January 6, 2004

Robert H. Leyse
P.O. Box 2850
Sun Valley, ID 83353

Dear Mr. Leyse:

I am writing to inform you that your amended request for rulemaking, submitted on November 28, 2003, fails to meet the minimum requirements for docketing as a petition for rulemaking under 10 CFR Paragraph (§) 2.802.

In an earlier letter, dated September 11, 2003, you had requested that the Nuclear Regulatory Commission (NRC) initiate a rulemaking to develop criteria for analyzing the effect of fouled or corroded fuel elements on the course of reactivity insertion accidents (RIAs). My letter of November 10, 2003, informed you that the original letter did not meet the requirement, stated in § 2.802(c)(1), that a petition for rulemaking must "[s]et forth a general solution to the problem or the substance of text of any proposed regulation or amendment, or specify the regulation which is to be revoked or amended...." I invited you to submit additional information that: (1) identified the section or sections of 10 CFR Part 50 which you believe to be in need of amendment, and (2) indicated how the regulations should be changed in order to address your safety concerns regarding the impact of fuel fouling on the severity of RIAs.

The NRC staff reviewed your amended submission of November 28 and has determined that neither the letter nor the attached supporting material addresses the defects earlier identified with respect to the original petition. The amended request does not identify a specific regulatory deficiency in current Part 50 requirements, and does not indicate what kind of analytic criteria would adequately address your concerns about the impact of fuel fouling on the severity of RIAs. It remains unclear what regulatory action you are requesting of the NRC, and how, in your view, Part 50 should be amended to accomplish that action. As such, your petition, as supplemented, is not eligible for docketing as a petition for rulemaking under § 2.802.

As provided under § 2.802(f), your petition is being returned as insufficient for docketing. This action does not prejudice your right to file a new petition at any time. Any questions about this matter may be directed to Michael Lesar, Chief, Rules and Directives Branch, by calling 301-415-7163 or by e-mail to MTL@nrc.gov.

Sincerely,

/RA/

William D. Travers
Executive Director
for Operations

Enclosure: As stated

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Sincerely,
/RA/

William D. Travers
Executive Director for Operations

Enclosure: As stated

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Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C., 20555-0001
Attn: Rulemakings and Adjudications Staff

November 28, 2003

Petition

1. Substance of Proposed Regulation

The specific regulatory deficiency in 10 CFR Part 50 is the lack of requirements that will limit the severity of Reactivity Insertion Accidents (RIAs) for light-water nuclear power reactors (LWRs) having fouled and/or corroded fuel elements.

2. Petitioner's grounds for and interest in the action requested:

Petitioner is aware that fouling and/or corrosion of fuel elements is significant and ubiquitous among light-water nuclear power reactors. This fouling and/or corrosion increases the severity of RIAs. The fouling and/or corrosion is a heat transfer barrier. This heat transfer barrier inhibits heat transfer from the fuel to the water and thus increases the severity of an RIA. This heat transfer barrier also leads to a substantial increase in the steady state operating temperature of the fuel at the initiation of the RIA and this is another factor that leads to an increased severity of the RIA.

Petitioner is aware that NRC is devoting substantial attention to the impact of high burnup on RIAs. However, there is no corresponding attention to the impact of severe fouling on RIAs even though fouling is ubiquitous in today's LWRs.

Actions required by NRC include:

- **Document the range of fouling and/or corrosion experiences and the fouling and/or corrosion characteristics of the fuel elements in the fleet of USA LWRs.** (On page 41965 of Federal Register: July 16, 2003, the NRC erroneously states, "The operating experience relative to significant crud deposits has been that the observed crud is powdery or fluffy." [Docket Nos. PRM-50-73 and PRM-50-73A] Mr. Robert H. Leyse; Denial of Petition for Rulemaking).
-
- **Publish the failure thresholds and failure consequences of the fuel elements under RIAs for the range of fouling experiences thus documented.** (The energy input that will rupture fuel cladding with fouled fuel elements in RIA events is likely much less than that implied by the criteria in existing regulatory guides. This energy decreases with increasing fouling and/or corrosion. The failure threshold now used by

the NRC for clean fuel is 170 cal/gm, although no loss of fuel is assumed below 280 cal/gm.)

- **Include the applicable fouling and/or corrosion characteristics of the fuel elements in NUREG-0800 requirements.** (For example, Draft Standard Review Plan 15.02, second sentence page 10, "The reviewer should confirm that the description of each accident scenario provides a complete and accurate description of the plant initial and boundary conditions and the accident progression.")

