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Operational Limitations of Light Water Reactors Relating to Fuel Performance*

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ABSTRACT

General aspects of fuel performance for typical Boiling and Pressurized Water Reactors are presented. Emphasis is placed on fuel failures in order to make clear important operational limitations. A discussion of fuel element designs is first given to provide the background information for the subsequent discussion of several fuel failure modes that have been identified. Fuel failure experiences through December 31, 1974, are summarized. The operational limitations that are required to mitigate the effects of fuel failures are discussed.

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1.0 INTRODUCTION

Reliable fuel performance is a necessity for achieving high plant availability and low energy costs. Plant restrictions due to fuel failures or actions required to reduce failures have led to lowered plant availability and capacity factors, resulting in a significant loss of power generation. In addition, fuel failures have the potential for (1) increasing radiation exposure to plant personnel and the public, and (2) compounding the problems of refueling, storage, waste management and disposal. Thus, it is very important that the problems of fuel failures be understood and that the operational restrictions concerning fuel failures be adhered to. The objective of this paper is to discuss the problems of fuel failures in light water reactors (LWR's) and the operational limitations that are required to reduce or avoid fuel failures.

Nuclear power plants are designed under the assumption that some fuel rods will fail and release radioactive fission products to the primary coolant system. Pursuant to this design practice, we shall focus our attention upon fuel failure problems for fuel performance evaluation. This approach is most likely to lead to sound operational procedures pertaining to fuel failures as well as to identify areas of potential improvement. Despite this negative approach, it should be pointed out that the success rate of LWR fuel has been generally high(99.8% or greater) and that there has been no impact on public health and safety from the fuel failures which have occurred.

As of the end of 1974, fuel performance data were available for 33 plants (14 BWR's and 19 PWR's) and constitute the bulk of the experience used in the present paper. The primary sources of fuel performance data for LWR's are vendors' topical reports, licensee reports, Nuclear Regulatory Commission's (NRC's) technical reports, trade journals, transactions of professional societies and special technical meetings.

Fuel element design aspects of both BWR's and PWR's are first discussed. Various modes of fuel failures are discussed next in conjunction with fuel failure experience in the past. Operational limitations pertaining to the most significant fuel failure modes are then discussed. A summary and important conclusions are also given.

2.0 FUEL PERFORMANCE

2.1 Fuel Descriptions

The evolution of the nuclear power industry has resulted in a variety of fuel designs. However, the general aspects of the fuel designs are similar in many ways for both BWRs and PWRs. For current generation LWR cores, the fuel rod is a hollow Zircaloy tube (cladding) into which cylindrical fuel pellets are stacked end to end. The cladding tube is sealed by welding end plugs into each end of the tube. The fuel is slightly enriched (typically 1 to 4 w/o U-235) UO_2 which is compacted into pellet form and then sintered at high temperatures with a resultant density of about 95% theoretical. Typical fuel rods are about 14-15 feet long containing an approximately 12-or 13-foot column of stacked pellets. The pellet columns are pressed downward by a spring in the top plenum of the tube. Figures 1 and 2 show the schematic of a BWR and PWR fuel rod, respectively.

The fuel rod is designed with radial and axial gaps within the cladding tube which are back-filled with helium. Fuel rods are characterized as "unpressurized" or "pressurized" depending on the backfill pressure of the gas (helium) at fabrication. The initial gap pressure is about atmospheric for "unpressurized" fuel rods and 200 to 400 psia for "pressurized" fuel elements.

A compilation of the important material properties of LWR fuels is given in Appendix A.

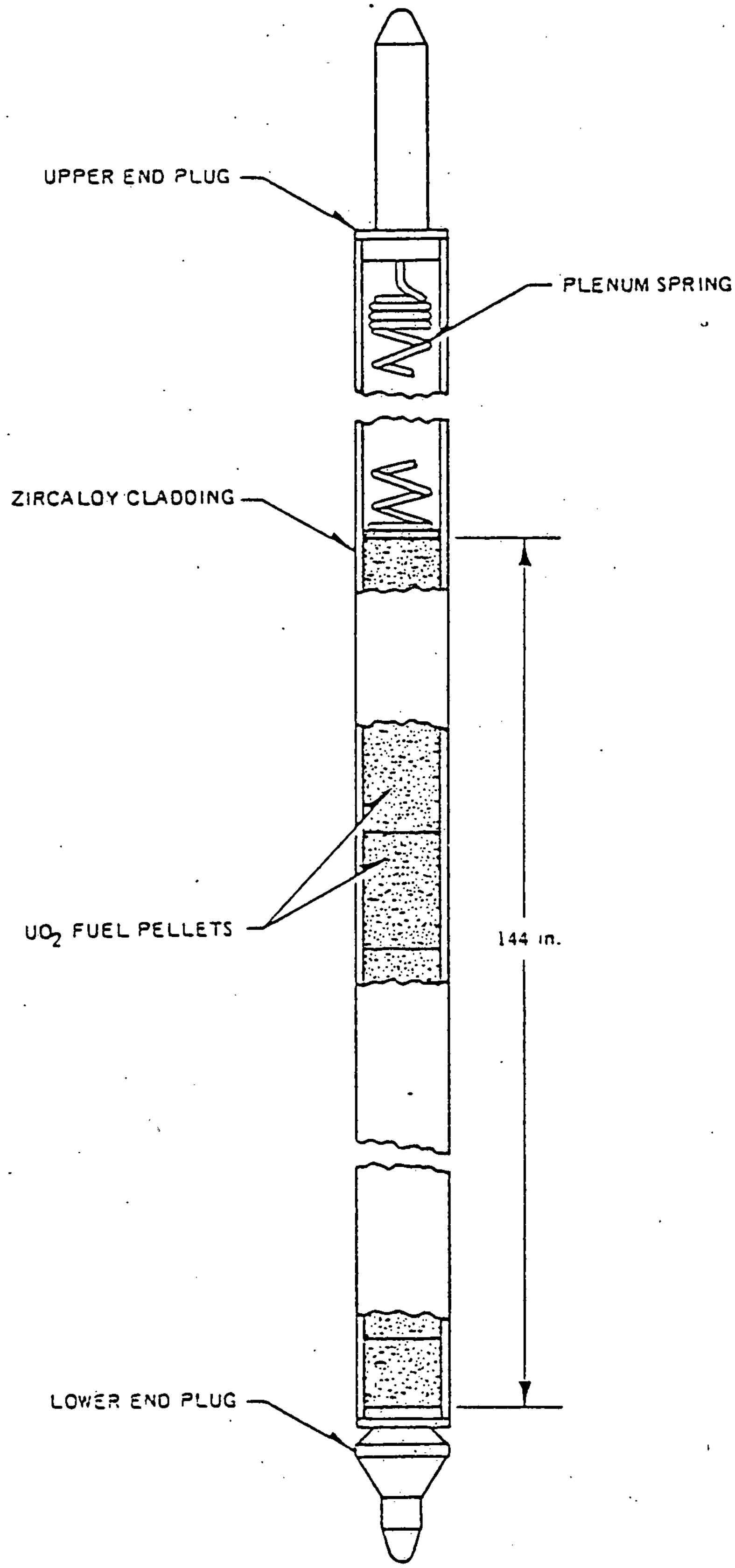
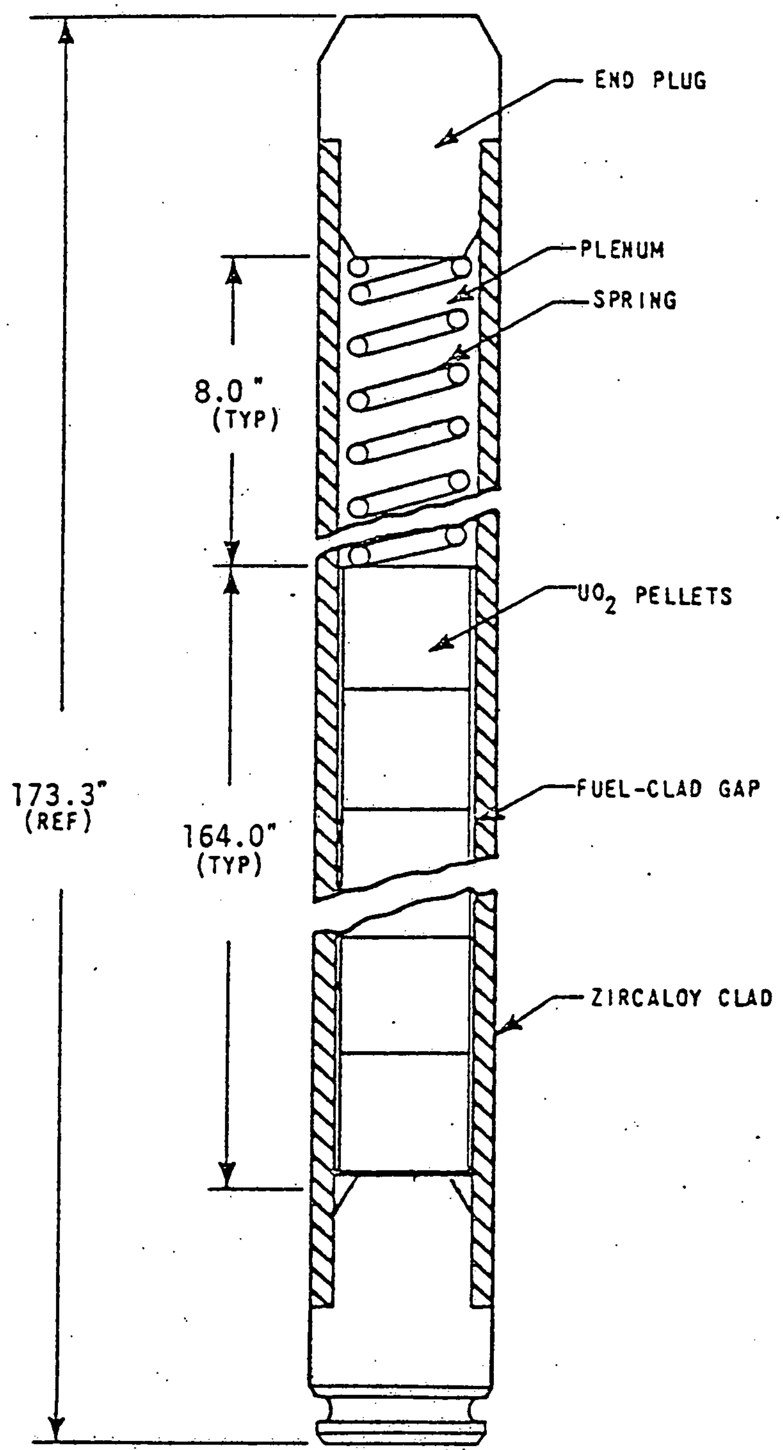


Figure 1. BWR Fuel Rod Schematic



SPECIFIC DIMENSIONS DEPEND ON DESIGN VARIABLES SUCH AS PRE-PRESSURIZATION. POWER HISTORY. AND DISCHARGE BURNUP

Figure 2. PWR Fuel Rod Schematic

Fuel rods are arranged in a square array to form a fuel bundle. They are uniformly spaced by means of a number of spacer grids. For BWRs, fuel bundles are also enclosed in a Zircaloy-4 channel box to prevent cross flow between fuel bundles. Figures 3 and 4 show the typical fuel assembly of a BWR and a PWR, respectively.

2.1.1 BWR Fuel Designs

As of December 31, 1974, there were 21 licensed BWR plants which had generated electrical power. Among them 20 were designed by General Electric while one (Genoa or LaCrosse) was designed by Allis-Chalmers. Some typical BWR fuel element design parameters are shown in Appendix B, Table B-2.

General Electric has utilized both stainless steel and Zircaloy-2 cladding in their fuel rod designs. Due to the advantages of improved neutron economy of Zircaloy-2 and the inherent problems of stainless steel cladding in the BWR water-steam environment, GE has concentrated primarily on Zircaloy-2 cladding, Zircaloy-2 cladding was used as early as 1960 in the Dresden 1 plant.

The basic features of the current GE fuel design have evolved as a consequence of the vast experience in the past 16 years. Fuel rod diameters in the range of 0.425 inch to 0.567 inch o.d. with cladding wall thickness from 30 to 35 mils and pellet-to-clad diametral gaps from 3 to 10 mils have

