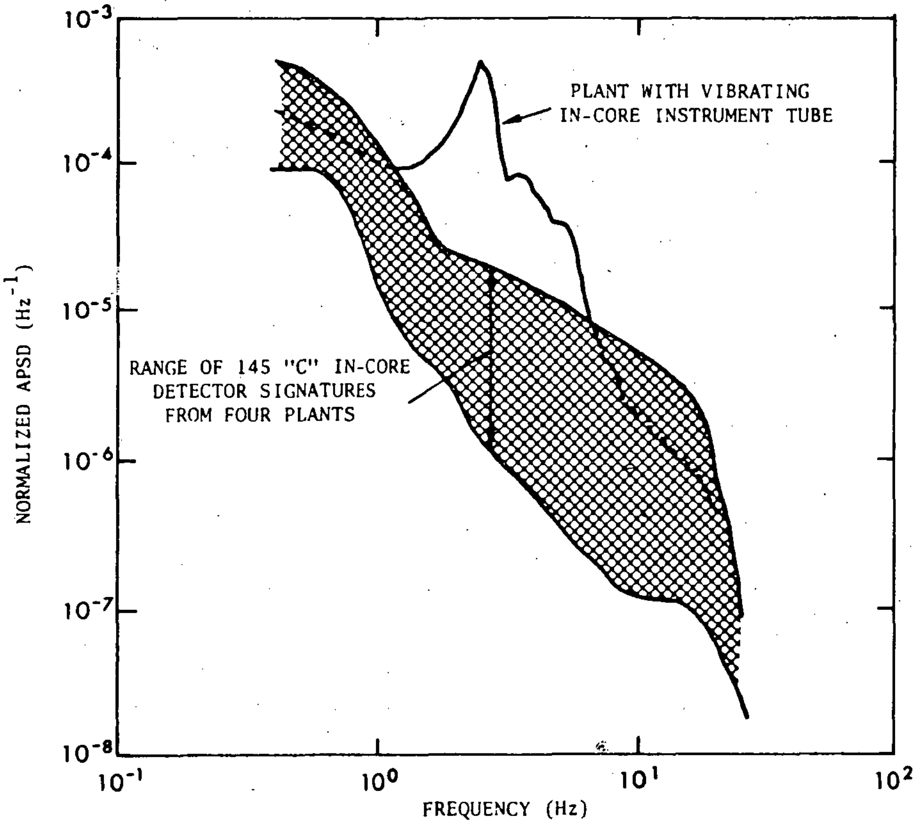


FIGURE 34. COMPARISON OF AN ABNORMAL SIGNATURE WITH A RANGE OF BASELINE SIGNATURES FOR BWR-4s.

(REFERENCE: NUREG/CR-0525)

researchers were called in to evaluate plant signals in order to determine the condition of the plant. At TMI-2, the ORNL researchers were able to show in both the forced and natural circulation phases that the plant was in the condition stated and most importantly that there was no bulk boiling.

Human Factors Research

The human factors research program is concerned with assessing the role of human errors in reactor operational safety. It includes specific studies in support of human error investigations and the development of associated training programs by the NRC Office of Inspection and Enforcement, the study of safety-related operator actions in support of the development of guides and standards by the NRC Office of Standards Development (SD), and a continuing review of the application of ergonomics in the design of nuclear power plants.

Human errors have been a significant cause of abnormal occurrences in nuclear power plants.† The contribution of human errors to the unavailability of safety systems and components was noted in the Reactor Safety Study.[32] The latter stated that "an actuarial data base for human error rates in nuclear power plants does not exist" and "in general, the design of controls and displays and their arrangements on operator panels in the nuclear plants studied in this analysis deviate from human engineering standards specified for the design of man-machine systems and accepted as standard practice for military systems." The report[100] to the American Physical Society by the Study Group on Light-Water-Reactor Safety recommended that "human engineering of reactor controls, which might significantly reduce the chance of operator errors, should be improved. We also encourage the automation of more control functions and increased operator training with simulators, especially in the accident-simulation mode." Human errors and the design of control rooms were also identified as a major area of concern by former General Electric employees, in testimony before the Joint Committee on Atomic Energy.[101]