3. Petitioner's statement of the specific issues involved; views with respect to those issues; relevant technical, scientific, or other data involved:

The applicable General Design Criteria , 10, 11, and 28 are found in Appendix A to Part 50 -- General Design Criteria for Nuclear Power Plants, and are as follows:

*Criterion 10 -- Reactor design***.** The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11 -- ***Reactor inherent protection*****.** The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 28 *-- Reactivity limits*. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include **consideration of rod ejection** (unless prevented by positive means), **rod dropout**, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

The following items are extracted from Draft Standard Review Plans 2.2 and 4.4 that purport to implement the General Design Criterion 10. These items include the word crud in bold face because that is the NRC's colloquial reference to fouling.

STANDARD REVIEW PLAN 2.2 *DRAFT Rev. 3 April 1996*

Page 2.2-1
This objective implements ***General Design Criterion 10****, and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDL).*

Page 4.2-3

*The SRXB reviews the thermal margins, the effects of corrosion products (**crud**), and the acceptability of hydraulic loads as part of its review responsibility for SRP Section 4.4.*

Page 4.2-5
*(d) Oxidation, hydriding, and the buildup of corrosion products (**crud**) should be limited, Allowable oxidation, hydriding, and **crud** levels should be discussed in the Safety Analysis Report and shown to be acceptable. These levels should be presumed to exist in paragraphs (a) and (b) above. The effect of **crud** on thermal-hydraulic considerations is reviewed as described in SRP Section 4.4.*

Page 4.2-11

<u>Operating Experience</u>

*Operating experience with fuel systems of the same or similar design should be described, including the maximum burnup experience. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed prior to gaining that experience need not be reviewed. Design criteria for fretting wear, **oxidation**, hydriding and **crud** buildup might be addressed in this manner.*

Page 4.2-11

<u>Prototype Testing</u>

In-reactor testing …

***Crud** formation*

Page 4.2-12

… needed as input to ECCS performance calculations.

<u>Analytical Predictions</u>

(a) Fuel Temperatures (Stored energy): Fuel temperatures and stored energy during normal operation are needed as input to ECCS performance calculations.

*Thermal conductivity of the fuel, cladding, **cladding crud**, and **oxidation** layers*

*Cladding **oxide** and **crud** layer thickness Cladding-to-coolant heat transfer coefficient*

Page 4.2-15

Post Irradiation Surveillance

For a fuel design like that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel rod failure, rod bowing or ***crud*** *deposition. There should also be a commitment in the program to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures. The surveillance program should also address the disposition of failed fuel.*

Pages 4.2-15 and 4.2-16

Technical Rational

... 10 CFR 50, 50.46 significantly reduces the possibility of a violent chemical reaction occurring between the Zircaloy cladding and the coolant ...

3. ... Compliance with GDC 10 significantly reduces the likelihood of fuel failures during normal operations or anticipated operational occurrences, thereby minimizing the possible release of fission products.

STANDARD REVIEW PLAN 4.4 *DRAFT Rev. 3 April 1996*

Page 4.4.8
The effects of ***crud*** *should be accounted for in the thermal-hydraulic design by including it in the CHF calculations in the core or in the pressure drop throughout the RCS. Process monitoring provisions should insure the*[45] *capability for the detection of a three percent*[46] *drop in the reactor coolant flow. The flow should be monitored every 24 hours.*

Page 4.4-9
The reviewer ensures that adequate account is taken of the effect of ***crud*** *in the primary coolant system, such as in the calculation of CHF in the core, heat transfer in the steam generators, and pressure drop throughout the RCS.*

Although fouling and or corrosion of fuel elements is addressed in these DRAFT Standard Review Plans 2.2 and 4.4, these DRAFT plans are grossly deficient in quantifying any aspect of fouling that impacts RIAs. The review plan on Page 4.2-12 of DRAFT 2.2 calls for inputs "... needed as input to ECCS performance calculations." Then it lists the following: "Analytical Predictions (a) Fuel Temperatures (Stored energy): Fuel temperatures and stored energy during normal operation are needed as input to ECCS performance calculations. Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers. Cladding oxide and crud layer thickness Cladding-to-coolant heat transfer coefficient." The

reference to ECCS performance calculations at least may force reference to the 17% oxidation limits of Appendix K and/or 50.59. However the 17% limit is grossly excessive in limiting the severity of RIAs.

Thus, in order to satisfy Criterion 10 of the General Design Criteria for Nuclear Power Plants, the NRC must quantify acceptable values for cladding-to-coolant heat transfer coefficients in limiting the severity of RIAs. The cladding must be defined as cladding that remains as metal. This means that the cladding-to-coolant heat transfer coefficient must include the resistance in series of the oxide layers in the tubing, the fouling deposits on the surface of the tubing and other factors such as porosity and delaminations within or between layers of oxide and/or fouling.

