

AGENCY PROGRAM PLAN FOR HIGH-BURNUP FUEL

Introduction

A goal of the NRC's Strategic Plan is to prevent radiation-related deaths and illnesses from civilian nuclear reactors by avoiding reactor accidents in which substantial damage is done to the reactor core. This is accomplished in part with a number of fuel damage criteria that serve as reactor "speed limits" to prevent postulated events from developing into severe accidents. Fuel damage criteria are the focus of this program plan.

In the 1970s when most of these criteria and related analytical methods (computer codes) were being established, high burnup was thought to occur around 40 GWd/t (average for the peak rod). Data out to that burnup had been included in data bases for criteria, codes, and regulatory decisions, and it was believed that some extrapolation in burnup could be made. By the mid 1980s, however, unique changes in pellet microstructure had been observed from vendor and international data at higher burnups along with increases in the rate of cladding corrosion (breakaway oxidation). It thus became clear that something new was happening at high burnups and that continued extrapolation of transient data from the low-burnup data base was not appropriate. Thus, on October 4, 1993, NRR issued a formal request to RES for assistance on high-burnup fuels, and that request initiated the first NRC research in this area in more than a decade.

The NRC's research that has been completed since that time and is planned for the near future will be described in this plan in the context of confirming previous decisions to permit fuel burnups in licensed reactors up to 62 GWd/t (average for the peak rod). Future approvals for extensions in burnup above the present limit will require additional research and analysis by the industry, and a licensing and research strategy for such approvals is outlined in the final section of this plan. The agency's Strategic Plan encourages the industry to develop codes, standards, and guides that can be endorsed by the NRC and carried out by the industry. This method of addressing extended burnup limits is incorporated in the discussion of the licensing strategy.

The Strategic Plan also incorporates an approach to focus on regulated activities that pose the greatest risk to the public. Therefore, a risk perspective has been developed and is applied for each of the issues described below.

Risk Perspective

During the past year, a list of potential high-burnup issues has been identified based on observed operational problems, experimental results from test programs, and an understanding of basic phenomena. This list is given in Table 1. and was discussed with the Commission on March 25, 1997.

Table 1. List of Potential High-Burnup Issues

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1. Cladding Integrity and Fuel Design Limits
 2. Control Rod Insertion Problems
 3. Criteria and Analysis for Reactivity Accidents
 4. Criteria and Analysis for Loss-of-Coolant Accidents
 5. Criteria and Analysis for BWR Power Oscillations (ATWS)
 6. Fuel Rod & Neutronic Computer Codes for Analysis
 7. Source Term and Core Melt Progression
 8. Transportation and Dry Storage
 9. High Enrichments (>5%)
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To help determine which issues warrant greater efforts for resolution, risk concepts have been employed. Of course, consideration of compliance and defense in depth also affect this determination, and a balance will be seen in later sections that discuss each issue.

In general, a reactor event sequence does not produce significant risk unless fuel melting and its resulting large fission product release are possible during the sequence. Therefore, the issue of cladding integrity and associated fuel design limits for normal operation (including anticipated operational occurrences), which relate to compliance with General Design Criterion 10 (GDC-10), are not significant from a risk point of view and are not included in probabilistic risk assessments (PRAs).

Control rod insertion (scram) must be capable of preventing fuel damage during normal operation including anticipated transients (GDC-26). Nevertheless, one class of events considered in regulation assumes that scram does not occur (anticipated transients without scram, ATWS), and that class of events is addressed separately below. Control rod insertion in combination with other reactivity control systems must also be sufficient to ensure coolable core geometry for postulated accidents (GDC-27). Thus, the risk associated with the failure of control rods to insert is significant, and control rod insertion also has strong compliance and defense-in-depth implications.

The large reactivity-initiated accidents (RIAs) have the potential to produce unacceptable fuel damage. These are the rod drop accident in a BWR and the rod ejection accident in a PWR. Probabilistic profiles have been developed at Brookhaven National Laboratory for these RIAs.

Risk assessments are not available because RIAs have very low probabilities of occurrence and have been considered outside the range of interest for modern PRAs.

An early NRC study (1975) examined the probability of a rod drop accident in a BWR that would result in an energy deposition of more than 280 cal/g (unacceptable fuel damage). That study was reexamined in 1986 and reaffirmed. If adjustments are made to that study to obtain the probability that a rod drop will result in an energy deposition of only 100 cal/g (a damage limit that might be more appropriate for high-burnup fuel), the resulting probability is $\sim 10^{-7}$. For reference, any generic safety issue with an associated core damage frequency of 10^{-7} or less would be dropped from further consideration by the NRC using a prioritization scheme based on principles of the Commission's safety goals.

Past studies are not available on failure probabilities for PWR rod ejection accidents, and no failures have occurred in control rod drive mechanisms in over 2400 reactor years of operation world wide; therefore, an estimate has been made. From this observation, it is estimated that the failure probability is no larger than $\sim 2 \times 10^{-4}$ per reactor year, which is consistent with estimates of the frequency of pipe breaks based on mechanistic models. It is further assumed that only half of the rods that could be ejected would result in prompt criticality, which is then assumed to result in unacceptable fuel damage. Further, prompt criticality is expected to happen only when the reactor is at hot zero power, which is less than 1% of the time. Combining these factors leads to an estimate of $\sim 10^{-6}$ per reactor year. This value is just within the range of interest for generic issue consideration.

Loss-of-coolant accidents also have the potential to cause unacceptable fuel damage and they have been studied extensively in recent PRAs. From the PRA data base, which was constructed from licensees' individual plant examinations, core damage frequencies are seen to have the following ranges (Table 2).

Table 2. Core Damage Frequencies per Reactor Year

Large Break LOCA	Small & Medium Break LOCA
PWR 3×10^{-5} to 1×10^{-7}	PWR 5×10^{-5} to 1×10^{-7}
BWR 2×10^{-6} to 1×10^{-9}	BWR 9×10^{-6} to 1×10^{-9}

Loss-of-coolant accidents are significant risk contributors in PRAs, and these frequency ranges are seen to be within the range of consideration for generic safety issues.

