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TECHNICAL BASIS FOR THE PROPOSED HIGH EFFICIENCY NUCLEAR FUEL PROGRAM

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Abstract

Greenhouse gas emissions from fossil fired electricity generating stations will dramatically increase over the next 20 years. Nuclear energy is the only fully developed technology able to supply large amounts of electricity without generation of greenhouse gases. However, the problem of noncompetitive economics and public concerns about radioactive waste disposal, safety, and nuclear weapons proliferation may prevent the reemergence of nuclear power as a preferred option for new electric energy generation in the U.S. This paper discusses a new research program to help address these issues, by developing fuel designs capable of burnup values in excess of 60 MWD/kgU. The objectives of the program are to

- Improve the reliability and robustness of light water reactor fuel, thereby improving safety margins.
- Significantly increase the energy generated by each fuel loading, thereby achieving longer operating cycles, higher capacity factors, and lower cost electric power.
- Significantly reduce the volume of spent nuclear fuel discharged for disposal by allowing more energy to be extracted from each fuel element prior to discharge.
- Develop fuel that is much more proliferation resistant.

1. INTRODUCTION

Trends in the world's population and energy use during the past century show dramatic and relatively parallel increases in both. These trends are expected to continue in the near future (at least the next 20 years), and the total world energy consumption in 2015 will be about 54% higher than it is today, led by growing demand in Asia [1]. The demand for electricity is expected to increase more rapidly than the demand for other forms of energy throughout the world and nearly double by 2015. Coal will be used to generate much of that electricity in the developing countries. In the industrialized world, there are also dramatic structural changes underway in the electric power industry to enhance competition in the generation segment of the business. This, along with ample natural gas supplies and relatively low gas prices, has made natural gas the preferred fuel for many power producers in the U.S. and elsewhere. These developments (increasing energy demand and increasing use of natural gas and coal) are expected to <u>increase</u> the amount of carbon emitted to the atmosphere from the world's electrical power plants by about 70% over the next 20 years [1].

Nuclear energy is the only fully developed technology able to supply large amounts of electricity without generation of greenhouse gases, and therefore should be a key element in the strategy to control greenhouse gas emissions. However, several problems cloud the future of nuclear power in the U.S. and need to be addressed for nuclear power to be a preferred electric power generation option. President Clinton's Committee of Advisors on Science and Technology (PCAST) [2] recently recommended that an enhanced national R&D effort is needed to improve energy technologies. Regarding nuclear power, PCAST stated that that *"the potential benefits of an expanded contribution from fission in helping address the carbon dioxide challenge warrant ...[finding] ...out whether and how improved technology could alleviate the concerns that cloud this energy option's future."*

The U.S. Department of Energy (DOE) has responded to the PCAST recommendations with two new programs: the Nuclear Energy Research Initiative (NERI) and the Nuclear Energy Plant Optimization (NEPO) program. The High-Efficiency Nuclear Fuel Program was developed by DOE and submitted as part of the FY 1998 budget, but did not receive funding from Congress. The authors expect that the program will compete favorably for funding under the Department's FY 1999 NERI program. The program will be a cooperative research and development program with industry to develop improved fuels that can be operated to higher burnups with greater safety margins.

The specific goals of the High-Efficiency Nuclear Fuel Program are to: (1) develop within 7 years light water reactor (LWR) fuel designs and cladding materials that can operate satisfactorily to about 25% higher burnup; (2) develop in 15 years LWR fuels that can be used about twice as long as current fuels; and (3) demonstrate that these new fuel materials and designs are superior to current LWR fuel during normal operation and during any accident that may occur. If the program is successful, use of these fuels will enable longer plant operating cycles and improved capacity factors, which would help lower operating costs; further reduce defect rates and provide improved margin during any off-normal or accident condition; reduce the amount of spent fuel that must be handled, stored, and placed in a repository; reduce the amount of low-level waste produced by the commercial nuclear power industry; reduce worker exposures; and significantly decrease the possibility of someone using LWR spent fuel for nuclear weapons material. The purpose of this paper is to discuss LWR fuel behavior during normal and design basis accident conditions and summarize the proposed research and development program. The benefits of the proposed program are discussed elsewhere [3].

2. LWR FUEL PERFORMANCE AT HIGH BURNUP

LWR fuel is currently limited to (a) about 62-MWD/kgU peak rod burnup by the USNRC because of concerns about high burnup fuel integrity, and (b) less than 5% enrichment because of the design and licensing of the fuel fabrication plants and handling and storage equipment. In addition, the control rod worths and other aspects of the core neutronics designs may limit the use of significantly higher burnup fuel. The issues which must be addressed when considering the use of LWR fuel at higher burnup include:

- Loss of cladding ductility and fracture toughness due to (a) excessive corrosion, hydrogen uptake, and zirconium-hydride formation, (b) neutron radiation damage, and (c) oxide spallation and zirconium-hydride blister formation.
- Excessive cladding growth.
- Increased fuel pellet-cladding mechanical interactions (PCMI) due to cladding creepdown, fuel swelling, and fuel-cladding diffusion bonding.
- Reduced fuel thermal conductivities and increased fuel temperatures due to plutonium and fission product buildup near the surfaces of the fuel pellets and the resulting formation of a porous rim.

- Increased fuel rod internal pressures due to the long irradiation times (more time for diffusion), larger inventory of fission products, and higher fuel temperatures.
- Runaway cladding oxidation.

