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THE EFFECT OF POWER CHANGE ON THE PCI FAILURE THRESHOLD

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ABSTRACT

Investigations of the PCI mechanism have led to the conclusion that the failure threshold is best defined by the power change (ΔP) during the ramp, rather than the final power achieved at the end of the ramp. The data base studied was comprehensive and includes a wide variety of water reactor systems and fuel designs. It has also been found that operating parameters have a more significant effect on failure susceptibility than fuel rod design variables. The most significant operating variable affecting the failure threshold was found to be the base irradiation history, indicating that fission product release and migration prior to the ramp (during base irradiation) is an important consideration. It can be shown that fuel irradiated at relatively higher linear heat ratings tends to fail at lower ΔP. This effect has also been independently verified by statistical analyses which will also be discussed. Industry out-of-pile internal gas pressurization tests with irradiated tubing in the absence of simulated fission product species and at low stress levels, also tends to indicate the importance of the prior irradiation history on PCI performance. Other parameters that affect the power ramping performance are the initial ramping power and the pellet power distribution which is a function of fuel enrichment and burnup.

1.0 INTRODUCTION

Investigations of the PCI mechanism have led to the conclusion that for fast ramps the failure threshold is best defined by the power change (ΔP) during ramping, rather than the final power achieved at the end of the ramp. This point of view will be supported first by a discussion of statistical analyses that have been performed on large comprehensive data bases. General observations from various out-of-pile and in-pile test programs also tend to support the importance of stress. A correlation of a limited data base will be presented to show the general trend of the PCI threshold as a function of base irradiation power level which indicate that fission product release and migration and the state of the fuel/clad prior to the ramp is an important consideration. A detailed discussion of the effect of the irradiation history will show that irradiation induced resolution affects are critical to PCI susceptibility. The effect of burnup is closely related to the fuel centerline temperature history prior to ramping. Specific examples of ramp test data will be presented to illustrate these effects.

The possibility that clad temperature during ramping, which affects the severity of the clad stress-time history (by affecting the clad stress relaxation rate), may be a PCI variable will also be discussed.

Finally, other parameters which affect the ramping performance will be addressed. These are burnup and enrichment as they affect the power distribution in the pellet which in turn affect pellet expansion for a given power change, and the initial ramping power. This latter parameter dictates the fuel temperature range over which a particular rod is ramped.

2.0 BACKGROUND

Previous PCI analysis published by Westinghouse involved a statistical analysis of a large data base (Reference 1). Typical results from that study are shown in Figures 1 and 2. The data base used encompassed 713 data points (117 failures and 596 non failures) representing a wide variety of water cooled reactor fuel (PWR, BWR, CANDU, and SGHWR). It was determined that the failure boundary could be well defined by the power change, ΔP, and that the failure threshold was sensitive to the irradiation power prior to the ramp, P_i. The results compared favorably (Reference 1) in terms of the form of the model (i.e. the key variables) and threshold trends to PROFIT (Reference 2), the only other known statistical PCI model. In this analysis, most of the data was obtained from the open literature and information on the detailed irradiation history was generally unknown. The value of P_i generally used was therefore the power just prior to the final ramp. It was therefore necessary to assume that this value also represented the mechanically conditioned power and the base irradiation history.

As a result of this study and the detailed evaluations of more complete data sets from ramp test programs such as INTERRAMP (Reference 3) and OVERRAMP (Reference 4), the conclusion is drawn that the PCI failure boundary is best defined by ΔP, which is a measure of the stress applied during the ramp, and the condition of the fuel/clad determined by the irradiation history prior to the ramp. This paper will deal with some aspects of the irradiation history which are considered important in the evaluation of PCI susceptibility.

3.0 OTHER OBSERVATIONS

In addition to the PROFIT model, there are several other observations from various industry experiments and test programs which in various ways tend to support the above conclusions.

a) Out-of-pile internal gas pressurization tests (Reference 5) with irradiated cladding in the absence of simulated fission product species and at low (below yield), constant stress levels have resulted in PCI type defects. The diametral strains at failure were <1% and the size and tightness of the defects were similar to those in iodine stress corrosion cracking tests. The test samples were exposed to varying amounts of fission products during prior irradiation as determined by rod puncture.

b) It is common practice to correlate PCI ramp test data with the ramp terminal level indicating that transient fission product release during the ramp plays a role in the failure mechanism. Good separation of failures and non-failures, however, is usually difficult to obtain by this procedure. The large fission gas releases obtained in non failed rods ramped to high power levels (and with large ΔP) indicate that transient release may be delayed so as not to affect the failure mechanism. Although short hold time transient fission gas release data is relatively scarce, the data available does tend to indicate that the amount of release in the relatively short times associated with PCI failures is small (Reference 6).

c) The Studsvik Demo Ramp II Project has also shown that cladding cracks initiate within the order of seconds rather than minutes at the terminal power level (Reference 7). This result also implies an importance of the irradiation history.