Finally, power oscillations in a BWR ATWS also have the potential to cause unacceptable fuel damage. Current licensing analyses show that fuel enthalpy remains below 280 cal/g and thus conclude that coolable geometry is maintained and there is no fuel dispersal. There is, however, concern that the 280 cal/g value may be too high, especially for high-burnup fuel. Probabilistic analyses do not exist with a lower value, but an upper bound on the probability of exceeding such a value would be the probability of occurrence of the ATWS in the first place. This probability is $\sim 1 \times 10^{-5}$. In general, ATWS events are found to comprise about 15% of the total risk for BWRs.

Plans to Resolve the Issues

In the following sections, each of the issues identified in Table 1 will be discussed. Risk perspectives will be discussed, along with compliance and other considerations, to help determine appropriate regulatory actions and research efforts. A near-term assessment of each issue will be described to show why, in some cases, it is satisfactory to wait 3-5 years for research results in order to achieve a more final resolution. Where applicable, related NRC research programs will be described along with their schedules. And finally, the expected final resolution of each issue will be outlined. Schedules for research programs and overall resource requirements associated with this program plan are given in the Attachment.

While not explicitly discussed below, it should be noted that the NRC staff's activities to address these issues involve significant external interactions. Much of the research is now done in cooperative programs. Some of these, like the Halden Project, are international projects in which we participate as a member. Others, like the collaboration with France, Japan, and Russia on reactivity accidents, are arranged with bilateral agreements. In other cases, EPRI and DOE participate in NRC research projects with memoranda of understanding. Technical discussions are maintained with the nuclear industry through daily regulatory activities, the Regulatory Information Conference, the Water Reactor Safety Information Meeting, ACRS meetings, and other special workshops.

1. Cladding Integrity and Fuel Design Limits

(a) Description of Issue

General Design Criterion 10 states the principle that specified acceptable fuel design limits (SAFDLs) should be met to assure that integrity is maintained in the first barrier for retention of fission products -- the fuel rod cladding -- during normal operation and anticipated operational occurrences. That is, cladding defects (also simply called cladding failures) should not occur under those conditions. The following list identifies some of the SAFDLs that are described in the Standard Review Plan (principally in SRP 4.2).

- Stress limits
- Fatigue lifetime
- Rod pressure limits
- Hold down spring capability
- Pellet moisture limits
- Uniform cladding strain less than 1%
- No pellet centerline melting
- Mechanical loads less than 90% of irradiated yield stress
- Avoidance of critical heat flux (limits on Departure from Nucleate Boiling Ratio, DNBR, and Minimum Critical Power Ratio, MCPR)

It is likely that some of these fuel design limits would be affected by pellet microstructural changes or the reduction in cladding ductility that accompany high burnup, thus affecting cladding integrity. Recent experience suggests that cladding failures, or cladding damage that might lead to failures, have occurred as the result of achieving higher burnups or attempting to reach higher burnups.

(b) Risk Perspective

There is no significant contribution to plant risk from cladding failures during normal operation and anticipated operational occurrences because only small fission product releases are possible without core melting. However, it appears that high burnup fuel design has contributed to an increase in the severity of fuel cladding failures that have occurred with gross fuel release during normal operation and during fuel handling. The attendant fuel particle contamination within and outside the plant are a safety concern.

(c) Near-Term Assessment

See Final Resolution.

(d) Related NRC Research

None.

(e) Final Resolution

Although burnup-related problems have occurred, there has been, nevertheless, an overall trend during normal operation of improved fuel performance in the past fifteen years. Figure 1 shows this improving trend in the number of fuel assemblies containing fuel rods with defects, normalized per giga Watt of generated electricity, for the period from 1980 to 1996. Figures 2 and 3 show the actual number of individual fuel rod failures during the last few years, and it can be seen from the totals for the most recent years that this average is about 1-2 fuel rods per reactor per year. Compared with the number of

fuel rods in a typical core (~50,000), this defect rate is very low.

Reactor coolant cleanup systems are designed to accommodate on the order of 1% fuel failures (about 500 rods), and off-site dose limits can be met at that level. Nevertheless, the intent of GDC-10 is to maintain the first barrier to fission product release (defense in depth), which means near-zero failures. From the discussion above, it is clear that the goal of near-zero failures is being met. NRR will continue to monitor industry performance in this area and assess the significance of fuel failures, trends, and root causes.

2. Control Rod Insertion Problems

(a) Description of Issue

In late 1995 and early 1996, several control rods failed to insert fully during scrams at two PWRs (South Texas and Wolf Creek). All of the affected control rods were positioned in high-burnup fuel assemblies. Upon inspection of the rods and fuel assemblies, the control rods were found to be in good condition, but the fuel assemblies were deformed. Related evidence was found in North Anna and at a number of plants of similar design in Europe.

(b) Risk Perspective

While the incomplete rod insertion events at Wolf Creek and South Texas resulted in loss of only a small amount of shutdown margin, these events could be precursors of events with more serious consequences. Review of the Wolf Creek data indicated that thimble tube distortion high in the core had the potential for control rods sticking at those locations. Continued operation under these conditions could have resulted in loss of significant shutdown margin.

(c) Near-Term Assessment

Very shortly after these incidents occurred, the staff issued Bulletin 96-01 requesting special actions to ensure compliance with the current licensing basis for the facilities with respect to shutdown margin and control rod drop times. Those actions included additional training for operators, a review of control rod operability based on the recent events, testing of rods starting at the next appropriate shutdown, and review of scram data for anomalous indications.

(d) Related NRC Research

None.

(e) Final Resolution

The root cause for the observed cases of control rod sticking was determined by Westinghouse to be the fuel assemblies' response at high burnup to several aspects of fuel design including creep, oxide thickness, operating temperature, holddown spring stiffness, thimble tube thickness, and dashpot dimensions. To verify the safety of operating plants, the staff considered a requirement for increased control rod testing frequency if control rods were located in high-burnup fuel assemblies. A supplement to Bulletin 96-01 was drafted to require such testing, and the draft was issued for public comment. While the draft was out for comment, additional technical information was provided and the industry indicated that it was redesigning the fuel assemblies and improving core management to eliminate the problem. Thus the staff has decided that the bulletin supplement is not needed. Utility awareness has been heightened, and it is believed that utilities have the means and ample motivation to avoid this problem in the future. The staff will continue to monitor industry performance in this area, and the issue needs to be addressed by the industry in any submittal for new fuel designs or burnup extensions.