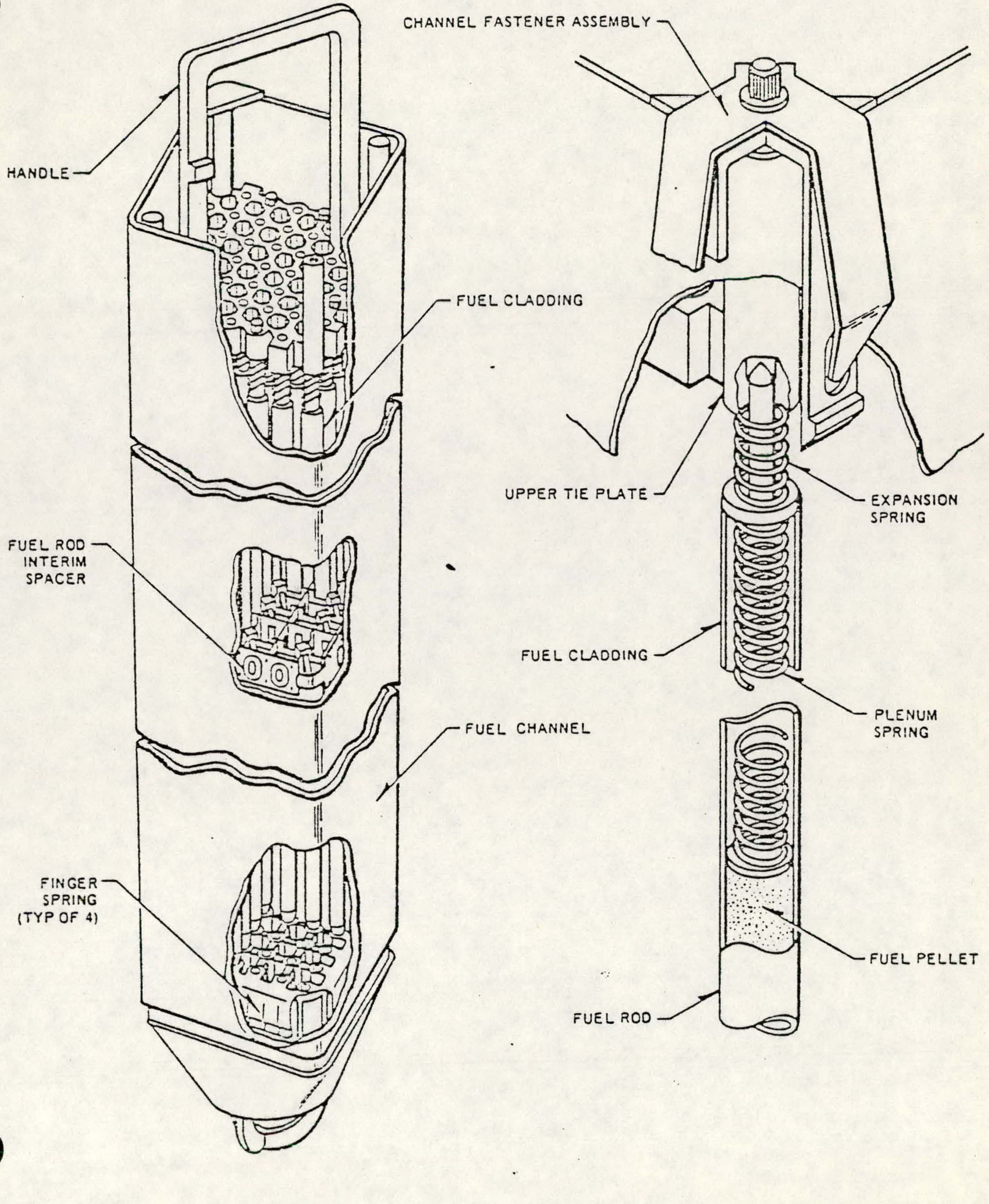


Figure 3. Fuel Assembly - GE BWR

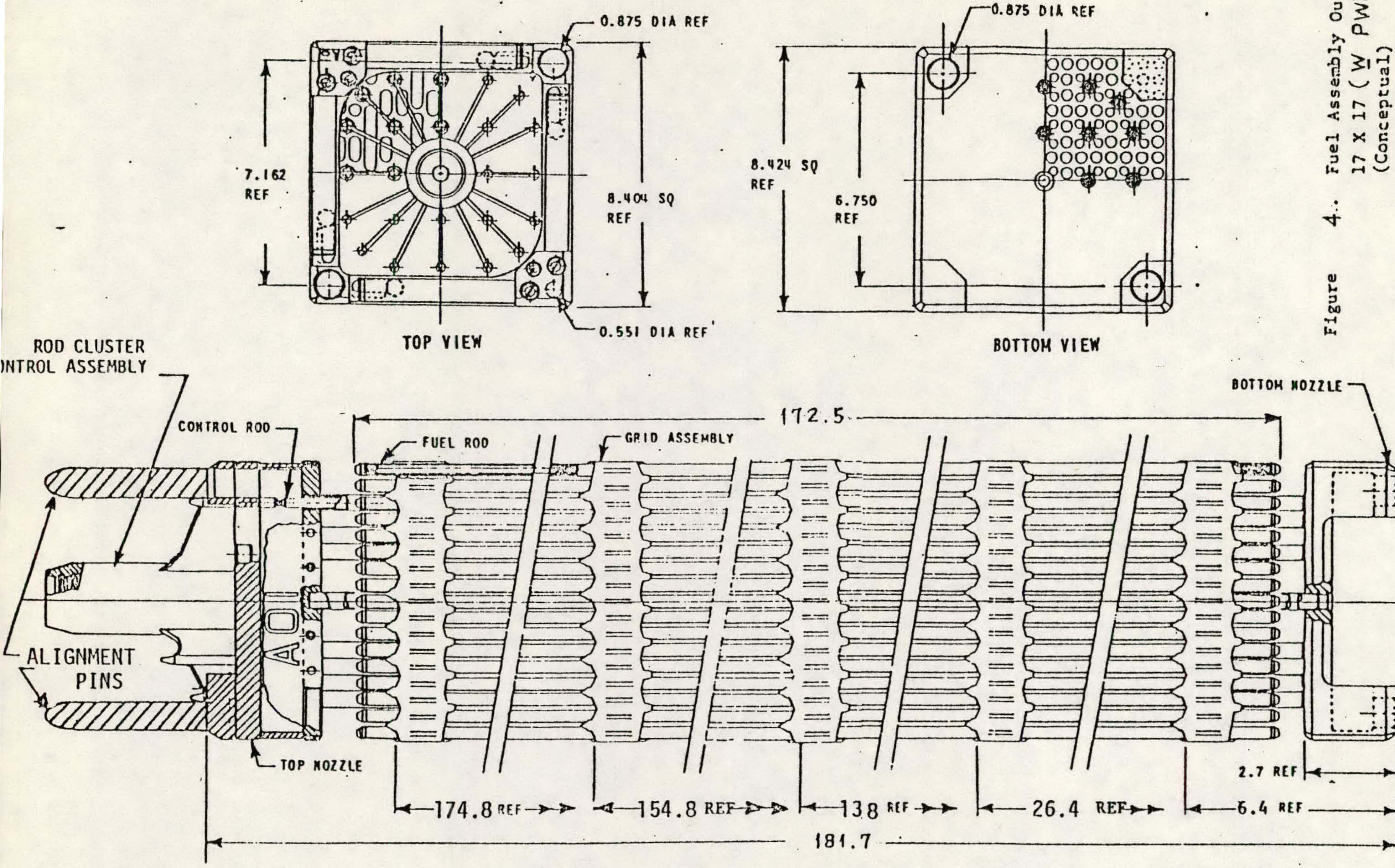


Figure 4. Fuel Assembly Outline 17 X 17 (W PWR) (Conceptual)

been used in production fuels. Active fuel column lengths have varied from 59.8 to 130.0 inches with fission gas plenum volume per unit of fuel volume from 0.013 to 0.100. The detailed information of the current BWR fuel designs is presented in Table 1.

The majority of GE BWR's utilized 7x7 fuel bundles. In order to reduce the linear heat generation rate (LHGR) per rod, the trend is to increase the total linear footage of fuel within the core. Therefore, the latest GE designs (BWR/6) utilize an 8x8 rod array; this type is also being used as reload fuel for most BWRs. Typical parameters for their "standarized" design are included in Appendix B, Table B-4.

BWR fuel rods are of the "unpressurized" type. Compared to PWR's, the cladding of BWR's is subjected to lower external pressure and operated at a lower temperature. These features tend to reduce the expected creep rate of the Zircaloy cladding such that the internal pressurization has not been deemed necesssry.

2.1.2 PWR Fuel Designs

As of December 31, 1974, there were 28 licensed PWR plants which had generated electrical power. Of these, 17, 7, and 4 were designed by Westinghouse, Babcock & Wilcox, and Combustion Engineering, respectively. Some typical PWR fuel element design parameters are shown in Appendix B, Table B-3.

TABLE 1

GENERAL ELECTRIC CURRENT FUEL DESIGN PARAMETERS

DESIGN PARAMETERS	8 x 8 (BWR/6)	7 x 7 (Browns Ferry-1)
Fuel Assemblies		
Number of Fuel Assemblies	848	764
UO_2 Rods per Assembly	63	49
Fuel Rod Pitch, in.	0.640	0.738
Overall Dimensions (including water gaps), in.	6.0 x 6.0	6.0 x 6.0
Fuel Weight (as UO_2), pounds	399,227	361,837
Number of Spacer Grids per Assembly	7	7
Overall Length, in.	176	175.83
Channel Wall Thickness, in.	0.120	0.080
Fuel Rods		
Number	53,424	37,436
Outside Diameter, in.	0.493	0.563
Pellet-to-Clad Gap, in	0.0045	0.006
Clad Thickness, in.	0.034	0.037
Clad Material	Zircaloy-2	Zircaloy-2
Active Fuel Length, in.	148.0	144.0
Fission Gas Plenum Length, in.	12.0	16.0
Fuel Pellets		
Material	UO_2 Sintered	UO_2 Sintered
Density at 95% Theroretical, gm/cc	10.32	10.42
Pellet Outside Diameter, in.	0.416	0.477
Pellet Length, in.	0.420	0.500
Water Rods		
Outside Diameter, in.	0.493	--
Inside Diameter, in.	0.425	--
Length, in.	160.78	--

In the earlier cores, Westinghouse utilized stainless steel for cladding. Thick stainless steel tubes are relatively inexpensive and provide high integrity. Later, consideration of fuel economy caused a switch to Zircaloy cladding for plants placed into operation in 1968 and 1969. In order to reduce the LHGR, the typical 14x14 and 15x15 fuel assemblies are being changed to a 17x17 rod array in the latest standarized design. General parameters for the 17x17 design are shown in Appendix B, Table B-4. While more detailed Westinghouse current fuel design parameters are summarized in Table 2.

The B&W design (Table B-3) has also changed from a stainless steel cladding to Zirzaloy-4. The active length of the core has been increased and the number of fuel rods per assembly increased to reduce the linear heat generation rate. In their proposed standarized design (Table B-4), which was later withdrawn, a 17x17 rod array is utilized; the fuel element design parameters are similar to the Westinghouse standarized design.

The Combustion Engineering design (Table B-3) has not changed significantly over the past few years. In their standarized design (Table B-4), a 16x16 fuel assembly has been employed to reduce the thermal duty (LHGR). The fuel element design parameters are also similar to the Westinghouse standarized design.

Both unpressurized and pressurized fuel rods have been

TABLE 2

WESTINGHOUSE FUEL DESIGN PARAMETERS

DESIGN PARAMETERS	164-INCH CORE[a]	RESAR 3 - AMEND 5 12 FOOT CORE[b]
Fuel Assemblies		
Design	RCC Canless	RCC Canless
Number of Fuel Assemblies	193	193
UO_2 Rods per Assembly	264	264
Rod Pitch, in.	0.496	0.496
Overall Dimensions, in.	8.426 x 8.426	8.426 x 8.426
Fuel Weight (as UO_2), pounds	253,675	222,739
Zircaloy Weight, lbs.	53,520	50,913
Number of Spacer Grids per Assembly	9-Type R	8-Type R
Loading Technique	3 region non-uniform (provision for up to 6 regions depending on fuel cycle)	3 region non-uniform
Fuel Rods		
Number	50,952	50,952
Outside Diameter, in.	0.374	0.374
Diametral Gap, in., Regions 1,2 and 3	0.0065	0.0065
Clad Thickness, in.	0.0225	0.0225
Clad Material	Zircaloy-4	Zircaloy-4
Fuel Pellets		
Material	UO_2 Sintered	UO_2 Sintered
Density (% of Theoretical)	95^2	95^2
Diameter, in., Regions 1, 2 and 3	0.3225	0.3225
Length, in.	0.530	0.530

(a) With densification effects
(b) Without densification effects

used in PWR designs. In order to reduce the inward creep of irradiated Zircaloy at operating conditions and to improve the fuel gap conductance, PWR fuel vendors began to pressurize fuel elements a few years ago and all current PWR fuel rods are of the pressurized type.

2.2 Fuel Design Considerations

The fuel designs of LWRs take into account possible fuel failures (safety considerations) and general fuel performance (general considerations).

The chief purpose of safety considerations is to contain the radioactive fission products within the fuel rod so that no significant amount of radioactivity will be released to the coolant. The safety considerations of the fuel design include:

(1) local overheating of the fuel cladding caused by boiling transition or departure from nucleate boiling (e.g., MCHFR $>$ 1.9 or MCPR $>$ 1.24 for BWRs, and MDNBR $>$ 1.3 for PWRs),

(2) local overstraining of the fuel cladding caused by pellet-to-clad interaction (e.g., less than 1% plastic strain of Zircaloy cladding),

(3) potential for chemical reactions, including hydriding and oxidation effects,

(4) densification effects,

(5) thermal cycling and fatigue (as aggravated by control rod movement),

(6) fretting corrosion,

(7) dimensional instability of the fuel and critical components during design lifetime (e.g., rod bowing), and

(8) radiation damage due to fission process.

In addition to the above safety considerations, the following general design criteria are also considered:

(1) A proper choice of fuel and cladding materials which

(a) are metallurgically compatible,

(b) will efficiently transfer heat from fuel to the coolant,

(c) will contain the radioactive fission products,

(d) will attain the desired fuel burnup and corrosion lifetime,

(e) will be capable of thorough evaluation and testing prior to use, and

(f) will yield optimum physics characteristics, compatible with the rest of the design.

(2) Ease of refueling to replace spent or damaged fuel elements.

(3) Reprocessing considerations.

(4) Economics of the design, fabrication, and operation of the fuel elements.

The reactor designer must necessarily assume that there will be some fuel failures, and factor this into the design of the shielding, containment, clean-up system, and radioactive treatment system. Considerable conservatism is built into these systems to assure that the requirements of 10 CFR 20, 50, and 100 are met.

2.3 Fuel Failures

As stated in the Introduction, this paper will concentrate primarily on fuel element failures rather than on successful performance even though the success rate has been quite high (99.8% or greater). This is done to highlight the important failure mechnisms which should help establish the operational limitations concerning fuel failures.

2.3.1 Definition of Fuel Failures

For the purpose of this paper, fuel failure is defined as:

(1) any breach (perforation or defect) of the fuel cladding,

(2) any structural change of the fuel assembly or its components, which requires abnormal maintenance or early replacement of the assembly or plant operating restrictions,

(3) any structural change of the fuel assembly or its components, which exceeds predicted limits of fuel performance.

2.3.2 Modes of Fuel Failures

The mechanisms of various fuel failures are very complex and not well established at present. A detailed discussion of fuel failure mechanisms is beyond the scope of the present paper. Instead, we shall discuss the various modes of fuel failures that have been observed to date.

From the past experience the following fuel failure modes have been identified:

(1) Internal Contamination (Hydriding)

(2) Pellet-Clad Interactions (PCI)

(3) Cladding Collapse (Densification)

(4) Fuel Rod Bowing

(5) Cladding Corrosion (Slow Chemical Reaction)

(6) Mechanical Damage

(7) Manufacturing Defects

(8) Waterlogging Rupture

These failure modes may not be entirely independent, but for the purpose of the present discussion, they are treated as separate phenomena. A summary of the first seven fuel failure modes is given in Appendix C, Table C-2 along with their associated applicable fuel failure experience compiled in Table C-1. These data are taken from Reference 1.

2.3.2.1 <u>Internal Contamination (Hydriding)</u>

Internal contamination is the introduction of a foreign material into the fuel rod, which may attack the cladding. For Zircaloy cladding, the predominant mechanism is hydriding that results from localized attack of the cladding by extraneous hydrogen in the fuel rod. The hydrogen impurities attack the inside surface of the cladding at various points to form blisters which may result in fuel rod perforation. Such failures generally occur relatively early in life (0 to 11,000 MWD/MT) of the fuel element.

Hydriding has accounted for a significant percentage of all the fuel failures reported to date, especially in BWRs. A significant number of hydriding failures also occurred early in the operation of the Beznau 1 and R.E. Ginna PWRs.

The following measures have been taken to mitigate the hydriding failure:

(1) New low-moisture specifications are established well below the threshold of hydriding failures.

(2) Strict procedural controls are utilized in manufacturing to prevent introduction of hydrogeneous impurities (e.g., oils, plastics, etc.) to the fuel element.

(3) Hot vacuum drying of each loaded fuel rod just prior to final end plug welding is employed to achieve a low level of moisture.

(4) Some kind of hydrogen getter material (a zirconium alloy in the form of small chips) is also added to the fuel rod to provide a further assurance.

2.3.2.2 Pellet-Clad Interactions (PCI)

Pellet-to-cladding interactions (PCI) had been identified as a fuel failure mechanism as early as 1971. Such failures have been predominantly associated with BWR cores. The PCI involves localized mechanical stress of the cladding adjacent to pellet cracks and ridges (interfaces). The fundamental driving force for PCI is the thermal expansion of UO_2 pellets due to heatup by the fission process. Fuel swelling due to accumulation of fission products also contributes to PCI. Cladding creep-down under external pressure is another phenomenon that can affect PCI.

Fuel pellets irradiated under normal operating conditions have been observed to undergo cracking. The development of pellet cracks is attributable to the radial thermal stresses in the fuel. Temperature differences as low as 60°C have been reported to be sufficent to cause cracking in UO_2 fuel.

The strain localization in the cladding due to PCI depends on local linear heat generation rate, initial pellet-to-cladding gap, burnup (or irradiation time), power (or thermal) cycling, design characteristics, and manufacturing variables. The greatest potential for PCI failure is later in life (6000 to 27,500 MWD/MT) when the clad ductility has been reduced.

As a result of the PCI failure mechanism, GE made changes to their basic fuel design for fuel fabricated in 1972 and later (e.g., modified 7x7 assembly design as used in Browns Ferry 1). Changes were specified for both the cladding and the pellet geometry. The cladding changes involved specifying a higher annealing temperature to achieve optimum uniformity of mechanical properties, and increasing cladding thickness to compensate for the lower yield strength. To minimize the potential for pellet ridging, a shorter, chamfered pellet with no dishing was specified. GE's latest design, consisting of 8x8 fuel assemblies, should further help reduce the PCI strain, due to the reduced linear heat generation rate (< 13.4 kw/ft.).

PWR fuel is not immune to the PCI failure either, but at a much lower percentage than BWR fuel. As an example, fuel defects due to PCI resulted during the Cycle 3 startup of Point Beach 1, in connection with a rapid rate of power increase after the refueling shutdown.

2.3.2.3 Cladding Collapse (Densification)

A phenomenon called "fuel densification" has been identified as the primary mechanism for cladding collapses that had occurred in some PWRs (e.g., R.E. Ginna and Beznau 1). Extensive studies of the densification problems concluded that cladding collapses resulted from the occurrence of axial gaps in the fuel-pellet column inside the fuel rods, and that the axial gaps resulted from the densification of the fuel. In those sections of fuel rods where axial gaps in the column had occurred, the inward creep of the cladding caused by external pressure was not arrested by the pellets and continued until essentially complete flattening had taken place.

It should be pointed out that the densification discussed above is different from the thermally induced densification of fuel that occurs during fuel irradiations at high temperatures. In the latter case, the structural changes are largely the result of the high temperature and steep thermal gradients that are present in uranium oxide fuel pellets during reactor operation. In the Ginna reactor core, the average linear heat generation rate in the hottest fuel rod is approximately 8.5 kw/ft, with a core-wide rod average of about 5 kw/ft. Thus, most of the fuel operated at temperatures at which little or no thermal restructuring occurred and at which thermally activated processes were insignificant. Experimental studies showed that the Ginna type densification had occurred by annihilation of pores. The following two mechanisms have been suggested for the annihilation of pores:

1. Re-solution-related mechanism. In this case, fission fragments passing close to the surface of a pore aid in trapping lattice vacancies in the surface. These vacancies migrate away from the pore and thus cause mass transfer of atoms to the pore, resulting in the disappearance of the pore.
2. Fission spike-related mechanism. In this case, interstitial atoms and lattice vacancies are generated by the fission process. The interstitial atoms, having a greater mobility at the temperatures of interest, quickly migrate to sinks, leaving an excess of vacancies in the lattice which increases the diffusion rates of the atoms in the fuel.

In general, fuel densification leads to increased stored energy in the fuel rod, increased linear heat generation rate, decreased heat transfer capability of the fuel rod, and increased potential for local power spikes.

It should be pointed out that cladding collapse was primarily associated with PWR plants, although gamma scanning of Oyster Creek 1 (BWR) fuel rods in 1973 also revealed significant axial gaps (not cladding collapses).

