NUREG-1749

Implications From the Phenomenon Identification and Ranking Tables (PIRTs) and Suggested Research Activities for High Burnup Fuel

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

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Implications From the Phenomenon Identification and Ranking Tables (PIRTs) and Suggested Research Activities for High Burnup Fuel

Manuscript Completed:September 2001
Date Published: September 2001

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ABSTRACT

Phenomenon identification and ranking tables (PIRTs) were developed by an international group of fuel experts for three postulated accident types that are important in plant safety analysis. Rankings were determined with regard to cladding damage and fuel dispersal for high-burnup fuel rods. Developing PIRTs is a structured way of obtaining expert opinions to help improve computer codes and to conduct experimental programs related to the accidents being considered. The PIRT tables and associated information are documented in three large reports, but those reports do not contain conclusions because of the way the activity was structured. In the present report, conclusions are reached based on rankings and rationales in the PIRT documents, transcripts from the PIRT meetings, and notes. Implications of the phenomenon rankings are discussed, and methods of resolving issues related to fuel damage limits are outlined. Resolution of these issues for approved fuel types and the current burnup limit of 62 GWd/t are expected to be completed in the 2003-2005 time frame within NRC's confirmatory research program. This report was prepared by NRC's Office of Nuclear Regulatory Research and does not necessarily represent the views of any individual PIRT expert or the group as a whole.

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FOREWORD

In the design and licensing of light-water reactors, it is postulated that a small set of low-probability accidents will occur, and it is required that the reactor be able to accommodate or mitigate their consequences without affecting the public health and safety. The most severe in this set of postulated accidents in terms of challenging both the reactor and its associated systems is the large-break loss-of-coolant accident. Small-break loss-of-coolant accidents are also postulated. The characteristics of these accidents serve to set the requirements for a number of the reactor's safety systems, including the emergency core cooling system and the design of the containment.

In addition to the loss-of-coolant accidents, the other important class of postulated accidents has been the reactivity accidents. These include PWR rod-ejection accidents, BWR rod-drop accidents, and BWR power oscillations without scram. In these accidents, energy is deposited in the fuel and causes rapid heating that may damage the fuel if the power burst is sufficiently energetic. Consideration of reactivity accidents has led to fast-acting reactor control systems as well as reactor core designs with inherently negative power and void coefficients.

In the mid 1990s, the NRC learned that regulatory criteria, which have been used to ensure benign behavior of these accidents, might not be adequate at high burnups. Further, there were questions at least in principle about the effect on these criteria of new cladding alloys being introduced by the industry. Faced with these concerns, the NRC took several actions to make sure that reactor safety is maintained, that public confidence is not eroded, and that no unnecessary regulatory burden is imposed.

One of the actions was the initiation of research programs to investigate the effects of high burnup and new cladding alloys. To ensure that these research programs were well planned and to get insights on resolving related issues, the NRC sought the advice of a large number of experts. This was done in the form of a structured elicitation process that was used to develop phenomenon identification and ranking tables (PIRTs) for the postulated accidents mentioned above. The PIRT information was then used to make sure that NRC's research programs, which were addressing the burnup and alloy issues, were well planned. Four reports collectively describe the results of this expert elicitation and the implications of the information received for follow-on NRC fuel research. The following is one of those reports, and this report makes reference to the others.

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1 INTRODUCTION

From August 1999 through October 2000, eight meetings were held with a group of experts on fuel behavior to develop phenomenon identification and ranking tables (PIRTs). That group was not constituted as a Federal Advisory Committee and therefore was unable to provide a consensus on any particular issue or to give the NRC advice from the group as a whole. Nevertheless, opinions and conclusions of individual members of the group were expressed and are recorded in three large reports that will be referred to here as the PIRT reports.[1-3] In the present report, conclusions are reached based on rankings and rationales in the PIRT reports, transcripts from the PIRT meetings, and notes. Implications of the phenomenon rankings are discussed, and methods of resolving issues related to fuel damage limits are outlined. This report was prepared by NRC's Office of Nuclear Regulatory Research and does not necessarily represent the views of any individual PIRT expert or the group as a whole. The background of this PIRT activity is described briefly below.

In 1994, NRC staff learned of data on high-burnup fuel that challenged one of the fuel damage limits that NRC had been using for years.[4] That situation then raised questions about the adequacy of other fuel damage limits used by NRC when they are applied to high-burnup fuel. Questions were also raised about the adequacy of computer codes that are used to demonstrate compliance with these limits for high-burnup fuel. Attention was later focused on three specific accident types for which NRC uses fuel damage limits to ensure that coolable core geometry is maintained.[5] Those accident types are (1) PWR rod-ejection accidents, (2) BWR power oscillations without scram, and (3) PWR and BWR loss-of-coolant accidents.

A technique had been in use since 1988 to aid in computer code development for accident analysis by identifying phenomena that occur during an accident and ranking them in importance in tabular form. These phenomenon identification and ranking tables (PIRTs) became a structured way of obtaining expert opinions to help improve computer codes. In 1999, it was decided to try to use this technique to not only address computer codes but to also address other research that is needed for the fuel damage limits themselves. For the fuel damage limits, the technique would be used to address tests and experiments that generated data on which the limits were based.

About two dozen experts were invited to develop the high-burnup PIRTs. Some experts were hired by NRC, some were sent from U.S. utilities and manufacturers, some came from the Electric Power Research Institute, and some came from foreign agencies that had been in contact with the NRC staff and the Advisory Committee on Reactor Safeguards. The three accident types mentioned above were addressed separately by the PIRT experts, and in each case the phenomena were separated into four groups, (A) plant transient analysis, (B) integral testing, (C) transient fuel rod analysis, and (D) separate-effect testing.

To develop a PIRT for a given accident sequence, all phenomena are identified that affect a certain outcome of, for example, a plant transient analysis (Category A) and the importance of each phenomenon is ranked with regard to that particular outcome. The outcome, which has to be specified, becomes the primary evaluation criterion for the PIRT. For the accidents being considered, there are two guiding principles with regard to outcomes, which are stated in terms of fuel damage limits in NRC's regulations. General Design Criterion 28 (GDC-28) states that reactivity accidents should neither (a) damage the reactor pressure boundary beyond limited local yielding nor (b) significantly impair the capability to cool the core.[6] GDC-35 states that the emergency core cooling system should function during a loss-of-coolant accident (LOCA) such that (c) fuel damage that could interfere with core cooling is prevented and (d) the reaction between cladding (metal) and water is limited to negligible amounts.[7]

GDC-28 for reactivity accidents is considered to be satisfied, according to Regulatory Guide 1.77, if the radially averaged fuel enthalpy at the peak location in the core does not exceed 280 cal/g.[8] From the original supporting data for this limit, it is seen that pressure pulses that could threaten the reactor boundary are generated by hot fuel particles that are expelled from the fuel rods and rapidly transfer heat to the water. It is also seen that coolable geometry of the fuel becomes impaired when the cladding fragments and fuel rods lose their rod-like geometry.

GDC-35 for LOCAs is considered to be satisfied, according to 10 CFR 50.46, if (1) the peak cladding temperature does not exceed 2200°F (1204°C), (2) cladding oxidation does not exceed 17% of the wall thickness, (3) total hydrogen generated does not exceed 1% of that possible by oxidizing all of the cladding in the core, (4) changes in core geometry (flow reductions due to ballooning) are such that the core remains coolable, and (5) long-term cooling is maintained.[9] The cladding temperature limit and the oxidation limit (1 and 2 above) are referred to as cladding embrittlement criteria. These criteria are used to ensure that the fuel rod will not fragment into several pieces with loss of fuel particles as this could interfere with core cooling.

The PIRT panel had a lengthy discussion about adapting the general design criteria to serve as primary evaluation criteria for the PIRTs. Difficulties were seen with this approach. In general, we have neither the calculational ability nor the experimental data to quantify the generation of pressure pulses or to describe the conditions under which dispersed fuel particles become uncoolable. Thus conditions for damaging pressure pulses and uncoolable core geometries could not be expected as outcomes of attainable analyses or experiments.