3. Criteria and Analysis for Reactivity Accidents

(a) Description of Issue

The specific accidents of concern are the rod drop accident in a BWR and the rod ejection accident in a PWR. For these postulated accidents, the NRC uses criteria to ensure that fuel rods remain coolable and that fuel particles are not dispersed into the coolant (280 cal/g peak fuel enthalpy) and to indicate the occurrence of cladding failure (DNB, MCPR, 170 cal/g peak fuel enthalpy) for the purpose of dose calculations. Tests in the French CABRI reactor in late 1993 with some highly degraded commercial fuel resulted in cladding failure at very low fuel enthalpies (~30 cal/g average for a fuel rod) and substantial fuel dispersal. Analysis of these and similar tests showed that failures were occurring in a partially brittle manner, as a result of the mechanical expansion of the pellets, rather than by dryout and overheating of the cladding as addressed by the current criteria. It thus appears that the current criteria may not achieve their purpose for high-burnup fuel.

(b) Risk Perspective

From the general discussion of risk presented above, it is seen that the frequency of occurrence of a BWR rod drop accident is below the range of interest for consideration as a generic issue whereas the frequency for a PWR rod ejection accident is just within that range. Nevertheless, their consideration is explicitly required by GDC-28. Therefore, it would seem appropriate to analyze these events more realistically (i.e., in a less conservative manner) than in previous analyses because of the low risk.

(c) Near-Term Assessment

Shortly after learning of the CABRI data in 1994, an assessment was made based on probability and power level, and the staff concluded that there was no safety concern requiring immediate regulatory action (Taylor memo to the Commissioners, 9/13/94). NRC and U.S. industry calculations, which were presented at an OECD conference (Cadache, 1995), further suggested that there would be no cladding failure (and hence no fuel dispersal) during these accidents, provided that heavily oxidized fuel with spalling is avoided.

(d) Related NRC Research

No test program of this kind has been in operation in the U.S. for over 15 years, so the NRC entered into formal agreements with France (CABRI test reactor), Japan (NSRR test reactor), and Russia (IGR test reactor) to obtain data from current programs. The NRC also initiated generic plant calculations (mentioned above) and an assessment (largely in house) of the test data and plant calculations. Results of this assessment were documented in a journal article and in Research Information Letter (No. 174, March 3, 1997). Based on those results, RES suggested tentative interim criteria shown in Table 3.

Table 3. Tentative Interim Criteria for RIAs

Oxide spalling:	none allowed
Cladding failure:	100 cal/g (enthalpy increase)
Coolability: ^(a)	280 cal/g ^(b) (enthalpy limit) <30 GWd/t
	No cladding failure >30 GWd/t

^(a)Loss of coolability is equated to fragmentation of the rod (several pieces) at low burnup and dispersal of fuel particles through cladding defects at high burnup.

^(b)There is evidence that the 280 cal/g value should be reduced to 230 cal/g, but this is not a high-burnup issue per se.

A fixed burnup limit was not given for these criteria because burnup does not seem to be the most important variable (it is oxidation). The data base for these criteria includes burnups to 64 GWd/t and oxide thicknesses to 130 microns (one test), with most oxide thicknesses below 80 microns; however, no new phenomena, like pellet microstructural changes or breakaway oxidation of the cladding, are expected just above these values. The main limitation appears to be that oxidation should not be so severe that spallation occurs because that introduces known phenomena that can cause localized embrittlement.

Although the test programs just mentioned provide valuable data for an interim assessment, these programs have also provided enough understanding of the related

phenomena to know that the current data base has substantial limitations (Table 4). As a result, there remains considerable disagreement among the international community as to what fuel enthalpy value constitutes an appropriate safety limit. In view of the data uncertainties, the staff does not believe that adopting revised criteria is appropriate at this time.

Table 4. Limitations of Current Data Base

Pulse:	too narrow in NSRR and most of CABRI tests
Cladding:	mostly obsolete varieties of Zircaloy
Coolant:	sodium in CABRI and stagnant water in NSRR
Temperature:	too low in NSRR
Pressure:	too low in CABRI and NSRR

To address these uncertainties in a cost-effective manner, the staff will participate in new programs through international agreements. In France, a new water loop will be constructed to test more current PWR cladding types with prototypical pulse widths, water as the coolant, and appropriate coolant flow to investigate cladding failure and the effects of dispersed fuel particles. In Japan, a new high-temperature, high-pressure capsule will be constructed to test more current PWR and BWR cladding types, and pulse-width effects will be cross checked with the French program.

Test schedules for NRC's participation in these RIA test programs are shown in the Appendix, and significant new results are not expected for 3-5 years. The costs of NRC's participation in these programs is highly leveraged.

(e) Final Resolution

Based on our current interpretation of the data, generic safety assessments, and the low probability of BWR rod drop and PWR rod ejection events, no reanalysis will be required for extant approvals. When significant results from the new test programs become available (3-5 years), this confirmatory assessment will be revisited and modified if necessary.

4. Criteria and Analysis for Loss-of-Coolant Accidents

(a) Description of Issue

For these postulated accidents, the NRC uses cladding embrittlement criteria (2200°F peak cladding temperature, 17% cladding oxidation) to ensure that coolable geometry is not lost; and related models must be used in safety analyses for oxidation kinetics, ballooning, rupture, and flow blockage to demonstrate that long-term cooling is

maintained. Additional analyses are performed to show that seismic and blowdown loads do not fragment the fuel or interfere with control rod insertion during and after a LOCA.