d) Out-of-pile SCC tests where the test samples incubate at temperature with iodine prior to the application of stress have shown shorter times to failure than standard SCC test where the stress, iodine and temperature are introduced simultaneously (Reference 8). This indicates that clad exposure to fission products prior to the application of stress can have an affect on the failure mechanism.

e) Fuel defects have also been detected during reload startups of commercial reactors (Reference 13). These type of defects occur at intermediate reactor power levels where fuel temperatures are relatively low and transient fission product release is not expected during the power increase. Although such defects have not as yet shown to be PCI failures with PIE, circumstantial evidence indicates strongly that they are. Their occurrence is directly related to power increases and is influenced by ramp rate, i.e. such defects can be prevented by a slow ramp rate. Also the ratio of the I-131 to I-133 coolant activity after such occurrences is relatively high which is an indication of small, tight defects. It is therefore concluded that high stress in a small percentage of rods (due to localized deconditioning during handling and the subsequent power increase) in combination with prior irradiation are sufficient to cause this type of PCI defects.

4.0 THE EFFECT OF BASE IRRADIATION HISTORY

The base irradiation history is considered to be important as it affects the amount of fission product release and the state of the fuel prior to any subsequent ramp the fuel might experience. Figure 3 is a plot of the fuel grain boundary fission product saturation burnup as a function of irradiation temperature and fission density published in Reference 9. The theoretical aspects of the derivation of this correlation which is related to the competition of diffusion and irradiation induced resolution will not be dealt with here but is discussed in Reference 10. Reference 11 also discusses the effects of resolution. Although these relationships were generated under constant irradiation conditions, they are believed to represent the trend of grain boundary saturation temperature as a function of burnup and fission density for rods irradiated at lower heat ratings, i.e., if fuel is irradiated at a relatively low heat rating and experiences a power increase, the saturation temperature (T_{sat}) is more likely to be reached at the higher heat rating if the accumulated burnup is greater and/or the fission density history (w/gmU) is lower. T_{sat} is that temperature above which increased fission gas release (and fuel swelling) can be expected as shown in the insert in Figure 3. In this paper the calculated fuel centerline temperature during irradiation prior to the final ramp for some ramp tested rods will be compared to the curves in Figure 3. Experimental evidence will be presented to show that in some cases, grain boundary saturation conditions existed. It can qualitatively be shown that PCI performance is related to the proximity of the fuel temperature to saturation conditions during base irradiation.

4.1 DISCUSSION

First some selected PCI data to illustrate the general trend of the threshold (ΔP) as a function of irradiation power (P_i) level will be discussed. These data are shown in Figure 4. Rods with low P_i are generally irradiated with fuel centerline temperature below T_{sat} and fission product release during base irradiation would be low. Conversely rods with very high P_i (>35 kw/m), fuel temperatures are greater than T_{sat} at appreciable burnup which would result in much higher fission product release (and fuel swelling) prior to the ramp so as to make the clad susceptible to failure at lower stress (i.e. lower ΔP).

Rod A (a W 17x17 rod that was irradiated in the BR3 reactor for 2 cycles to a burnup of ~35 MWD/KgU and ramped at Studsvik in R2) and the OVERRAMP rods were irradiated at intermediate power levels and as can be seen from Figure 4 there is apparent scatter in the results for this particular data set. In the case of OVERRAMP the failure threshold was well defined for each group of rods but there were differences in the thresholds between groups. All these rods were, however, preconditioned at 30.0 kw/m in the R2 reactor before the final ramp and analysis indicated that fuel temperatures during the preconditioning phase were close to saturation conditions for some groups. These results lead to an explanation for the apparent scatter in this data. The data in Figure 4 is discussed below in greater detail.

4.2 FUEL TEMPERATURE LESS THAN T_{sat}

The KAHL tests (Reference 12) shown in Figure 4 were BWR rods irradiated at ~ 11.0 kw/m to a burnup of ~ 15 MWD/KgU. The rods were at ~ 15 to 17 kw/m for a short period of time between 8-10 MWD/KgU and were preconditioned for 1 hour at 15 kw/m. Fuel centerline temperature was well below Tsat during the entire base irradiation history. The rods survived large power increases.

4.3 FUEL TEMPERATURE GREATER THAN T_{sat}

The power histories for the INTERRAMP rods (Reference 3) oscillated between high power periods (of about 3 MWD/KgU duration) and low power periods (of about 2 MWD/KgU duration). The rating during the high power periods ranged from ~35 kw/m (for the low power rods) to ~39 kw/m (for the high power rods). T_{sat} was estimated as a function of burnup from Figure 3 based on the fission density (w/gm). Analysis showed that the fuel centerline temperature was above T_{sat} during the second high power period for the low burnup (~10 MWD/KgU) rods and for the second, third and fourth high power periods for the high burnup (~20 MWD/KgU) rods. A rough estimate of the fission product release during the base irradiation of the INTERRAMP rods can be obtained from rod HR2. This rod was ramped to 38.0 kw/m which is in the range of base irradiation levels during the high power periods. The measured release was ~3.5%, which correspond to the rod average. INTERRAMP metallography showed that some regions of the clad inside were covered by a thicker Zr-oxide film than normal and occasional adhering or bonded microsized UO_2 fragments indicating the existence of intimate and firm interfacial contacts during base irradiation (Reference 3). The ΔP failure threshold for the INTERRAMP data was estimated to be ~9.0 to 10 kw/m. This threshold was based on the difference between the ramp terminal level and the rating during the peak high power period. A small correction was made for clad creepdown during subsequent low power operation.