Zircaloy was originally chosen for cladding the fuel in nearly all LWRs because of its low neutron cross section and relatively good corrosion resistance. However, thick oxide layers are often found on zircaloy-clad fuel rods, especially pressurized water reactor (PWR) fuel rods, irradiated to burnups of 50 to 60 MWD/kgU. A cross section of a zircaloy clad PWR fuel rod irradiated to about 60 MWD/kgU is shown in FIG. 1. The oxide layer thickness was about 150 μm before the oxide began to spall off. The relatively lower temperatures at the locations where the oxide spalled off resulted in hydrogen diffusion down the temperature gradient and the formation of blisters of zirconium-hydride. The remainder of the cladding wall thickness contains numerous zirconium-hydride platelets. There is very little ductility or toughness left in such material.

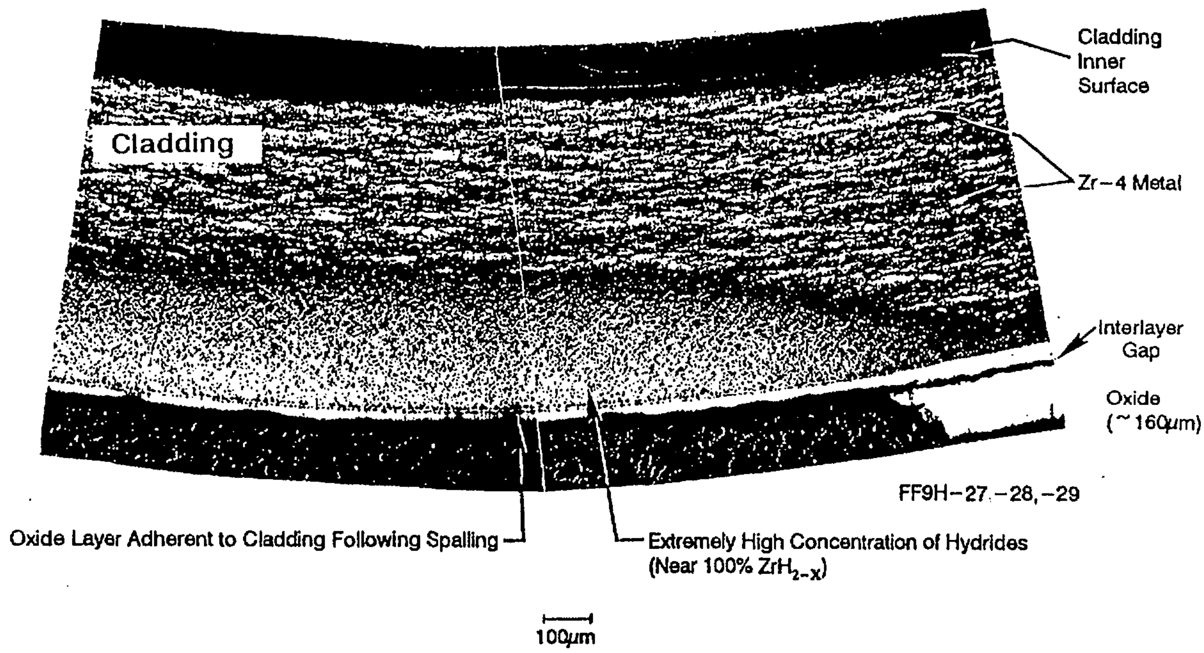

FIG. 1 Zirconium hydride concentration resulting from operation with spalled oxide.

In addition to the effects of corrosion, the ductility of zircaloy cladding is significantly reduced by neutron radiation. For example, the total plastic elongation at burst of zircaloy tubes irradiated to fast fluences above 10 x 10^{21} n/cm^2 (E>1Mev) is sometimes as low as 1/2 to 1% (compared to 15 to 20% for unirradiated material).- values which might result in cladding failure during modest power increases. Other problems with the use of zircaloy as cladding and structural material in LWRs include (a) excessive thimble growth resulting in bowing which restricts full insertion of control rods and (b) runaway cladding oxidation likely due, in part, to poor water chemistry.

PCMI failures have also occurred in some LWRs, especially boiling water reactors (BWRs) where control rod movement results in a significant power change in nearby fuel rods. Higher burnup will result in additional cladding creepdown, fuel swelling, and fuel-cladding diffusion bonding - phenomena which should result in more-severe PCMI during power changes.

Fortunately, the international nuclear fuel vendors have been developing and testing advanced fuel cladding and structural materials for, in some cases, over 20 years. The result has been new products which appear to be much more resistant to corrosion and hydrogen uptake and PCMI than standard zircaloy. FIG. 2 is a plot of oxide thickness versus burnup for a variety of alternative Westinghouse fuel rod cladding materials [4]. The low-tin ZIRLO material exhibits about 1/4 the corrosion of standard zircaloy [5]. Other fuel vendors have also developed new cladding materials (zirconium alloys) which show promise for use at higher burnup [6-11]. Some of these materials not only have much better corrosion resistance, but exhibit less growth and creep. Cladding liners have also been developed for BWR fuel rods to help protect against PCMI failure.