The criteria, models, and analyses being used today were based on data from unirradiated cladding, yet the burnup process will likely have an effect. High-burnup fuel rods can accumulate heavy oxide coatings during normal operation and experience some loss of ductility (embrittlement) from related hydrogen absorption. In a few cases, measured oxide levels have approached the 17% limit locally. Further, the enhanced "breakaway" oxidation that is observed in some cladding types suggests that the oxidation kinetics at high temperature would be increased. Thus it is likely that the criteria and models for LOCA analysis will be affected at high burnup, although it is not clear that high-burnup fuel will become limiting.

(b) Risk Perspective

Core damage frequencies for LOCAs are as high as $\sim 5 \times 10^{-5}$ for PWRs and $\sim 9 \times 10^{-6}$ for BWRs, and while these numbers are quite small on an absolute scale they represent significant risk contributors. The compliance issue is also strong for LOCAs as the embrittlement criteria mentioned above are written directly into the regulations (10 CFR 50.46). Thus the question of high-burnup effects on LOCAs is being given a high priority.

(c) Near-Term Assessment

Unlike the reactivity accidents, there is no evidence at this time that the criteria or analytical conclusions for LOCAs are deficient. Current embrittlement criteria, which are conservative for fresh fuel, may be adequate at high burnup provided that the initial oxide accumulation is taken into account. Preliminary tests in France indicate that this will be the case. The amount of oxidation that is predicted to occur during a LOCA transient is often small compared with the 17% limit, so the remaining margin may accommodate a large amount of initial oxidation from normal operation. Further, changes at high burnup in cladding ductility are likely to be in the favorable direction (less deformation and flow blockage with less ductility). Fuel vendor calculations also show that high-burnup fuel has lower peaking factors and is less limiting for LOCA analysis than fresh fuel, so in general there may be more margin all around to accommodate changes due to burnup.

However, in one recent case involving Westinghouse fuel rods with a burnable absorber, analysis showed that the 17% limit might be exceeded for a LOCA at some time in the future if the expected amount of oxidation at high burnup were included. Those cases are being addressed on a case-by-case basis to ensure continued compliance with regulatory requirements. An information notice is being drafted to alert all licensees of these high-burnup effects and their potential impact on the requirements of 10 CFR 50.46.

Changes may also be needed in allowable structural loads for earthquakes during and after a LOCA because the strength and ductility of high-burnup cladding will not be the same as for fresh material. Elastic analyses are usually performed and resulting loads are compared with ASME allowable values. Analyses for fresh fuel show ample margins, and the increased strength at high burnup would seem to enlarge those margins. But this method of assessment presumes that the material being analyzed is ductile, whereas substantial losses in ductility occur at high burnup for some heavily oxidized fuel rods. These effects are being addressed in the research program described below.

(d) Related NRC Research

Fuel behavior during loss-of-coolant accidents is assessed with embrittlement criteria and several types of analyses. (a) The initial stored energy in the fuel is calculated with the NRC's FRAPCON-3 code or similar vendor and licensee codes. (b) During the transient, the amount of oxidation and the peak cladding temperature are calculated for comparison with the embrittlement criteria, and the deformation of the rod (amount of ballooning) is calculated to provide related flow blockage. NRC's FRAPTRAN code can calculate these quantities, and models of these phenomena are usually built into vendors' systems codes. (c) Systems codes like the NRC's TRAC-P and TRAC-B codes calculate the entire plant transient, including the long-term cooling phase. (d) Finally, finite-element structural mechanics codes are used to calculate the fuel assembly and core response to seismic and LOCA loads.

The FRAPCON-3 and FRAPTRAN codes will be discussed in a following section (Issue 6). Plant systems codes, which describe thermal, hydraulic, and neutronic behavior of the reactor, are not in the scope of this program plan. Nevertheless, fuel-related models developed in this program will be fed into the plant systems codes as appropriate. Embrittlement criteria and fuel behavior correlations for LOCA analysis are being investigated in a program at Argonne National Laboratory (below), and fuel structural response is also being addressed in that program.

In FY97, a major program was initiated to establish a data base for LOCA criteria and models utilizing typical high-burnup fuel from U.S. power reactors. The program is being carried out in the hot cells at Argonne National Laboratory and will also provide fundamental mechanical properties, measured under temperature and rate conditions that are applicable to a wide range of postulated transients and accidents. Cooperation on obtaining and preparing fuel rods for the tests is being obtained from EPRI and DOE, and collaboration on technical matters is also being obtained from France, Japan, and Russia. A detailed test plan has been prepared for this program and is being followed. Experimental techniques are being developed at this time, and fuel rod acquisition is expected in 1998. Test schedules are shown in the Appendix.

(e) Final Resolution

Confirmation of existing criteria and models at current burnup levels, or an indication of need for revision, will be available from the new database starting around 2000.

5. Criteria and Analysis for BWR Power Oscillations (ATWS)

(a) Description of Issue

The 280 cal/g criterion that is used as a limit for reactivity accidents is also used for BWRs to demonstrate the absence of cladding fragmentation, fuel dispersal, and related phenomena during an anticipated transient without scram (ATWS) with power oscillations. Since test results for the reactivity accidents show that fuel dispersal can occur at much lower fuel enthalpies for high-burnup fuel, the 280 cal/g limit may not ensure coolable geometry for high-burnup fuel subjected to the power oscillations.

(b) Risk Perspective

From the discussion above, the probability of a BWR ATWS with power oscillations is $\sim 1 \times 10^{-5}$. Although this would be a conservative estimate of the probability of causing unacceptable fuel damage (level unknown), it is high enough that this issue should be pursued.

(c) Near-Term Assessment

The ATWS oscillation transient is much slower than the reactivity-accident pulse. More heat transfer will occur such that expansion-driven stresses on the cladding are reduced and cladding temperatures are increased, thus reducing the likelihood of cladding failure. Further, fuel fragmentation and dispersal may be reduced or eliminated without the prompt expansion of fission gases on grain boundaries. Therefore, questions about the adequacy of the 280 cal/g limit do not necessarily imply unacceptable fuel damage for such power oscillations. Because of the low probability of this event, there is no immediate safety concern, and research activities have been initiated to address this situation.