Furthermore, the quantified acceptable values should be based on certified data that is reported without disclaimers. (Examples of deficient certification from Appendix K: Example 1, On page 58 of ANL-6548, Baker and Just disclose Part VII. APPLICATION TO REACTOR HAZZARDS ANALYSIS*. The asterisk leads to the following footnote:
<u>*This discussion is of a preliminary nature: work in this area is continuing.</u>
Example 2, The introductory page to NUREG-17 includes the following warning:
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the Energy Research and Development Administration/United States Nuclear Regulatory Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.
Example 3, The cover page to WCAP-7665 includes the following warning:
***This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.*) (The next example, Example 4, is not in Appendix K, however it is under review by NRC. EPRI Report 1002865, *Topical Report on Reactivity Initiated Accidents: Bases for RIA Fuel Rod Failures and Core Coolability Criteria* has a three paragraph DISCLAIMER OF WARANTIES AND LIMITATION OF LIABILITIES.)**

Returning to the matter of cladding-to-coolant heat transfer coefficients, Mr. Deshon of EPRI referred to "boiling chimneys" during his presentation to the ACRS Reactor Fuels Subcommittee, September 30, 2003. On page 132, Deshon asserts that these boiling chimneys enhance heat transfer from the cladding to the coolant when the thickness of the fouling is up to a thickness of 20 microns. Next, on page 133 refers to a flake with a thickness of 125 microns with "... very large voids in

the crud, representing these boiling chimneys." Now, it is unlikely that a chimnied layer having a thickness of 20 microns will enhance the heat transfer from the cladding to the cooling water, and it is very unlikely that a porous layer of 125 microns will be anything other than a significant barrier to heat transfer. Deshon presented no experimental data to prove the enhancement of heat transfer. Here is an example of conjecture, rather than certified heat transfer data, leading to conjectures that the ACRS appeared to accept based on assertion rather than certified test data.

Following are examples of documentation that is relevant to the evaluation of Reactivity Insertion Accidents (RIAs). In no case is there any disclosure that accounts for the impact of fouling and/or corrosion of fuel elements on the course of RIAs. Some of the examples acknowledge the existence of fouling and/or corrosion, but none disclose that fouling and/or corrosion has the potential to become a substantial heat transfer barrier that increases the severity of RIAs to an extent that obviates compliance with Criterion 10 of the General Design Criteria for Nuclear Power Plants.

I. On June 12, 2002, the Nuclear Energy Institute provided the NRC with EPRI Report 1002865, *Topical Report on Reactivity Initiated Accidents: Bases for RIA Fuel Rod Failures and Core Coolability Criteria.* This report purports to provide, "Revised acceptance criteria (that) have been developed for the response of light water reactor (LWR) fuel under reactivity initiated accidents (RIA). Development of these revisions is part of an industry effort to extend burnup levels beyond currently licensed limits. The revised criteria are proposed for use in licensing burnup extensions or new fuel designs."

II. Following is NRC's description of its research on high burnup fuel:

HIGH-BURNUP FUEL RESEARCH

NRC's research on high-burnup fuel is focused on postulated events that involve significant fuel damage. These events have the potential to cause fuel melting, if not kept within certain bounds, and melting could produce a large fission product release and significant consequences (hence risk). While there may be many pathways leading to such events, there are only two ways to melt fuel. One is to lose the coolant and the other is to get excessive power in the fuel. Selected design-basis accidents are postulated to serve as bounding examples of these kinds of events, and fuel damage limits are used to ensure that coolable core geometry is not lost, thus avoiding significant consequences.
Two main design-basis accidents are postulated to provide this bounding protection. One is a reactivity accident and the other is a loss-of-coolant accident (LOCA). Particular varieties of these design-basis accidents are evaluated for PWRs and BWRs. The fuel damage limits that are supposed to ensure a coolable core are 280 cal/g peak fuel enthalpy (Regulatory Guide 1.77) for the reactivity accidents and 17% cladding oxidation and 2200^{o}F peak cladding temperature for the loss-of-coolant accidents (10 CFR 50.46). Related evaluation models are used to demonstrate that the regulatory criteria are not exceeded.

A few years ago, we learned from experimental work on reactivity accidents in France and Japan that cladding failure accompanied by fuel dispersal (loss of geometry) could occur at fuel enthalpies below 100 cal/g for high-burnup fuel. This is far below the current regulatory criterion. Further, the failures were occurring by a brittle mechanical fracture mechanism rather than the high temperature process seen in the original studies on fresh fuel, from which the current criterion was derived. In addition, we learned from work in Russia that different cladding alloys may behave in a completely different manner depending on their ductility. To complicate matters, the U.S. industry is now using three distinctly different PWR cladding alloys and two different BWR cladding types, all of which may have different mechanical properties in their highly irradiated state.