The Westinghouse Zorita Program (References 13, 14) involved the irradiation of high enrichment test rods at high power and to high burnup. Post irradiation examinations indicated that several rods failed due to PCI. During reactor operation a local transient occurred during which several of the high power rods experienced a power increase of ~5.5 kw/m after which an increase of reactor coolant activity was detected.

Estimated fission product release based on similar rods that did not fail was ~8 to 12%. Metallography showed significant bonding over a large portion of the circumference even at axial positions away from the ramped location.

Although no detailed analysis was done for the CANDU "ripple" defects (Reference 15), the pre-ramp irradiation level was high and the rods failed after very low power changes.

4.4 INTERMEDIATE FUEL TEMPERATURE

The OVERRAMP rods and rod A were irradiated at intermediate power levels between 13 to 27 kw/m. As can be seen from Figure 4 there is apparent scatter in the ramp test results as discussed previously. The analysis of the irradiation histories of these rods which offer an explanation of the ramp results is discussed below.

4.4.1 ROD A

The design of rod A is similar to the Westinghouse 17x17 design but with a length of ~1 meter and an enrichment of 8.26 w/o. The rod was irradiated for two cycles in the BR3 reactor at Mol, Belgium to a burnup of ~35 MWD/KgU at linear ratings between 13 to 20 kw/m at the peak axial location. After shipment to Studsvik, Sweden, the rod was preconditioned at ~30.0 kw/m in the R2 reactor prior to ramping. Shown on Figure 3 is the calculated fuel centerline temperature during the preconditioning phase. The fission density vried between 28 to 42 W/gmU during base irradiation. Interim examinations were performed after preconditioning and prior to final ramping. It is significant that neutron radiography indicated partial dish filling. This is evidence of gaseous swelling indicating that the fuel did operate near or actually above the saturation temperature. It is believed that this observation represents evidence that there is validity in this type of analysis (of comparing fuel temperatures to the curves on Figure 3). Rod A was subsequently preconditioned again at 30.0 kw/m and was ramped to 37.5 kw/m. It failed at this relatively low ΔP (7.5 kw/m) as shown in Figure 4. It is believed that the failure of this rod is related to the conditions in the fuel rod established during preconditioning.

4.4.2 OVERRAMP DATA

The OVERRAMP (Reference 4) rods were irradiated at ratings between 13 to 27 kw/m but were subsequently preconditioned at 30.0 kw/m for 72 hours in the R2 reactor prior to final ramping. The predicted fuel centerline to temperature during preconditioning for the various rod groups is plotted on Figure 3. The groups that performed relatively poorly during ramping (as shown in Figure 4) are indicated by (•). As can be seen there is a clear relationship of the PCI performance and the fuel temperature during preconditioning. For example groups 5, 6, and 7 had very similar fission density histories. However group 5 was irradiated at relatively higher temperature during base irradiation, including the preconditioning phase and this group failed at lower ΔP. The fission density for the OVERRAMP rods varied between 28 to 46 W/gmU during base irradiation. Rod groups 1 to 4 operated at relatively lower fission density at the end of base irradiation just prior to preconditioning which may have resulted in a lower grain boundary saturation temperature for these particular rods. Results for two lower burnup groups are also shown in Figure 3. Fuel temperatures were well below T_{sat} and rods in these groups survived large power changes.

Based on the dish filling observed for rod A and the fuel temperature evaluations shown in Figure 3 it is believed that the PCI performance of OVERRAMP rods is related to the degree of grain boundary saturation experienced during irradiation prior to ramping, particularly during the preconditioning phase. Possible mechanisms for this relationship are discussed in Section 4.5. Upon inspection of the fuel temperature calculations, the reason for the relatively poor performance (i.e. high temperature at 30 kw/m) of some of the rod groups appears to be related to one or more of the following: (a) larger as-built gap, (b) slower creepdown of the clad (which was normalized to profilometry measurements) during base irradiation and (c) higher enrichment which results in a flatter pellet power shape and relatively greater fuel temperatures at higher burnup (this will be discussed in Section 6.0).

4.5 SUMMARY OF IRRADIATION HISTORY EFFECT

This analysis leads to some interesting conclusions pertaining to the effect of burnup on PCI. Upon inspection of Figure 3, it would appear that high burnup fuel would not be highly susceptible to PCI if the power history was such that fuel temperature was monatonically decreasing so as to remain well below the grain boundary saturation temperature (as would normally be the case for high burnup power reactor fuel) and, in the case of a ramp test, the preconditioning level is not too high. The effect of fission product resolution at low temperature would prevent grain boundary saturation from occurring. On the other hand, at low burnup, the PCI threshold can actually be lower if fuel temperatures are above T_{sat} as was the case in the INTERRAMP Project.