2.3.2.4 Fuel Rod Bowing

Bowing of fuel rods has been observed in Westinghouse designed PWR cores. There are two distinct types of rod bowing: interference bow and non-interference bow. The first type is typical of an earlier Westinghouse fuel assembly design which uses stainless steel guide thimbles, Zircaloy clad fuel rods, and a small fuel rod to adapter plate gap. Rod bow resulted from axial interference of the fuel rods and adapter plate, due to the differential thermal contraction of fuel rods and thimbles during reactor cooldown.

The second type of rod bow is non-interference related and its mechanism is not well understood. However, the frequency of significant bow of this type is small, is generally region-dependent, and is skewed in magnitude toward the bottom of the fuel assembly.

It should be noted that rod bow has rarely resulted in fuel rod perforation or breakage. One such case was an early Dresden 1 core. In this case, rod bowing led to accelerated corrosion which eventually caused some fuel rods to fail.

In general, the effects of rod bowing are factored into the design methods of the core. Westinghouse studies concluded that the DNBR reductions and power peaking increases due to rod bowing are adequately accounted for by the present design practice and that fretting and corrosion of bowed rods are negligible.

2.3.2.5 Cladding Corrosion

Cladding corrosion is the buildup of corrosion products on the clad surface and is highly dependent on the extraneous materials in the primary coolant. Some early commerical BWR's (Big Rock Point, Dresden 1, Humboldt Bay 3) had fuel failures due to crud buildup on the fuel rods. Crud buildup can cause overheating of the cladding at local power spikes, accompanied by accelerated oxidation and hydriding, which lead to eventual rod failures. Careful selection of materials in the feedwater system and improved primary coolant water chemistry control are effective measures for reducing failures of this type.

2.3.2.6 Mechanical Damage

Mechanical damage includes such things as mishandling of fuel at the plant, failure of fuel handling equipment, and fretting and wear of fuel elements. Fretting and wear have resulted from such causes as foreign materials loose in the coolant and vibration of fuel rods against grid springs. Channel box wear and breakage observed in some BWR's are also in this category, resulting from flow-induced vibration of temporary burnable curtains or instrument tubes. The use of flow loops for pretesting fuel designs plus lessons learned from experience have significantly reduced failures of this category.

2.3.2.7 Manufacturing Defects

Fuel failures may also result from such manufacturing defects as defective materials, Zircaloy tubing flaws, welding flaws (e.g., defective welding of the end plugs), inadequate inspection, enrichment mixups, and mechanical handling damage. However, the number of failures which can be traced to manufacturing defects has been quite small.

Failures of this type cannot be totally eliminated, but should remain very low in number provided that stringent quality assurance programs are maintained and continually updated as necessary.

2.3.2.8 Waterlogging Rupture

Fuel failure can also result from "waterlogging" in case of cladding defects (e.g., micropores or pinholes). Water entering the micropores or pinholes during low power operation is subsequently converted to steam under high power operation. The trapped steam due to its poor heat transfer capability, can cause local overheating of the cladding which can lead to gross distortion of the cladding (e.g., rupture).

Pinholes are eliminated during production by 100% leak check of assemblies. The leak detector system consists of a high vacuum system and a mass spectrometer. The fuel bundle is placed in the vacuum chamber and evacuated to very low (vacuum) pressure.

After the vacuum pressure is attained, the mass spectrometer tuned to the helium mass range is switched into the system. The output meter of the spectrometer will indicate the presence of any helium gas in the chamber. This production procedure is considered to preclude the potential for a waterlogging rupture throughout the fuel cycle.

2.4 Fuel Failure Experience

Appendix C, Table C-1 provides a compilation of fuel failure data for LWR's up to December 31, 1974. This table corresponds to the fuel failure summary categorization presented in Table C-2.

The three most important fuel failure modes having significant impact on plant operation are hydriding, pellet-clad interactions (PCI), and fuel densification. Consequently, they form the basis of discussions on the operational limitations concerning fuel failures to be presented in the following section.

3.0 OPERATIONAL LIMITATIONS

3.1 Operational Limitations Concerning Hydriding

Since hydriding is most likely to occur in a high temperature environment, the chief means of reducing the hydriding fuel failures from the utility standpoint is to reduce the reactor power level. Another means is by way of early refuelings (replacement by new fuel rods with hydriding fix).

Some examples of power reductions or early refuelings due to hydriding failures are given in Table 3. It should be mentioned that, for Quad Cities 1 and Dresden 3, pellet-clad interactions (PCI) also contributed a significant amount of failures. Further limitations may be necessary at times to limit the rate of power changes and to limit the stress on the fuel rods.

3.2 Operational Limitations Concerning PCI

The PCI fix has been approached from the standpoint of design changes to the fuel assembly, and the use of "fuel preconditioning". The former has been discussed in Subsection 2.3.2.2. The latter is in the form of procedural controls for a slow ascent to full power, which preconditions the fuel for subsequent normal full power generation. These procedures, for General Electric designed BWR's, are designated Pre-Conditioning Interim Operating Management Recommendations (PCIOMR).

Although the PCI mechanism is not yet fully understood, it has been observed that PCI is correlated to the rate of power change, fuel exposure (burnup), and the control rod movement. Rapid change of power level and frequent movement of control rods tend to contribute to the PCI failures. Consequently, in PCIOMR, GE imposes a limit on the rate of power change to be less than 0.06 kw/ft/hr.

Table 3

EXAMPLES OF PLANT RESTRICTIONS OR EARLY REFUELINGS CAUSED BY HYDRIDING

Plant	Reactor Type	Comments
Dresden 2	BWR	During Cycle II, (5/29/71 to 2/19/72), the unit was base loaded at approximately 60% power, to minimize off-gas activity release rate.
Dresden 3	BWR	As a result of fuel failures and off-gas limitations, the first refueling (originally scheduled for Fall of 1973) occurred in the Spring of 1973.
Monticello	BWR	During late 1973 and until refueling shutdown in the first quarter of 1974, power was restricted at times to reduce off-gas release rate.
Nine Mile Point 1	BWR	During first part of 1971, above normal off-gas activity caused the reactor power to be limited until the fuel was replaced.
Quad-Cities 1	BWR	The plant was administratively limited in power level at times starting in last half of 1973 to maintain stack rates at acceptable levels.
Vermont Yankee	BWR	During 1972, excessive gaseous release activity levels resulted in power reductions until the scheduled mid-January 1973 shutdown.
		The plant was administratively limited to lower power during 1974 until refueling shutdown in October due partly to excessive off-gas activity at the steam jet air ejector.

3.3 Operational Limitations Concerning Densification

Fuel densification can lead to fuel rod collapse, with a potential for cladding perforation and local power spikes. Since densification results from irradiation which is a function of power level and since the primary coolant system pressure tends to contribute to the cladding collapse, the densification fix from the operational point of view has been approached by a reduction of power level and/or system pressure. However, such reductions lower the plant efficiency.

Several PWR's operated at derated power levels for a part of 1973 due to densification problems. Table 4 lists the PWR plants and the restrictions imposed by the densification concern initiated in 1973.

While densification problems were most significant in PWR's, evidence of fuel densification was also noted in a BWR (Oyster Creek 1) during the Spring 1973 refueling. The USAEC staff's review indicated that changes in the operating conditions were necessary to assure that calculated peak cladding temperature following postulated loss of coolant accidents (LOCA) would not exceed 2300°F, taking into account fuel densification effects. On August 24, 1973, the Director of Regulation issued orders to the licensees of 10 BWR plants (Oyster Creek 1, Dresden 2 and 3, Quad Cities 1 and 2, Nine Mile Point 1, Millstone 1, Monticello, Pilgrim 1, and Vermont Yankee) modifying the licenses by ammending the technical

TABLE 4

PWR PLANT RESTRICTIONS DURING 1973

DUE TO DENSIFICATION

Plant	Restrictions
H.B. Robinson 2	Power level limited to 94.8% until July.
Point Beach 1*	Power level limited to 75% from March to May.
Point Beach 2	Approval of 2000 psia operation in December.
R.E. Ginna 1	Power level limited to 83.3% until July. Approval of 2000 psia operation in October.
San Onofre 1	Power level limited to 90% for part of July and August.
Surry 1 and 2	Power level limited to 92% at beginning of life. By year end, they were allowed to operate to 97.6% and 96.2%, respectively, with both units reaching 100% in the first half of 1974. Both units received approval in July, 1973 for 2000 psia operation.
Turkey Point 3 and 4	Power level limited to 93% at beginning of life. By year end, Unit 3 was allowed to operate up to 98%, reaching 100% in March 1974. Unit 4 reached 100% in March 1974. Unit 3 and Unit 4 operating pressures were reduced to 1885 psia in December 1973 and January 1974, respectively.

*Reactor coolant system reduced from 2235 psia to 2000 psia in May 1974 during beginning of Cycle 3.

specifications. These new limitations were met at several reactors without a reduction from rated power through modification of control rod pattern, and at some BWR's a decrease in cycle length. Several plants were limited in power by other considerations.

In November 1973, General Electric submitted the topical report NEDO-20181, in which the AEC approved a modified GE model for fuel densification. Limits on the maximum allowable planar LHGR (called MAPLHGR) and the peak LHGR were relaxed on December 28, 1973 for the affected BWRs. This allowed increased operational flexibility, including possible full power operation, for any BWR which had previously been limited to less than 100% rated power.

3.4 Operational Limitations Concerning Channel Box Wear

Channel box wear in BWRs can be caused by flow-induced vibrations of temporary control curtains and the incore instrument tubes. In August 1973, extensive wear on the corners of some fuel assembly channel boxes was observed during an inspection of fuel at the BWR KKM reactor (Swiss). The most significant channel box wear, including through-wall penetrations, corresponded to the location of temporary curtain stiffeners. The severe wear was caused by rubbing of the curtains against the channel boxes due to the impingement on the curtains of high velocity jets of water flowing through the bypass flow holes in the lower core plate. Channel box wear was also observed at Vermont Yankee in September 1973.

In 1975, excessive channel box wear was also discovered in several BWR-4's such as Duane Arnold, Browns Ferry 1, Vermont Yankee, and Fukushima 2 (Japan). Several cracked channel boxes were also observed at Fukushima 2. The wear and cracking of these channel boxes were caused by the vibration of instrument tubes due to turbulence created by the cooling water flowing through the bypass holes in the plate supporting the reactor core.

Since coolant flow turbulence is the primary cause of the vibrations, General Electric recommended reactor power and core flow reductions as a measure to mitigate the effects of channel box wear. The U. S. Nuclear Regulatory Commission (NRC) also ordered in 1975 a reduction of reactor power and core flow to those BWR plants plagued by the instrument tube vibrations. These measures are negative approaches, resulting in a significant loss of plant capacity.

A more positive approach for a channel wear fix has been the plugging of the bypass holes to prevent creation of flow turbulence. Tests have been performed by GE to develop a plug design for the bypass holes to eliminate the cause.

4.0 CONCLUSIONS

Despite the very low rate of fuel failures (about 0.2% or less) in commercial LWR's, plant restrictions due to fuel failures or actions taken to mitigate failures have resulted in undesirable lower plant availability and capacity factors. However, the fuel failures experienced to date have not had any adverse effect on the public health and safety.

Fuel failures have been caused by hydriding, pellet-clad interactions (PCI), densification, fuel rod bowing, accelerated corrosion, mechanical damage, and manufacturing defects. Of these, hydriding, PCI, densification, and channel box wear have had the most impact on plant performance, and have consequently led to various operational limitations such as power deration, flow reduction, pressure reduction, early refueling, reduced rate of power change, control rod manipulation, and other procedural controls (e.g., PCIOMR).

Past fuel failure experiences seem to suggest that BWR's were more prone to fuel failures than PWR's. Although the new 8x8 fuel designs of GE BWR/6, with the reduced thermal duty, have the potential for significantly reducing fuel failures, it is prudent to assume that BWR plants will probably continue to experience hydriding and PCI failures, at least until the older fuel has been replaced by the later fuel designs.

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APPENDIX A - LWR FUEL MATERIAL PROPERTIES

I. TABLE A-1

This table is a compilation of the physical and thermal properties of various fuel pellet materials employed in LWRs.

II. TABLE A-2

This table is a compilation of the physical and thermal properties of various fuel cladding materials used in LWRs.

III. Figure A-1

This figure presents the experimental and correlated data of the thermal conductivity of UO_2 compiled by Westinghouse.

IV. TABLE A-3

This table lists the important compositions of the Zircaloy cladding materials.

TABLE A-1

MATERIAL PROPERTIES OF LWR FUEL PELLETS

Material	Property	Value	Units	Remark	Ref.
UO_2	1. Theoretical Density	10.96	gm/cc		Exxon
	2. Melting Point	5080 - 0.00635E	^oF	E in MWD/STU	GE
		5080 - 0.00580E	^oF	E in MWD/MTU	W
		2840 ± 40	^oC		Exxon
	3. Thermal Expansion Coefficient (Linear)	$3.8\times10^{-6}+8.95\times10^{-10}T$	$^oF^{-1}$	T in ^oF	GE
		10.1×10^{-6}	$^oC^{-1}$	25-1000^oC	Exxon
	4. Thermal Conductivity (at 95%TD)	$\frac{3978.1}{(692.61+T)}+6.024\times10^{-12}(T+460)^3$	Btu/hr-ft-^oF	T in ^oF	GE
		$\frac{1}{(11.8+0.0238T)}+8.775\times10^{-13}T^3$	W/cm-^oC	T in ^oC	W
		$\frac{3825.0}{(129.41+T)}+6.080\times10^{-11}\ T^3$	W/M-^oK	T in ^oK	GE
	5. Specific Heat, C_p (Constant P)	$134.0+0.18T-3.1\times10^{-5}T^2$	J/kg-^oK	T in ^oK (1300$^oK\le T\le$2300^oK)	
		$0.0726+3.33\times10^{-6}T-1.542\times10^4(T+460)^{-2}$	Btu/lb_m-^oF	T in ^oF	
		$0.0711+5.994\times10^{-6}T-4.76\times10^3T^{-2}$	Btu/lb_m-^oF	T in ^oK (300$^oK\le T\le$1500^oK)	
	6. Young's Modulus	1930	kilobars		Exxon
	7. Shear Modulus	770	kilobars		Exxon
	8. Poisson's Ratio	0.302			Exxon
	9. Resistance to Thermal Shock	Good			
	10. Heat of Fusion	2.74×10^5	J/kg	@3150^oK	
	11. Heat of Vaporization	2.10×10^6	J/kg	@2470^oK	
	12. Viscosity (Liquid phase)	$7.5-8.0\times10^{-3}$	Pa-s	3070^oK-3270^oK	

TABLE A-1 (Continued)

MATERIAL PROPERTIES OF LWR FUEL PELLETS

Material	Property	Value	Units	Remark	Ref.
PuO_2	1. Theoretical Density	11.45	gm/cc		Exxon
	2. Melting Point	2400 ± 30	^{o}C		Exxon
	3. Thermal Expansion Coeff. (Linear)	10.9×10^{-6}	$^{o}C^{-1}$		Exxon
	4. Thermal Conductivity	0.023	$w/cm\text{-}^{o}C$	@1000^{o}C	Exxon
	5. Resistance to Thermal Shock	Faily Good			
UO_2-PuO_2 (20w/oPu)	1. Theoretical Density	11.04	gm/cc		Exxon
	2. Melting Point	2810 ± 25	^{o}C		Exxon
	3. Thermal Expansion Coefficient (Linear)	10.3×10^{-6}	$^{o}C^{-1}$		Exxon
	4. Thermal Conductivity (@ 95% TD)	0.021	$w/cm\text{-}^{o}C$	$T > 1600^{o}C$	Exxon
	5. Resistance to Thermal Shock	Good			
	6. Young's Modulus	1400 ± 100	kilobars		Exxon
	7. Shear Modulus	550 ± 50	kilobars		Exxon
	8. Poisson's Ratio	0.280 ~ 0.290			Exxon