Instead, the PIRT panel utilized a more conservative approach that did not appear to introduce undue conservatisms. For reactivity accidents (relatively low cladding temperature), the primary evaluation criterion for the PIRTs was taken to be cladding failure with significant fuel dispersal. For LOCAs (relatively high cladding temperature), the criterion was taken to be cladding fragmentation. It seemed clear that if there was no fuel dispersal and no cladding fragmentation, there would be no damaging pressure pulses and no loss of coolability; the general design criteria would be satisfied. Impairment of cooling due to ballooning and flow blockage was not emphasized by the PIRT panel. Cladding failure, fuel dispersal, and cladding fragmentation are amenable to analysis and testing, so these are practical outcomes that the PIRT panel could deal with.

Further, the measure of cladding failure with significant fuel dispersal for reactivity accidents was still thought of as fuel enthalpy (but not the value in Regulatory Guide 1.77), and the measure of cladding fragmentation (embrittlement) for LOCAs was still thought of as peak cladding temperature and cladding oxidation (but not necessarily the values in 10 CFR 50.46). By keeping these measures in mind, it was easier to have specific discussions and to attempt to stay close to the current licensing philosophy.

Finally, it should be mentioned that the PIRT panel adopted a very liberal definition of a phenomenon in their attempt to address codes and tests in a practical way. Just about anything that might affect the outcome of a calculation or an experiment was included in the ranking tables and called a phenomenon.

2 PWR ROD EJECTION ACCIDENTS

The specific accident considered was a postulated rod-ejection accident from a hot zero power condition in Three Mile Island, Unit 1 (TMI-1).[1] The fuel was UO_2 in Zircaloy-4 cladding in assemblies with a 15x15 configuration. The affected fuel assemblies were assumed to have a burnup of 62 GWd/t (average in the peak rod) because this is the current burnup limit imposed by NRC. During the phenomenon identification and ranking, PIRT panel members also considered the applicability of their rankings to (a) plutonium-bearing mixed-oxide (MOX) fuel, (b) other cladding alloys like ZIRLO and M5, (c) different PWR reactor types, and (d) extended burnups up to 75 GWd/t, a level being considered by the industry. The only significant effects noted were six phenomena that would have been ranked higher in importance for MOX fuel, and those were mostly related to plutonium inhomogenieties.

A qualitative plot of fuel rod power and cladding temperature during this accident is shown in Fig. 1. The power pulse is prompt and very large with the peak power being hundreds of times

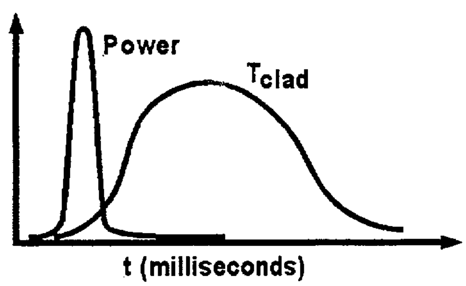

Fig. 1 Qualitative plot of fuel rod power and cladding temperature for a PWR rod-ejection accident

normal reactor power, but the pulse is very narrow with a width of about 10-30 msec at half maximum. Rapid heatup of the UO_2 fuel pellet gives rise to strong negative Doppler feedback that reduces reactor power to a low level. Heat transfer occurs from the pellets to the cladding such that the cladding also heats up, but cladding heatup is somewhat delayed. At hot zero power, the cladding is around 300°C and very little cladding temperature rise occurs before the expanding pellets exert their maximum stress on the cladding. For Zircaloy cladding with significant corrosion (oxidation), the cladding is somewhat embrittled by hydrides in the outer rim and cladding failure can occur at a relatively low fuel enthalpy by a pellet-cladding mechanical interaction (PCMI). The low corrosion niobium-bearing alloys have more ductility at low temperatures and may fail by a LOCA-like mechanism (see below) after surviving the PCMI. Such failures would occur at a later time at a higher cladding temperature and a higher fuel enthalpy.

2.1 Implications of the Phenomenon Rankings

Based on the phenomenon rankings in Tables 3-1 through 3-4 in the PIRT report (Sect. 3 of Ref. 1), the rationales given in Appendices A-D of that report, transcripts from the PIRT meetings, and notes, the following observations can be made.

2.1.1 Plant Transient Analysis

Phenomena in this category were ranked in relation to the question "Is the code-calculated outcome sensitive to this input parameter or model?" in relation to cladding failure with significant fuel dispersal. The following stand out as having high importance.

- Ejected control rod worth
- Fuel cycle design
- Pin peaking factors
- Fuel temperature feedback
- Delayed neutron fraction
- Heat capacities of fuel and cladding

Code developers need to make sure that these phenomena are represented well in their plant transient codes. The PIRT panel members indicated that these phenomena are well known, so it should be straight forward to make any code improvements that are needed. However, further observations can be made that will be useful in resolving the issues for rod-ejection accidents.

The last three items in the list are fundamental properties over which one has no control, but the first three in the list are properties of the core that the designer can alter. These three are closely related and can be thought of together as core design. There are at least two implications of this observation. First, core design changes could be made to reduce the energy deposited in the fuel if necessary to remain under a regulatory limit. While this may be costly in terms of fuel cycle length, it is at least possible to control the inherent reactivity of the core. Second, in a safety assessment, the worth of the ejected rod and perhaps other core design parameters may have utility as a substitute for a fuel enthalpy limit because of their high importance. For example, suppose it was found that you had to have a control rod worth greater than, say, $2 to reach the enthalpy limit of, say, 100 cal/g. Then it might be advantageous to use the $2 rod worth as a limit for plant assessment This potential substitution is the basis for Step 4 (Sec. 2.2.4 below), which may avoid the need for individual 3-D plant transient calculations.

There is one phenomenon, not shown in this list, that is often thought to be very important, but in fact was not ranked as highly important. That is the rate of reactivity insertion. This would be related to the rate of ejection of the control rod. The rate of reactivity insertion was considered to be relatively unimportant as long as the reactivity was inserted within a few hundred milliseconds, i.e., before the UO_2 heated up and broadened the resonance absorptions (Doppler effect). Thus, knowing the exact speed of ejection of the control rod is not necessary, and this suggests that results may not be especially sensitive to the details of the particular reactivity transient.

2.1.2 Integral Experimental Testing

Phenomena in this category were ranked in relation to their effect on the outcome of pulse reactor tests with regard to cladding failure with significant fuel dispersal. The list of phenomena was long and only those of high importance that have major implications for Step 1 (Sect. 2.2.1 below) are shown in the following list. Others address more specific aspects of selecting test rods and performing the tests, and they will of course be of value to the experimenter.

- Burnup of test rod
- Hydrogen distribution in cladding of test rod
- Agglomerates in test rod (MOX only)
- Coolant heat transfer conditions during the test
- Pulse width during the test

Burnup, as a separate parameter from oxidation, was considered to be of high importance for the following reasons.

> "The behavior of high burnup fuel is determined by the condition of both the fuel and the cladding. Although some processes occurring in fuel may saturate at a moderate burnup, others may not saturate but continue to grow with burnup. The database for high burnup fuel is significantly smaller than for low and moderate burnups. Therefore, it is important to select fuel rod specimens that have burnup levels representative of the levels that will occur in plants."

This suggests that testing should be done in the burnup range of interest rather than using lower burnup data and assuming that oxidation-related phenomena alone will determine the outcome. This result shows the experts' reluctance to accept extrapolation. Although burnup was rated high in importance, the power history while accumulating the burnup was considered to be of only medium importance. Thus there should be a wide latitude in selecting test specimens with acceptable power histories as long as the burnup and oxidation-related parameters (next item) are adequately covered.