In an effort to determine whether improvements could be made, NRC contracted in 1976 with the Aerospace Corporation for a study of control-room displays and operator performance; the final report[102] was issued in March 1977. This study identified a number of instances in which the design of control rooms in operating nuclear plants was not based on optimum human engineering principles. It did conclude that control-room designs and associated operating procedures, utilizing current training and licensing practices for nuclear operators, are sufficient to ensure safe operation. Improvements in control-room design, operator training, and operating practices to increase the margins of safety were recommended. Other recommendations for improved human engineering in control centers were presented.

Advanced control-room designs utilizing graphic displays (CRTs) in conjunction with computers are being proposed for the next generation of nuclear plants. Typical examples are the General Electric Company's NUCLENET[103] and the advanced control-room design developed by Combustion Engineering, Inc. These control rooms provide for more informative displays for system diagnosis and hence should contribute to safer and more efficient plant operation. Safety criteria and guides for assessing advanced control-room designs will have to be prepared by NRC after the completion of a detailed technical review of the requirements for human engineering.

The NRC is a member of the Halden Program* Group, which is conducting a number of studies on automatic process supervision and control in nuclear power plants. The use of advanced control-room displays, computer operation, and diagnostics is being studied and demonstrated experimentally at the Halden reactor facility.

Training programs in MORT (Management Oversight and Risk Tree), RSOS (Reported Significant Observation Studies), risk management, and accident investigation techniques have been given to DOE contractors and employees for several years. The NRC currently does not have an ongoing program for training inspectors and licensee reviewers in the principles of ergonomics. Many of the techniques and principles presented in manuals and training programs for the above-mentioned topics would, with minor changes, be applicable to nuclear power plant safety studies and investigations.

The NRC participates in the development of industrial guides and standards. An industrial standard (ANS 51.4, ANSI N660) has been drafted and issued for trial use in evaluating safety-related operator actions.[104] As noted in this standard, "there are now no generally accepted criteria for safety related operator actions." The standard proposes criteria for determining the time to be allowed for manually initiating the operation of safety systems. However, a firm technical base is lacking for the criteria proposed, and the need for research in this area is highlighted. As a result, NRC is sponsoring a modest research program to develop the needed technical base. The first phase of this research has recently been completed[105] and shows that it is possible to develop information on the response time of reactor operators to various operational incidents. Figure 35 shows the process used by ORNL researchers to obtain and evaluate information on operator response. For one event, inadvertent safety injection, it was possible to obtain sufficient data on specific operation actions to perform a graphical analysis using probability plotting. (See Figure 36). This very preliminary work suggests that the mean times to respond are within the range of values proposed in the draft of ANSI N660 as released for trial use and comment. An expansion of this and related work is anticipated.

A general review of the role of human errors as reported in NRC licensee event reports and DOE abnormal occurrence reports is being made at INEL. This study is expected to categorize the role of human errors and identify potential future research that would contribute to reducing the potential for human errors in the operation of nuclear power plants.

† See the NRC Annual Reports for 1976, 1977, and 1978.

* An experimental program conducted by the Organization for Economic Cooperation and Development (OECD) at the Halden BWR facility in Norway. (See also discussion of Halden on page 13.)