Early experimental work in France on loss-of-coolant accidents
does not show that the oxidation and peak cladding temperature limits are inadequate, but the situation is very clouded. In extreme cases, fuel rods in commercial reactors have accumulated nearly 17% cladding oxidation during normal operation. The extent to which this reduces the amount of oxidation that can be tolerated during the accident is not known, and thus the NRC is currently taking the conservative position that the sum of transient and steady-state oxidation should be limited to 17%. Even if the oxidation and peak cladding temperature "embrittlement criteria" of 10 CFR 50.46 can be shown to be adequate at high burnup for all cladding types, it is likely that the evaluation models for oxidation, ballooning, and rupture will be affected.

Fortunately, design-basis accidents are unlikely and current analyses generally contain conservative margins. Therefore, in its high-burnup program plan, the NRC concluded that there was time to resolve these issues with long-range research programs (3-5 years). To accomplish this, we engaged in a number of formal agreements to gain access to international programs and we initiated some of our own work. A list of current NRC research activities on high-burnup fuel is shown below.

1. *ANL (NRC) Hot Cell LOCA Tests of Fuel Rods and Mechanical Properties of Cladding*
2. *PNNL (NRC) Steady-State and Transient Fuel Rod Codes and Analysis*
3. *BNL (NRC) Neutron Kinetic Codes and Analysis of Plant Transients*
4. *Halden (Norway) Reactor Tests of Fuel Rods in Steady State and Mild Transients*
5. *Cabri (France) Reactivity Accident Tests of Fuel Rods and Related Programs*
6. *NSRR (Japan) Reactivity Accident Tests of Fuel Rods and Related Programs*
7. *IGR (Russia) Reactivity Accident Tests of Fuel Rods and Related Programs*

Information on the test program at Argonne National Laboratory (ANL) can be found at http://www.et.anl.gov/aghcf. Information on code development at Pacific Northwest National Laboratory (PNNL) can be found at http://www.nrc.gov/RES/FRAPCON3. Recent information on the other programs was presented at the NRC Water Reactor Safety Information Meeting on October 26, 1999, and can be found in the meeting transactions, NUREG/CP-0168 (October 1999), starting on p. 37.

NRC's research on high-burnup fuel is focused on postulated events that involve significant fuel damage. These events have the potential to cause fuel melting, if not kept within certain bounds, and melting could produce a large fission product release and significant consequences (hence risk). While there may be many pathways leading to such events, there are only two ways to melt fuel. One is to lose the coolant and the other *is to get excessive power in the fuel. Selected design-basis accidents are postulated to serve as bounding examples of these kinds of events, and fuel damage limits are used to ensure that coolable core geometry is not lost, thus avoiding significant consequences.*

Acceptance criteria for extended fuel burnup *Increasing economic: competition among electric power suppliers is causing the nuclear power industry to pursue longer operating cycles with longer use of fuel elements before they are replaced (called higher burnup). Higher burnup leads to certain changes in fuel characteristics including higher cladding oxidation, that leads to embrittlement of the cladding and, higher fuel pellet fission gas release, which leads to higher internal fuel cladding pressure. A number of fuel damage criteria are used as limits in core-reload licensing, and the adequacy of these limits must be reestablished for higher burnup fuel. NRC regulatory criteria on fuel behavior were developed when the burnup limit was about 40 Giga Watt-day/metric tonne (Gwd/t) and it was thought that some extrapolation of fuel behavior with higher burnup could be made. However, international test data on reactivity-initiated accidents, along with observed increases in the rate of cladding corrosion at higher burnup, suggest that establishing regulatory acceptance criteria on the basis of extrapolated data is not consistent with the experimental evidence. Accordingly, transient fuel performance data at higher burnups is needed to confirm current acceptance criteria. In addition to confirming the adequacy of fuel behavior under the currently approved burnup limit of 62 GWd/t, the staff must be prepared to respond in a timely fashion to industry plans for further burnup extensions. During FY 2002, cooperative agreements with foreign countries will continue to provide the relevant data to properly assess the behavior of high burnup fuels. The data will be used to maintain safety by either confirming the adequacy of existing regulatory requirements or providing the basis for revising these requirements. The test and analysis results from this research will provide the bases needed for both confirming existing fuel burnup limits and approving anticipated requests by the nuclear power industry to further increase fuel burnup limits. This research will result in better models for predicting fuel behavior under accident conditions and will ensure that appropriate safety limits for high burnup fuels are established and maintained. The outcomes of the research will be more realistic NRC safety decisions, the elimination of important knowledge gaps, and a reduction in the uncertainty of NRC decisions aimed at maintaining safety of high burnup fuels. This is a several year research program dependent on cooperative agreements and use of foreign test facilities. Thus, some results will not be obtained until 2004. Click here for more details on the Extended Fuel Burnup*