(d) Related NRC Research

BWR ATWS oscillations are being analyzed in house with the FRAPTRAN transient fuel rod behavior code. These calculations will attempt to estimate fuel enthalpies, cladding stresses, and fission gas behavior so that the fuel duty can be compared to that during a reactivity pulse transient (e.g., rod drop accident). Only approximate results can be expected at this time because NRC's fuel rod codes have historically been designed for thermal calculations rather than mechanical calculations, and such improvements in the

codes will not be available for a couple of years.

Inquiries are also being made about the possibility of performing BWR-type oscillations in test reactors in several foreign programs. There is no facility in the U.S. that could perform such tests without large startup costs. Specific research on cladding fragmentation, fuel dispersal, and fission product release is needed to support the assessment of this postulated event.

(e) Final Resolution

The final course of action will depend on the results of ongoing analyses and other factors such as interactions with the industry and international research organizations.

6. Fuel Rod & Neutronic Computer Codes for Analysis

(a) Description of Issue

NRC uses FRAPCON-3, a steady-state fuel behavior computer code, to audit similar vendor codes that are used to calculate LOCA stored energy, end-of-life rod pressure, gap activity, and to perform other licensing analyses. FRAPTRAN, a transient code, is also used by NRC for special calculations and to interpret test results. Although the vendors were using fuel codes that had been updated for high-burnup applications, at the time that reviews were being done of vendor requests to go to 62 GWd/t, NRC's codes had not been updated for about 10 years and had been validated out to only about 40 GWd/t (rod average). Thus NRC's ability to deal with high-burnup fuel issues has been hampered by outdated analytical tools.

For reactor power calculations, neither the industry nor the NRC is, as a rule, using 3-D neutronics codes. Postulated accidents like the rod ejection in a PWR, the rod drop in a BWR, and the BWR ATWS power oscillations are very localized in nature and cannot be analyzed well without 3-D kinetics codes. While some industry 3-D codes have been submitted for NRC review, most licensing codes do not have this capability or involve overly simplifying assumptions. NRC also occasionally uses its own 3-D neutronics codes for special analyses, but those codes are not coupled with the NRC's principal thermal-hydraulic codes. To accommodate the reduction in resistance to failure that fuel cladding experiences at high burnups, 3-D licensing analyses may be needed to avoid penalties associated with the current conservative kinetics models. Such 3-D codes would require NRC review and approval.

In addition to the transition to a more dimensional kinetics analysis, there are several specific features of the kinetics codes that may need to be modified to address localized high-burnup effects. One is the local power peaking during rapid power pulses (critical or prompt critical) that may not be treated conservatively by codes that use fuel rod

bundles as the smallest calculational node rather than individual fuel rods. Another is the reduction in the delayed neutron fraction that results from the buildup of plutonium isotopes at very high burnups. These and other high-burnup code features need to be examined carefully.

(b) Risk Perspective

The NRC's and the vendors' fuel codes can be used for a range of applications including safety analyses for LOCA. Because of the risk significance of LOCAs, these codes are therefore important from a risk perspective. The neutronics codes can also be used for a range of applications including the rod drop and rod ejection accidents, which are not particularly large risk contributors. However, this same capability is also needed to analyze the BWR ATWS, which has potentially greater risk significance, and other power transients.

(c) Near-Term Assessment

The need for improved NRC fuel rod codes was identified early, and a major part of that work has now been completed. For the neutronic codes, a long lead time will be required to prepare codes for NRC review and to conduct a review if such codes are submitted by the industry.

(d) Related NRC Research

The steady-state fuel code, which is used most frequently by NRC in licensing activities, has been updated as FRAPCON-3 and is applicable to about 65 GWd/t (rod average). A peer review was conducted, and the code and its documentation were issued in December 1997. Work is currently underway to upgrade the transient fuel code, FRAPTRAN, and this phase of the work will continue in FY98-99. To date, improvements in both codes have focused on thermal analysis, and additional improvements are needed in the mechanical models. Further updates of the thermal models will also be made as additional data become available at higher burnups and for higher concentrations of burnable poisons. Detailed work plans have been developed for FRAPCON-3 and FRAPTRAN, and these plans are being followed. Schedules for the code-related research are shown in the Appendix.

The NRC's TRAC-P and TRAC-B plant systems codes do not currently have 3-D kinetics models, so the RAMONA-4B neutronics code has been used by NRC for independent 3-D studies of the BWR rod drop accident with high-burnup fuel. To improve NRC's plant analysis capability, Purdue University's PARCS 3-D kinetics code is being coupled with TRAC-P and TRAC-B to provide the full three-dimensional capability. To estimate the effects of local power peaking within fuel bundles, the PARCS code is being compared with the Russian BARS code, which has pin-to-pin modeling. The effects of

local power peaking, delayed neutron fraction, and other high-burnup effects will be assessed in an ongoing program at Brookhaven National Laboratory.

(e) Final Resolution

Maintaining steady-state and transient fuel codes and updating them for new burnup ranges and new fuel and cladding materials will be a continuing activity. The fuel vendors will also continue to update their codes as new data become available for higher burnups and modified fuel designs. However, resolution of this issue will occur for the fuel codes when the current FRAPTRAN update has been completed to install the high-burnup thermal models developed for the recently updated FRAPCON-3 code.

For the 3-D neutronics codes, resolution will be largely achieved when NRC's new 3-D capability, with the coupled PARCS code, becomes available in 1998. There will be, nevertheless, continuing activities to maintain this capability and to keep the codes updated for new fuel materials and new operating conditions. Industry submittals of new 3-D codes for high-burnup applications would require a long lead time for staff review.

7. Source Term and Core Melt Progression

(a) Description of Issue

During a severe accident, the progression of the accident sequence is strongly dependent on the way molten material develops in the core. Radiological releases, in turn, are determined by the progression of the accident. Estimated releases for a spectrum of severe accidents have been used to develop the recent NUREG-1465 source term. This source term, however, may not be applicable for fuel irradiated to high burnup levels (in excess of about 40 GWd/t as noted in NUREG-1465). It is known that at higher burnups the gap inventory will increase, fuel particle behavior will be different, and the isotopics will shift. It is also known that cladding becomes more brittle at higher burnups, potentially resulting in earlier cladding failure and fuel relocation during a severe accident.