TABLE A-2

MATERIAL PROPERTIES OF LWR FUEL CLADDING

Material	Property	Value	Units	Remark	Ref.
<u>Zircaloy-4</u>	1. Density	6.4875	gm/cc		
	2. Melting Temperature	3375	^oF		<u>W</u>
	3. Thermal Expansion Coeff. (Linear)	$1-3x10^{-6}$	$^oF^{-1}$		
	4. Thermal Conductivity	$4.14+1.044x10^{-2}T$ $+5.276x10^{-6}T^2$ $+1.536x10^{-9}T^3$	Btu/hr-ft-^oF	T in ^oF	ORNL-TM-4712 GEMP-482
		$0.5307+2.0446x10^{-2}T$ $+1.7707x10^{-5}T^{-2}$ $+1.5503x10^{-8}T^3$	W/m- ^oK	T in ^oK	ORNL-TM-4712
	5. Specific Heat, C_p (Constant P)	$0.068+1.33x10^{-5}T$	Btu/lbm-^oF	T in ^oF	ORNL-TM-4712
		143.95 + 0.05569 T	J/kg-^oK	T in ^oK	ORNL-TM-4712
		200.736	J/kg-^oK	$T > 1020^oK$	ORNL-TM-4712
		0.0863	Btu/lbm-^oF	$T > 1376^oF$	ORNL-TM-4712
<u>Zircaloy-2</u>	1. Density	6.550	gm/cc		GE
	2. Melting Temperature	3375	^oF		
	3. Thermal Expansion Coeff. (Linear)	3×10^{-6}	$^oF^{-1}$		GE
	4. Thermal Conductivity	9.0 - 10.0	Btu/hr-ft-^oF	$600^oF \leq T \leq 800^oF$	GE
		$4.14 + 1.044 \times 10^{-2}T$ $+ 5.276 \times 10^{-6}T^2$ $+ 1.536 \times 10^{-9}T^3$	Btu/hr-ft-^oF	T in ^oF	ORNL-TM-4712
	5. Specific Heat, C_p (Constant P)	$0.068 + 1.33 \times 10^{-5}T$	Btu/lbm-^oF	T in ^oF	ORNL-TM-4712

TABLE A-2 (Continued)

MATERIAL PROPERTIES OF LWR FUEL CLADDING

Material	Property	Vale	Units	Remark	Ref.
Stainless Steel	1. Density (100% TD)	8.0	gm/cc	304 or 316 SS	
	2. Melting Point	1700 ± 34	^oK	AISI - 304	
	3. Thermal Expansion Coeff. (Linear)	1.85×10^{-5}	$^oC^{-1}$	304 or 316 SS	BMI-1900 (May, 1968)
	4. Thermal Conductivity	$0.1383 + 1.568 \times 10^{-4}T - 1.446 \times 10^{-9}T^2$	W/cm-^oC	T in ^oC	LA-5868-MS (Jan. 1975)
		$10.54 + 13.54 \times 10^{-3}T$	W/m-^oK	T in ^oK (300-1100^oK)	
	5. Specific Heat, C_p	$41.9+2.43T-4.51\times10^{-3}T^2 +3.76\times10^{-6}T^3-1.13\times10^{-9}T^4$	J/kg-^oK	T in ^oK (300-1100^oK)	
		670	J/kg-^oK	T at 1450^oK	
Fission Gas	Thermal Conductivity	$7.321\times10^{-5}+1.258\times10^{-7}T -1.106\times10^{-11}T^2$	W/cm-^oC	T in ^oC	LA-5868-MS
Helium	Thermal Conductivity	$1.494\times10^{-3}+3.063\times10^{-6}T -1.888\times10^{-10}T^2$	W/cm-^oC	T in ^oC	LA-5868-MS

TABLE A-3

COMPOSITION OF ZIRCALOY CLADDING

Isotope	Zircaloy-4	Zircaloy-2
Zr	98.20%	Balance
Fe	0.21%	0.07 - 0.20 %
Cr	0.10%	0.05 - 0.15 %
Ni	0.004%	0.03 - 0.06 %
Sn	1.40%	1.20 - 1.70 %
Hf	100 ppm	
Al	20 ppm	
C	95 ppm	
Si	58 ppm	

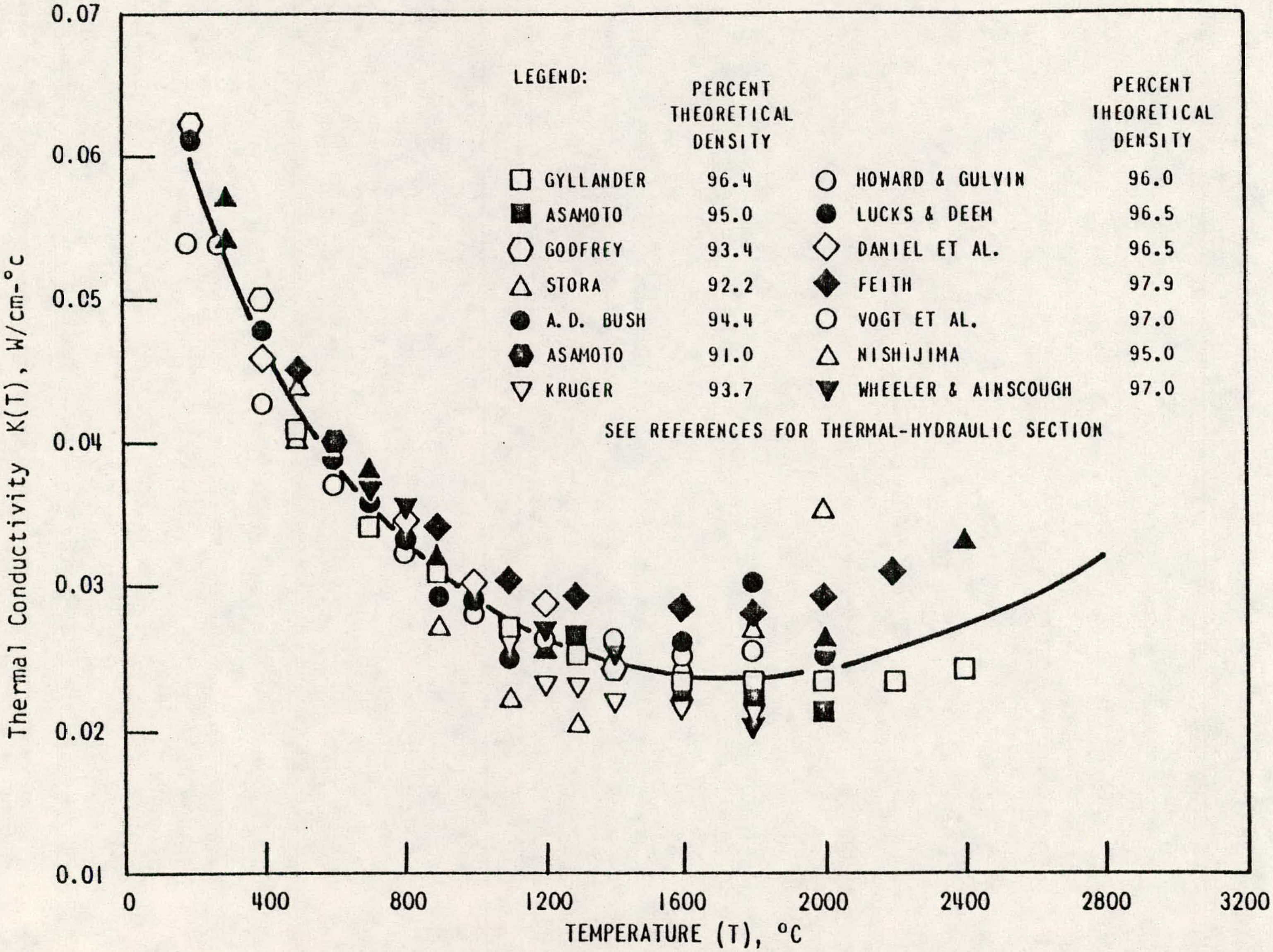

Figure A-1. Thermal Conductivity of UO_2 (Data Corrected to 95% Theoretical Density)

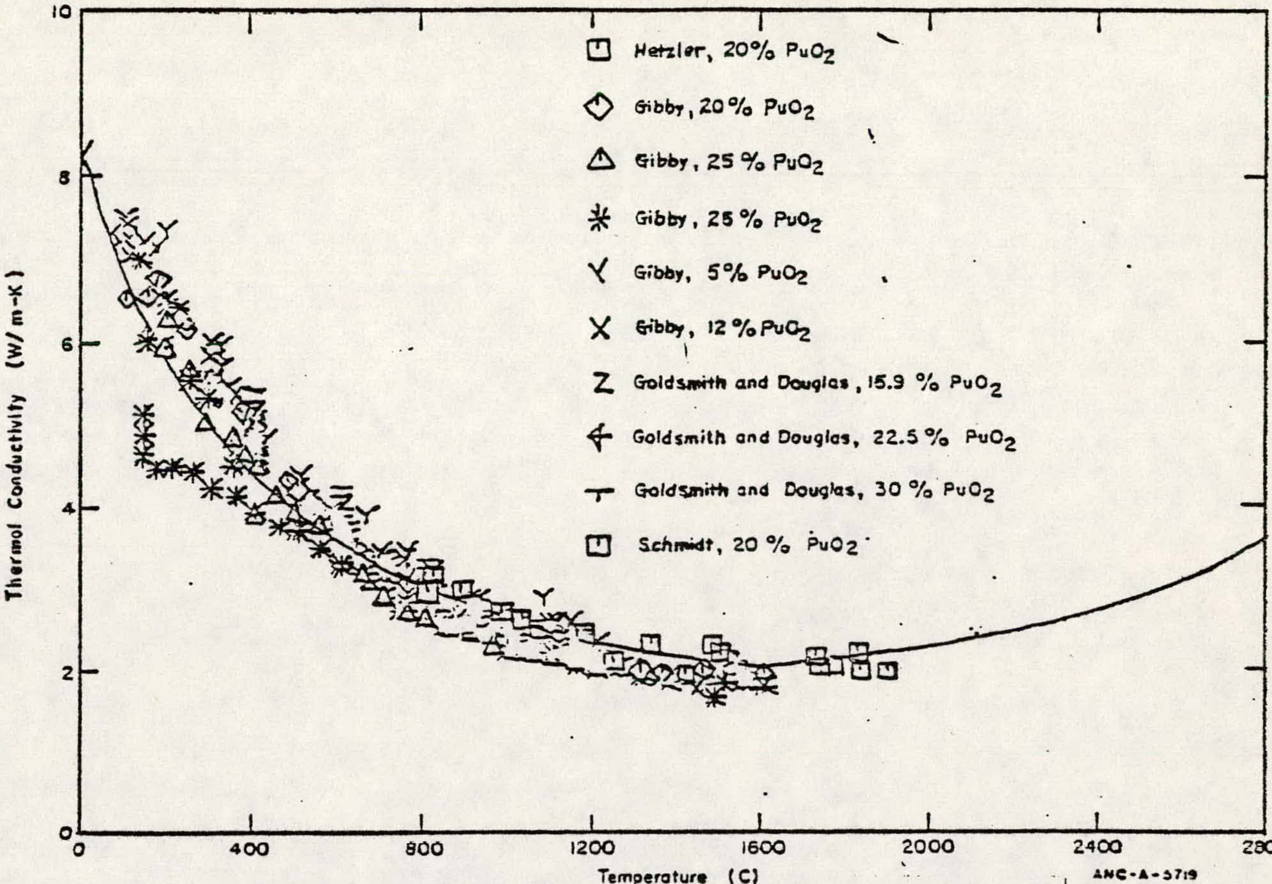

Fig. A-2. Comparison of measured mixed oxide conductivity with predictions (ANCR-1263) for 96% theoretical density MO_2.

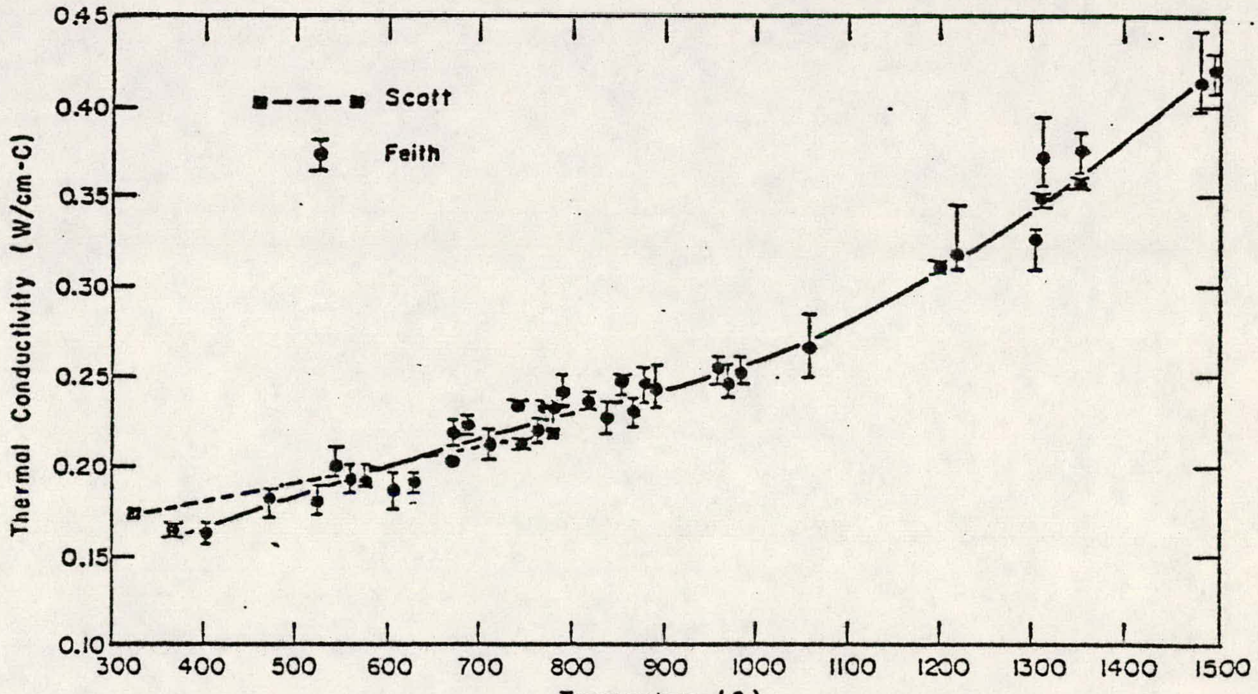

Fig. A-3. Thermal conductivity of zircaloy-4 as a function of temperature.

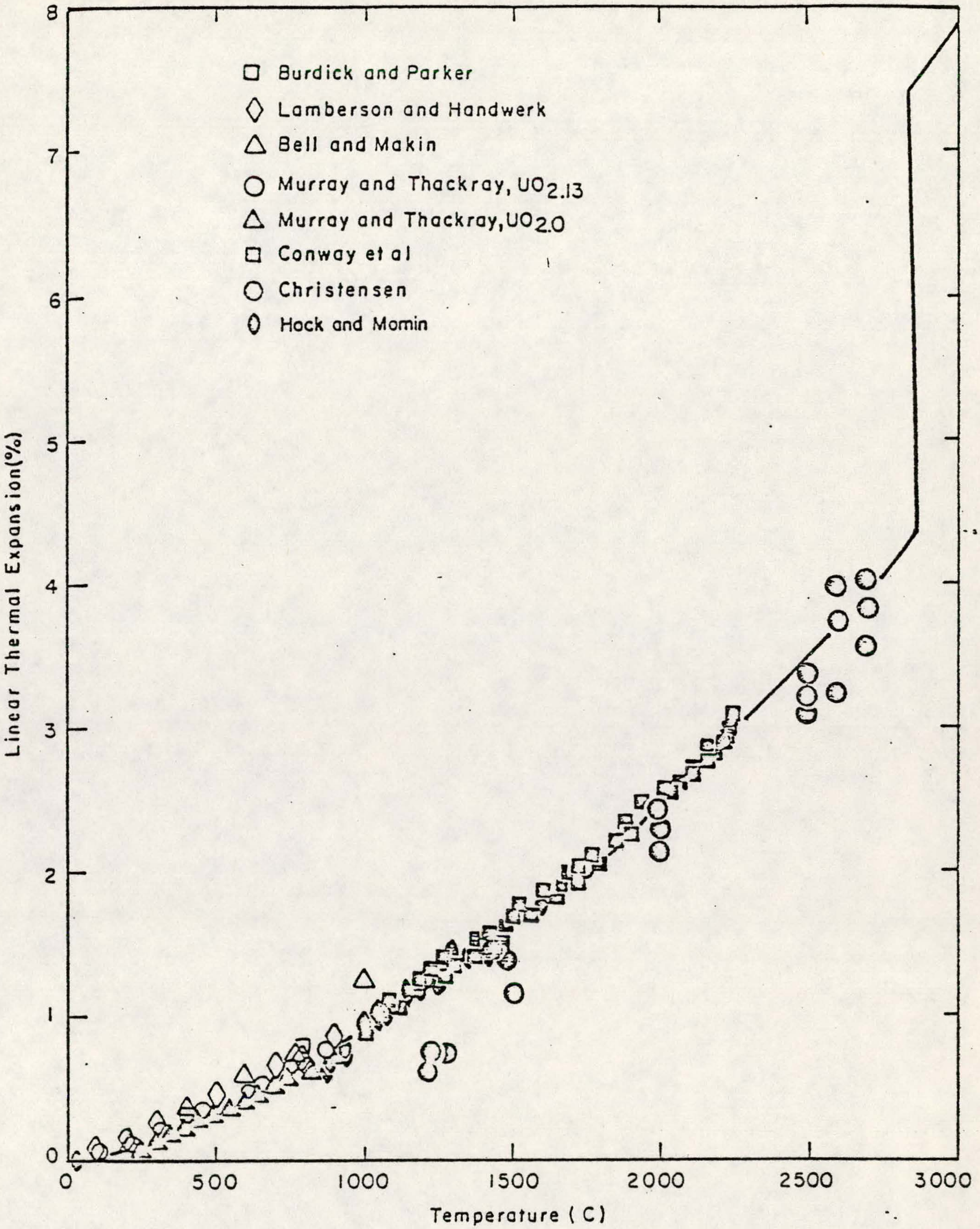

Fig. A-4. Comparison of UO_2 thermal expansion data with that calculated from FHTEXP subcode.