III. ACRS Focuses on High-Burnup Fuel, Ignores Impact of Fouling

The Honorable Richard A. Meserve
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
Dear Chairman Meserve:

SUBJECT: CONFIRMATORY RESEARCH PROGRAM ON HIGH-BURNUP FUEL

During the 496th meeting of the Advisory Committee on Reactor Safeguards, October 10-12, 2002, we met with representatives of the NRC's Office of Nuclear Regulatory Research (RES) to discuss their confirmatory research program on high-burnup fuel, as well as research they do to support safety regulation of dry cask storage of spent fuel including high-burnup fuel. We also met with representatives of the NRC's Office of Nuclear Reactor Regulation to discuss their plans to review an EPRI topical report on the response of high-burnup fuel to reactivity insertion events. Our Subcommittee on Reactor Fuels met on October 9, 2002, to review these topics in detail and to discuss with representatives of EPRI their work to define fuel failure criteria and coolability criteria for high-burnup fuel exposed to reactivity transients. We also had the benefit of the referenced documents.

CONCLUSIONS

- ***RES has a well-organized and leveraged program of confirmatory research on the behavior of high-burnup fuel under the conditions of reactivity insertion events in pressurized water reactors, design-basis loss-of-coolant accidents (LOCAs), and anticipated transients without scram in boiling water reactors. RES has also undertaken research on creep of high-burnup fuel cladding to support safety regulation of dry cask fuel storage.***
- ***A consensus has emerged that the energy input that will rupture fuel cladding in a reactivity insertion event is much less than that implied by the criteria in existing regulatory guides and decreases with increasing fuel burnup at least above 40 GWd/t.***
- ***RES is nearing resolution of the issues of reactivity insertion events in high-burnup fuel and has initiated experimental investigations of high-burnup fuel under conditions of design-basis LOCAs. We remain concerned that the time-temperature conditions used in the study of high-burnup fuel during design-basis LOCAs may not reveal phenomena unique to high-burnup fuel.***

DISCUSSION

There are economic and societal incentives to use nuclear fuel to higher levels of burnup. Burnup levels now approved exceed the data bases underlying the models that are used to predict fuel behavior under upset and design-basis accident conditions. French and Japanese tests of high-burnup fuel have shown cladding failure and even fuel dispersal during reactivity insertions at energy levels substantially below the criteria found in Regulatory Guide 1.77.

RES has undertaken a research program to confirm that the current limit on fuel burnup (62 GWd/t) ensures adequate protection of the public health and safety. A research program of experimental and analytic research involving the collaboration of NRC, EPRI, and numerous foreign partners has been organized. Risk-informed methods have been used to select issues of high-burnup fuel to investigate. The program addresses high-burnup fuel behavior under conditions of design-basis LOCAs and boiling water reactor anticipated transients without scram, as well as reactivity insertion events in pressurized water reactors.

RES has upgraded the fuel behavior computer code (FRAPTRAN) and neutron transport code (PARCS) available for regulatory analysis of high-burnup fuel. It has also completed detailed phenomena identification and ranking studies for high-burnup fuel under a variety of conditions. In addition, RES has participated with its foreign

partners in the continued experimental study of reactivity transients in high-burnup fuel.

Analyses of data on high-burnup fuel behavior during reactivity transients have progressed in many quarters, including within the RES program and independently by EPRI. It is now broadly accepted that the energy input necessary to fail fuel in a reactivity transient is much less than the criterion in Regulatory Guide 1.77. At least for burnups greater than 40 GWd/t, the energy needed to fail fuel decreases with increasing fuel burnup. This sensitivity to sudden energy inputs is thought to be attributable to embrittlement of the fuel cladding. RES is also showing with realistic analyses that design-basis reactivity transients do not produce energy inputs of the magnitude and speed necessary to fail cladding embrittled at burnups less than 62 GWd/t.

RES anticipates that with the aid of 2 or 3 additional inpile tests in France's CABRI reactor and tests at elevated temperatures in Japan's NSRR, it will be able to quantitatively characterize the degradation of the capacity of fuel to sustain sudden energy inputs with increasing fuel burnup. RES is pursuing both empirical and mechanistic pathways to develop this characterization.

Controversy still exists within the reactor fuel community on whether distinct burnup-dependent criteria should be developed for fuel cladding rupture and for the energy input sufficient to cause loss of coolable configuration of the fuel. RES currently supports a single criterion for fuel failure that would also be conservative for coolability, whereas EPRI has proposed the continued use of separate criteria.