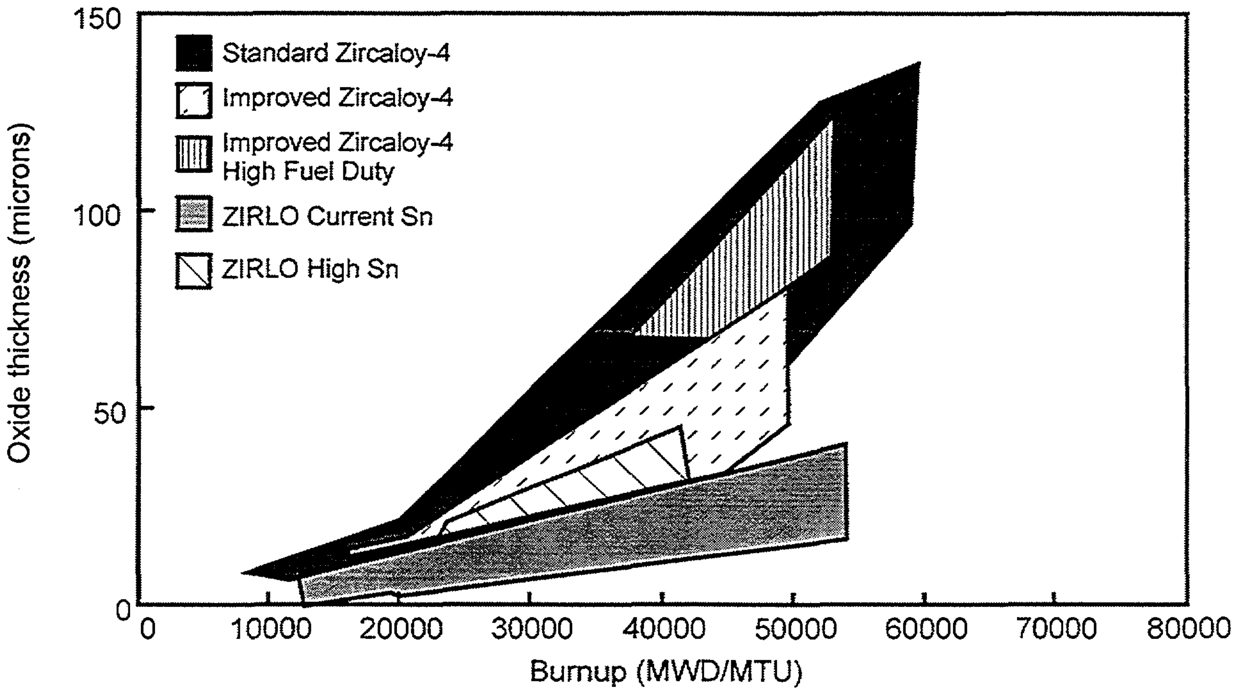

FIG. 2 Oxide thickness versus burnup for alternative Westinghouse fuel rod cladding materials [4].

There has been much less development of the LWR fuel form over the years, which is primarily pressed and sintered UO_2. There has been some minor changes in the pellet diameter-to-length ratios, the dish shapes, amount of chamfer, fuel density, and fuel grain size, primarily to minimize PCMI and fuel densification and gas release. Some LWR fuel rods have plenums on both ends of the rods, some have a top plenum only, and various end plug shapes are used. However, more work could be done to develop fuel designs which better retain the fission products within the fuel, have a more uniform rod internal pressure, and minimize fuel-cladding mechanical interactions.

3. FUEL PERFORMANCE UNDER TRANSIENT CONDITIONS

If a fuel type designed for high-burnup operation is to be utilized, then the performance of that fuel under postulated accident scenarios must meet relevant regulatory criteria. Two important classes of design-basis accidents for LWRs in the U.S. are the reactivity-initiated accident (RIA) and the loss-of-coolant accident (LOCA). Recent evidence indicates that the evolution of characteristics and properties of LWR fuel and cladding materials during extended irradiation may degrade the ability of fuel rods to withstand failure under RIA and LOCA conditions; e.g., [12]. Therefore, the proposed program includes provision for assessing and demonstrating the RIA and LOCA behavior of the fuel designs developed in this program. The following sections address considerations relevant to each accident class.

3.1 Reactivity-initiated accidents

Two principal regulatory criteria are used in the U.S. to assess safety under postulated RIAs [12]. To ensure core coolability after an RIA and to preclude energetic dispersal of fuel particles into the coolant, the peak fuel-rod enthalpy is limited to 1170 kJ/kg fuel (280 cal/g fuel). To allow calculations of radiological releases, other values are used to indicate cladding failure: critical heat flux values related to departure from nucleate boiling or, for low-power accidents in BWRs, 711 kJ/kg fuel (170 cal/g fuel). These regulatory criteria were established using test data from fresh fuel or from fuel with relatively low burnup (i.e., $\leq$ 5 MWD/kgU). In Japan, the cladding failure criterion is more conservatively set at 356 kJ/kg fuel (85 cal/g fuel) for irradiated fuel, reflecting current knowledge that the RIA failure resistance of some fuel designs is significantly degraded with burnup [13].

Results of recent simulated RIA tests of higher-burnup fuel rods indicate that fuel failure is possible at fuel enthalpy values considerably lower than the U.S. Nuclear Regulatory Commission (USNRC) criteria values. Therefore, the issue continues to receive considerable international attention [14]. Initially, the consequences of low-enthalpy failures were thought to be sufficiently minimal as to pose no concern for public health and safety [12]; however, further evaluation of the observed test fuel failures has indicated that fuel dispersal from low-enthalpy failures may raise questions regarding core coolability. For example, personnel from the USNRC are now considering establishment of a new criterion of 418 kJ/kg fuel (100 cal/g fuel) as both a cladding failure threshold and a fuel dispersal threshold [15]; additional testing would be required to establish a coolability criterion above the cladding failure threshold. Ongoing test programs at the CABRI facility in France and the Nuclear Safety Research Reactor (NSRR) in Japan are intended to determine the conditions and phenomena that lead to RIA fuel failure and to assess post-failure fuel dispersal. Although the programs at each location have included tests of several rods which did not result in failure, it is instructive to review the parameters and conditions for the tests which did result in fuel failure. A summary of information from those tests is provided in Table I.