Hydrogen absorption is a consequence of the corrosion (oxidation) process. Hydrogen distribution, or hydride distribution, was thought to be more important than average concentration. This distribution would be related to any spalling, pellet gaps, and temperature gradient across the cladding during normal power operation. It would be important to not disturb that distribution in preparation and preconditioning of test specimens.

For MOX fuel, the selection of rods with Pu-rich agglomerates was considered to be of high importance. Local Pu agglomerates produce locally high fission rates. This in turn produces locally high burnups and locally high temperatures. The locally high temperatures produce locally enhanced fission gas release (bubbles on grain boundaries). Thus gas expansion for this type of fuel is different than for uniform UO_2 and produces a different loading on the cladding. This finding confirms the need to test MOX fuel rods.

Two rationales were given for the high ranking of coolant heat transfer conditions during the test. One is related to the fact that the primary evaluation criterion goes beyond simple cladding failure to include significant fuel dispersal. Coolant conditions become very important when examining conditions beyond failure and including dispersal and pressure pulse generation. The other is that as much as 25% of the total deposited energy can be conducted out to the coolant, again leading to the conclusion that coolant conditions are important. Such heat transfer would affect cladding temperatures and hence the mechanical properties of the cladding. These rationales suggest that testing in a pressurized water loop should be performed rather than relying entirely on results from the currently available low pressure sodium loop and stagnant water capsules.

Pulse width was considered to be of high importance in conducting tests because results from Cabri are believed to have shown such an effect. Nevertheless, the exact shape of the pulse was considered to be of only medium importance as long as the integrated power has the same energy content as the ideal Gaussian-like pulse. This conclusion suggests that an irregularly shaped pulse, which may result from artificially broadening or narrowing the natural pulse of a test reactor, should not be a problem.

Finally, the experts did not believe the particular cladding alloy to be very important. All cladding alloys are principally zirconium with about a percent of tin and/or niobium added to improve the strength. It is believed that some differences in test results might be found because of different mechanical properties of the various alloys, but that characterization of the mechanical properties with burnup would allow extrapolation of the behavior to the other alloys. According to this finding, testing could be switched from Zircaloy to newer cladding materials like ZIRLO and M5 if no changes are made in the fuel pellets, and the results can be added to the existing Zircaloy data base with minor adjustments as contemplated in Steps 2 and 3 (Sect. 2.2.2 and 2.2.3 below).

2.1.3 Transient Fuel Rod Analysis

As with plant transient analysis, phenomena in this category were ranked in relation to the question "Is the code-calculated outcome sensitive to this input parameter or model?" in relation to cladding failure with significant fuel dispersal. This category of phenomena addresses fuel rod code improvement and validation (see Step 3, Sect. 2.2.3 below). High ranked input phenomena such as gap size, power distribution, and condition of oxidation (spalling) seem rather obvious. Some of the rankings for the analytical models were not so obvious, however.

Pellet-cladding contact (gap closure) models are clearly important, yet current models may have been derived to optimize temperature predictions rather than the mechanical loading. This might indicate the need for model improvements in current codes.

The stress-strain response of the cladding was considered to be of high importance as would be expected. However, strain rate effects, anisotropy, and biaxiality were considered to be of medium to low importance based on known results and the availability of adequate models. Adequate stress-strain data for cladding on high-burnup fuel rods are not currently available and are the subject of several ongoing research programs.

The mechanical properties of fuel pellets were considered to be of only moderate importance in relation to the loading applied to the cladding. The importance would be higher in regard to fragmentation and dispersal after cladding failure. This suggests that relatively simple models for pellet deformation may be adequate.

2.1.4 Separate Effect Testing

At the time the PIRT discussions were taking place for the PWR rod-ejection accident, all of the attention for this group of phenomena was given to the low-temperature mechanical properties of the cladding because those are needed to model the pellet-cladding mechanical interaction (PCMI). It was later recognized that the typical PCMI failures observed in Zircaloy cladding might not occur in alloys with niobium because of their low corrosion (and assumed high ductility) during normal operation. In those cases, cladding temperature would increase and high-temperature failure mechanisms would be similar to those that occur during a LOCA. It was agreed that separate effect testing for high-temperature behavior would be considered during discussions of LOCAs (see Sect. 4.1.4 below).

This category of phenomena for separate-effect testing was thus limited to mechanical properties testing. Phenomena in this category were ranked for specimen selection and test conditions, and the rankings did not contain surprises. For specimen selection, the rankings again emphasize the condition of the cladding oxide (spalling or delamination) and the hydrogen distribution rather than just the amount of oxygen or hydrogen. For test conditions, the rankings emphasize temperature, stress state imposed on the specimen, tensile specimen design, and burst specimen design, confirming the emphasis being given in NRC's program at Argonne National Laboratory.

2.2 A Method to Resolve Issues for Rod-Ejection Accidents

The above observations and implications from the PIRTs suggest a path for resolving issues associated with rod-ejection accidents. Since the current burnup limit is 62 GWd/t, the following discussion will be limited to that burnup. Similar or related methods could be chosen by the industry to address burnup extensions from 62 GWd/t up to, say, 75 GWd/t, but that choice will be left to the industry. To follow this path for resolution up to 62 GWd/t, some additional data would be needed to establish new regulatory limits for fuel damage and validated analytical tools to ensure that under accident conditions these limits will not be exceeded. This path is outlined below.

2.2.1 Step 1

An empirical correlation can be determined for fuel enthalpy increase as a function of cladding oxide thickness for a wide range of oxide thicknesses and a range of fuel burnup to ~62 GWd/t. This correlation would bound the fuel enthalpies that can be experienced with reasonable assurance that no fuel dispersal will occur. The correlation would be at some enthalpy increment above that associated with cladding failure if adequate data exist to justify such an increment; otherwise, the correlation would bound the fuel enthalpies that correspond to no cladding failure.

Figure 2 shows the current data base. The numbers associated with points on the plot refer to tables in Ref. 10, and all tests performed since the publication of Ref. 10 are explicitly labeled on the plot. The most useful testing to date has been performed with fuel having Zircaloy cladding and burnups up to 65 GWd/t (specimen burnup). No tests have been performed with ZIRLO-clad fuel; one rather mild test was performed with M5-clad fuel with relatively low corrosion; and the numerous tests with Russian VVER-clad fuel (similar cladding composition to M5) have all had very low corrosion and sufficient ductility to avoid PCMI cladding failure. Therefore, the initial correlation that would be developed would be for PCMI failure of Zircaloy-clad fuel. Adjustments for ZIRLO and M5 can be made as described in Step 2. Examination of Fig. 2 reveals the need for some additional test data for the following reasons:

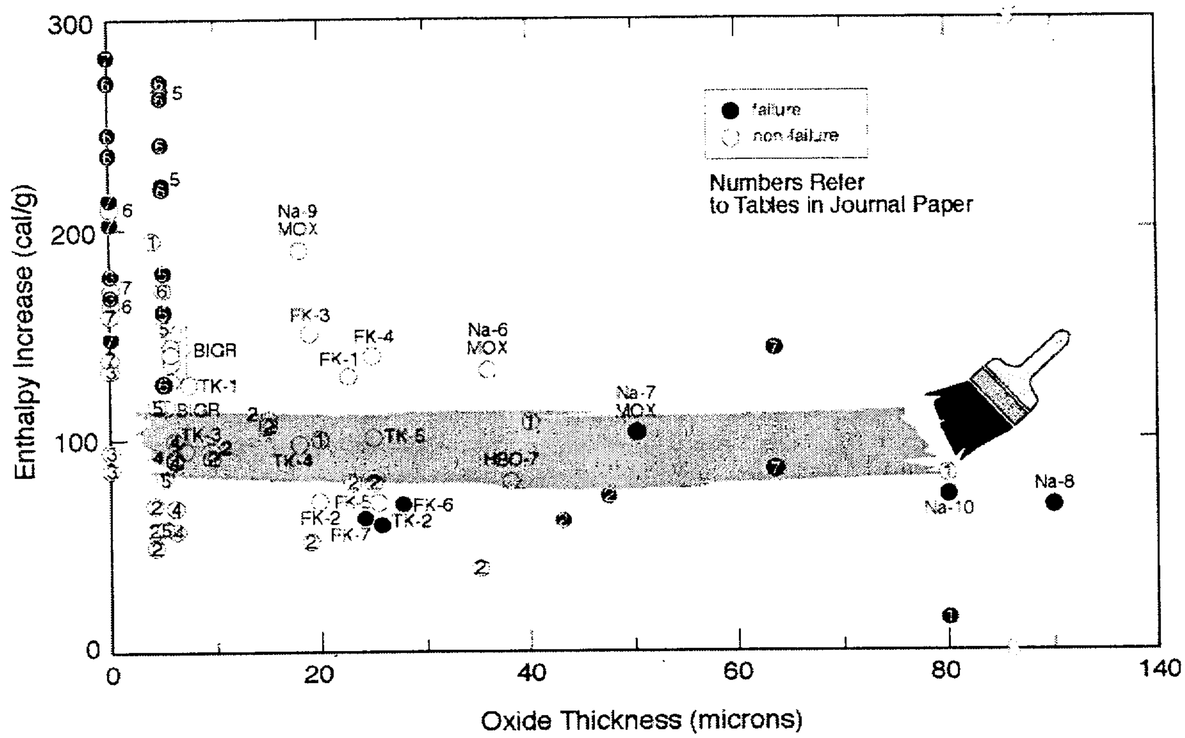

Fig. 2 Fuel enthalpy as a function of oxide thickness for tests described in Ref. 10 (solid symbols indicate cladding failure; open symbols indicate no failure)

(i) Cabri tests, REP-Na8 and REP-Na10 suggest that pulse width, perhaps in combination with spalling or some other factors, caused the earlier test, REP-Na1, to fail at an abnormally low fuel enthalpy. There is other evidence that REP-Na1's unusual pre-conditioning caused some hydride redistribution that resulted in embrittlement.[11] Although the latter is disputed, it is likely that REP-Na1 will be disregarded as non typical. A couple of additional broad-pulse data points from Cabri with heavily oxidized cladding (Zircaloy, ZIRLO, or M5 as long as they are heavily oxidized) would increase confidence in disregarding REP-Na1.

(ii) Cabri tests, REP-Na1 (dispersal), REP-Na8 (no dispersal), and REP-Na10 (no dispersal) taken together suggest that fuel dispersal will not occur from failed cladding unless the fuel enthalpy is at least 30 cal/g above the cladding failure level or, perhaps, the pulse width is less than 30 msec. This is a very limited data set for such an important conclusion and, in fact, PIRT panel members have altered their views recently on the causes fuel dispersal. Additional data and understanding would be needed to select an enthalpy limit that was greater than the cladding failure threshold.

(iii) Cladding failures observed in NSRR tests generally occurred at lower enthalpy values than in Cabri and it is believed that the main cause of this is the atypically low temperature and hence lower cladding ductility in the NSRR tests (20 °C vs. 300 °C for the event of interest). Several tests in the proposed high-temperature capsule for NSRR could provide a basis for adjusting upwards the NSRR data points on the plot. If increased reliance can be placed on NSRR data points on this plot, the quality of the correlation will improve significantly. Also, these additional tests in NSRR may reveal more details about dynamic fission gas expansion and its role in fuel response at extended burnups if the adjusted NSRR failure enthalpies remain below those from Cabri with its broader pulse width.

(iv) Additional testing in a water loop such as being constructed for Cabri could be done to see if all important phenomena have been accounted for in the earlier tests that had less typical environments. It is likely that additional cladding heatup will be experienced compared with the sodium loop and this should result in failure at higher energies and hence show larger margins. Tests in a water loop could be considered to be confirmatory and might lead to further modifications (increases) in the regulatory damage limits.

(v) Finally, at low oxidation levels with new corrosion resistant fuel, sufficient ductility may be present in the cladding to survive the PCMI. In these cases, failure would not occur until a DNB-related high temperature mechanism became active. The threshold for DNB is believed to be in the range of 60-115 cal/g fuel enthalpy change for a rapid power pulse.[10] Failure would be at an even higher level, but the effects of DNB can only be reproduced accurately in a water loop. The tests in a water loop that were mentioned in (iv) would show whether the PCMI picture and related enthalpy limits are altered for any of the new cladding types.

2.2.2 Step 2

The use of advanced cladding alloys like ZIRLO and M5 are an integral part of the industry's strategy to go to high burnups; therefore, the correlation outlined in Step 1 must be adjusted to apply to these alloys as well. Conversely, should tests be performed with ZIRLO and M5 cladding, those test results would need to be adjusted to contribute to the Zircaloy correlation described above.

It is likely that one can analytically reconcile data for Zircaloy, ZIRLO, and M5 based on the relative ductility of these alloys. Therefore, mechanical properties (especially uniform and total elongation) are needed for this purpose and they must be measured for all of these cladding types under conditions representative of a rod-ejection accident. Such measurements on irradiated cladding can be made in NRC's program at Argonne National Laboratory with continued cooperation from the industry. These mechanical properties will be needed to do the calculations described below.

2.2.3 Step 3

For an increase in ductility of, say, ZIRLO compared with Zircaloy, calculations could be done with NRC's transient fuel rod code, FRAPTRAN, to determine the corresponding increase in fuel enthalpy that could be tolerated before cladding failure occurred. Suppose one finds from testing that some measure of ductility (an index to be defined for this purpose) of ZIRLO is 0.5% greater than that of Zircaloy under the same test conditions (burnup, oxidation, etc.). One would then calculate the cladding strain corresponding to the empirically determined failure enthalpy for Zircaloy, and in another calculation for ZIRLO one would look for the enthalpy corresponding to a somewhat larger cladding strain for which the ductility index was 0.5% greater for the ZIRLO case (details to be worked out later). The resulting difference in enthalpy values would correspond to the difference in ductilities.

To have confidence in these calculations, the code needs to be validated with integral test data from Cabri and NSRR, and the code must incorporate appropriate mechanical properties. Hence FRAPTRAN code improvement and validation for rod-ejection transients would be an important element of the process.

2.2.4 Step 4

Finally, to apply the empirical fuel enthalpy correlation to a plant safety analysis requires the use of a plant transient analysis code. In the case of NRC's confirmatory assessment up to 62 GWd/t burnup, that code would be the PARCS 3-D kinetics code coupled with the TRAC thermal-hydraulics code. The PARCS-TRAC code would therefore have to be validated for rod-ejection transient analysis at high burnup.

Although it is recognized that high-burnup fuel, during a reactivity accident, would not achieve as high a fuel enthalpy as low-burnup fuel, it is nevertheless reasonable to assess what could be achieved considering that assembly locations can vary. Accordingly, NRC would need to perform a confirmatory assessment at 62 GWd/t that is applicable to all currently licensed PWRs, so a range of core design parameters would have to be encompassed. To avoid doing 3-D calculations for each present and future core design, some bounding scheme of parameters could be developed. There is the potential that one could associate a control rod worth, and perhaps some other related core design parameters, with the fuel enthalpy limit based on generic calculations. This idea comes from the PIRT rankings discussed above in Sec. 2.1.1 (Plant Transient Analysis). Then, particular core designs could be compared with that control rod worth and its related parameters to determine if the enthalpy limit would be met without doing a specific 3-D plant transient calculation for each core. Further research would be needed to confirm this as a viable option. Plant-specific 3-D analyses could always be done if needed in certain cases.