The reasons that fuel temperatures near or above T_{sat} during base irradiation affect PCI performance are believed to be related to one or more of the following phenomenon:

a. Increased release and migration of fission product species. The greater the release before ramping, the lower the stress threshold.

b. Fuel swelling as evidenced in the neutron radiography of rod A. Swelling would result in tighter pellet/clad contact and possibly higher pre-ramp clad stress.

c. Fission product species might be stored at grain boundaries that would be released immediately during the ramp. The quantity of such species would be a function of the irradiation history.

It is believed that the considerations presented here as to the effect of the base irradiation are important to the understanding of the PCI failure mechanism. Currently these evaluations are primarily qualitative and there are uncertainties, particularly since predictions of fuel temperature are involved. It is important to understand and resolve the effects of operational variables such as the power history before conclusions are drawn as to the effects of design variables. It is recommended that in future ramp test programs, more emphasis should be placed on interim PIE to study fuel characteristics that exist prior to final ramping.

5.0 CLAD TEMPERATURE DURING RAMPING

Some degree of success has been obtained in separating failure and non-failure ramp test data using the clad inner surface temperature during ramping (i.e. at the ramp terminal level) as a variable. For example, some of the poorer performing rods in the OVERRAMP project were also ramped at lower clad temperature which may have also affected their performance. The statistical analysis (Reference 1) determined clad temperature to be a variable with a decrease in the (ΔP) failure threshold of ~4.5 kw/m for a decrease in clad temperature of 50°C. It is possible that reduced clad temperature will reduce the stress relaxation rate to such an extent so as to result in a more severe stress-time history, thereby accelerating crack propagation and influencing the failure probability. It could be that the PCI failure boundary is defined by both clad temperature and power history as shown in Figure 5. Unfortunately the effect of irradiation history cannot be quantified accurately enough to separate the effects of the two variables based on the ramp test data currently available. It is planned to utilize the forthcoming data from the DOE High Burnup PWR Ramp Test Program at Petten (Reference 16) to resolve this issue. This program will supply 88 additional data points with rods similar to those in the OVERRAMP Project. It is hoped that when this data is considered, there will be enough rods with similar power histories to verify whether or not clad temperature is a variable.

6.0 EFFECT OF ENRICHMENT AND BURNUP

It is the intent of this section to describe the effect of as-built enrichment on fuel performance during ramp tests. It is believed that this is an important consideration in the evaluation of certain data sets, such as OVERRAMP, where a considerable portion of the tests involved high enrichment (8.26 w/o) rods irradiated in the BR3 reactor.

Figure 6 shows the calculated radial profile in the pellet of the key fissile isotopes for enrichments of 8.26 (in BR3) and 3.0 w/o (in typical PWR spectrum) and at burnups of 0 and ~20,000 MWD/MTU. As can be seen the amount of U-235 depletion at 20,000 MWD/MTU is approximately the same for the two enrichments. However due to the much larger initial concentration of U-235, the relative effect of the Pu-239 buildup is much less for the higher enrichment. In this case the absolute magnitude of the Pu-239 is also lower for the higher enrichment due to the lower fast flux in the BR3.

This has an effect on the pellet power distribution at higher burnup. The upper curves in Figure 7 show the pellet power distribution at 20,000 MWD/MTU for the two enrichments. The large effect of the plutonium buildup for the lower enrichment can be seen. The higher relative centerline power for the higher enrichment results in higher centerline fuel temperature. For the case of zero burnup, the opposite is true, as can be seen from the lower curves in Figure 7. The high enrichment power distribution is steeper due to the higher initial fissile (U-235) concentration. One way of illustrating the effect on fuel temperature is shown in Figure 8 which shows the power depression factor F, which can be derived from the pellet power distribution, as a function of enrichment and burnup.

It can be seen from Figure 8 that after less than 5 MWD/kgU, higher fuel temperatures would be expected for the high enrichment case. Figure 9 shows the fuel centerline temperature during a power increase following a base irradiation at 25 kw/m to a burnup of 25 MWD/KgU. It can be seen that the higher enrichment results in higher fuel temperatures. Of particular importance in the evaluation of ramp test data is the effect on pellet expansion. Figure 10 shows the effect of enrichment on pellet expansion for a power increase of 15 kw/m (30 to 45 kw/m) for burnup greater than 20 MWD/Kg. Due to the flatter pellet power shape and higher fuel temperatures, pellet expansion is greater for the 8.26 w/o enrichment irradiated in BR3. At 35 MWD/KgU for example, pellet expansion is ~8% greater than for the typical enrichment (3 w/o).