Recently, RES initiated out-of-pile tests of individual fuel rod segments under conditions of design-basis LOCAs. The objective is to replicate with high-burnup fuel the investigations of fresh cladding behavior that were the bases for the so-called "embrittlement" criteria specified in 10 CFR 50.46 and Appendix K. These tests involve monotonic heatup of fuel rods to a limiting temperature (2200 F) and monotonic cooling and quenching. We remain concerned that other safety-significant phenomena, such as spallation of pre-existing oxide from the cladding, may be important for high-burnup fuel and may be revealed only when more complicated, and realistic, time-temperature conditions are used in the tests. The single rod segment tests will not reveal features of high-burnup fuel behavior that arise when multiple rods are present.

The behavior of high-burnup fuel during anticipated transients without scram has become a topic of particular interest as power uprates have shortened the time available for operators to respond to the transients in boiling water reactors. Analyses done to date show that high energy inputs can occur during power oscillations in these transients, but the power inputs occur too slowly to produce intense pellet-clad mechanical interactions that threaten cladding integrity. This analytic finding appears to be substantiated by a recent test in Japan's NSRR.

We conclude that the confirmatory research program for high-burnup fuel is well-designed and is making good progress in light of the challenges of inpile and out-of-pile tests with fuel irradiated to high levels of burnup. We remain supportive of this program.

We recognize that this confirmatory research program is not addressing the risk consequences of taking fuel to high levels of burnup. These consequences will be examined in planned studies of high-burnup fuel in beyond design-basis accident conditions. We look forward to hearing about RES plans to explore this important aspect of high-burnup fuel in nuclear power plants.
Dr. William J. Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

George E. Apostolakis
Chairman

References:

1. *Memorandum dated July 6, 1998, from L. Joseph Callen, Executive Director for Operations, NRC, to the Commissioners, Subject: Agency Program Plan for High Burnup Fuel.*
2. *Nuclear Energy Institute, EPRI Report 1002865, "Topical Report on Reactivity Initiated Accidents: Bases for RIA Fuel Rod Failures and Core Coolability Criteria," June 12, 2002.*
3. *U. S. Nuclear Regulatory Commission, NUREG/CR-6742, "Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel," August 2001.*
4. *U. S. Nuclear Regulatory Commission, NUREG/CR-6743, "Phenomenon Identification and Ranking Tables (PIRTs) for Power Oscillations Without Scram in Boiling Water Reactors Containing High Burnup Fuel," August 2001.*
5. *U. S. Nuclear Regulatory Commission, NUREG/CR-6744, "Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel," August 2001.*
6. *U. S. Nuclear Regulatory Commission, NUREG/CR-6739, Vol. l, "FRAPTRAN: A Computer Code for the Transient Analysis of Oxide Fuel Rods," August 2001.*
7. *U. S. Atomic Energy Commission, Regulatory Guide 1.77, "Assumptions Used for Evaluating A Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.*
8. *Letter dated March 14, 2002, from George E. Apostolakis, ACRS Chairman, to William D. Travers, Executive Director for Operations, NRC, Subject: Confirmatory Research Program on High-Burnup Fuel.*

IV. Following is the extensive EPRI press release ; the Robust Fuel Program:

An industrywide, EPRI-led strategic initiative—the Robust Fuel Program—is playing an important technical support role in the nuclear power industry's response to recent fuel performance problems.

The low cost of nuclear fuel is a key factor that makes nuclear power plants economically competitive with other low-cost sources of electricity generation. Yet fuel-related problems can have a direct impact on nuclear plant operations and adversely affect a plant's cost of producing electricity. By extending the burnup (energy extraction) and cycle life of nuclear fuel, companies that operate the nation's nuclear power plants can save as much as $300 million a year in reduced fuel purchase costs and costs for spent fuel storage, transportation, and disposal.

Despite recent gains in plant performance and production, continued efforts by the nuclear power industry to strive for defect-free performance of nuclear fuel have not been fully successful in preventing problems with fuel and reactor core performance in recent years. The emerging trend is drawing increasing attention and visibility within the industry, reflected in a series of Significant Operating Experience Reports issued to reactor owners by the Institute of Nuclear Power Operations (INPO), the most recent in June 2003.

Follow-up discussions organized by INPO with industry leaders and technical experts are defining specific actions that industry organizations such as EPRI, INPO, the Nuclear Energy Institute (NEI), electric utilities, and nuclear fuel vendors should take to address fuel performance issues. In addition to the impact that fuel performance challenges can have on the economic competitiveness of nuclear plants, the industry is also mindful of the potential safety significance. Fuel cladding constitutes the innermost of three engineered barriers to prevent the release of fission products.