2.3 Summary and Schedule for Resolution of Issues for Rod-Ejection Accidents

Cores can be designed and operated to limit the worth of ejected rods and, therefore, realistic criteria can be found to prevent fuel dispersal and thereby ensure that reactivity accidents do not degrade into core-melt events. A realistic fuel enthalpy criterion is not yet available because of limitations of the present data base, mainly in Cabri and NSRR. Nevertheless, tests in Cabri's sodium loop and in NSRR's stagnant water capsules can probably give a satisfactory enthalpy criterion in the burnup range covered by the tests, but the criterion may have to be conservative because of the limited cladding temperature rise and the absence of interactions with water. Those results should be confirmed in a more typical water loop.

Two final tests in Cabri's sodium loop are scheduled for 2002 and then there will be a major delay while the reactor is upgraded and the water loop is installed. A high-temperature capsule is also being constructed for NSRR and several tests might be available by the end of 2002 or

2003. Therefore, a new plateau of information should be reached in 2003, and that presents an opportunity to attempt resolution of the issues for rod-ejection accidents for fuel burnups to 62 GWd/t.

The final two tests in Cabri's sodium loop will be conducted with a ZIRLO-clad fuel rod and an M5-clad fuel rod. Results from these tests will require scaling between those two cladding types and Zircaloy to incorporate the two data points in the Zircaloy data base (and to apply the resulting fuel enthalpy correlation to ZIRLO and M5). To accomplish this, NRC's FRAPTRAN code will have to demonstrate its capability to analyze rod-ejection accidents, and mechanical properties data will have to be available for high-burnup ZIRLO-clad fuel and M5-clad fuel. The recently completed FRAPTRAN validation and peer review included Cabri and NSRR tests in its assessment, so FRAPTRAN should be ready to perform these calculations. Mechanical properties of the ZIRLO and M5 cladding from the Cabri test rods are expected to be measured in the Cabri program, so those data should be available in 2002.

Finally, to cover all core designs in operating reactors, it may be possible to convert the enthalpy criterion to some limiting core parameters, especially control rod worth. This would permit a simple screening of core design parameters in operating reactors to determine compliance with the enthalpy criterion. The development of screening criteria (i.e., the limiting core parameters) is currently being explored with the PARCS code at Brookhaven National Laboratory, and some measure of success is expected. If this turns out not to be satisfactory, a selected set of core designs would be analyzed by NRC to hopefully confirm that the peak fuel enthalpy achievable does not exceed the criterion. This can probably be completed by early 2003.

3 BWR POWER OSCILLATIONS WITHOUT SCRAM

The specific accident considered by the PIRT panel was postulated for LaSalle, Unit 2, which is a BWR/5.[2] LaSalle-2 has fuel bundles of the 8x8 type, and the fuel has a cladding of Zircaloy-2 with a soft zirconium liner. The core was assumed to contain fuel with burnups up to 62 GWd/t (average in the peak rod). During the phenomenon identification and ranking, PIRT panel members also considered the applicability of their rankings to (a) fuel bundles with different geometries (e.g., 9x9 or 10x10), (b) fuel manufactured by different vendors, (c) different BWRs (e.g., BWR/2 through BWR/6), and (d) extended burnups up to 75 GWd/t. Few of the phenomenon rankings were affected, and those that were affected would have little effect on interpreting the results.

The plant was assumed to be operating at 84% power when both recirculation pumps tripped and there was a failure to scram. Core flow decreased and power decreased rapidly to about 40% power, and a few minutes later flow oscillations began. Because there were changes in the void fraction during the flow oscillations, there were also power oscillations. A qualitative plot of fuel rod power and cladding temperature during these oscillations is shown in Fig. 3.

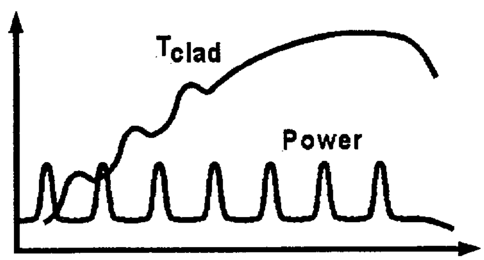

Fig. 3 Qualitative plot of fuel rod power and cladding temperature for BWR power oscillations without scram

The power pulses are relatively small, with a peak power on the order of ten times normal reactor power, and the pulses are relatively wide with a width of about 300 msec at half maximum. The time between pulses is about 2 sec, and this is too short for all the heat to be transferred out of the fuel. Therefore, fuel and cladding temperatures increase with each pulse until some terminal condition is achieved or the oscillations are terminated. During each pulse, there will be a pellet-cladding mechanical interaction (PCMI) when the pellets expand against the cladding, but the cladding temperature and hence cladding ductility will increase with each pulse. It is not known whether the repeated pulses would cause PCMI failure of the cladding or if failure would ultimately occur by a high temperature LOCA-like mechanism.

3.1 Implications of the Phenomenon Rankings

The PIRT panel's discussions of Category A (plant transient analysis) and D (separate effect testing) were straight forward and addressed calculations and tests of a type that are routinely performed. The discussions of Category B (integral experimental testing) and C (transient fuel rod analysis), however, explored new territory and were very creative. To complete PIRT tables in these categories, the experts had to logically think through the detailed fuel response in a way that had not been done before. From these PIRT discussions, the following picture emerged.

BWR cladding during the oscillations could in principle experience (a) PCMI due to thermal expansion of the fuel pellets, (b) possible failure and fuel dispersal from PCMI, (c) continued heat-up of the cladding on the back side of each pulse, (d) further heat-up from subsequent pulses because the period of the pulses is less than the time constant of the fuel rod, (e) eventual dry out, (f) failure to rewet, (g) high temperature oxidation, (h) cladding melting, (i) fuel melting, and (j) fragmentation of embrittled cladding upon quenching when the transient is terminated.

From the experts' discussion (Sect. 2.3.2 of Ref. 2), it is clear that they do not expect PCMI failures during the BWR power oscillations. Temperature had just been reduced before the oscillations, tending to open the gap; the pellet temperature increase during each pulse would be small; and kinetic gas expansion effects would be small or non-existent because of the slower nature of the BWR power pulses. This expectation is seen again in the rankings and rationales in Category C (Appendix C of Ref. 2). For example, hydrogen concentration and distribution, which are so important for PCMI failure, are ranked as low importance because the hydrogen would not affect the mechanism of fuel melting. Throughout the Category C rationales, you can see that the experts were focusing only on the high temperature effects leading ultimately to fuel melting.

From further discussion, it was found that the experts believe that cladding embrittlement would occur before cladding melting or fuel melting took place. Because of the potential long duration of the high temperature part of the transient, the fuel would behave in a similar manner to that during a LOCA. At high burnups, the pressure differential across the cladding would be positive because of the high fuel temperatures and high fission gas releases, although the magnitude of the pressure differential may be different from that during a LOCA. Thus ballooning would be expected, but rupture may not occur at the same temperature (and hence in the same metallurgical phase region) as during a LOCA. Nevertheless, ballooning and rupture should be calculable from the same data base being produced for LOCA analysis.

The net result is that the BWR oscillations appear to create a thermal transient that could have limits set on oxidation and cladding temperature just as for a LOCA (i.e., 2200°F peak cladding temperature and 17% equivalent cladding oxidation, with any suitable modifications for high burnup or different alloys).

There was a question whether oxidation would become so rapid during this event that there would be a runaway temperature escalation. During a LOCA high-temperature transient, the only source of power (other than heat from the oxidation reaction) is decay heat such that the cladding could remain at a relatively stable high temperature for a relatively long time. During the BWR oscillations, however, the average power level remains around 40% of full reactor power and rapid temperature escalation might be possible. Several points were made to suggest otherwise. Runaway conditions would be determined by heat balances, and in this event there would be a lot of steam present to remove heat from the fuel. Experimenters have measured the oxidation rate in steam at temperatures all the way up to cladding melting without experiencing runaway temperature escalations. A peak cladding temperature limit of 2200°F (1204°C) is well below the oxide phase transformation to a cubic structure, where a sharp increase in the rate of reaction is observed. Thus it was concluded that there would not be a runaway situation. It can be noted, though, that NRC's experimental program to investigate oxidation kinetics for high-burnup fuel could provide further confirmation for this conclusion.