TABLE I. Summary of Information from Selected Recent RIA Tests Resulting in Failed Fuel

Test ID	Fuel Type	Fuel Burnup MWD/kgU	Cladding Oxide Thickness (μm)	Pulse Width @ FWHM (ms)	Peak Fuel Enthalpy at Failure (kJ/kg fuel)	Fuel Dispersal
NSRR Tests [12,13,16]:						
HBO-1	UO_2	50	43	4.4	250 (60 cal/g)	yes
HBO-5	UO_2	44	103	4.4	301 (72 cal/g)	small
TK-2	UO_2	48	23	4.4	250 (60 cal/g)	small
CABRI Tests [17,18]:						
REP-Na1	UO_2	64	80	9.5	125 (30 cal/g)	yes, 6g fuel
REP-Na7	MOX	55	50	40	501 (120 cal/g)	TBD
REP-Na8	UO_2	60	130	80	347 (83 cal/g)	No

The conclusions emerging from the ongoing RIA testing programs in France and Japan indicate that failure of high-burnup fuel rods is induced by PCMI, exacerbated by the condition of the fuel and cladding after high-burnup, steady-state irradiation [12]. Temperature-driven expansion of fission gas in the oxide fuel matrix is more pronounced with increasing burnup, and provides a driving force for radial expansion of the fuel as the fuel temperature increases (in addition to the intrinsic thermal expansion of the fuel). During an RIA transient, the radial expansion of the fuel places a hoop stress on the cladding. If the cladding has experienced significant degradation due to corrosion, as is often the case with zircaloy cladding materials, then the ability of the cladding to withstand the PCMI-induced stresses will be reduced. Specific cladding degradation phenomena include cladding oxidation and/or spallation, which effectively reduces the thickness of the cladding wall available to withstand the PCMI loading, and embrittlement due to hydrogen uptake and irradiation damage. Uptake of hydrogen into the cladding and movement of hydrogen down the temperature gradient to form zirconium hydride precipitates within the cladding wall is considered to be a key phenomenon leading to some of the failures listed in the table above. Development and implementation of cladding alloys that resist corrosion, and radiation embrittlement at burnup values well above 62 MWD/kgU, will be important to licensing fuel designs intended for ultra-high burnup.

The decreased grain size and locally higher fission content that forms near the periphery of a fuel pellet during prolonged irradiation (i.e., the rim effect) also facilitates fragmentation of fuel debris during a transient. Fuel grain decohesion occurs as fission gas bubbles at the grain boundaries (a characteristic of the rim structure) pressurize with increasing temperature and expand, resulting in loose particles or fragments of fuel material. Fragmented fuel debris can be released through a cladding breach into the coolant where, depending on the degree of fuel release, coolant channel blockage can be a concern. Fuel designs that avoid formation of this rim structure, or that otherwise mitigate fuel fragmentation are desirable for achieving ultra-high burnup reliability.

Although the failures of the test fuel rods listed in Table I are considered to be induced by PCMI, other phenomena can contribute to failure at different conditions. For example, if a fuel rod withstands a RIA to fuel enthalpy values higher than those indicated in the table, then failure due to internal fission gas pressurization may occur. Because fuel irradiated to high burnup retains a correspondingly higher fission gas inventory, an increase in temperature can lead to release of an amount of fission gas into the fuel rod plenum that is sufficient to balloon the cladding. Furthermore, as cladding temperatures increase during a RIA transient, the yield or ultimate strength of the cladding material decreases, perhaps to the point of failure under internal pressurization or PCMI.

RIA variables that most affect fuel failure during a RIA are cladding temperature and fuel temperature (which affects fuel expansion and/or fission gas release). The reactor parameters that control these variables are power pulse width, typically expressed as the full-width at half-maximum (FWHM) value of reactor power as a function of time, and energy deposition into the fuel, typically manifested as fuel enthalpy rise expressed in units of calories per gram of fuel. Time-dependent cladding and fuel temperatures are also determined by coolant flow conditions.

Three-dimensional neutronics and thermal-hydraulics calculations of the response of a LWR core to a RIA generally predict energy insertions of less than 418 kJ/kg (100 cal/g) in any fuel rod and power pulse FWHM values ranging from 70 to 100 msec, although one recent calculation resulted in 30 to 80-msec power pulses [19]. Most RIA tests of high-burnup fuel have been performed with excessively-narrow power pulses (<10 msec FWHM), although the more recent tests in CABRI have been performed with a power pulse that approximates an RIA pulse in a PWR. Narrow power pulses deposit energy into the fuel in a non-prototypic manner, exacerbating fuel-cladding mechanical interactions that may lead to failure. More specifically, narrow power pulses 1) induce fuel pellet expansion at a faster rate, leading to a higher PCMI-induced strain rate in the cladding, and 2) induce relatively higher temperatures in the rim region of the fuel pellet (due to energy deposition occurring at a faster rate than heat transfer mechanisms can remove energy), intensifying the expansion phenomena that cause PCMI and thereby raising the peak hoop stress in the cladding. Furthermore, the peak stress on the cladding occurs before the cladding can heat up to temperatures at which the cladding will more easily yield in a ductile manner; thus, brittle failure may be artificially induced. These tests have also been performed with non-prototypic coolant conditions, using sealed capsules containing stagnant air or water coolant or using loops with flowing sodium, leading to non-prototypic pre-test temperatures and pressure. Furthermore, the fuel rods that have failed in those tests were clad with standard zircaloy alloys; these alloys exhibit much more severe oxidation and hydriding behavior (which render a fuel rod more susceptible to failure by fuel-cladding mechanical interaction) than do the improved zirconium-based alloys that are in current use in many nuclear power plants (NPPs).