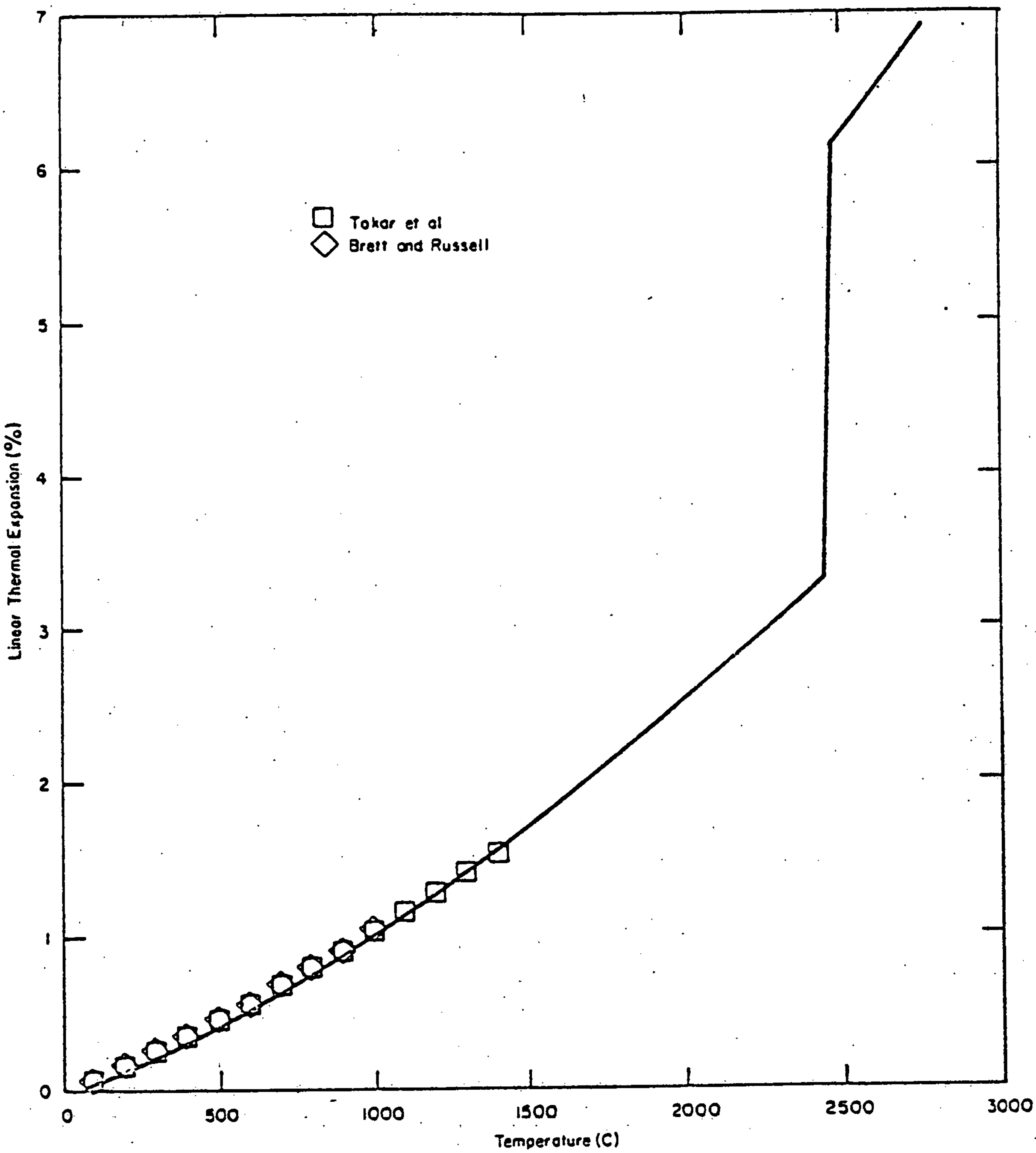

Fig. A- 5. Comparison of PuO_2 thermal expansion data with that calculated from FHTEXP subcode.

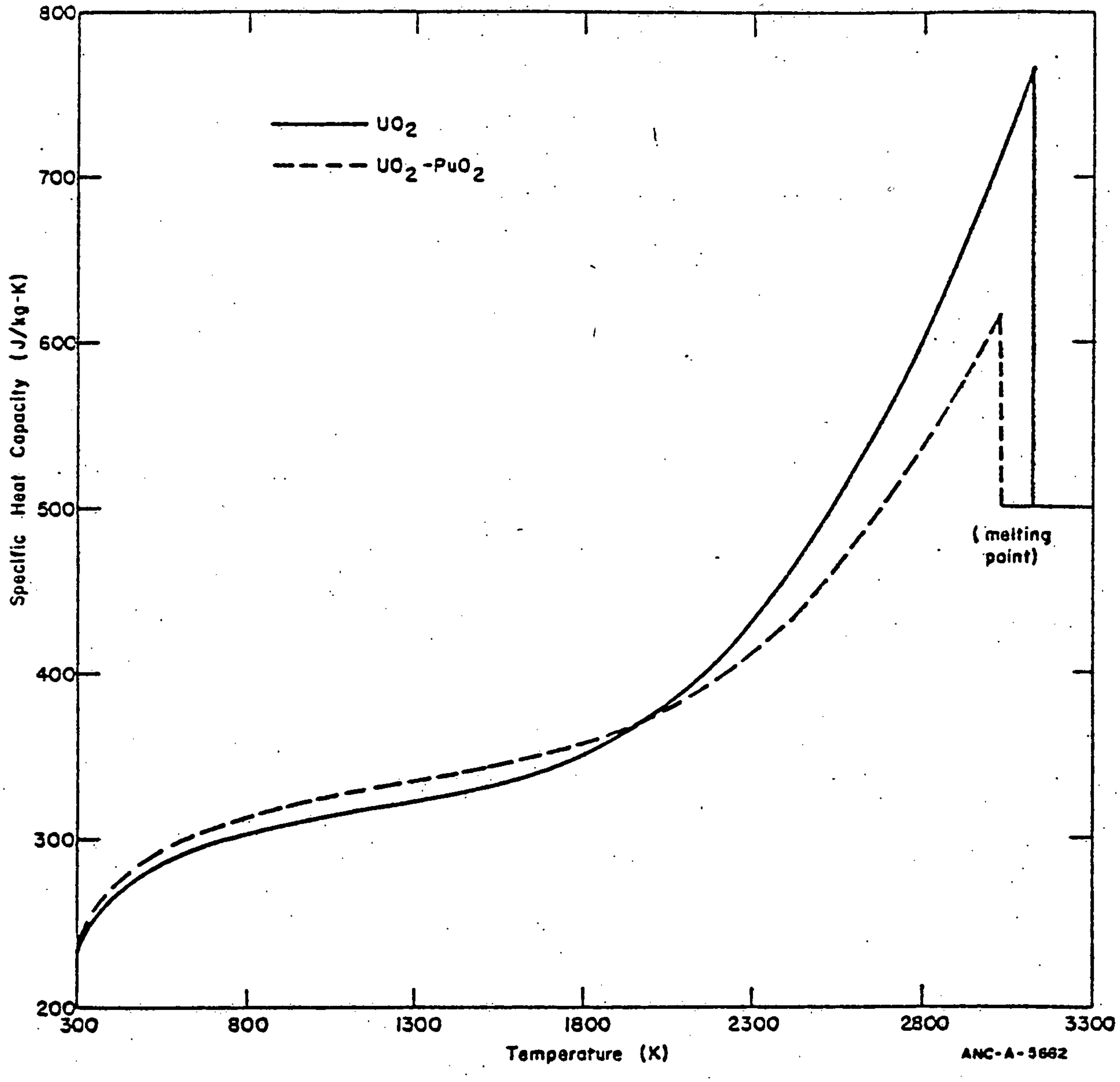

Fig. A- 6. Temperature dependence of mixed oxide specific heat capacity calculated by FCP for UO_2 and $UO_2 - PuO_2$.

APPENDIX B - DESIGN DATA

I. TABLE B-1

This table is a compilation of miscellaneous plant data of the 49 plants which have generated electricity as of December 31, 1974. The plants are listed in order of date of first electrical generation. The listing includes 21 BWR's and 28 PWR's.

II. Table B-2

This table lists some typical BWR fuel element design parameters for the original cores installed in various plants. The data shows the evolution of fuel element design over a number of years.

III. Table B-3

This table lists some typical PWR fuel element design parameters for the original cores installed in various plants. The data shows the evolution of fuel element design over a number of years.

IV. Table B-4

This Table lists some fuel element design parameters for the standarized designs submitted by the 4 LWR nuclear steam system suppliers. It should be noted that the design submitted by Babcock & Wilcox was subsequently withdrawn.

TABLE B-1

MISCELLANEOUS PLANT DATA

(LISTED IN ORDER OF DATE OF FIRST ELECTRICAL GENERATION)

Plant	Principal Owner (Utility)	Reactor Type	Nuclear Steam System Supplier[a]	Architect Engineer[b]	Docket Number	No. of Fuel Assemblies	No. of Control Assemblies	Date of First Electrical Generation	Thermal Power[c] (MWt)	Electrical Capacity[c] (MWe-Net)
Dresden 1	Commonwealth Edison Co.	BWR	GE	Bechtel	50-010	464(max)	80	4/15/60	700	200
Yankee (Rowe)	Yankee Atomic Electric Co.	PWR	W	S&W	50-029	76	24[d]	11/10/60	600	175
Indian Point 1	Consolidated Edison Co. of New York, Inc.	PWR	B&W	O/Vitro	50-003	120	21[e]	9/16/62	615	265
Big Rock Point	Consumers Power Co. of Michigan	BWR	GE	Bechtel	50-155	84(max)	32	12/8/62	240	75
Humboldt Bay 3	Pacific Gas & Electric Co.	BWR	GE	Bechtel	50-133	184(max)	32	4/18/63	240	65
San Onofre 1	Southern Calif. Edison & San Diego Gas & Electric Co.	PWR	W	Bechtel	50-206	157	45[f]	7/16/67	1347	430
Haddam Neck	Connecticut Yankee Power Co.	PWR	W	S&W	50-213	157	45[f]	8/7/67	1825	575
Genoa	Dairyland Power Cooperative	BWR	AC	S&L	50-409	72	29	4/26/68	165	50
Oyster Creek 1	Jersey Central Power & Light Co.	BWR	GE	B&R	50-219	560	137	9/23/69	1930	640

TABLE B-1 (Cont'd)

Plant	Principal Owner (Utility)	Reactor Type	Nuclear Steam System Supplier[a]	Architect Engineer[b]	Docket Number	No. of Fuel Assemblies	No. of Control Assemblies	Date of First Electrical Generation	Thermal Power[c] (MWt)	Electrical Capacity[c] (MWe-Net)
Nine Mile Point 1	Niagara Mohawk Power Corp.	BWR	GE	0	50-220	532	129	11/9/69	1850	625
R.E. Ginna 1	Rochester Gas & Electric Co.	PWR	W	Gil	50-244	121	33[g]	12/2/69	1520	490
Dresden 2	Commonwealth Edison Co.	BWR	GE	S&L	50-237	724	177	4/13/70	2527	809
H.B. Robinson 2	Carolina Power & Light Co.	PWR	W	Ebasco	50-261	157	53[g]	9/26/70	2200	700
Point Beach 1	Wisconsin Elec. Power Co. & Wisconsin-Michigan Power Co.	PWR	W	Bechtel	50-266	121	37	11/6/70	1518	497
Millstone 1	Northeast Nuclear Energy Co.	BWR	GE	Ebasco	50-245	580	145	11/29/70	2011	652
Monticello	Northern States Power Co.	BWR	GE	Bechtel	50-263	484	121	3/5/71	1670	545
Dresden 3	Commonwealth Edison Co.	BWR	GE	S&L	50-249	724	177	7/22/71	2527	809
Palisades	Consumers Power Co. of Michigan	PWR	Comb	Bechtel	50-255	204	45[g]	12/31/71	2200	700

TABLE B-1 (Cont'd)

Plant	Principal Owner (Utility)	Reactor Type	Nuclear Steam System Supplier[a]	Architect Engineer[b]	Docket Number	No. of Fuel Assemblies	No. of Control Assemblies	Date of First Electrical Generation	Thermal Power[c] (MWt)	Electrical Capacity[c] (MWe-Net)
Quad-Cities 1	Com. Edison-Iowa-Illinois Gas & Electric	BWR	GE	S&L	50-254	724	177	4/12/72	2511	800
Quad-Cities 2	Com. Edison-Iowa-Illinois Gas & Electric	BWR	GE	S&L	50-265	724	177	5/23/72	2511	800
Surry 1	Virginia Electric & Power Co.	PWR	W	S&W	50-280	157	53[g]	7/4/72	2441	788
Pilgrim 1	Boston Edison Co.	BWR	GE	Bechtel	50-293	580	145	7/19/72	1998	664
Point Beach 2	Wisconsin Elec. Power Co. & Wisconsin-Michigan Power Co.	PWR	W	Bechtel	50-301	121	37	8/2/72	1518	497
Vermont Yankee	Vermont Yankee Nuclear Power Corp.	BWR	GE	Ebasco	50-271	368	89	9/20/72	1593	514
Turkey Point 3	Florida Power & Light Co.	PWR	W	Bechtel	50-250	157	53[g]	11/2/72	2200	693
Maine Yankee	Maine Yankee Atomic Power Corp.	PWR	Comb.	S&W	50-309	217	85[g]	11/8/72	2440	790

TABLE B-1 (Cont'd)

Plant	Principal Owner (Utility)	Reactor Type	Nuclear Steam System Supplier[a]	Architect Engineer[b]	Docket Number	No. of Fuel Assemblies	No. of Control Assemblies	Date of First Electrical Generation	Thermal Power[c] (MWt)	Electrical Capacity[c] (MWe-Net)
Surry 2	Virginia Elec. & Power Co.	PWR	W	S&W	50-281	157	53[g]	3/10/73	2441	788
Oconee 1	Duke Power Company	PWR	B&W	O/Bechtel	50-269	177	69[h]	5/6/73	2568	886
Turkey Point 4	Florida Power & Light Co.	PWR	W	Bechtel	50-251	157	53[g]	6/21/73	2200	693
Fort Calhoun	Omaha Public Power District	PWR	Comb.	GHDR	50-285	133	49[g]	8/25/73	1420	457
Indian Point 2	Consolidated Edison Co. of New York, Inc.	PWR	W	UE&C	50-247	193	61[g]	6/26/73	2758	873
Zion 1	Commonwealth Edison Co.	PWR	W	S&L	50-295	193	61[g]	6/28/73	2760*	893*
Brown's Ferry 1	Tennessee Valley Authority	BWR	GE	O	50-259	764	185	10/15/73	3293	1065
Oconee 2	Duke Power Company	PWR	B&W	O/Bechtel	50-270	177	69[h]	12/5/73	2568	886
Prairie Island 1	Northern States Power Company	PWR	W	PS&E	50-282	121	33[g]	12/4/73	1650	530
Zion 2	Commonwealth Edison Co.	PWR	W	S&L	50-304	193	61[g]	12/26/73	2760*	893*

* Represents 85% capacity.