One other effect should be noted which was not accounted for in this analysis. The evaluation of the fissile and power distribution for the high enrichment rods was done in the BR3 environment, i.e. BR3 coolant conditions, neutron spectrum, etc. While this will give the proper fissile isotopic distribution, the power distribution will not be the same as when the rod is tested in R2 since the R2 spectrum is different. The R2 neutron spectrum is less thermal which results in a flatter power distribution. The effect of enrichment on fuel temperatures and pellet expansion may therefore be slightly greater in the R2 than shown in Figures 9 and 10.

7.0 EFFECT OF INITIAL RAMPING POWER

Another effect to consider when evaluating ramp test data, particularly at different preconditioning levels, is the initial ramping power. The initial ramping power dictates the fuel temperature range over which a rod is ramped, which in turn affects the thermal expansion of the pellet due to the temperature dependence of the thermal conductivity and the coefficient of thermal expansion of UO_2. Pellet diametral expansion (ΔD) can be expressed simply as:

$$\frac{\Delta D}{\Delta P} = \frac{\Delta T}{\Delta P} \times \frac{\Delta D}{\Delta T}$$

Since the thermal conductivity of UO_2 decreases monotonically as a function of temperature to ~1700°C, the change in fuel temperature for a given power change, $\Delta T/\Delta P$, is less at lower temperature or initial power. Likewise the coefficient of expansion, $\Delta D/\Delta T$, is lower at lower temperatures. Figure 11 shows the results of a parametric study performed with an enrichment of 3 w/o to illustrate these effects. Shown is the power change required for a diametral expansion. $\Delta D/D$ equal to 0.4% as a function of initial power and burnup. At these higher burnups there is complete pellet/clad contact so the effects of UO_2 properties discussed above are dominant and there is relatively little effect due to gap closure. For an initial ramping power of 25.0 kw/m, an 8% larger power increase is required to give the same pellet expansion as a ramp starting from 30.0 kw/m. Also a small benefit is seen at very high burnup. This is due to the increased plutonium buildup at the pellet surface which results in a steeper pellet power shape with a lower relative power at the center. As noted in the previous section, this type of shape results in less pellet expansion for a given power change. (This effect is also seen in the bottom curve in Figure 10.)

8.0 RESIDUAL CLAD STRESS AFTER PRECONDITIONING

Another important consideration in the evaluation of certain ramp test data is the residual clad stress after preconditioning. The ΔP values for the OVERRAMP data plotted in Figure 4 were determined by subtracting the preconditioning power level, 30.0 kw/m, from the ramp terminal level. The local heat ratings of the rods, however, were considerably lower than 30.0 kw/m at the end of the prior base irradiation. The clad would therefore experience some stress at the beginning of preconditioning which would relax during the 72 hour hold time at 30.0 kw/m. It is felt that the residual clad stress at the end of preconditioning represents an important credit when evaluating the effective severity or true ΔP of the ramp. In order to estimate this effect, stress calculations were performed which resulted in an equivalent ΔP adder of ~5 kw/m which could be added to the quantity, RTL minus 30 kw/m, to determine the effective ΔP of the ramps.

9.0 SUMMARY

The following is a summary of the important points presented in this paper.

1. For fast ramps, the PCI threshold is best defined by the power change, ΔP.

2. Statistical analysis and observations from test programs and out-of-pile tests have been presented which tend to support this view.

3. A key variable which determines the threshold is the base irradiation (fuel temperature) history prior to ramping. Fuel that is irradiated above grain boundary saturation temperature for fission product release tends to fail at lower stress, i.e. lower ΔP.

4. Other factors which affect ramping performance are the pellet power distribution, which is a function of enrichment and burnup, and the initial ramping power which determines the temperature range over which the fuel is ramped.

5. It is recommended that in future ramp test programs that more emphasis be placed on PIE (e.g. neutron radiography, metallography and fission gas measurements) to study fuel characteristics that exist prior to final ramping.

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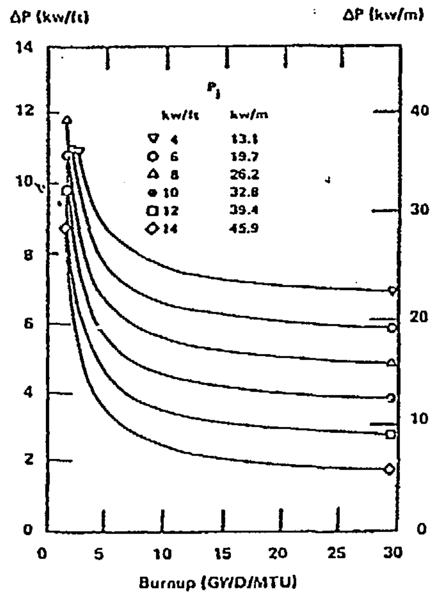

Figure 1. Predicted Change-In-Power (ΔP) Failure Boundaries as a Function of Burnup