The tests performed to date have been suitable for the stated objectives of those test programs, which is to study PCMI during the early stages of the transient [17], and the results reported from those test programs thus far have provided tremendous insight into the mechanism of PCMI-induced failure. However, any additional tests to be performed for determining the RIA behavior of high-burnup fuel should more closely simulate postulated RIA events, using 30 to 100-msec power pulse widths, inducing peak fuel enthalpies of 83.7 to 837 kJ/kg fuel (20 to 200 cal/g), employing prototypic coolant conditions and test rods with the improved cladding alloys in current use. A prototypic fuel temperature during a test is ensured by a prototypic power distribution through the fuel (which is determined by the duration and magnitude of the power pulse and by the neutron spectrum) and by prototypic coolant conditions. Coolant flow conditions during a test are best provided by a water loop that supplies flow to the test fuel at prototypic temperatures and flow rates.

3.2 Loss-of-coolant accidents

The U.S. regulatory criteria pertaining to LOCAs are intended to ensure core coolability through the duration of and beyond the accident. Specific criteria include a limit on the maximum cladding temperature attained during a LOCA of 1204°C and a limit to the cladding oxidation corresponding to a 17% cladding wall thickness reduction. Furthermore, the core must remain coolable after the accident, which implies that failed fuel debris cannot cause an unacceptable reduction or blockage of the coolant flow.

The behavior of LWR fuel during a LOCA is influenced by somewhat different factors than those that influence LWR fuel behavior during an RIA. The sudden depressurization that occurs during the large break LOCA combined with a degraded heat transfer associated with the loss of coolant is expected to cause cladding ballooning and burst. The magnitude of the ballooning will be influenced by the absolute temperature (zircaloy is more ductile in either its alpha or beta phase and significantly less ductile when changing phases), the rod internal pressure, and the temperature distribution around and along the rod. Because of cladding creepdown, fuel swelling, and fuel-cladding diffusion bonding, high burnup tends to promote large, long balloons during a LOCA which result in more flow blockage that occurs with low burnup or fresh fuel. In addition, the choice of alloying elements can influence the cladding ballooning behavior.

The core coolability is also influenced by the cladding oxidation and embrittlement. There is some evidence that zircaloy cladding with a heavy load of hydrogen may oxidize much faster during a LOCA than normal zircaloy. Furthermore, fuel material from the outer periphery (or rim) can fragment due to decohesion induced by thermal stresses during fuel quenching, which occurs as liquid coolant is re-contacted with the fuel. As with the RIA mechanisms described, fuel fragmentation may release fuel material into the coolant, leading to a coolant flow blockage if the release is sufficiently severe.

Further study of fuel and cladding properties is required to understand the implications of high-burnup irradiation on fuel behavior during LOCAs. The cladding ballooning and rupture and fuel fragmentation characteristics for high-burnup fuel with various cladding materials must be assessed to determine the conditions at which failure is expected and the consequences of failure under specified conditions. The oxidation behavior of irradiated cladding must be assessed under conditions similar to those encountered in LOCAs. Development of a modeling capability to predict fuel behavior under transients (either LOCAs or RIAs) will require more complete mechanical properties data for irradiated material at relevant stress and strain conditions.

A joint program funded by the USNRC and the Electric Power Research Institute (EPRI), with participation of the U.S. DOE, has been initiated to address the needs described above [20]. The study currently is addressing a limited number of fuel designs and cladding types with a burnup limit of 60 MWD/kgU. The program, which is being conducted in a hot cell, is comprised of two major tasks. The first task consists of engineering tests of fuel rods and cladding under LOCA conditions. These tests will determine the kinetics of oxidation of selected zircaloy cladding alloys, determine the integral LOCA behavior of fuel rod segments, and tests to determine rod stiffness and ductility for assessing resistance to seismic loading during and after a LOCA event. The second task will determine the post-irradiation mechanical properties of selected, current cladding alloys; emphasis will be placed on mechanical property testing at stress and strain conditions relevant to both LOCAs and RIAs. The program discussed below will be complementary to and coordinated with the USNRC program.

4. PROGRAM DESCRIPTION AND PRODUCTS

The High Efficiency Nuclear Fuel Program will include a three-part approach:

- Determine the useful life of the best fuel currently in commercial NPPs
- Develop a better fundamental understanding of the life-limiting degradation mechanisms at high burnup
- Design and test advanced and innovative LWR fuel forms.

A preliminary program schedule, which shows the relative timing of the major activities, is presented in FIG 3.