TABLE B-1 (Cont'd)

Plant	Principal Owner (Utility)	Reactor Type	Nuclear Steam System Supplier[a]	Architect Engineer[b]	Docket Number	No. of Fuel Assemblies	No. of Control Assemblies	Date of First Electrical Generation	Thermal Power[c] (MWt)	Electrical Capacity[c] (MWe-Net)
Peach Bottom 2	Philadelphia Electric Co.	BWR	GE	Bechtel	50-277	764	185	2/19/74	3293	1065
Kewaunee	Wisconsin Public Service Corp.	PWR	W	PS&E	50-305	121	33[g]	4/8/74	1650	541
Cooper Station	Nebraska Public Power District	BWR	GE	B&R	50-298	548	137	5/10/74	2381	778
Duane Arnold	Iowa Electric Light & Power	BWR	GE	Bechtel	50-331	368	89	5/19/74	1658	569
Three Mile Island 1	Metropolitan Edison Company	PWR	B&W	Gil	50-289	177	69[h]	6/19/74	2535	819
Arkansas 1	Arkansas Power & Light Company	PWR	B&W	Bechtel	50-313	177	69[h]	8/17/74	2568	850
Brown's Ferry 2	Tennessee Valley Authority	BWR	GE	O	50-260	764	185	8/28/74	3293	1065
Peach Bottom 3	Philadelphia Electric Co.	BWR	GE	Bechtel	50-278	764	185	9/1/74	3293	1065
Oconee 3	Duke Power Company	PWR	B&W	O/Bechtel	50-287	177	69[h]	9/18/74	2568	886
Rancho Seco	Sacramento Municipal Utility District	PWR	B&W	Bechtel	50-312	177	69[h]	10/13/74	2772	913

TABLE B-1 (Cont'd)

Plant	Principal Owner (Utility)	Reactor Type	Nuclear Steam System Supplier[a]	Architect Engineer[b]	Docket Number	No. of Fuel Assemblies	No. of Control Assemblies	Date of First Electrical Generation	Thermal Power[c] (MWt)	Electrical Capacity[c] (MWe-Net)
Edwin I. Hatch 1	Georgia Power Company	BWR	GE	SSI	50-321	560	137	11/11/74	2436	786
Prairie Island 2	Northern States Power Company	PWR	W	PS&E	50-306	121	33[g]	12/21/74	1650	530
Calvert Cliffs 1	Baltimore Gas & Electric Company	PWR	Comb	Bechtel	50-317	217	85[g]	12/30/74	2560	845

Notes to TABLE B-1:

(a) Nuclear Steam Systems Suppliers:

GE = General Electric Company; W = Westinghouse; B&W = Babcock & Wilcox; Comb = Combustion Engineering; AC = Allis Chalmers.

(b) Architect Engineers:

S&W - Stone & Webster; O = Owner; S&L - Sargent & Lundy; B&R = Burns & Roe; Gil = Gilbert Associates; GHDR = Gibbs & Hill and Durham & Richardson;

SSI = Southern Services, Inc.; PS&E = Pioneer Services & Engineering Company; UE&C = United Engineers & Constructors.

(c) Authorized power levels.

(d) The core also includes 8 fixed Zircaloy cruciform shim rods.

(e) The core also includes 16 fixed Zr-2 filler rods.

(f) Includes safety and regulatory rods.

(g) Includes full and part length rods.

(h) Includes 8 axial power shaping rods.

TABLE B-2

TYPICAL BWR FUEL ROD DESIGN PARAMETERS (ORIGINAL CORES)

Plant	Nuclear Steam System Supplier	Date of First Electrical Generation	Fuel Pellet Diameter (Inches)	Fuel Rod Diameter, OD (Inches)	Clad Thickness (mils)	Clad Material	Active Fuel Length (Inches)	Fuel Assembly Description
Humboldt Bay 3	General Electric	4/18/63	0.420	0.463	19	304 SS	79	7 x 7 fuel rod array. 49 fuel rods per fuel assembly. 172 assemblies in the core.
Dresden 2	General Electric	4/13/70	0.488	0.563	32	Zr-2	144	7 x 7 fuel rod array. 49 fuel rods per fuel assembly. 724 assemblies in the core.
Millstone 1	General Electric	11/29/70	0.488	0.570	35.5	Zr-2	144	7 x 7 fuel rod array. 49 fuel rods per fuel assembly. 580 assemblies in the core.
Browns Ferry 1	General Electric	10/15/73	0.488	0.562	32	Zr-2	144	7 x 7 fuel rod array. 49 fuel rods per fuel assembly. 764 assemblies in the core. Also contains gadolinia bearing rods.
Genoa (Also called LaCrosse or LACBWR)	Allis-Chalmers	4/26/68	0.350	0.396	20	348H SS	83	10 x 10 fuel rod array. 100 fuel rods per fuel assembly. 72 assemblies in the core.

TABLE B-3

TYPICAL PWR FUEL ROD DESIGN PARAMETERS (ORIGINAL CORES)

Plant	Nuclear Steam System Supplier	Date of First Electrical Generation	Fuel Pellet Diameter (Inches)	Fuel Rod Diameter, OD (Inches)	Clad Thickness (mils)	Clad Material	Active Fuel Length (Inches)	Fuel Assembly Description
Haddam Neck	Westinghouse	8/7/67	0.3835	0.422	16.5	304 SS	121.8	15 x 15 fuel rod array. 204 fuel rods per fuel assembly. 157 assemblies per core, including 4 with Zr-4 clad.
			0.3669	0.422	24.3	Zr-4	120.0	
Surry 1	Westinghouse	7/4/72	0.3659 0.3649	0.422	24.3	Zr-4	144	15 x 15 fuel rod array. 204 fuel rods per fuel assembly. 157 assemblies per core.
Prairie Island 1	Westinghouse	12/4/73	0.3659	0.422	24.3	Zr-4	144	14 x 14 fuel rod array. 179 fuel rods per fuel assembly. 121 assemblies per core.
Indian Point 1 (original core,* supplied by Babcock & Wilcox)	Babcock & Wilcox	9/16/62	0.260	0.304	20.5	304 SS (boron modified)	98.5	14 x 14 fuel rod array. 195 fuel rods per fuel assembly. 120 assemblies per core.

*This core contained UO_2-ThO_2 fuel material.

TABLE B-3 (Cont'd)

Plant	Nuclear Steam System Supplier	Date of First Electrical Generation	Fuel Pellet Diameter (Inches)	Fuel Rod Diameter, OD (Inches)	Clad Thickness (mils)	Clad Material	Active Fuel Length (Inches)	Fuel Assembly Description
			0.313	0.3415	12.0			14 x 14 fuel rod array.
Indian Point 1			0.280	0.3415	28.5	304 SS	101.5 (avg.)	173 fuel rods per fuel assembly.
(Core B- supplied			0.297	0.3415	20.0			120 assemblies per core.
by Westinghouse)								
Oconee 1	Babcock & Wilcox	5/6/73	0.370	0.430	26.5	Zr-4	144	15 x 15 fuel rod array. 208 fuel rods per fuel assembly. 177 assemblies per core.
Palisades	Combustion Engineering	12/31/71	0.359	0.4135	24	Zr-4	132	15 x 15 fuel rod array. 212 or 208 fuel rods per fuel assembly. 204 assemblies per core.
Calvert Cliffs 1	Combustion Engineering	12/30/74	0.3795	0.440	26	Zr-4	136.7	14 x 14 fuel rod array. 176 or 164 fuel rods per fuel assembly. 217 assemblies per core.

TABLE B-4

STANDARIZED FUEL ROD DESIGN PARAMETERS

Reference	Reactor Type	Docket Number	Fuel Pellet Diameter (inches)	Fuel Rod Diameter OD (inches)	Clad Thickness (mils)	Clad Material	Active Fuel Length (inches)	Fuel Assembly Description
General Electric GESSAR	BWR	50-447	0.416	0.493	34	Zr-2	148	8 x 8 fuel rod array. 63 fuel rods and 1 water rod per fuel assembly. 732 fuel assemblies per core. (177 control rods.)
Westinghouse RESAR-41	PWR	50-480	0.3225	0.374	22.5	Zr-4	164	17 x 17 fuel rod array. 264 fuel rods per fuel assembly. 193 fuel assemblies per core. (61 full length and 8 part length rod cluster control assemblies).
Babcock & Wilcox B-SAR-241 (this application (was withdrawn by B&W on December 3, 1974)	PWR	50-481	0.324	0.379	23.5	Zr-4	143	17 x 17 fuel rod array. 264 fuel rods per fuel assembly. 241 fuel assemblies per core. (76 full length control rod assemblies plus 8 axial power shaping rod assemblies).
Combustion Engineering CESSAR	PWR	50-470	0.325	0.382	25	Zr-4	150	16 x 16 fuel rod array. Average of 236 fuel rods per fuel assembly. 241 fuel assemblies per core. (81 full length plus 8 part length control element assemblies).

APPENDIX C - FAILURE EXPERIENCE

I. TABLE C-1

This table summarizes the failure data for those commercial light water reactors which have generated electricity as of December 31, 1974. The approximate failure frequency together with the failure type, where known, are described.

II. TABLE C-2

This table summarizes the categories of failure together with their associated applicable items from Table C-1 above.

TABLE C-1

FUEL FAILURE DATA FOR COMMERCIAL LIGHT WATER REACTORS

(PLANTS ARRANGED IN ALPHABETICAL ORDER)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Date
1a	Arkansas 1	PWR	Inspection of the 40 fuel assemblies containing orifice rod assemblies (ORA's) revealed that the ORA's were binding in the guide tubes of 7 of the fuel assemblies. This event occurred on 12/28/73, prior to initial fuel loading.	Fabrication misaligment combined with the small clearance between the orifice rod and the guide tube nut I. D.	12	2/74
2a	Big Rock Point	BWR	Prior to April 1969, there were no known failures and there were two suspected leakers in Type B fuel assemblies. (See item 2b below).	Failure (?) type unknown	13	Spring 1971
2b	Big Rock Point	BWR	During April 1969 refueling outage, dry sipping revealed 7 leaky fuel assemblies (4 type B and 3 Type E). Inspection of 2 Type B and 2 Type E leaker fuel assemblies revealed 5 and 9 failed fuel rods, respectively. (Also noted were 8 failed fuel rods in centermelt development fuel assembly D-50.)	The observed fuel rod failures were of the same character in all fuel types inspected and were limited to $\leq$20 in. of active fuel length in any given rod. The fuel rod failures resulted from heavy buildup of crud scale that caused the cladding surfaces to overheat to abnormally high temperatures (i.e., accelerated corrosion due to crud).	13	Spring 1971
2c	Big Rock Point	BWR	Following Cycle 7, 100% of the core was sipped and 19 leaky fuel assemblies identified (5 Type B, 11 Type E, and and 3 Type EG).	Examination of the Type B and E leaker assemblies indicated failures are predominantly crud-related (i.e., accelerated corrosion due to crud). The Type EG fuel failures gave indication of early-life hydriding.	6	5/72
2d	Big Rock Point	BWR	Several fuel assemblies failed. (See item 2e below).	Premature failure of several E fuel assemblies.	14	8/71

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
2e	Big Rock Point	BWR	Following Cycle 8, 100% of the core was sipped and 17 leaky fuel assemblies identified (5 Type B, 11 Type E, and 1 Type EG).	Examination indicated that the Type B and E leaker assemblies are predominantly of the crud-related failures previously described (i.e., accelerated corrosion due to crud). The Type EG failures appeared to be divided roughly between crud-related and early-life hydride failures.	6	5/72
2f	Big Rock Point	BWR	Thirty one of 84 fuel assemblies were found to have failed.	The failed assemblies consisted of 4 types of experimental bundles.	15	8/72
2g	Big Rock Point	BWR	Cobalt target rods in 4 fuel assemblies became unlocked.	Fuel inspection determined that several of the cobalt target rods had become unlocked in four fuel assemblies. The loose cobalt rods were removed and the fuel assemblies recharged into outer rows in the core. Analysis shows power peaking will not occur; also change in flow distribution will not have a large effect. Unlocking resulted from insufficient force in the spring that locked the rods in position. Modification made that increases force required to unlock target rods (i.e., installed auxiliary spring which has locking force of 18 lb).	16 17	March and April 1973, resp.
2h	Big Rock Point	BWR	One failed fuel rod.	Tie rod from "E" type fuel bundle unexpectedly found on spent fuel pool floor.	18	11/73
2i	Big Rock Point	BWR	Twenty three of 84 fuel assemblies, examined by sipping early in 1973, contained failed fuel rods.	Probable cause was accelerated corrosion. Evidence of internal hydriding was not observed.	19	7/73
2j	Big Rock Point	BWR	Dry sipping of all 84 assemblies during 11th refueling (starting 3/23/74) showed 9 assemblies were leakers.	Most probable cause was accelerated cladding corrosion induced by crud spalling and the resulting localized heating. Crud buildup on one-cycle assemblies was minimal.	20 21	8/74 and 8/74, resp.

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
2k	Big Rock Point	BWR	Off gas rates continued at high levels. Power derated to 63 MWe in May 1974. After encountering other plant problems in June, decision was made to refuel once again. Dry sipping of 71 assemblies showed 15 leakers.	Unreported. However, it is likely to be similar to item 2j above.	20	8/74
2ℓ	Big Rock Point	BWR	During nondestructive fuel examination of fuel which had undergone irradiation, but not yet received its final goal exposure, an anomalous peak in gamma activity was discovered.	Probable cause is a pellet of higher density which had been mislocated in a rod; most likely the pellet was overlooked during cleanout of equipment during enrichment runs.	22	11/74
3a	Dresden 1	BWR	Of 77,184 fuel segments, 22 failed (<0.1%) Ten of these failed during fourth operating cycle (May 1965-February 1967).	Five fuel segments failed because of accelerated corrosion due to bowing and 5 failed because of internal corrosion due to end plug stringers. Twelve others (which were operated well beyond original design burnups) failed due to inadequate space for expansion and fission gas release.	13	Spring 1971
3b	Dresden 1	BWR	Of 400 fuel assemblies, 5 failed. The 5 leakers in 400 assemblies could result from imperfections in 5 out of 13,000 fuel rods (<0.1% defects.)	Underwater inspection of 4 of these 5 assemblies revealed no fuel rod failures. The fifth assembly has one fuel rod with a cracked bottom end-plug weld.	13	1971
3c	Dresden 1	BWR	Twenty nine fuel assemblies failed (3 type III B, 19 Type III F, 7 Type V).	Of the fuel rod failures, approximately half due to brittle longitudinal cladding cracks caused by strain localization and half due to internal hydriding.	13,6 23	Spring 1971, May 1972 and April 1972 resp.

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
3d	Dresden 1	BWR	Sipping results at end of Cycle 6 (Sept. 1969) indicated 29 leaking fuel assemblies and at end of Cycle 7 (Sept. 1971) another 20 leaking fuel assemblies. In the 49 assemblies, 58 failed fuel rods noted (~0.4% of the 14,472 fuel rods of Type III B, III F, and V reload fuel).	The 58 failed fuel rods had brittle longitudinal cracks characteristic of pellet-to cladding interaction mechanism (longitudinal crack-strain localization failures).	6, 23	May 197 and Apri 1972, resp.
3e	Dresden 1	BWR	During Fall 1973 refueling, 46 of 464 assemblies were identified as leakers, by combination of in-core and out-of-core wet sipping.	Most probable cause was pellet-clad interaction.	24	12/74
4a	Dresden 2	BWR	Significant offgas release observed as early as first week of May 1970 during operation and testing at 50% of rated power. A total of 131 fuel assemblies were sipped out of the core and 27 assemblies identified as failed on basis of sip signals. Two other fuel assemblies remained out of core on basis of visual inspection results. (see item 4d below)	Four fuel assemblies dissassembled and fuel rods examined. Defects observed were minor and were primarily small blisters on individual rods. The blisters indicate highly localized chemical reaction in the cladding; the localized points of reaction are brittle. Some failed fuel was located in areas of the core considerably removed from the high probability suspect areas defined by flux tilting. Cause of the fuel failures has not been determined at this time, but it is most likely due to an abnormal condition introduced during fuel manufacturing (internal hydriding). The 29 defective fuel assemblies were replaced with identical assemblies that had been fabricated for Dresden 3.	25	7/70
4b	Dresden 2	BWR	Off-gas began increasing in May 1970. In March 1971, 215 fuel assemblies removed. (See item 4e below)	Investigative work in June 1970 indicated that leaky fuel rods were caused by zirconium hydriding from inside of the fuel rod due to an unidentified hydrogeneous material from an unidentified source. The 215 fuel assemblies were removed based on their confirmed leakage, other suspicious data, or statistical evaluation performed to determine potential leakers.	26	6/71

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Date
4c	Dresden 2	BWR	Of 724 fuel assemblies, 69 identified as leaker assemblies and the prospect exists of considerable additional failures or incipient failures still remaining in core. (See items 4d and 4e below).	Early-life failures caused by internal hydriding caused by an initial hydrogen impurity inadvertantly introduced during fabrication. The specific impurity or exact means of introduction could not be determined. Initial fuel load was not vacuum outgassed.	6, 23	May and April 1972, resp.
4d	Dresden 2 (Cycle 1)	BWR	Twenty eight leaker assemblies; of the 28 assemblies, 19 each had at least 1 perforated fuel rod, 4 each had at least 1 defective fuel rod, and 5 each had at least 1 questionable fuel rod. It was estimated early in 1971 that 60-70 fuel assemblies were causing the off-gas problem.	Internally initiated, localized hydriding of cladding caused by some unspecified hydrogenous impurity or impurities (trace amounts of alcohol and other hydrocarbons found) most likely introduced in the manufacture. Also 1 assembly was damaged during the outage and was replaced.	27	11/73
4e	Dresden 2 (Cycle 1A)	BWR	Forty one of 668 sipped fuel assemblies identified as leakers. Thirty five of the 37 Dresden 2 type and 4 of 4 Dresden 3 type fuel assemblies each had at least 1 perforated fuel rod.	Fuel rod failures caused by hydrogenous impurities during manufacture. In addition to the 41 leakers, 174 other assemblies also replaced in attempt to minimize further hydriding failures.	27	11/73
4f	Dresden 2 (Cycle II)	BWR	Of 239 fuel assemblies sipped, 2 of the 215 CY (reload) type and 1 of 7 DN (original core) type fuel assemblies were identified as leakers. Questionable fuel rods (?no.) and 5 defective fuel rods were replaced with other sound discharged fuel rods.	Two of 5 defective rods revealed blisters of type attributed to internal hydriding; other 3 showed nothing unusual.	27	11/73
4g	Dresden 2 (Cycle III)	BWR	Off-gas activity during 1974 indicated several rod failures. (See item 4h below).		27 24	11/73 and 12/74 resp.
4h	Dresden 2	BWR	During Fall 1974 refueling (end of Cycle III), 615 assemblies were wet sipped out-of-core. Thirty eight defective assemblies were detected.	Results are preliminary at this time. Data is still being evaluated.	28	1/75