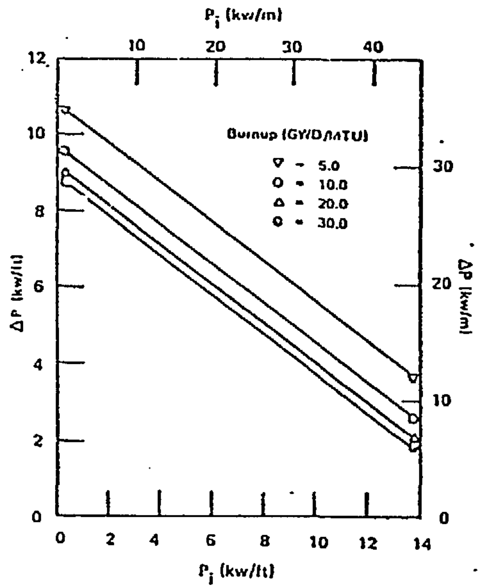

Figure 2. Predicted Change-In-Power (ΔP) Failure Boundaries as a Function of Irradiation Power (P_i)

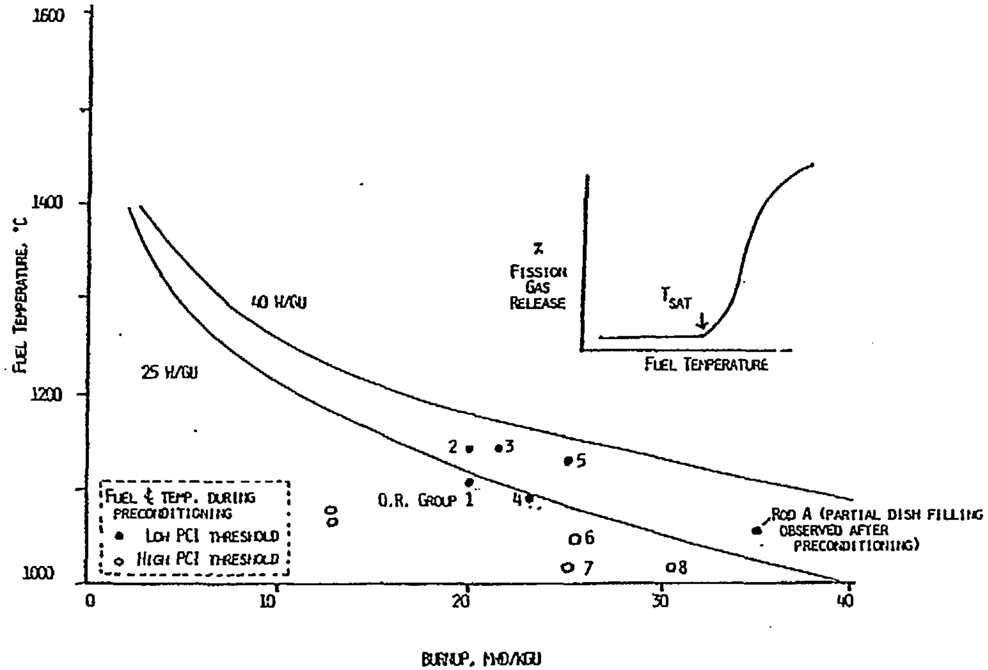

FIGURE 3

GRAIN BOUNDARY SATURATION BURNUP AS A FUNCTION OF TEMPERATURE AND FISSION DENSITY (REFERENCE 9)