The research with the commercial spent fuel will include characterization of the condition of modern fuels that have been burned in commercial NPPs to the current USNRC limits, further irradiation of these fuels in a test reactor to ultra-high burnups, and design-basis accident testing to demonstrate that this fuel will meet USNRC licensing criteria. Further irradiation and power ramp testing of PWR fuel designed and fabricated by the U.S. fuel vendors will start 12 to 18 months into the program and will be performed primarily in the Advanced Test Reactor (ATR) in Idaho. PWR conditions can be exactly reproduced in reasonable-size loops in the ATR, and fuel previously irradiated in a commercial reactor can be safely driven to much higher burnup in the ATR and then power-ramp tested. Lead-use assembly (LUA) irradiations of BWR rods will be conducted in selected commercial NPPs, and power ramp testing of BWR fuel may also be done in the ATR. The design-basis accident testing of the ultrahigh-burnup material from the ATR and LUAs will be done in the Transient Reactor Test Facility (TREAT) in Idaho and in the Argonne and Idaho National Engineering and Environmental Laboratory (INEEL) hot cells. Both loss-of-coolant (LOCA) and reactivity-initiated accident (RIA) tests will be conducted. A complete set of results, including appropriate computer models, should be available in about 6 to 7 years.

The development of advanced fuels will be accomplished primarily by the commercial fuel vendors in collaboration with fuel experts from DOE's laboratories. This fuel will also be irradiated in the ATR. The ATR irradiations will continue for about 4.5 years and include

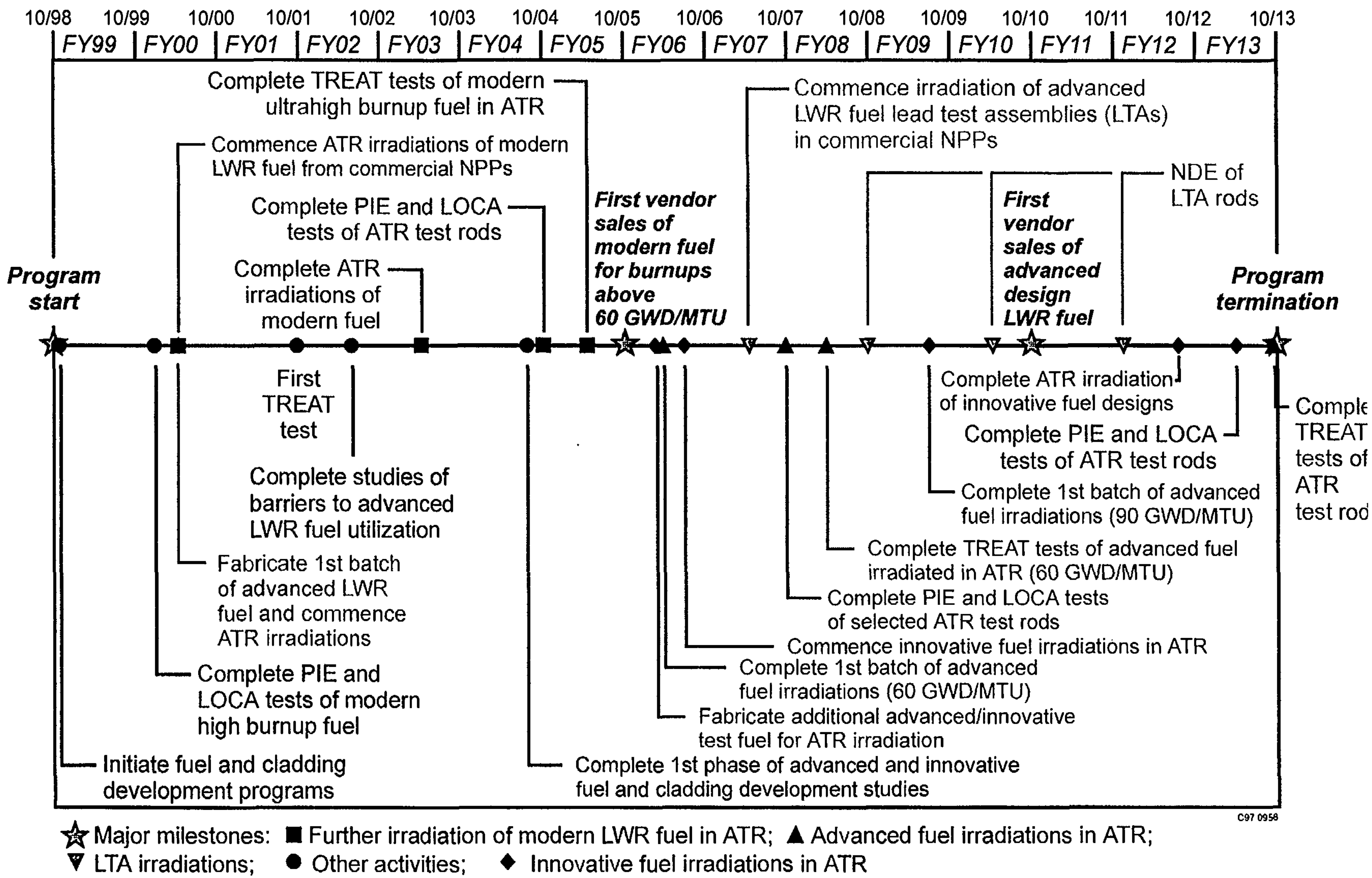

FIG 3. High-Efficiency Nuclear Fuel Program 15-Year Schedule.

power ramp tests at appropriate intervals. Some of the fuel will then be removed and examined, tested in TREAT, and further examined in the Argonne and INEEL hot cells. The remainder of the advanced and innovative fuel designs will continue irradiation in the ATR to ultrahigh-burnups (about 100 MWD/kgU) and then be tested in TREAT and the hot cells. Irradiation of advanced LWR fuel LUAs in commercial NPPs will begin after about 7 or 8 years of development and testing.