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
5a	Dresden 3	BWR	As a result of quality control audit, it was found that a small number of fuel rods (5 in 10,000) contained pellets of 2.4% enrichment instead of 1.4% enrichment.	Deficiencies in quality control program related to fuel fabrication.	29	4/71
5b	Dresden 3	BWR	As a result of fuel failures and off-gas limitations, first refueling moved from Fall of 1973 to Spring 1973. One hundred three of 724 assemblies were identified as leakers or suspect leakers.	Most probable causes are hydriding and pellet-clad interactions. Fifty two new assemblies and 51 reconstituted assemblies were installed.	30	4/74
5c	Dresden 3	BWR	Second refueling began 3/11/74. In-core and out-of-core sipping showed 27 definite leaker assemblies plus 6 probable defective assemblies.	Unreported; however, most probable causes are those of items 5b above. The 33 assemblies plus 11 high exposures assemblies (all of the 7x7 design) were replaced by new assemblies of the 8x8 design.	31	8/74
5d	Dresden 3	BWR	On October 31, 1974, a sudden increase in off-gas radiation occurred, indicating that several fuel rods has ruptured.	Most probable cause of failure is pellet-clad interaction, due to allowing rapid local power changes to occur. The plant has been limited to lower power levels since 10/31/74 to reduce the off-gas rates.	32	1/75
6a	Duane Arnold	BWR	Prior to fuel loading, during fuel bundle inspection, it was found that one Type I bundle had a lower tie plate without the required orifice. In addition, a Type II bundle was discovered with a lower tie plate containing a Type I orifice.	Poor inspection at the fabrication facility.	33	12/73

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
7a	Fort Calhoun 1	PWR	On a fuel rod basis, the failure rate is <0.01%.	Unreported	34	6/74
8a	Genoa	BWR		Fuel rods of 2 assemblies found to be bowed in bottom quarter. Bowing of one may be enough to effect future behavior. Cause of bowing not known-unless locked-in tube-drawing stresses were released by tubes standing in 540°F water.	35	6/69
8b	Genoa	BWR		Bowed fuel rods in 13 fuel assemblies. Bowed rods on side of assembly adjacent to fully withdrawn control rods.	36	11/69
8c	Genoa	BWR		Several fuel assemblies removed from reactor had fuel rods that were significantly bowed.	37	4/70
8d	Genoa	BWR		Bowing of fuel pins first observed in May 1969. It was determined that shroud locking rings had been unlocked during previous operation. This condition caused the fuel assemblies to be improperly seated and produced twisting and stressing of the fuel assemblies.	38	1/71
8e	Genoa	BWR	Fission product leakage occurred in several fuel assemblies. One fuel rod was severed.	The fission product leakage resulted in a stack release of I-131 in excess of technical specification limits. Inspection of one fuel assembly (No. 64) revealed a severed fuel rod.	39	8/72
8f	Genoa	BWR	Five fuel assemblies may have cladding failures. (See item 8g below)	When failed-fuel-element-detection system placed in service, results indicated that cladding failures may have occurred on 5 fuel assemblies.	40	12/72

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
8g	Genoa	BWR	Reactor refueled twice in 1973. Visual and dry sipping examinations were used to detect defective assemblies. Twenty and 23 defective assemblies were removed in April and November 1973, respectively.	Clad defects believed to be caused by pellet-clad interactions. In the April examinations, 2 of the defective assemblies had severely bowed rods. In November, 1 of the failed assemblies had 2 badly bowed rods.	24, 41, 42	12/74, 4/73, ar 11/73, resp.
				There is some preliminary evidence that accelerated corrosion may have contributed to some failed fuel rods.	28	1/75
9a	Haddam Neck	PWR	Two fuel assemblies.	Fuel assembly difficult to latch; examination showed radial vane of spider assembly, which holds absorber rods, broken from spider. A second fuel assembly was also found to have another severed vane.	43	5/70
9b	Haddam Neck	PWR	Coolant activity indicates existence of a few leaking fuel rods since the first reactor cycle. (See item 9c below)	Fuel failure type unknown.	44	4/72
9c	Haddam Neck	PWR	During the Fall 1973 refueling, representative inspections of fuel assemblies revealed no abnormalities. Concentration of radioactive fission products in coolant was indicative of a few minor defects in a few fuel rods.	Unknown.	24	12/74
10a	H. B. Robinson 2	PWR	Rod-control cluster failure in one fuel assembly.	Vane for rod-control cluster in a fuel assembly separated from the spider nut during operation. Failure occurred in braze joint; no cause found and no other failure was found.	45	5/73
10b	H.B. Robinson 2	PWR		During the past few months, flattened fuel rods have been observed in Region I fuel (unpressurized). No collapsed cladding observed in other regions which contain pressurized fuel.	46	10/73

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
10b (Cont'd)				Reactor refueled in Spring, 1973, with visual inspection of all fuel during core unload and TV inspection of periphery of fuel assemblies in fuel pit. Twenty four of 53 A type assemblies of unpressurized fuel showed one or more collapsed rods. Also in Region 1, 2 cases of severe bowing and 2 instances of failed cladding were noted in the unpressurized fuel. The 53 Region 1 assemblies were replaced.	47, 24	8/73 and 12/74, resp.
10c	H. B. Robinson 2	PWR	One grid strap on one bundle failed. (See 10d below)	Two small sections of a fuel assembly spring clip grid strap made of Inconel were discovered in steam generator during a routine shutdown in November, 1973. Normal reactor coolant would readily carry the grid strap sections into the steam generator channel head. The spring clip grid pieces came from a single corner area of one grid; hence, six fuel rods are partially unsupported at the one grid location. Most likely explanation is that the grid edge caught on some portion of an adjacent assembly as the affected assembly was being inserted into its core position during refueling operations. Results suggest that the grid pieces are from previously irradiated fuel. During forthcoming refueling outage, comprehensive fuel inspection to be conducted to determine location of damaged fuel assembly and affect, if any, on surrounding fuel assemblies.	48	12/73
10d	H. B. Robinson 2	PWR	(Continuation of item 10c above).	During refueling, starting about 5/6/74, the fuel assembly was identified as No. C-08. The pieces came from its 6th grid from the bottom. No apparent damage to other assemblies.	49	8/74

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
10e	H. B. Robinson 2	PWR	During Cycle 2, number of blips per monitored assembly (an indication of densification) increased to about 2.0. Increased iodine activity indicated some fuel clad failures.	Unknown.	49	8/74
10f	H. B. Robinson 2	PWR	During refueling starting about 5/6/74, examination indicated some bent nozzle springs in three Region 4 assemblies.	Mechanical damage.	49	8/74
11a	Humboldt Bay 3	BWR	Three leaky Type II fuel assemblies detected by sipping.	Failure type unknown. The 3 leaker fuel assemblies had exceeded their design exposure.	6	5/72
11b	Humboldt Bay 3	BWR	Eleven leaker Type III fuel assemblies identified.	The failed fuel rods in the leaker fuel assemblies exhibit the characteristics of early-life hydride failures.	6	5/72
11c	Humboldt Bay 3	BWR	Sixteen of 86 assemblies dry sipped during Fall 1973 refueling were identified as leakers.	Unreported.	24	12/74
11d	Humboldt Bay 3	BWR	Refueling started in October 1974. Sixty assemblies were selectively dry sipped. Eleven leakers were identified.	Unreported. Elements were all in high power density regions.	50, 28	11/74, and 1/75, resp.
12a	Indian Point 1	PWR	One fuel assembly with broken top nozzle.	After loading a spent fuel assembly into shipping cask and while trying to disengage the loading tool which would not release, the top nozzle was broken from the fuel assembly. Fuel rods were not damaged. Causes of nozzle and grapple failures are being investigated.	51	8/70
12b	Indian Point 1	PWR	Coolant activity has indicated one or two leaking fuel rods. (See item 12d below).	Failure type unknown.	44	4/72

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
12c	Indian Point 1	PWR	Top nozzles on two fuel assemblies became separated from the perforated stainless steel cans.	In both cases, tack welds holding the can to the nozzle failed during refueling and spent-fuel cask loading operations.	52	6/73
12d	Indian Point 1	PWR	Plant was not operated in 1973. Coolant activity level indicated approximately 1 fuel rod failure.		24	12/74
13a	Kewaunee	PWR	On 9/4/74, primary coolant activity level increased suddenly. Confirmed to be a leaking rod.	No apparent cause can be identified at this time. No indications of clad creep.	53	9/74
14a	Maine Yankee	PWR	During receipt inspection of fuel, a condition of non-contact between some Zircaloy grid spring fingers and the fuel rods was noted.	Believed to be caused by excess lateral loads applied to the fuel rods during handling or shipping.	54	1973
14b	Maine Yankee	PWR	One fuel assembly replaced because of damaged grids. One fuel assembly had to be modified.	Basket containing in-core loading detector was being removed and caught under hold-down plate of an adjacent fuel assembly, lifting it off its 4 alignment pins and damaging 2 spacer grids. The fuel assembly was replaced with a spare. Two diagonally located support-plate alignment pins were found to be out of alignment (fabrication error); a fuel element had to be modified by enlarging the pinholes before it would fit properly.	55	11/72
14c	Maine Yankee	PWR	Higher than average coolant activity indicates that some fuel rods have failed. (See items 14d and 14e below)	Failure type not indicated yet. Reactor may be shut down in June 1974 (orginally scheduled for refueling next year) to correct condition.	56	5/74
14d	Maine Yankee	PWR	On a fuel rod basis, the failure rate is <0.1%. (See items 14b and 14e)		34	6/74
14e	Maine Yankee	PWR	Plant shutdown on 6/28/74 (earlier than anticipated) due to high iodine release rates. All assemblies sipped and 43 leakers were identified (41, 1 and 1 in Regions B, A and C, respectively).	Most likely hydriding and/or pellet clad interactions; analysis is continuing. In addition, problems were identified concerning fuel pin bowing and spacer-grid damage; the causes for these were not reported. The bowing of fuel pins resulted in some fuel loading problems.	57, 28,	8/74, and 1/75, resp.

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
14f	Maine Yankee	PWR	Licensee data indicates a factor of 10-15 increase in I-131 levels in primary coolant system during last two months of 1974. Gross primary coolant activity has increased from 1% to 6% of Technical Specification limit. Average energy of primary coolant sample has been drifting downward which would be indicative of fuel failures.	Unknown at this time. Licensee is planning to reduce power level to 80% until scheduled refueling in May, 1975. Licensed power level is presently 95%.	58	1/75
15a	Millstone 1	BWR	Off-gas trend suggests some fuel rod failures exist in the core (has 508 fuel assemblies). (See item 15b below)	Failure type unknown; no fuel inspection to date. The fuel rod failures are suspected to be early-life hydride failures. Only a portion of the initial core fuel assemblies contain fuel rods which have been vacuum outgassed.	6	5/72
15b	Millstone 1	BWR	Of 112 fuel assemblies discharged, 105-110 leakers determined by sipping out of core.	Unreported.	59	1/73
15c	Millstone 1	BWR	Plant restricted frequently to 80% power due to off-gas activity. Refueling started in Summer 1974; of about 460 assemblies dry sipped, approximately 25 were leakers. Some visual examinations also performed.	Data, and its interpretation, are not complete at this time.	28	1/75
15d	Millstone 1	BWR	During refueling, while transferring an unchanneled spent fuel bundle from a fuel preparation machine to a spent fuel rack in the fuel pool, the bundle fell from the main grapple to the floor of the spent fuel pool. No release of activity was measured, even though the bundle was damaged	Design deficiency of the grapple.	60	9/74
16a	Monticello	BWR	Offgas trend suggests that some fuel rod failures have occurred in the core (has 484 fuel assemblies). (See item 16b below)	No fuel inspection performed yet. Fuel rod failures probably due to early-life hydriding. Initial core fuel loaded in Monticello was not vacuum outgassed during fabrication.	6	5/72

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Date
16b	Monticello	BWR	During the first refueling shutdown during Spring 1973, sipping identified 25 out of 484 assemblies as leakers. A total of 163 fuel rods were rejected.	Predominant failure mechanism in the relatively low exposure Cycle 1 fuel was hydriding. Failed assemblies replaced by 20 Type B 7x7 assemblies and 5 reconstituted assemblies. Off-gas rates at end of 1973 were indicative of several additional failures.	61	7/74
16c	Monticello	BWR	During Cycle 2, power was administratively reduced to reduce stack off-gas activity. (See item 16d below)		61	7/74
16d	Monticello	BWR	During refueling, starting about 3/15/74, in-core and out-of-core wet sipping identified 83 leaking assemblies out of 484.	Limited visual inspection indicated that pellet-clad interaction was the predominant failure mechanism. For Cycle 3, 116 8x8 assemblies plus 7 reconstituted assemblies were inserted in the core.	61	7/74
16e	Monticello	BWR	During Cycle 3, power was administratively limited at various levels to reduce stack off-gas activity.	Not known at this time. Consideration is being given to sipping and replacement of defective assemblies prior to the end of design life. (As of December 31, 1974, a refueling is planned for early 1975).	61	7/74
17a	Nine Mile Point	BWR	Above normal off-gas activity indicated increasing fuel rod leakage. (See item 17b below)	Maximum reactor power will be limited until fuel is replaced.	62	6/71
17b	Nine mile Point	BWR	100% of core (532 fuel assemblies) sipped and 38 leaky fuel assemblies identified.	The leaker fuel assemblies showed predominant failure characteristics of early-life cladding hydride attack; however, 10 of the leaker assemblies had fuel rod failures attributed to fretting wear from debris trapped in spacers. Of the 38 leaker fuel assemblies, 22 were repaired (failed rods replaced) and 14 of the 22 recharged into reactor. Additional leakers were replaced in April, 1972.	6	5/72
17c	Nine Mile Point	BWR	During the Spring 1973 refueling, 104 of 532 assemblies were identified as leakers.	Unreported.	24	12/74

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
17d	Nine Mile Point	BWR	Refueling started 3/29/74. Wet sipping of assemblies in core identified 28 leakers.	Unreported.	63	8/74
18a	Oconee 1	PWR	Coolant activity levels observed correspond to fission gas escape through small pinholes. (See item 18b below)	There has been very little change in activity level as a function of time since startup. In January 1973, one-half of fuel was replaced with prepressurized fuel rods.	64, 65	6/74 and 1/74, resp.
18b	Oconee 1	PWR	During the Fall 1974 refueling, visual examinations and physical measurements were made on a few fuel assemblies, in accordance with the Technical Specifications.	Results are preliminary at this time, but no defective assemblies were detected. Approximately one-third of the higher burnup assemblies were discharged from the core. Difficulties were encountered with fuel handling equipment.	28	1/75
19a	Oyster Creek	BWR	During sipping operations to detect fuel leakers, 1 fuel assembly was found to have been installed 90 degrees counter-clockwise from its proper position. The reactor had been operating 6 months in this condition.	Personnel and/or procedure deficiencies. The fuel assembly had been improperly loaded into the core and 4 administrative checks had failed to discover the situation.	66	5/72
19b	Oyster Creek	BWR	100% of core (560 fuel assemblies) sipped and 44 leaky fuel assemblies identified.	Fuel rod failures identified predominantly had characteristics of early-life hydride attack. Of the 44 leaker fuel assemblies, 20 were repaired (i.e., failed rods replaced) and recharged into reactor.	6	5/72
19c	Oyster Creek	BWR	100% of core (560 fuel assemblies) sipped during each outage. Bundle failure (activity release to coolant) is due to only a few perforated rods among the 49 in an assembly. Fuel rod failure rate <0.5% even for earliest cycles. (See item 19d below)	Relationship between fuel assembly and fuel rod failure frequency indicates some positive correlation in fuel rod behavior within an assembly. Observed clustering of failures is felt due to similarity in operating environment within an assembly rather than casual failure interaction mechanisms between rods.	67	6/73
19d	Oyster Creek	BWR	In the Spring 1973 refueling, 77 of 560 assemblies were identified as leakers.	Unreported	24	12/74

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
19e	Oyster Creek	BWR	Refueling outage started 4/13/74. In-core sipping procedures identified 27 leakers out of 560 assemblies.	Unreported.	68	8/74
20a	Palisades	PWR	After inspection of control blade upper end fittings, one blade bound slightly as it was being reinstalled into the core. The binding was caused by bent guide rod nut capture devices on two adjacent Type A fuel bundles. Further inspection revealed 20 bent nut capture devices in the core, all on Type A fuel.	Design and/or procedure deficiences. The 0.060 inch thick nut capture devices bend very easily. Also, it is difficult to insert the blade by a crane without the blade catching on the edges of fuel assemblies.	69	12/71
20b	Palisades	PWR	On a fuel rod basis, the failure rate is <0.1%.	Unreported.	34	6/74
20c	Palisades	PWR	During an extended shutdown period, fuel was being stored in a spent fuel pool. The fuel was eventually reloaded into the reactor. One fuel bundle was damaged and had to be replaced.	Damage caused by handling.	70	8/74
21a	Pilgrim 1	BWR	The first core utilized temporary poison curtains in the bypass regions (zones between the channel box assemblies). Coolant flow through bypass flow holes caused curtain vibration which resulted in damaged fuel channels. During the late December, 1973 shutdown, the channels were inspected. Damage observed ranged from slight to through-wall wear. All damaged channels were replaced. (See similar item 31c).	Design deficiencies.	71, 72	11/73 and 2/74, resp.