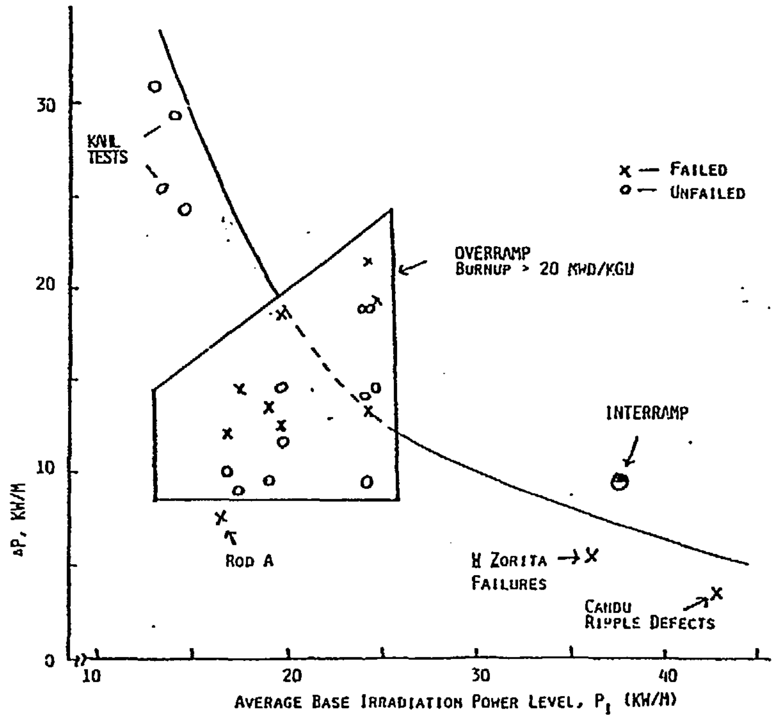

FIGURE 4

TREND OF PCI FAILURE THRESHOLD (ΔP) WITH BASE IRRADIATION POWER LEVEL

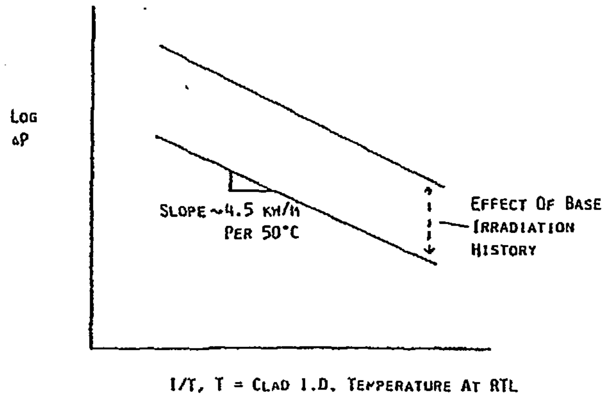

Figure 5
PCI Failure Space Defined By Clad
Temperature And Base Irradiation History