In parallel with the irradiation of the advanced design fuels in the test reactors, laboratory research to study the metallurgical and environmental factors that affect the degradation (corrosion and radiation hardening) of the fuel cladding will be conducted. Analysis to evaluate fuel rod designs that better retain the fission products within the fuel, have a more uniform rod internal pressure, and minimize fuel-cladding mechanical interactions will also be conducted.

The products available for use in the commercial nuclear power industry will include:

1. A thorough evaluation of the useful life of the best fuel currently being sold by the participating fuel vendors at the end of 7 years. We expect that a number (maybe all) of the latest product lines can be used at burnups above the current USNRC limit of 62 MWD/kgU.
2. Development and initial testing of advanced ultra-high burnup fuel designs, sufficient for the vendors to start selling lead fuel assemblies at the end of 7 or 8 years.
3. A well-documented, physical understanding of the metallurgical and environmental factors that affect cladding (and assembly structural material) degradation. Improved fuel and fuel assembly predictive models and a solid technical basis for the sale and licensing of advanced fuels will be developed at the end of 15 years.

5. TEST FACILITIES

We assume that data will be needed to license and use LWR fuel to higher burnups from the following types of tests: steady state irradiations, power ramp tests, and design-basis accident tests, primarily LOCA and RIA tests. The irradiations and tests will be designed to demonstrate reliability and, where necessary, determine failure behavior and thresholds. Hot cell examinations before and after testing will document the cladding, fuel, and bundle performance and provide data for modeling. Use of DOE's test facilities are critical to obtain timely and fully representative results, enabling the program to have its maximum benefit. The two key test facilities and the planned transient tests are discussed in the following subsections.

5.1 Extended Burnup Irradiation in the Advanced Test Reactor.

Irradiation in the ATR offers the quickest means to extend burnup beyond the values allowed in a commercial reactor. The ATR core has a serpentine fuel arrangement, which provides nine flux traps. The flux traps in the corner lobes of the serpentine arrangement are almost entirely surrounded by fuel, which allows the power in a single lobe to be adjusted somewhat independently of the rest of the core. The power in a lobe at full reactor power can be maintained at levels of 17 to 60 MW. A pressurized water loop will be installed into the flux trap in the northeast lobe of the ATR, which will be designed to operate at standard PWR conditions (327°C, 15.5 Mpa). The irradiation test space will be approximately 4 inches in diameter, sufficient to accommodate a bundle with 32 17 × 17 PWR-type rods (i.e., a 6 × 6 array without corner positions). A proposed fuel rod arrangement for testing is shown in cross section in FIG. 4, which also shows typical linear power values (in kW/ft) for high-burnup (i.e., >60-MWD/kgU) LWR fuel rods. The proposed arrangement includes two positions for irradiation of PWR guide tubes of advanced design or material composition, which could be instrumented, if necessary, to help verify irradiation conditions in the test bundle.

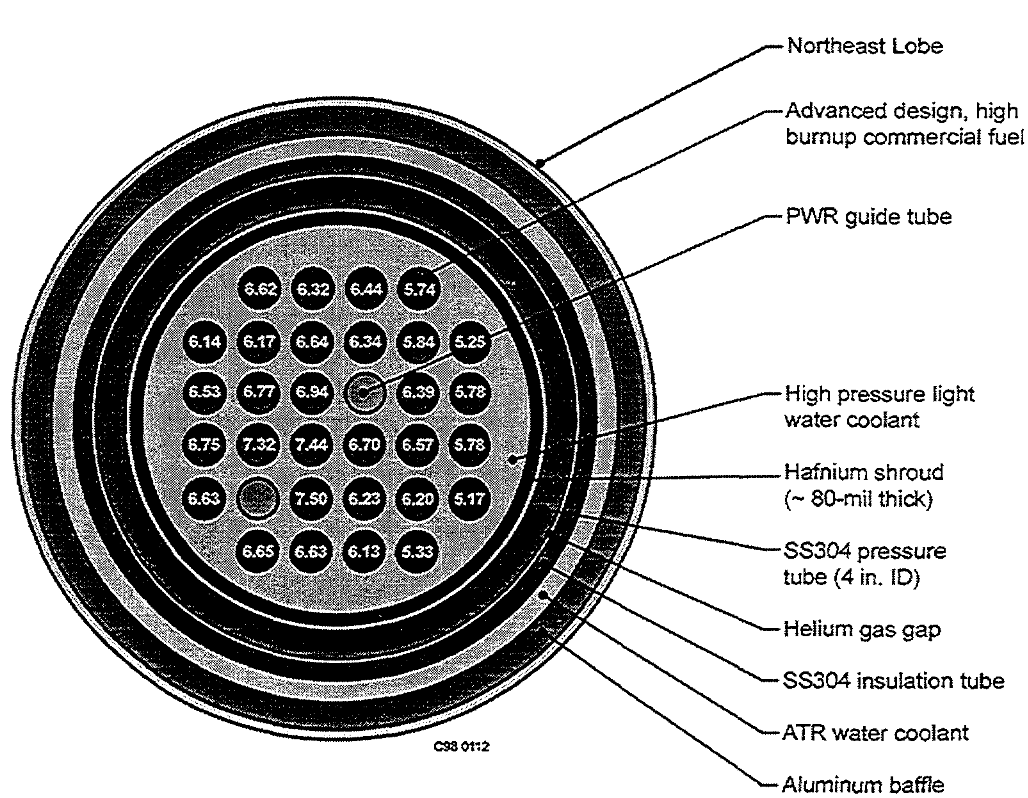

FIG. 4 ATR test bundle cross section. The powers shown were arbitrarily set at a peak of 7.5 kw/ft.