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Date
21b	Pilgrim 1	BWR	During transfer, an irradiated fuel bundle became detached from grapple and fell about 20 feet in the spent fuel pool. Grapple hook apparently was not completely latched under handle of the fuel element. There was no measurable release of activity. The nose piece and the nose piece end of the fuel channel were crushed; there were no indications of broken fuel rods.	Design and/or procedure deficiencies of the grapple. Subsequently, a switch was installed on the grapple to indicate closure of the hook by activating a light on the bridge console. The immediate fix was additional administrative controls requiring visual monitoring of grapple hook closure.	73	1/74
21c	Pilgrim 1	BWR	During refueling outage, fuel sipping began on 1/18/74. Sixteen fuel assemblies showed indications of cladding perforations. In addition, 4 other assemblies were damaged.	Unreported. The 20 7x7 design assemblies were replaced by new 8x8 design assemblies.	74	8/74
21d	Pilgrim 1	BWR	From 12/17/74 through 12/31/74 station operation limited to about 95% of rated power due to high airborne effluent release rates and unexplained perturbations in the Augmented Off-Gas-System.	Not known at this time.	28	1/75
22a	Point Beach 1	PWR	Low-level coolant activity observed from beginning indicating one or two leaking fuel rods. (See item 22c below)	Failure type unknown.	44	4/72
22b	Point Beach 1	PWR	Seventy fuel rods in 26 unpressurized fuel assemblies showed indications of collapse, representing a collapse ratio of 3.5%. (See item 22e below)	Examination performed by binocular observation. At time of shutdown, core had 13,000 effective full power hours. Prepressurized rods exhibited no evidence of collapse.	75	10/72
22c	Point Beach 1	PWR	Of 105 fuel assemblies sipped, 23 were leakers and 1 was suspect.	Weak relationship found between leaky fuel assemblies and those with collapsed fuel rods. No correlation was found between collapses and core location, burnup, or fuel-assembly inserts.	76	11/72

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
22d	Point Beach 1	PWR		During the past few months, flattened fuel rods have been observed in Region I fuel (unpressurized). No collapsed cladding observed in other regions which contain pressurized fuel.	46	10/73
22e	Point Beach 1	PWR	Twenty five fuel assemblies with failed rods (collapses and leaks); 6 fuel assemblies with collapsed sections have no leaks.		59	1/73
22f	Point Beach 1	PWR	During the June 1974 startup of Cycle 3, higher than expected main coolant radioactivity indicated some rod defects.	Tentatively attributed to pellet-clad interaction in conjunction with a rapid rate of reactor power increase after the refueling shutdown.	4	11/74
22g	Point Beach 1	PWR	On site examinations performed during late 1972 revealed some bowed fuel rods.	Measurements of 3 bowed rods showed the bow to be a few tens of mils from a true centerline. Examinations of the rods showed no signs of abnormalities due to operation of these rods in the bowed condition.	4	11/74
23a	Point Beach 2	PWR	During the Fall 1974 refueling, about 48 higher burnup assemblies were discharged from the core. The remaining assemblies were reinserted after visual inspection. No defective assemblies were observed.	Examinations are not complete at this time. In one of the assemblies reinserted into the core, a small chip was found and was removed.	28	1/75
24a	Quad-Cities 1	BWR	Release rate for I-131 exceeded several times during a 5-day period. (See item 24b below)	Defective fuel elements will be replaced during upcoming refueling outage.	77	7/73
				The plant was administratively limited in power level at times, starting in the last half of 1973, to maintain stack rates at acceptable levels.	78	2/74
24b	Quad-Cities 1	BWR	During refueling outage starting 3/31/74, in-core and out-of-core sipping identified 29 leaker assemblies out of 724.	Cladding hydriding and pellet-clad interactions.	79, 80	5/74 and 8/74, resp.

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
25a	R. E. Ginna 1	PWR	Coolant activity increases observed in March 1970. Leaks were confined to 32 fuel assemblies in Region 3. Replacing the 12 worst leaker assemblies with fresh ones reduced activity to about half the level prior to outage.	Leaky fuel assemblies identified by visual examinations and leak testing. Evaluation of observations suggested local hydriding resulting from fuel-contained moisture as the likely cause of the leaks; it was later confirmed that source of leaks was moisture contained in the fuel.	44, 81	4/72 and 9/71 resp.
25b	R. E. Ginna 1	PWR	End plug separated from fuel rod.	During refueling operations, one fuel element would not bottom properly, protruding 1/2 in. above other core assemblies. Four days later an end plug from a Region 3 fuel assembly was retrieved from the bottom core plate. Plug to be examined to see why it separated from the fuel rod; expected reason is severe internal hydriding.	82	10/72
25c	R. E. Ginna 1	PWR	Fuel rod end-plug recovered. No indication of fuel deterioration observed after 48 fuel assemblies were replaced with other assemblies.	About 13 days required for replacement of 48 unpressurized fuel assemblies and recovery of a fuel rod end-plug from the lower core-support plate.	83	2/73
25d	R. E. Ginna 1	PWR	0.4%, based on primary coolant activity early in first cycle; went additional 400 days before further defects indicated.	Study, without visual examination, indicated most probable cause to be internal hydriding due to moisture which was later confirmed. Prior to this, modifications in fuel production had been introduced to eliminate this since it had been expected. During Spring refueling, collapsed rods observed with collapsed sections ranging from 4-8 cm in length and are the result of gradual creepdown of cladding over an unsupported length due to high differential pressure.	84	12/72
25e	R. E. Ginna 1	PWR		During Cycle 1 refueling, a large number of fuel rods were observed to be in interference with the top nozzles and a few of these rods were bowed. Rod interference and bowing were due to larger-than-expected Zircaloy growth during irradiation.	85	6/73

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
25f	R. E. Ginna 1	PWR		Flattened fuel rods observed in Regions I, II, and III (all unpressurized fuel) during the Cycle 1 refueling in 1972.	46	10/73
25g	R. E. Ginna 1	PWR		Some fuel rods collapsed because of in-reactor densification of fuel. Densification phenomenon reported by AEC to occur at linear heat rates as low as 1 to 2 kW/ft (33 to 65 W/cm).	86	10/73
25h	R. E. Ginna 1	PWR		Fuel failures and collapsed cladding noted. Final 48 non-pressurized fuel assemblies discharged from core.	59	1/73
25i	R. E. Ginna 1	PWR	During 1973, coolant activity was indicative of some fuel failures ($\sim$0.05%).	Unknown.	24	12/74
26a	San Onofre 1	PWR	Coolant activity implies existence of one or two leaking fuel rods during second operating cycle.	Visual examination of fuel discharged during second refueling outage disclosed two damaged fuel rods, which corroborated radiochemistry results. No other anomalies were found.	44	4/72
26b	San Onofre 1	PWR	All fuel assemblies in the core were visually inspected during the June 1973 refueling. No anomalies were noted.		24	12/74
27a	Surry 1	PWR	During first half of 1974, I-131 activity level in primary coolant indicates about 2-4 defective fuel rods (See item 27b below)	Cause unknown at this time. In addition, primary pressure was reduced to preclude fuel collapse. Densification induced power spikes observed in all regions of the core. The number is increasing, but all spikes are relatively small.	87, 88	8/74 and 3/74, resp.
27b	Surry 1	PWR	During the Fall 1974 refueling, visual (binocular) inspection was performed on all 157 assemblies. TV inspection performed on 12, 20, and 12 Region 1, 2 and 3 assemblies, respectively. No defects were observed. Very little crud present. Slight bowing was observed in	No specific failures noted. All examinations are not yet complete. Eighty four higher burnup assemblies were replaced. Number of power spikes decreased in second half of 1974.	28	1/75

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Date
27b (cont'd)			some assemblies. I-131 activity levels at end of cycle 1 indicated 3-4 defective rods.			
28a	Surry 2	PWR	During 1974, I-131 activity level in primary coolant indicates about 1 defective fuel rod.	Cause unknown at this time. Densification induced power spikes observed in all regions of the core. However, the total number has not appeared to increase in the first 6 months of 1974; one additional spike observed in last half of 1974.	87, 28	8/74 and 1/75, resp.
29a	Turkey Point 3	PWR	Coolant activity increased caused by fuel-cladding defects (failure rate very small, perhaps ~0.01%). (See item 29b below)	Unknown.	89, 24	7/73 and 12/74, resp.
29b	Turkey Point 3	PWR	During the Fall 1974 refueling, 157 assemblies were visually (binocular) inspected. Sipping was not done due to equipment problems. Some bowing of fuel rods was observed.	Results incomplete. Some sipping of discharged assemblies may be performed later (about one-third of core discharged). Some trouble in refueling encountered due to the bowed fuel rods.	28	1/75
30a	Turkey Point 4	PWR	One fuel assembly dropped during initial fuel loading.	Fuel assembly dropped 4 or 5 inches (cable clamps did not grip cable) while being raised to the vertical position. Skeleton of fuel assembly replaced before assembly was loaded.	90	4/73
31a	Vermont Yankee	BWR	Because of excessive gaseous release activity levels, power level was reduced until mid-January 1973 shutdown. Vendor indicated possibility that 183 fuel rods out of 18,302 could fail. (See item 31b below)	Cause of activity release is believed to be fuel cladding perforations due to internal hydriding of the zirconium. Cause of hydriding is excessive moisture in fuel rod as a result of inadequate vacuum outgassing during fuel rod fabrication.	91	12/72
31b	Vermont Yankee	BWR	Fifty four of 368 fuel assemblies idenfied as leakers by sipping. Of the 54, 51 had perforated and/or defective fuel rods (an average of 7 rods per assembly). Three hundred seventy defective fuel rods in 51 leaking assemblies have been replaced.	Cause of failure is thought to be internal hydride attack of the Zircaloy cladding. In 8 fuel assemblies which showed no indication of failure by sipping, examination revealed that an average of 4 fuel rods per assembly were defective.	92, 93	Apr. and Feb, 73, resp.

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Date
31c	Vermont Yankee	BWR	Of 368 fuel assemblies, 14 fuel assemblies were hydrided. Of 53 fuel channels, 19 had cracks, holes, and worn spots. An additional 20 leakers were removed in the Fall of 1973.	Hydriding noted on 14 fuel assemblies. Temporary neutron-poison curtains believed to have vibrated and rubbed against channels because of rapid water flow. One hundred fuel channels were replaced because of wear holes and cracks; flow holes in channels were plugged. (See similar item 21a).	65	1/74
31d	Vermont Yankee	BWR	Plant was administratively limited to lower power during 1974 due to excessive off-gas activity at the steam jet air ejectors. (See item 31e below)	Problem attributed to "faulty cladding". Probably caused by hydriding.	94	8/74
31e	Vermont Yankee	BWR	During the Fall 1974 refueling, 328 assemblies were replaced by the new 8x8 design. The remaining 40 (of the improved 7x7 design, including getter, etc.) were wet sipped out-of-core and no defects found. These 40 (having about 1 year of exposure) were reinserted into the core.	See item 31d above.	28	1/75
32a.	Yankee (Rowe)	PWR	Two assemblies removed.	Two Zircaloy-clad test assemblies removed in 1966 because of grid and clip failures. Corrections were made to later test assemblies.	95	3/69
32b	Yankee (Rowe)	PWR		Removal of 4 Zircaloy-clad test fuel assemblies proposed because inspection indicates length change of fuel rods greater than expected.	96	8/69
32c	Yankee (Rowe)	PWR	One fuel assembly damaged.	Crane operator mispositioned spent fuel assembly and damaged fuel assembly and refueling equipment.	97	9/69
32d	Yankee (Rowe)	PWR	Possibility of only one pinhole sized leak.	Failure (?) type unknown. Reactor coolant activity seems to indicate that the pinhole had either sealed itself or that no defect has existed and activity increase represented some uranium contamination on the fuel rod surface.	44	4/72

TABLE C-1 (Cont'd)

Item No.	Plant	Reactor Type	Approximate Failure Frequency	Failure Type	Reference No.	Reference Date
32e	Yankee (Rowe)	PWR	One damaged fuel assembly will not be reused; the fuel assembly was not ruptured.	When upper core barrel was lifted, a fuel assembly stuck to it and was lifted 8 ft above core. While trying to reinsert fuel assembly, it was dislodged and fell several inches to top of core adjacent to its orginal position. Upper nozzle and upper fuel assembly wrapper sheet were damaged. The fuel assembly will not be reused. The hang-up was caused by a small foreign object locking the assembly in place; marks were found on the upper core-support plate.	98, 99	Dec. and Nov., 1972 resp.
32f	Yankee (Rowe)	PWR	One fuel assembly slightly damaged.	The fuel assembly was slightly damaged while the upper core barrel was being removed. A new pressurized fuel assembly was used as the replacement.	100	3/73
32g	Yankee (Rowe)	PWR	During refueling shutdown starting 5/10/74, 12 Core X predetermined fuel assemblies were given close surveillance. Some crud, discoloration, and abrasions were noted.	No failures were noted. The 12 assemblies were considered acceptable for continued second and third cycles of operation in Core XI.	101	11/74
33a	Zion 2	PWR	During initial fueling, fuel assembly B62P was dropped during handling. No visible damage, but the fuel assembly was replaced by another.	Personnel error. Fuel assembly may have been set in a flow hole instead of a fuel aligment pin.	102	12/73

TABLE C-2

FUEL FAILURE SUMMARY CATEGORIZATION
(See Notes 1, 2 and 3 below)

Category	See Item No. in Table C-1
Internal Contamination: (21 items)	2c, 2d(e), 3c, 4a(c,d), 4b (c,e), 4f, 5b, 5c(?), 11b, 14c(d,e)(?), 15a(b)(?), 16a(b) 17a(b), 19b, 24a(b), 25a, 25b(c)(?), 25d, 31a(b), 31c, 31d(e).
Manufacturing Defects: (9 items)	1a, 2ℓ, 3a, 3b(?), 5a, 6a, 9a, 10a, 14b.
Mechanical Damage: (18 items)	4d, 10c(d), 10f, 12a, 12c, 14b, 14e, 15d, 17a(b), 20c, 21b, 21c, 30a, 31c, 32a(?), 32c, 32e, 32f.
Fuel Cladding Interactions: (11 items)	3c, 3d, 3e(?), 5b, 5c(?), 5d, 8f(g)(?), 14c(d,e)(?), 16c(d), 22f, 24a(b).
Accelerated Corrosion: (7 items)	2a(b), 2c, 2d(e), 2i, 2j, 3a, 8f(g)(?).
Fuel Rod Bowing: (11 items)	8a, 8b, 8c, 8d, 8f(g)(?), 10b, 14e, 22g, 25e, 27a(b), 29a(b).
Cladding Collapse: (6 items)	10b, 22b(e), 22d, 25d, 25f, 25g(h).
Other:	
Miscellaneous Design Deficiences: (5 items)	2g, 2h(?), 20a, 21a, 32b.
Unknown or Unreported: (36 items)	2f, 2k, 4f, 4g(h), 7a, 8e, 9b(c), 10b, 10e, 11a, 11c, 11d, 12b(d), 13a, 14f, 15a(b), 15c, 16c(d), 16e, 17a(b), 17c, 17d, 18a(b), 19c(d), 19e, 20b, 21c, 21d, 22a(c), 25h, 25i, 26a, 27a(b), 28a, 29a(b), 32d.

Notes:

1. Items 14a, 19a, 23a, 26b, 32g, and 33a are not included above since no failures were evidenced.

2. To avoid duplication, some items have been combined. For example, 24a(b) is considered one item for purposes of this table; 24a noted an increase in gaseous radioactive effluent while 24b denotes the results of fuel inspection for the same core.

3. Due to lack of specific data, certain of the failures for BWR fuel attributed to hydriding (internal contamination) may have been initially caused by fuel-cladding interactions.