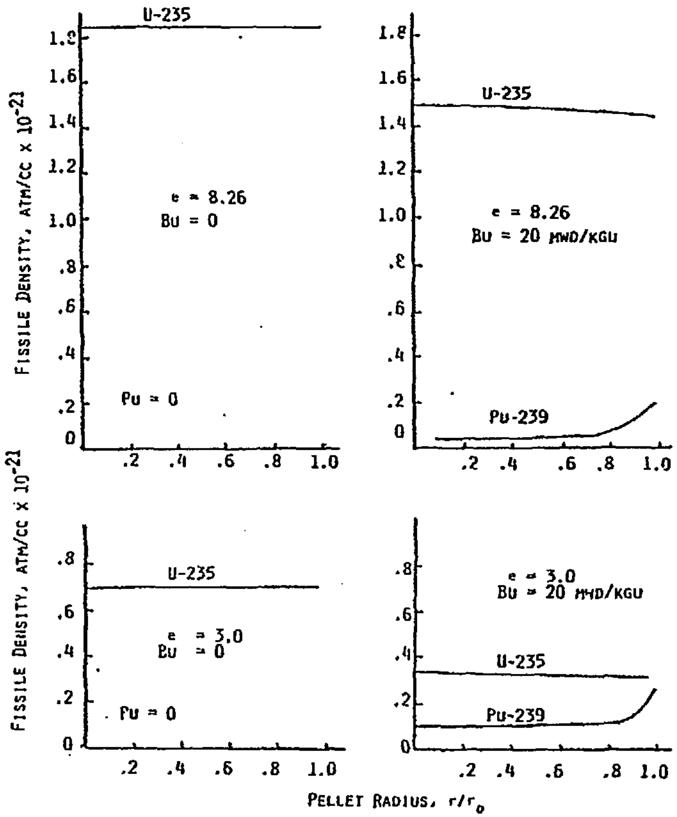

Figure 6
Pellet Fissile Distributions

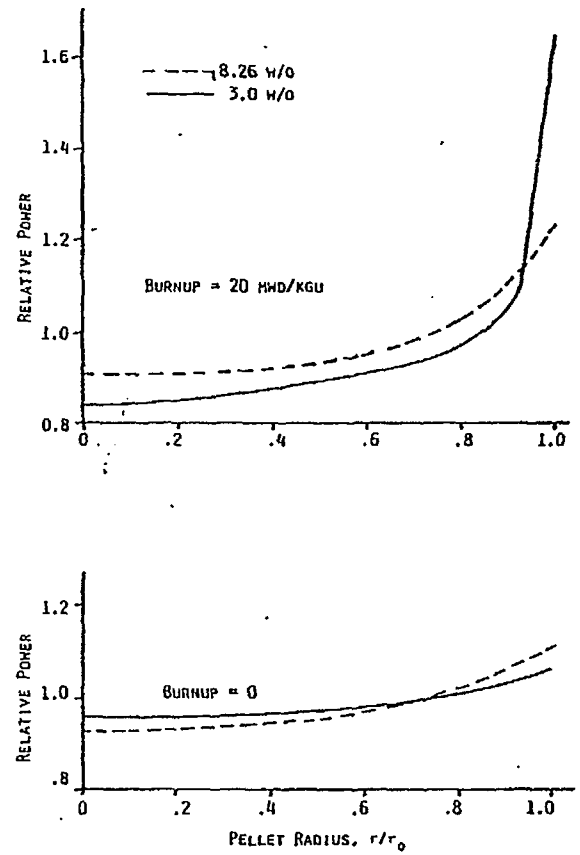

Figure 7
Pellet Power Profiles

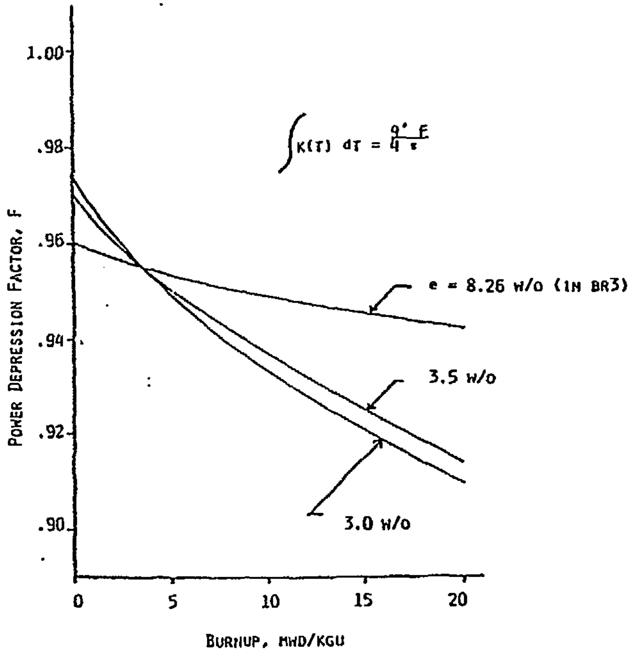

Figure 8
Power Depression Factor vs Burnup, Enrichment

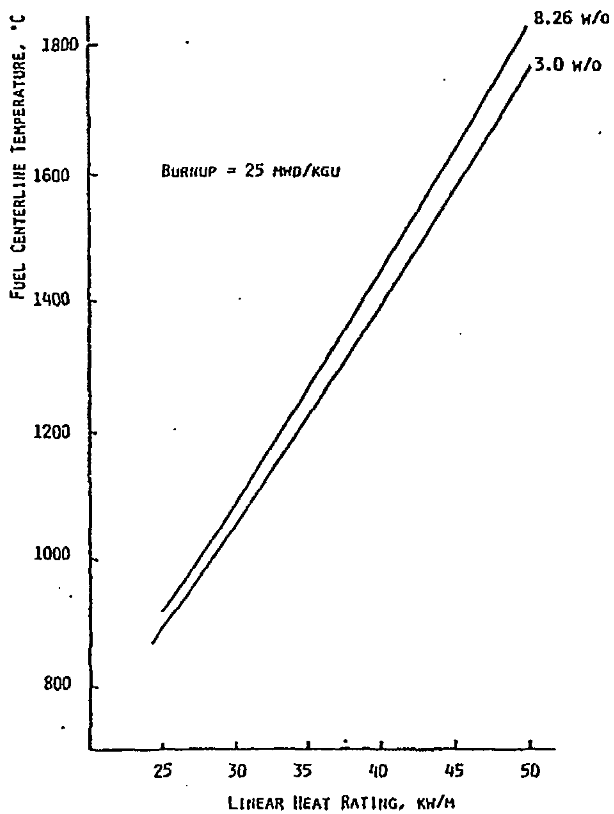

Figure 9
Fuel Centerline Temperature As A Function Of LHGR And Enrichment

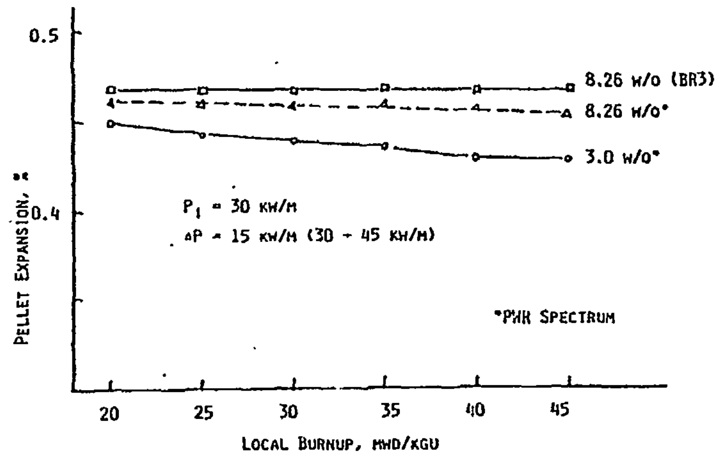

Figure 10
Pellet Expansion As A Function Of Burnup And Enrichment

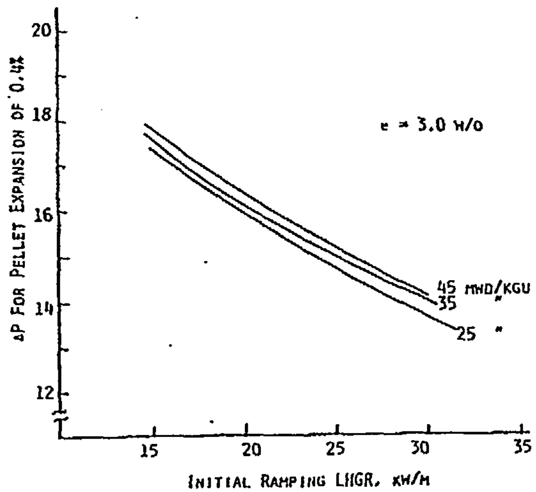

Figure 11
ΔP Required For Pellet Expansion of 0.4%