Although the experts spent a lot of time exploring experimental conditions in Category B for the low temperature portion of the oscillations, those postulated tests can be thought of as tests to confirm that PCMI does not contribute to cladding failure with significant fuel dispersal during BWR oscillations.

3.2 A Method to Resolve Issues for BWR Power Oscillations without Scram

NRC has performed a few calculations but has not yet done any significant research to address the response of fuel (high burnup or low burnup) to BWR oscillations.[12] Therefore, a plan to resolve related issues will include activities in all categories to improve computer codes for analyzing this event and to determine appropriate fuel damage limits. Following those activities, NRC could undertake a series of parametric calculations to, hopefully, demonstrate that currently licensed BWRs would remain below appropriate fuel damage limits in the event of power oscillations without scram for burnups up to 62 GWd/t. Particular elements of a plan could be as follows.

3.2.1 Category A (Plant Transient Analysis)

NRC's new fully 3-dimensional code, TRAC-M coupled with PARCS, could be readied for these BWR calculations. As indicated by the PIRT experts, improvements might be needed in models for subcooled boiling, dry out, film boiling, rewet, bypass void fraction due to direct moderator heating, core flow blockage (and gap conductance). As soon as this code could give preliminary results, those results should be provided to assist efforts in the other categories. Eventually, the plant transient codes would be used to perform parametric plant calculations to provide a basis for resolving the issues.

3.2.2 Category B (Integral Testing — Low Temperature)

It should be technically feasible to perform tests in the Nuclear Safety Research Reactor (NSRR, JAERI, Japan) to try to confirm that PCMI failures would not occur as a result of small repeated power pulses that would take place during BWR oscillations. Since it was thought that such tests would not result in failure, conservative test conditions could be used as long as they also showed no failures. A test that would probably be sufficient to investigate the mechanical response of the cladding would be a series of several pulses (4.5 msec wide) with ~15 cal/g fuel enthalpy rise at intervals of hours or days with the capsule temperature at its design temperature of ~300°C. Broader pulses, shorter intervals, and escalating temperatures would be more prototypic, but the conditions just described would be possible within the ongoing NSRR program and should be adequate.

3.2.3 Category B (Integral Testing — High Temperature)

It should be technically feasible to perform tests in the Halden reactor (Norway) to determine if LOCA criteria would prevent cladding failure with significant fuel dispersal. An enveloping cladding temperature history could be produced by varying power in the rod to achieve dry out and subsequent temperature escalation. These integral tests could be run up to the LOCA cladding temperature and oxidation limits to see if the fuel rod maintains structural integrity and retains its fuel upon termination of the transient. This test could also indicate any tendency for rapid temperature escalation below the temperature limit.

The high-temperature testing envisioned by the PIRT experts could also be performed with electrically heated bundles in a loop to make sure that code predictions of dry out and rewetting are adequate. Review of existing data will be required to determine if the available data are sufficient.

3.2.4 Category C (Transient Fuel Rod Analysis)

FRAPTRAN can be used for these calculations. Most of the code areas of importance for fuel rod analysis during BWR oscillations are areas of importance for all transients (thermal expansion, gap closure, heat resistances, heat transfer coefficients, etc.). Because FRAPTRAN was derived from earlier codes that were originally developed for LOCA analysis, it is especially well suited for this analysis.

However, at the present time FRAPTRAN is a single-rod code and not a single-channel code, so time-varying dryout and rewet conditions of the oscillations cannot currently be handled. Cooperative work on FRAPTRAN is being done by the Finnish Center for Radiation and Nuclear Safety (STUK) to incorporate thermal-hydraulic models by merging their GENFLO code with NRC's FRAPTRAN code. This will result in a single-channel version of FRAPTRAN that should be able to handle this transient.

3.2.5 Category D (Separate Effect Testing)

A full range of mechanical properties is being measured at Argonne National Laboratory with Limerick BWR fuel rods at high burnup and with similar archive tubing. This should be sufficient for the mechanical part of the analysis with FRAPTRAN. Ballooning, rupture, and oxidation kinetics measurements under LOCA conditions will also be made on the Limerick cladding specimens, and these should be directly applicable to the BWR power oscillations. The oxidation kinetics measurements should also provide additional information on any tendency (or lack thereof) for rapid temperature escalations.

3.3 Summary and Schedule for Resolution of Issues for BWR Power Oscillations

BWR power oscillations without scram appear to provide a greater challenge to the fuel during the high-temperature excursion after dryout rather than during the low-temperature mechanical interaction with the pellets (PCMI). The absence of PCMI failure can be confirmed with one or two repeated-pulse tests in NSRR. Such tests are being considered by JAERI and could be run in the 2002-2003 time fame, depending on JAERI priorities. Tests of the high-temperature excursion can be performed in the Halden reactor to confirm that LOCA-type criteria are sufficient to limit fuel damage and to get further information on dryout and rewet conditions. Such tests are being discussed by the Halden Program Group. Conceptual elements of a Halden test might be defined in 2001, followed by construction of test apparatus that would take at least a year. Testing might be done in the 2003-2004 time frame.

Nevertheless, analysis is the most important aspect of resolving the issues related to BWR power oscillations. The plant transient code and the fuel rod code are going to have to describe dryout and rewet (and the eventual failure to rewet) during the oscillations such that cladding oxidation and temperature can be calculated. This may require significant upgrade of dryout and rewet models for oxidized fuel cladding. Given good heat transfer information, however, calculations with the combined TRAC-M and PARCS plant-system code should be straight forward. Calculations of local oxidation and cladding temperature with FRAPTRAN also require further developments. The initial FRAPTRAN code was completed in 2001, but it does not contain cladding-to-coolant heat transfer models. Cooperative work with the Radiation and Nuclear Safety Authority (STUK) in Finland appears to offer the improvements that are needed, and that work could be completed by about the end of 2002. Peak cladding temperatures and total cladding oxidation could then be compared with the same criteria that will be used for LOCA analysis (see next section). Overall resolution, with plant calculations and confirmatory reactor testing, could be completed in the 2004-2005 time frame.

4 LOSS-OF-COOLANT ACCIDENTS

No specific plant or break size was postulated for the PIRT rankings although presentations were made on a PWR large-break LOCA (double-ended guillotine break in a cold leg), a PWR small-break LOCA (three-inch break), and a BWR LOCA (double-ended break in the suction side of the recirculation line).[3] Again, the fuel was assumed to have Zircaloy cladding and burnups to 62 GWd/t (average in the peak rod). PIRT panel members also considered the applicability of their rankings to (a) fuel array, pellet design, and MOX, (b) cladding type and manufacture, (c) reactor type, and (d) extended burnup to 75 GWd/t. Some effects were noted, but no interesting trends were revealed.

Reactor power drops quickly during this accident when the coolant (moderator) is lost, but the fuel pellets have stored heat because of their heat capacity and radionuclide decay continues to provide an additional heat source. Consequently, the cladding temperature increases with time and the fuel pellet temperature decreases with time as the fuel and cladding temperatures tend to equilibrate. A qualitative plot of fuel rod power and cladding temperature is shown in Fig. 4.