(b) Risk Perspective

Since risk is the product of probability and consequence, understanding core melt progression and having a source term are necessary even to determine risk for various events. The severe accident source term is also used for analysis of consequences of a LOCA, and LOCA is a risk-significant design-basis accident. Thus, determining the effect of high burnup on source terms and core melt progression is itself very important from a risk perspective.

(c) Near-Term Assessment

The main effects that might impact source terms and core melt progression at high burnup are (a) a reduction in the amount of unoxidized zirconium in the core, (b) embrittlement of the fuel cladding, (c) an increase in the release of fission gases from fuel pellets during normal operation, (d) fragmentation of fuel pellets, and (e) a shift in the spectrum of fission products produced as plutonium fission becomes more important.

A reduction in the amount of unoxidized zirconium metal in the core could diminish the severity of core melt and subsequent ex-vessel phenomena by lowering the reaction heat from metal oxidation. However, the amount of preoxidation of the cladding will be less than 17% of the wall thickness because of the restrictions of 10 CFR 50.46 and it likely to be much lower than that for newer cladding alloys, so this beneficial effect would be small. Non-molten fuel relocation may occur due to cladding embrittlement, particularly for scenarios involving delayed reflood or depressurization, but this is not expected to significantly affect the overall outcome of uninterrupted core melt accidents. Gap activity comprises only a small part of the source term so that even large changes in gap activity would not have a big effect on the source term. (However, some licensing analyses use only gap activity, e.g., the fuel handling accident, and for those, consideration will need to be given to the increased gap activity resulting from the use of higher burnup fuel.) Fuel fragmentation has been observed at high burnup, but it appears that dispersal of fragments occurs by washout and there may be no means to get that material into the atmosphere as aerosol particles. In contrast, particulate releases included in the source term are lifted from the core as high temperature gases that condense as aerosol particles. The source term itself consists of release fractions and therefore would not be affected by isotopic shifts. Those shifts would be accounted for in the generation analysis (e.g., with the ORIGEN code) and any changes are expected to be small.

Considering these factors, it is unlikely that high burnup will have a significant effect on source terms or core melt progression.

(d) Related NRC Research

The applicability limitation of 40 GWd/t, mentioned in NUREG-1465, came from the data range of the HI and VI fission product tests at Oak Ridge National Laboratory. Similar measurements on higher burnup fuel specimens are being made by CEA (France) at Grenoble, and results from those tests are available to the NRC through NRC's Cooperative Severe Accident Research Program. An assessment of the effects of high burnup on core melt progression and the source term, utilizing recent French data, was scheduled for FY98, but funding is no longer available for this work.

(e) Final Resolution

The current source term is considered to be adequate for the foreseeable future.

8. Transportation and Dry Storage

(a) Description of Issue

Two aspects of transportation and dry storage of spent fuel that might be affected by high burnups are the nuclide inventory and long-term cladding integrity. The nuclide inventory in turn affects shielding, heat sources, and potential releases of activity. As in reactors, the spent fuel cladding is the first barrier for retention of fission products. The cladding's integrity affects potential releases of fission products and the ability of licensees to safely retrieve the spent fuel for ultimate disposal.

(b) Risk Perspective

A dry cask PRA has been initiated at Brookhaven National Laboratory and could provide a risk perspective for this issue, although the funding to complete this work is not available.

(c) Near-Term Assessment

This issue addresses future actions that are now under consideration. Vendors of spent fuel casks have applied for storage of fuel with burnups up to 65 GWd/t (average for the peak rod), which is well above the burnup level for which current methods and assumptions have been approved.

(d) Related NRC Research

Research will be defined in FY98 to address the two topics of nuclide inventory and long-term cladding integrity.

(e) Final Resolution

Final resolution depends on the outcome of this future research and will occur when spent fuel casks are approved for fuel with high burnups.

9. High Enrichments (>5%)

(a) Description of Issue

To date, the validation of criticality safety codes, and associated cross section libraries,

for LWR fuels has concentrated on enrichments less than 5%. Neither benchmarks of code performance nor the bases for extrapolating code performance in the enrichment range of 5-10% have been well established. Moving into this range will require care because the physics of criticality begins to change as enrichments reach 6% and beyond, where single moderated assemblies can go critical and criticality of weakly moderated or unmoderated systems becomes possible. Enrichments above 5% will require redesign of some fuel fabrication and handling equipment and fuel transportation packages. The possibility of recriticality during severe accident core melt sequences should also be addressed as this could alter the progression of such accidents.

(b) Risk Perspective

Risk studies have not been performed.

(c) Near-Term Assessment

This is an emerging issue. Some enrichment facilities and fuel fabricators have formally stated the intent to go to enrichments greater than 5%. (Aspects of the same criticality validation issue have already arisen in the ongoing downblending of surplus HEU to 5% enrichment. One downblending facility recently received an infraction for having failed to validate its criticality analysis methods in the 5-10% enrichment range.)

(d) Related NRC Research

Ongoing research at ORNL on the ranges of applicability of criticality validation is aimed in part at helping address this issue. Needs for any additional research in this area, such as analytical benchmark studies, new experimental benchmark data, and severe accident considerations, will be defined in FY98.

(e) Final Resolution

Final resolution depends on the outcome of this future research and will occur when higher enrichments are approved.

Licensing and Research Strategy

The data and analyses described above (some of which will not be completed for 3-5 years) are intended to provide confirmation of acceptable fuel behavior for current fuel designs up to the present burnup limit. To obtain higher burnup limits, additional data and analyses of a similar nature will have to be provided. To avoid the need for extensive confirmatory work in the future, sufficient data and analyses will have to be provided prior to receiving NRC approvals.