High-tech, but R&D cut

Driving the more demanding operating conditions for nuclear fuel are increasing burnup of discharged fuel, longer operating cycles, increased fuel peaking and enrichment, plant uprates, new water chemistry environments, fuel design changes, and the introduction of new materials. Taken together, these conditions have in some cases resulted in reduced operating margins and increased uncertainties regarding fuel performance problems, says Rosa Yang, an EPRI senior technical leader who manages the Robust Fuel Program, an industry-wide strategic initiative launched in 1998.

"Historically, nuclear fuel has operated very well, and when you've had something operate well, you tend to want it to operate even better, especially in a restructured electricity market environment. Consequently, companies that operate nuclear power plants are driving the fuel harder today than was the case 10 to 15 years ago," Yang explains. "Nuclear fuel is a high-technology product, but because of excess manufacturing capacity and extremely competitive pricing, fuel manufacturers have limited resources to perform the desired research and development (R&D) to meet the demands of the new, more aggressive operating environment.

"The nuclear fuel industry needs to be doing more R&D to develop better products," Yang adds. "We have new fuel products, but sometimes we get into trouble when new products are not adequately tested."

Trends cross reactor types

Fuel performance problems have occurred with both pressurized water reactors (PWRs) and boiling water reactors (BWRs), albeit with different timelines and differences related to core and fuel designs.

PWRs, many of which went to higher initial fuel enrichment levels (4.5% and higher) in the mid-1990s, have experienced varying degrees of axial offset anomaly (AOA), or a downward shift of the reactor core power profile, as a result of a buildup of corrosion-product deposits commonly known as crud on fuel surfaces that, in turn, is caused by increased fuel duty. EPRI has successfully demonstrated at two PWR plants an ultrasonic method for removing crud deposits to mitigate AOA. Although occurrences of AOA persist, its frequency and severity are down significantly from recent trends. Grid-rod fretting—the metal spacers that separate individual rods in fuel assemblies—continues to be the main root cause of PWR fuel failures.

For BWRs, the principal industry concern today is the relatively large increase in the number of reported fuel failures beginning in 2002, after more than a decade of steadily decreasing failure rates. Over a 17-month period from January 2002 to May 2003, 19 BWRs reported fuel failures and 10 were operating with failed fuel in June 2003. At some BWRs, excessive bowing of the fuel channels can increase control blade insertion times. The industry is investigating a number of open technical issues posed by the BWR fuel failures. Although nuclear plants are designed to operate with some failed fuel, plant owners pursue a constant operating goal of keeping personnel exposure to radiation as low as reasonably achievable.

For both reactor types, the role of new water chemistry regimes introduced to lower in-plant radiation exposures to plant personnel or to support extended plant operation are being assessed with respect to fuel performance. For PWRs, these regimes include elevated coolant pH using lithium and zinc injection and for BWRs include the addition of noble metal chemicals and zinc.

Supporting industry's response

EPRI's Robust Fuel Program (RFP) is playing an important technical support role in the nuclear power industry's response to recent fuel performance problems. The program's 19 U.S. and nine international nuclear plant-operating member companies collaborate closely with INPO, NEI, the U.S. Nuclear Regulatory Commission's Office of Research, and the five major nuclear fuel manufacturers, both on an executive committee level and through four technical working groups (fuel and water chemistry; response to transients; high burnup properties to ensure margins; and failure mechanisms and mitigation).

The focus of EPRI's RFP spans fundamental research on fuel and cladding; poolside and hot cell examinations to assess margins of modern high-duty PWR and BWR fuels; fuel failure root cause investigation; plant demonstrations of mitigation measures; development of non-destructive fuel inspection techniques; assessing the impact of water chemistry changes on fuel performance; and proactively addressing regulatory and licensing issues.

As part of the program's efforts to investigate the root cause of fuel failures, two member company shipments of failed rods to shielded, hot cell examination facilities are anticipated in the fourth quarter of 2003, and the program stands ready to assist additional member companies in such investigations. As might be expected, the planning and logistics of such investigations can become complicated affairs.

The program is currently developing ultrasonic cleaning technology (UT) for BWR fuel, modifying the technology successfully used for PWRs. Once tests and evaluation to address BWR-specific issues are completed, UT technology for BWR applications is expected to be available in early 2004.
The RFP is proactively collecting data from hot cell examinations to ensure adequate operating margins under high duty conditions. The program typically has spent 20%-30% of program funding to obtain such data, because no data had been available for high-duty fuel. The data are crucial to ensure fuel reliability and performance margins, and can also be used for burnup extension.
"EPRI's Robust Fuel Program provides an efficient way of obtaining industry fuel performance data and developing means for improving fuel performance and reliability," says Yang. The program promotes increased awareness of fuel-related changes in plant operations by urging members to insist on adequate testing of all design and manufacturing changes; evaluate the impact of operating and chemistry changes; and perform fuel inspections to confirm the impacts.
Yang notes that INPO cited EPRI's Robust Fuel Program Technical Requirements for Nuclear Fuel Performance (110689) in its June SOER as a useful guide for evaluating such changes. INPO recommends that utilities conduct periodic assessments to determine the strength of fuel vendor processes.
"The current U.S. fleet of 103 nuclear power plants provides 20% of the nation's electricity with no pollutant emissions or greenhouse gases," notes EPRI Chief Nuclear Officer Dave Modeen. "It's critical that the nuclear industry stays on top of fuel performance issues that can directly affect nuclear plants' economic competitiveness as a highly reliable, safe, and emissions-free source of low-cost electricity."
For more information, contact Rosa Yang, ryang@epri.com, 650-855-2481.