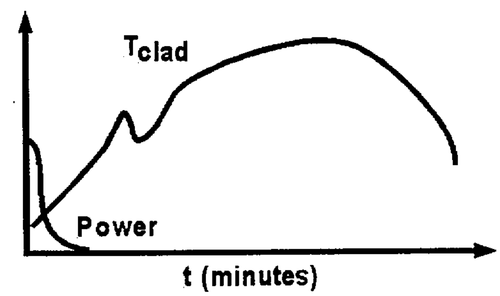

Fig. 4. Qualitative plot of fuel rod power and cladding temperature for a loss-of-coolant accident

As the cladding temperature reaches about 800°C, ballooning of the cladding will take place, but the ballooning process becomes unstable and rupture follows quickly. Following rod burst, the cladding temperature continues to rise to as high as 1200°C, and most of the cladding oxidation will take place at the higher temperatures. During this ascent in temperature, pellet fragments can move into the ballooned region and increase the heat source in that region. At the end of the high temperature period, cooldown and quenching will occur. Because of reductions in ductility during the oxidation process, the thermal shock during quenching may fragment the cladding, or other mechanical loads may fragment the cladding after it has been quenched.

4.1 Implications of the Phenomenon Rankings

At the outset, resolution of the LOCA issues appeared to be a straight forward matter of determining if the fuel damage limits in 10 CFR 50.46 and the evaluation models in Appendix K remain valid for high-burnup fuel. NRC had well developed programs underway before the PIRT elicitations were initiated to experimentally test the fuel damage limits and to modify NRC's computer codes. It was expected that the PIRT outcome would confirm the course of action already being taken, but the result was not exactly as expected.

4.1.1 Plant Transient Analysis

Two observations can be made from the assessment of the PIRT rankings for plant transient analysis, Category A (Sect. 3.4.1 of Ref. 3). First, there are a small number of fuel-related models in the plant transient codes that need to be scrutinized because they are thought to be of high importance and are not well understood. These are (a) gas pressure, (b) rod free volume, (c) cladding temperature, (d) burst criteria, (e) location of burst and blockage, and (f) time-dependent gap-size heat transfer. This result was expected. Second, there are thermal-hydraulic models in the Category-A PIRT table that have nothing to do with high-burnup fuel that were ranked as highly important and not well understood. These include (a) film boiling over a wide void fraction, (b) rewet, (c) rod-to-spacer-grid thermal-hydraulic interaction, and (d) spacer-grid rewetting and droplet breakup.

4.1.2 Integral Experimental Testing

The rankings in Category B (Sect. 3.4.2 of Ref. 3) on integral testing highlighted important and poorly known phenomena that were, for the most part, already well recognized. These included oxidation plateau temperature, quench rate, fuel versus no fuel in the test specimen, fuel relocation into the ballooned section, and cladding chemistry (oxygen and hydrogen content). It is noteworthy that fuel relocation had an importance ranking as high as any and the lowest knowledge ratio of all the highly ranked phenomena. It is also noteworthy that fuel relocation received a rather low importance ranking for plant transient analysis and for transient fuel rod analysis, reflecting different views of the two subgroups of experts.

4.1.3 Transient Fuel Rod Analysis

In the PIRT rankings in Category C (Sect. 3.4.3 of Ref. 3) on transient fuel rod analysis, only a small number of transient fuel-related phenomena were identified as being of high importance and poor understanding. These were heat resistances in the gap, heat resistances in the oxide, cladding oxidation magnitude, size of burst opening, burst criteria, and time of burst. However, a large number of input parameters such as initial gap size, initial gas pressure, etc. were ranked as highly important and poorly known. This indicates that attention should be given to steady-state codes such as FRAPCON in addition to addressing the transient codes such as FRAPTRAN. Further, there were transient thermal-hydraulic models that were considered to be of high importance and not well known just as there were such thermal-hydraulic models identified in the plant transient codes.

4.1.4 Separate Effect Testing

The experts discussions of Category D on separate effect testing were very creative and produced some surprises. First, the experts conducted a brainstorming session to identify candidate separate-effect tests that could help address regulatory criteria and models, and then they proceeded to rank the tests according to their importance. Thus, before any phenomenon ranking was done, the postulated tests themselves were ranked with the following result.

1. High temperature cladding oxidation
2. Quench behavior (after oxidation)
3. Phase relations of cladding (before and after oxidation)

4. Mechanical properties (before and after oxidation)
5. Seismic Response (after oxidation)
6. Fuel relocation into the ballooned region

During the discussion of experimental testing, one of the experts reviewed the history of the derivation of LOCA criteria in the U.S.[13] This review made it clear that NRC's embrittlement criteria (2200° F and 17%) were in fact based on post-quench ductility tests by Hobson rather than on integral tests with ballooning, rupture, oxidation, and quenching. Hobson's ring-compression tests were used to obtain the criteria whereas the integral tests were used for confirmation. The commission and the staff had been reluctant to rely on integral tests alone because of the lack of assurance that mechanical constraints in the tests were representative of constraints imposed by fuel assembly spacer grids and neighboring rods. Therefore, the commission chose to require non-zero ductility at a relatively low temperature (275°F) after the Zircaloy cladding had been oxidized at a high temperature (2200°F), and this requirement produced the embrittlement criteria from Hobson's data.

As a result of this discussion, the PIRT experts added a group of tests on mechanical properties at low temperature (after oxidation) to the original mechanical properties in the above list. The original group of tests were high temperature tests (before oxidation) that addressed ballooning and rupture. Thus, the post-quench ductility tests ended up at a lower priority in the above list than the quench tests. This appears to be the wrong relative priority for these two tests and resulted from the evolution of the discussions and the failure to go back and re-prioritize the entire list.

Because a number of phenomena were ranked for each of the six postulated tests, the ranking tables contain a lot of detailed information that will not be summarized here, but can be studied by experimenters. It is of interest, though, to look at the importance ranking of one phenomenon that is common to all of those tests and that is the alloy type in the selection of specimens for the tests.

Alloy type was considered to be very important (Importance Ratio >75) for four of the six tests. For quench tests, the Importance Ratio was only 60 as the result of a single vote for low importance on the basis that data show no significant impact of alloy type on quench test results. Since very few quench tests have been performed on high-burnup fuel and the effects of axial constraints cannot be readily assessed, it is likely that this ranking is too low. For fuel relocation tests, the Importance Ratio for alloy type was 60 and the discussion focused on the likelihood of different balloon sizes for different alloys. It seems clear that the alloy type would not directly affect a fuel relocation test, given the same size of ballooning deformation for different alloy specimens. The overall implication is that alloy type should have a strong effect on the outcome of all of the tests listed above except for the fuel relocation test, and for that test the alloy type would have an indirect effect on the outcome if it affected the magnitude of ballooning deformation. This implication suggests that it is important to test all significantly different cladding types such as Zircaloy, ZIRLO, and M5, which were the examples of different alloy types discussed by the experts.

4.2 A Method to Resolve Issues for Loss-of-Coolant Accidents

NRC has in place a well developed experimental program of integral and separate-effect tests at Argonne National Laboratory with cooperation from EPRI. The transient fuel rod code, FRAPTRAN, is also under development at Pacific Northwest National Laboratory for application to high-burnup fuel. Modifications to these programs will be made based on the results of this PIRT. Further, NRC's plant transient codes could be improved in the areas indicated by these PIRT rankings.

4.2.1 Category A (Plant Transient Analysis)

NRC's TRAC-M code is currently being developed and improved with new models, particularly for rewet at high temperatures, which was one of the highly ranked, poorly known phenomena identified in this PIRT. Related tests are being planned in the Rod Bundle Heat Transfer Facility at The Pennsylvania State University. The rather large number of thermal-hydraulic models indicated for further attention in the PIRT can be incorporated in the current plans for TRAC-M improvements. The fuel-related models needing further attention can also be addressed in the ongoing code improvement effort.

4.2.2 Category B (Integral Testing)

The phenomena that were indicated for additional consideration in the PIRT report had already been recognized during the planning for the integral LOCA criteria test in NRC's program at Argonne National Laboratory, so no significant modification of that test is needed. However, the emphasis on fuel relocation in the PIRT discussions has re-focused our attention on that subject.