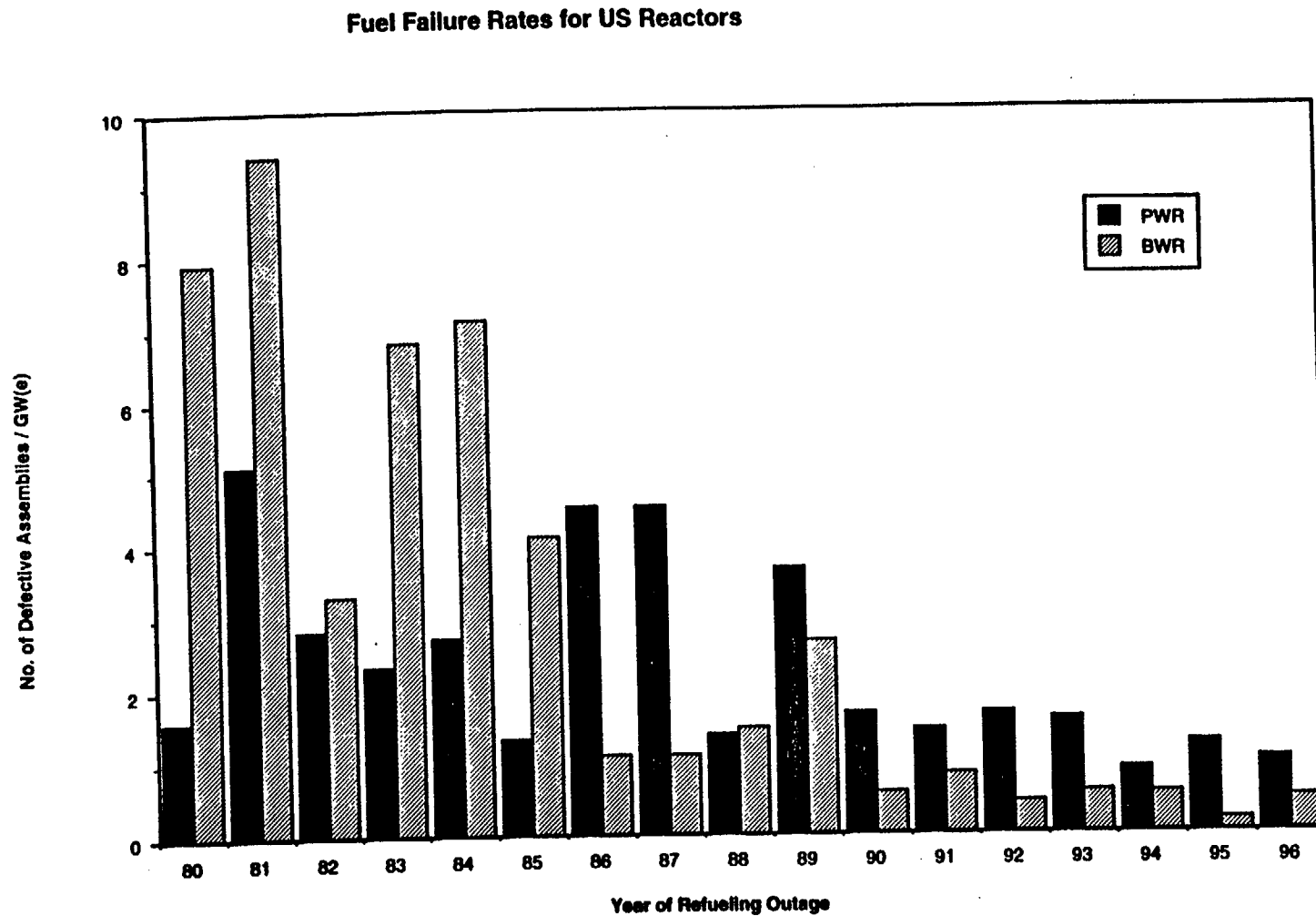
In the past, the NRC has always performed the research needed to define regulatory criteria, and the industry has performed research to develop methods of demonstrating compliance with those criteria. In recent years, NRC's research budget has declined to a level that the NRC can no longer support such research. Thus, if the industry wants further burnup extensions, it will have to develop a data base for revised (or confirmed) regulatory criteria. The staff will make it clear to the industry that such research must be non-proprietary, to ensure that resulting criteria are fully scrutable, and the NRC staff must have full access to those research programs. If NRC resources are available, the NRC will actively participate in those research programs; however, the industry will be expected to take the lead in this work.

In accordance with the NRC's Strategic Plan, the staff will also encourage the industry to develop codes, standards, guides -- and, by inference, fuel damage criteria -- that can be endorsed by the NRC and carried out by the industry. Fuel behavior would have to be addressed during normal operation, transients, and postulated accidents, and, at a minimum, the high-burnup issues identified above would have to be covered. A program for monitoring fuel performance should be included in the industry proposal.

Also in line with the Strategic Plan, these codes, standards, guides, and criteria could be focused on events that pose the greatest risk to the public, based on probabilistic risk assessment concepts and other approaches for determining high- and low-risk activities. If found acceptable, these codes, standards, guides, and criteria would be incorporated in a regulatory guide and endorsed by the staff. The review, public comment, and issuance process would likely take 12-18 months from receipt of a comprehensive industry proposal. Demonstration of compliance with the provisions of the guide would follow for a particular fuel design and burnup limit.

To develop a data base necessary to justify further burnup extensions, suitable fuel rod specimens would have to be available for testing under transient and accident conditions. For this purpose, the NRC would encourage the irradiation of lead test assemblies (LTAs) with typical burnup histories up to the proposed licensing limit and positioned in near-limiting core locations. NRC would further encourage the irradiation of segmented test rods in the LTAs to facilitate subsequent testing. The NRC would also consider limited cooperation with the industry in the data phase of such test programs as that would make important data available to the NRC for its own independent assessment.