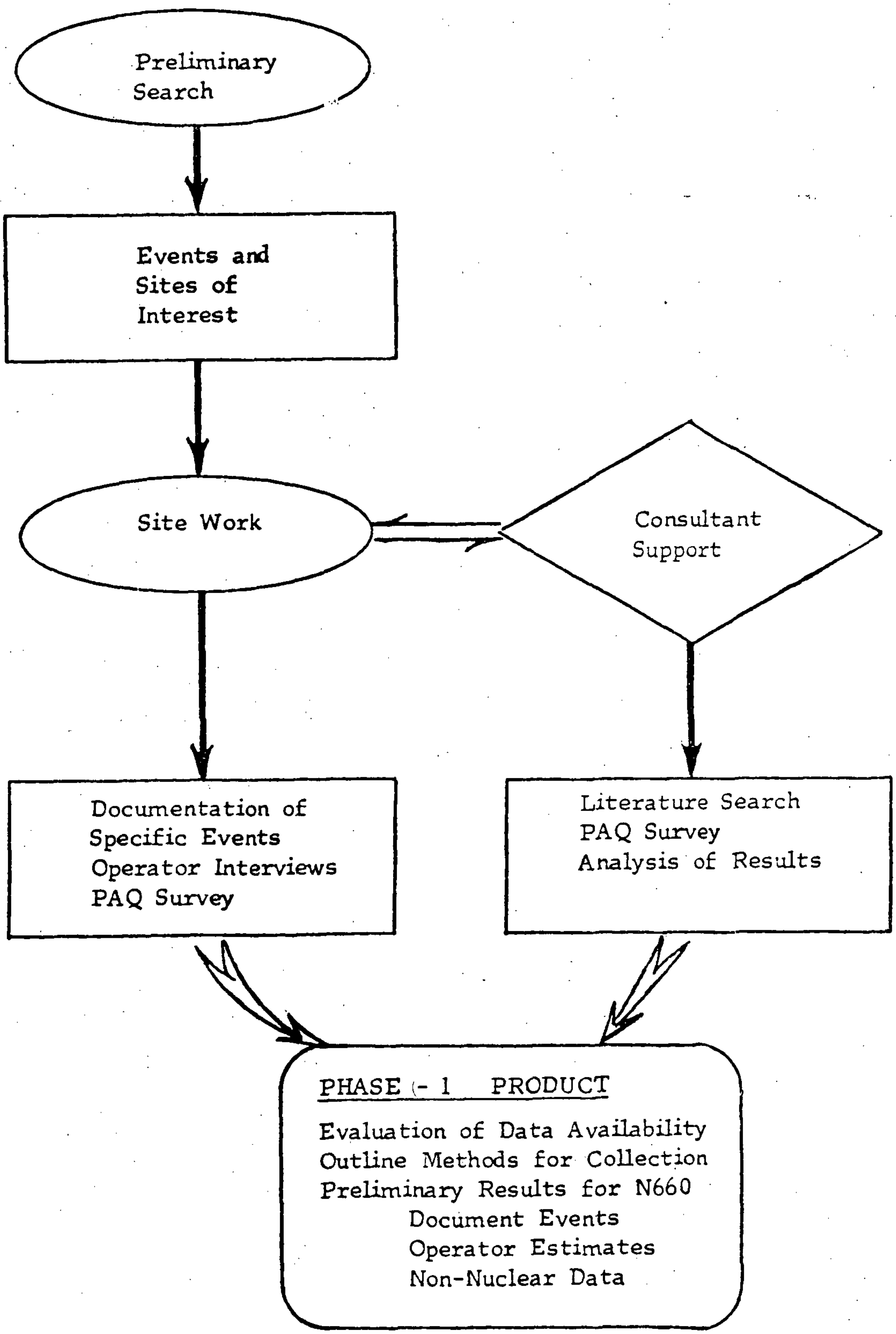


FIGURE 35. WORK FLOW DIAGRAM ON SAFETY RELATED OPERATOR ACTION

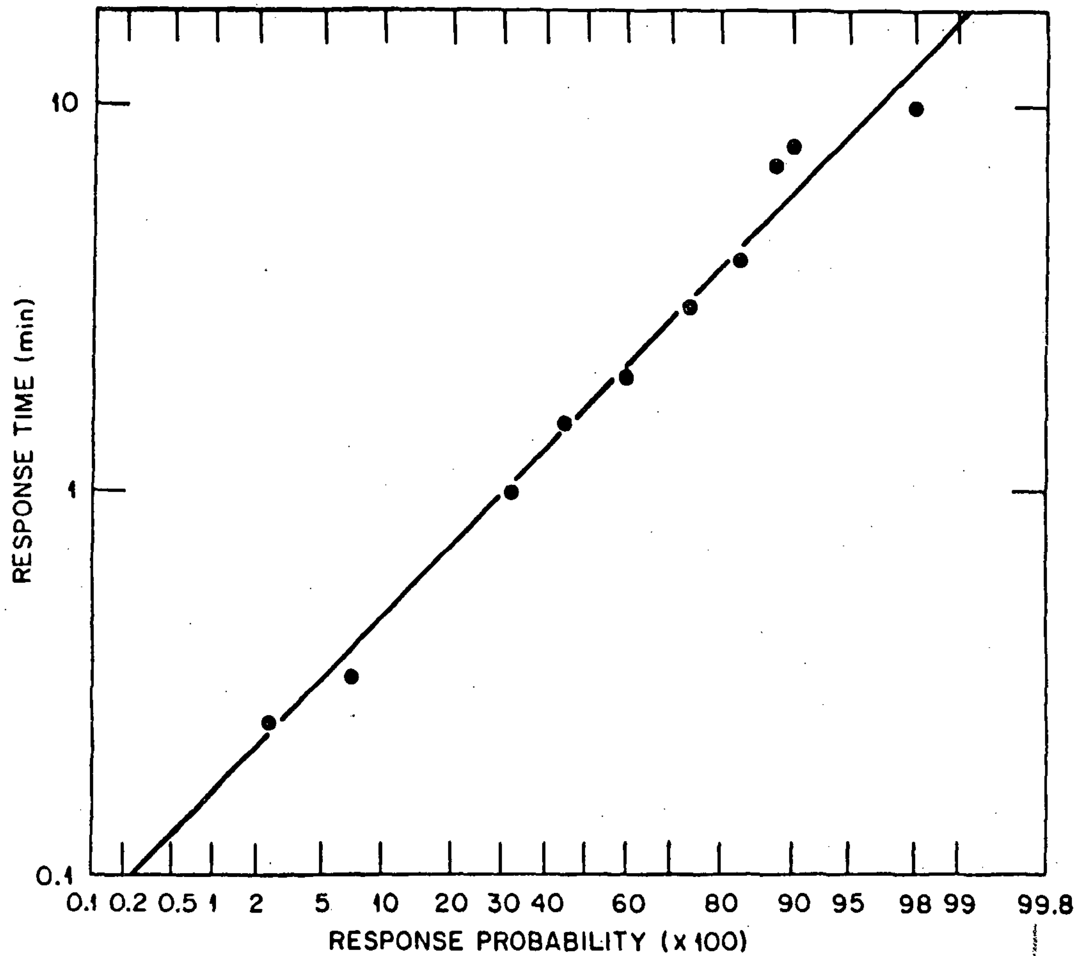


FIGURE 36. INADVERTENT SAFETY INJECTION PROBABILITY PLOT

(REFERENCE: NUREG/CR-0901)

CONCLUSION

The NRC research programs in fuel behavior, metallurgy and materials, and operational safety have been developed in response to defined regulatory needs. These research programs have produced a considerable body of data of use to NRC, the public and industry in quantifying the safety margin of nuclear power plants. More information is expected in the future.

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*This list contains only recent NSIC publications. A complete list may be obtained from the Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee 37830.

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The NRC light-water reactor safety-research program is part of the NRC regulatory program for ensuring the safety of nuclear power plants. This paper summarizes the results of NRC-sponsored research into fuel behavior, metallurgy and materials, and operational safety. The fuel behavior research program provides a detailed understanding of the response of nuclear fuel assemblies to postulated off-normal or accident conditions. Fuel behavior research includes studies of basic fuel rod properties, in-reactor tests, computer code development, fission product release and fuel meltdown. The metallurgy and materials research program provides independent confirmation of the safe design of reactor vessels and piping. This program includes studies on fracture mechanics, irradiation embrittlement, stress corrosion, crack growth, and nondestructive examination. The operational safety research provides direct assistance to NRC officials concerned with the operational and operational-safety aspects of nuclear power plants. The topics currently being addressed include qualification testing evaluation, fire protection, human factors, and noise diagnostics.

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