First, it is noted that NRC has an existing Generic Issue (GI #92) called Fuel Crumbling During LOCA.[14] That generic issue was evaluated in 1984 and given a low priority ranking. This meant that there was insufficient risk-based justification for starting a major re-review of existing safety analyses. However, the assessment that resulted in the low priority ranking was based on an assumption that coolable geometry was maintained even though the cladding temperature rose above 2200°F. Because 2200°F is the temperature above which coolable geometry is not assumed in most safety analyses, this priority ranking may be incorrect.

Second, the Halden Reactor Project is planning a LOCA test that would look at unexpected fuel behavior, channel blockage, azimuthal thermal gradients, the effects of the collapse of fuel columns and the consequent problems associated with cooling, and the effects of cladding axial constraint on quenching strength. The NRC staff had not taken an active part in planning for that test partly because the perceived focus on fuel relocation was seen as a low priority. Results of the PIRT ranking and the review of NRC's generic safety issue alter this perception, and NRC staff are now actively participating in the final planning for the proposed Halden test.

4.2.3 Category C (Transient Fuel Rod Analysis)

The NRC's FRAPTRAN code is an evolution of previous codes that were developed principally for LOCA analysis. Therefore, FRAPTRAN is well suited for LOCA analysis and has recently been upgraded for high-burnup applications. The detailed insights from the PIRT ranking will be used by the FRAPTRAN code developers at Pacific Northwest National Laboratory to adjust their code improvement program. Special attention will be given to model changes that might be needed to account for axial variations in heat generation due to relocation of fuel particles into the ballooned sections. PIRT rankings relevant to the NRC's steady-state fuel rod code, FRAPCON, will also be used by the code developers at PNNL.

4.2.4 Category D (Separate Effect Testing)

Separate-effect testing related to LOCA is included in NRC's program at Argonne National Laboratory and in foreign national programs to which NRC has access. While most of the results of the PIRT ranking in this category confirm the elements of those programs, there are several results that suggest further actions

First, no special-effect testing has been planned on fuel relocation at ANL. However, relocation was ranked high for integral tests (Sect. 4.1.2), lower for analysis (also Sect. 4.1.2), and lowest of six important separate-effects tests. In light of this variable PIRT result, NRC staff will pay close attention to possible fuel relocation in the integral LOCA tests to be conducted at ANL and in the LOCA tests that are planned in the Halden reactor. Results from those tests will be examined before making any plans for separate-effect testing on fuel relocation, which may or may not be needed.

Second, the reminder that current LOCA criteria were based primarily on Hobson's ductility measurements will have a significant impact on the test program at ANL. Post-quench ductility tests are being added to the ANL test matrix and these will include ring-compression tests like Hobson's. It may be possible, however, to perform more precise tests than Hobson's ring-compression tests, and other tests are also being considered for this purpose. Such tests will be run at 275°F after high-temperature oxidation, just as Hobson's tests were done. Consideration will be given to switching to these separate-effect tests for primary assessment of the LOCA criteria for high-burnup fuel and making the currently planned LOCA criteria integral test a confirmatory test as was done originally.

Finally, there is a strong indication from the PIRT rankings that LOCA-related testing at ANL should be extended to ZIRLO and M5 alloys; only Zircaloy cladding is being tested at the present time. However, if the primary means of assessing LOCA regulatory criteria becomes a post-quench ductility test, as discussed above, then the integral test being performed for Zircaloy-clad fuel might be skipped altogether. Because the ductility test requires only short specimens of de-fueled cladding and a simple test procedure, whereas the integral test requires long specimens of fueled cladding and a very complicated procedure (heatup, ballooning, burst, and quench), testing of the other alloys could be cheaper and quicker.

4.3 Summary and Schedule for Resolution of Issues for Loss-of-Coolant Accidents

After the PIRT discussions were concluded and the draft PIRT reports were written, some older reports were discovered on post-quench ductility of Alloy E-110, the Zr-1%Nb cladding used in Russian VVERs.[15,16] Since Framatome's M5 cladding, which was recently introduced in the U.S., has the same nominal Zr-1%Nb composition, those older reports took on a significance that was not recognized before the PIRT activity. The reports show that E-110 loses its ductility after about 6% cladding oxidation in contrast to the 17% oxidation number that is used for

Zircaloy and is a limiting value in NRC regulations.[9] Nevertheless, limited results presented very recently by Framatome show M5 post-quench ductility values that are like those for Zircaloy rather than E-110, and Westinghouse has presented data that show similar Zircaloy-like results for ZIRLO, which also contains ~1% Nb.[15] No mechanistic understanding has yet been reached as to why the E-110 and M5 results are so different when the alloy compositions are so similar, so it would be desirable to conduct systematic tests on these alloys in the same laboratory to understand this situation. Such measurements on unirradiated cladding specimens could be made in NRC's program at ANL within the present year, 2001, provided material is made available for testing.

Testing of high-burnup BWR fuel with Zircaloy-2 cladding (with a liner) is underway at ANL at this time, and testing of PWR fuel with Zircaloy-4 cladding will begin soon. A full range of tests will be performed including tests related to embrittlement criteria (10 CFR 50.46), tests related to evaluation models (Appendix K), and tests to measure the mechanical properties. Testing of fuel rods with Zircaloy-2 cladding should be completed in 2002. Approximately two years will be required for equivalent testing for each of the PWR cladding types, Zircaloy-4, ZIRLO, and M5, although some parallel testing may be possible. At the present time, fuel specimens with ZIRLO and M5 cladding types are not available at the laboratory, but efforts are underway to obtain such fuel.

Code developments mentioned above can easily be made within two years, so analytical capability should not be limiting with regard to resolving these LOCA issues. Some of the code activities are underway at this time.

Integral testing in the Halden reactor with emphasis on fuel relocation are scheduled to be completed in 2002 or 2003. Two realistic tests are planned. One with a very high burnup rod (low end-of-life power), but with a peak cladding temperature just sufficient to ensure ballooning and rupture. The other with a mid-burnup rod (time of maximum power for a high-burnup design) at a peak cladding temperature of about 1200°C. Basic design of the test rig was done in 2000 and specimen acquisition is taking place in 2001. Testing might be done in 2002 depending on completion of test rig construction and reactor scheduling.

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202	U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-1749
2. TITLE AND SUBTITLE Implications From the Phenomenon Identification and Ranking Tables (PIRTs) and Suggested Research Activities for High Burnup Fuel		3. DATE REPORT PUBLISHED MONTH: September YEAR: 2001 4. FIN OR GRANT NUMBER
5. AUTHOR(S) R.O. Meyer		6. TYPE OF REPORT Technical 7. PERIOD COVERED *(Inclusive Dates)* 9/99 to 6/01

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

Phenomenon identification and ranking tables (PIRTs) were developed by an international group of fuel experts for three postulated accident types that are important in plant safety analysis. Rankings were determined with regard to cladding damage and fuel dispersal for high-burnup fuel rods. Developing PIRTs is a structured way of obtaining expert opinions to help improve computer codes and to conduct experimental programs related to the accidents being considered. The PIRT tables and associated information are documented in three large reports, but those reports do not contain conclusions because of the way the activity was structured. In the present report, conclusions are reached based on rankings and rationales in the PIRT documents, transcripts from the PIRT meetings, and notes. Implications of the phenomenon rankings are discussed, and methods of resolving issues related to fuel damage limits are outlined. Resolution of these issues for approved fuel types and the current burnup limit of 62 GWd/t are expected to be completed in the 2003-2005 time frame within NRC's confirmatory research program. This report was prepared by NRC's Office of Nuclear Regulatory Research and does not necessarily represent the views of any individual PIRT expert or the group as a whole.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*		
cladding	power oscillations	13. AVAILABILITY STATEMENT: unlimited
control-rod ejection accident	primary evaluation criterion	14. SECURITY CLASSIFICATION *(This Page)* unclassified
expert elicitation		*(This Report)* unclassified
LOCA		15. NUMBER OF PAGES
		16. PRICE

NRC FORM 335 (2-89)

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