FIG. 1 Trend in US Fuel Failure Rates



**Fig. 2 Causes of Fuel Failure in US PWRs
Over the 1991-1996 Time Period**

Number of failed rods

Failure cause	1991	1992	1993	1994	1995	1996
Handling Damage	3			1	3	
Debris	110	28	10	7	12	1
Baffle Jetting						
Grid Fretting	46	71	102	13	55	113
Primary Hydriding		23				
Crud/Corrosion					10	
Clad Creep Collapse					1	
Other Fabrication	1	7	2	1	20	4
Other Hydraulic			1			
Inspected/Unknown	14	52	30	44	16	9
Uninspected	23	17	22	3	13	3
Totals	197	198	167	69	130	130

**Fig. 3 Causes of Fuel Failure in US BWRs
Over the 1989-1996 Time Period**

Number of failed assemblies (same as rods)

Failure cause	1989	1990	1991	1992	1993	1994	1995	1996(pre)
CILC	52	5	3					
Fabrication	3	3	1	1	1	2		
PCI		1			2		2	2
Debris	2	2	17	2	6	4		1
Inspected/Unknown		4	3	9	7	9	2	11
Totals	57	15	24	12	16	15	4	14
Total FAs discharged	4020	3759	2872	4150	3974	3893	4684	3336

HIGH BURNUP FUEL RESEARCH SCHEDULES

Issue 1. Cladding Integrity and Fuel Design Limits

None

Issue 2. Control Rod Insertion Problems

None

Issue 3. Criteria and Analysis for Reactivity Accidents

CABRI Test Reactor (France)

FY 98	FY 99	FY 00	FY 01	FY 02
Sodium Loop Tests				
Water Loop Tests				
Water Loop Construction & Installation				

NSRR Test Reactor (Japan)

FY 98	FY 99	FY 00	FY 01	FY 02
Ambient Pressure, Temperature Capsule				
High Pressure, Temperature Capsule				

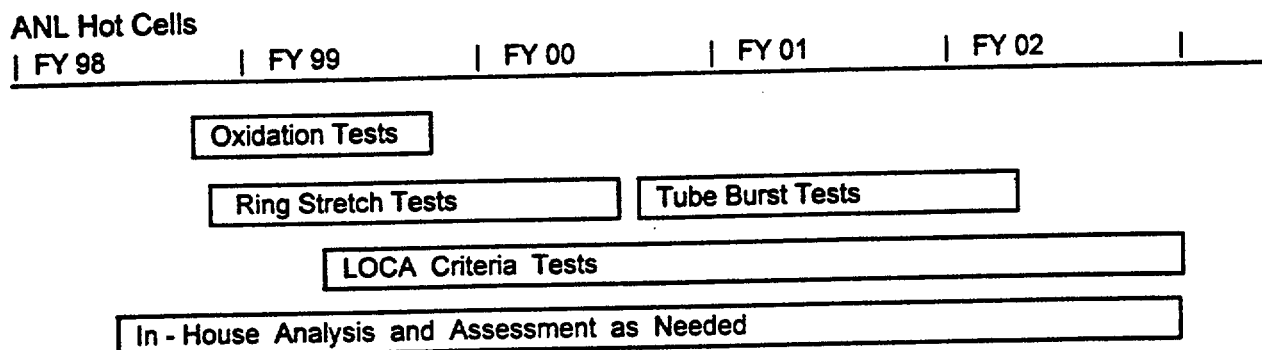
IGR Test Reactor and RIAR Hot Cells (Russia)

FY 98	FY 99	FY 00	FY 01	FY 02
Document IGR				
Zr - Nb Mechanical Properties				
Neutronics and Fuel Codes				

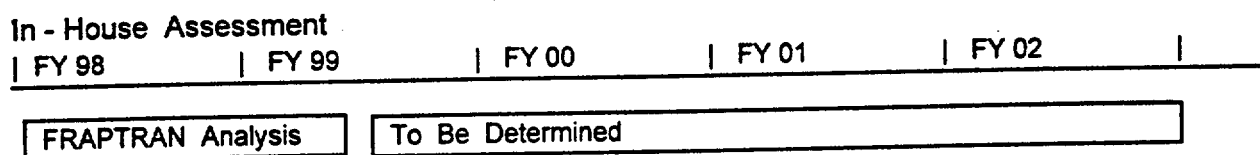
In - House Assessment

FY 98	FY 99	FY 00	FY 01	FY 02
In - House Analysis and Assessment as Needed				

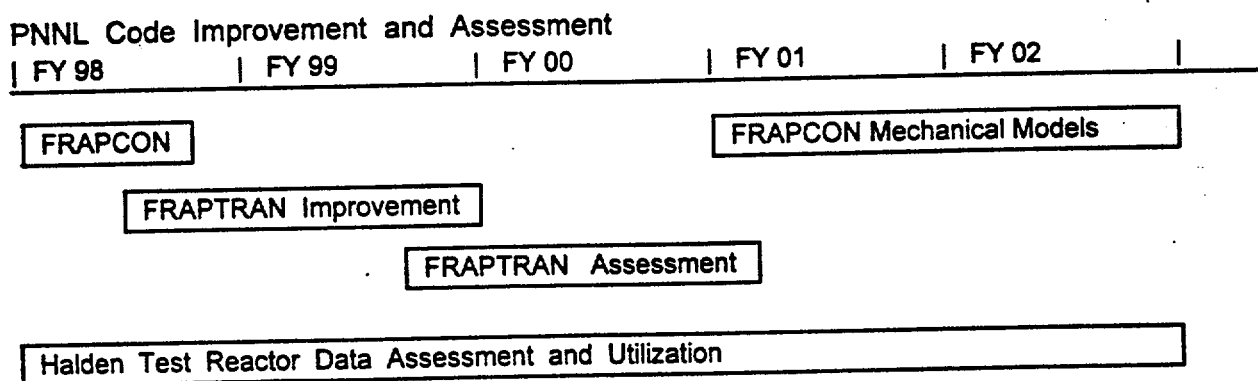
Issue 4. Criteria and Analysis for Loss-of-Coolant Accidents



Issue 5. Criteria and Analysis for BWR Power Oscillations (ATWS)



Issue 6. Fuel Rod and Neutronic Computer Codes for Analysis



Issue 7. Source Term and Core Melt Progression

To Be Determined

Issue 8. Transportation and Dry Storage

In - House Assessment

FY 98	FY 99	FY 00	FY 01	FY 02
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Define Work

Issue 9. High Enrichments (>5%)

In - House Assessment

FY 98	FY 99	FY 00	FY 01	FY 02
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Define Work