Particular attention has been given to configuring the proposed irradiation vehicle to provide conditions similar to those in a PWR. The loop will be equipped with a pressurizer and purification and makeup water systems, which will allow real-time control of water chemistry, including prototypic, time-dependent variation of boric acid, and lithium hydroxide concentrations through a cycle. The hafnium shroud shown in the cross section of FIG. 4 is included to reduce the total neutron flux in the test bundle to values typical of PWRs, thus allowing prototypic linear power values. As shown in FIG. 5, calculations thus far have shown the neutron energy spectrum in the ATR test bundle to be very similar to that of a PWR. Further work is being performed to tailor the beta and gamma flux in the ATR bundle to that of PWRs, to ensure that the corrosion behavior exhibited by the test rods is the same as expected in a PWR environment.

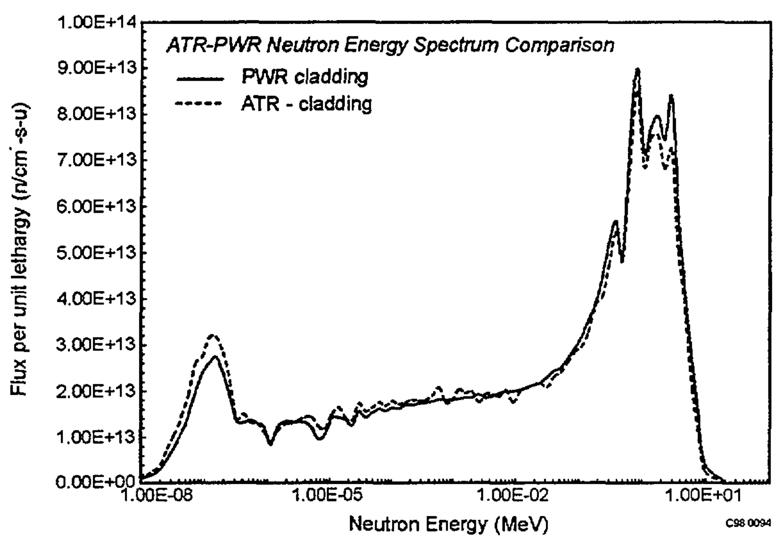

FIG. 5. Comparison of the neutron energy spectrum in typical high-power, four-loop PWR fuel cladding and in the ATR test bundle fuel cladding.

5.2 Transient Tests.

It is anticipated that properly simulated RIA tests of the ultrahigh-burnup fuel developed for this program will be necessary for licensing. Because it is particularly important to perform the RIA tests at prototypic conditions, use of TREAT is proposed. A description of TREAT and a discussion of its suitability for doing this work appear elsewhere [21]. The TREAT RIA tests will be a series of 2- or 3-rod tests at increasing energy insertion until the fuel rod failure threshold is identified. We expect failure of high-burnup (60-MWD/kgU) LWR fuel, with the newer cladding materials and modest zirconium oxide layers, at energy (enthalpy) insertions above that now calculated to be possible in a LWR (i.e., at 150 to 200 cal/g fuel). It is more difficult to predict the failure thresholds for 90-MWD/kgU fuel. But, it is likely that if the new designs and materials perform as well during normal operation as we expect, the failure thresholds during an RIA may still be above what is possible in a LWR core.

The expected outcome of a LOCA strongly depends on flow blockage and cladding embrittlement. Regulatory limits include the 10 CFR 50.46 cladding embrittlement criteria and the Appendix K evaluation models (or best-estimate substitutions). There are also requirements to ensure control rod insertion. Our objective is to show that any flow blockage and cladding embrittlement during an Appendix K LOCA in a modern high-burnup core will be essentially the same as, or less severe than, in a low-burnup core. Therefore, we will conduct oxidation studies and quenching, structural response, and biaxial burst tests. This work can be conducted in hot cells and will be similar to the work the USNRC, EPRI, and DOE are currently sponsoring with older design fuel [20].

6. SUMMARY

Although reemergence of nuclear power as a preferred option for new electric energy generation in the U.S. is a long-term prospect, the problems of global warming, coupled with the uncertain outlook for alternative energy sources and certain global population and energy growth, make the maintenance of the nuclear option in the United States a necessary objective. The proposed High Efficiency Nuclear Fuel Program is part of a larger effort by the DOE to address the problems associated with nuclear power (and identified by PCAST and others) of noncompetitive economics, spent fuel disposal, safety, and nuclear weapons material proliferation. Other DOE programs will address the issues associated with aging management and license renewal of the current plants, generation optimization for the current plants, advanced proliferation-resistant power technologies, etc. The High Efficiency Nuclear Fuel Program is an important part of this overall program and addresses all four of the key problem areas through the development of fuel designs capable of reliable performance to burnups in excess of 60 MWD/kgU.

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