V. A corrosion product flake from PWR fuel exceeded 100 microns in thickness.

*See Nuclear Energy, 2001, **40**, No. 2, Apr., 123-135 "Axial offset anomaly: coupling PWR primary chemistry with core design," Frattini, P. L. and others.*
This report discloses that more than 20 US cores have exhibited axial offset anomaly (AOA). "The root cause of AOA is corrosion product deposition in the upper spans of fuel assemblies as a result of sub-cooled nucleate boiling."

VI. NRC Information Notice 97-85 clarifies Axial Offset Anomaly (AOA):

Axial offset (AO) is a measure of the difference between power in the upper and lower portions of the core. This difference must remain within limits established in the technical specifications to ensure that both SDM and clad local peaking factors are not exceeded. Exceeding these limits could result in the reactor fuel exceeding 10 CFR 50.46 limits on fuel clad temperature (1204C). If the reactor approaches these limits, compensatory measures, including a power reduction, must be taken to maintain the reactor within its operational limits.

VII. *Following is an EPRI press release that describes Ultrasonic Fuel Cleaning:*

Palo Alto, Calif. —March 19, 2003 —A new ultrasonic nuclear fuel cleaning technology developed by EPRI removes deposits from reload fuel, allowing higher fuel utilization and reducing worker exposure rates. The process can also lower feed fuel costs and holds promise for additional savings related to spent fuel inventory.

Nuclear plant operators can use higher-enrichment fuel assemblies and longer burn-up cycles to increase megawatt-hour output from reactor cores. However, extended fuel duty can increase the buildup of corrosion-product deposits, shifting the power profile toward the bottom of the core in the phenomenon known as axial offset anomaly (AOA). Severe cases of AOA can require derating of a plant late in the fuel cycle to maintain a safe shutdown margin.

"Removing the corrosion products before reloading the fuel can reduce the risk of AOA in the subsequent fuel cycle, and the lowered corrosion product deposits will reduce the amount of activated material that would otherwise contribute to personnel radiation exposure," says EPRI's Paul Frattini, who co-invented the process and managed its initial development. The EPRI-patented method has been used successfully at AmerenUE's Callaway plant and at the South Texas Project Unit 2 pressurized-water-reactor plants.

According to Ameren's Gail Gary, the core at the Callaway plant remained free of AOA throughout the fuel cycle for the first time in the eight most recent cycles after one fuel cycle in which all reload fuel was ultrasonically cleaned.

"We are including the EPRI fuel cleaning process in an aggressive program of AOA mitigation," Gary said. "We expect to be able to eliminate a 4 ℉ average core temperature reduction imposed earlier as a precaution to minimize AOA." Each degree of recovered core temperature restores approximately 4.5 MW of generating capacity over a fuel cycle.

While AOA has not been a problem for the South Texas Project, the utility purchased ultrasonic fuel cleaners for each of its two units as a proactive measure for corrosion product control after replacing steam generators and uprating both units. All reload fuel for the Unit 2 reactor was cleaned in October 2002. Reload fuel for Unit 1 will be cleaned in April 2003. STP calculates that the fuel cleaning will allow higher fuel utilization and achieve the benefits of lowered worker dose rates.

"STP will save over $1 million per core in fuel costs because each core can be loaded with four fewer feed assemblies," said Dave Hoppes, STP fuel engineering supervisor. Hoppes projects an added savings of $250,000 per cycle in present-value dollars as a result of reduced spent fuel inventory and related handling, storage, and disposal costs. He adds that preliminary results promise reduced in-plant radiation levels, producing significant additional long-term dollar benefits.

For more information on the ultrasonic fuel cleaning process, contact Paul Frattini, at 650-855-2027 or pfrattin@epri.com.

EPRI, headquartered in Palo Alto, Calif., was established in 1973 as a non-profit center for public interest energy and environmental research. EPRI's collaborative science and technology development program now spans nearly every area of power generation, delivery and use. More than 1,000 energy organizations and public institutions in 40 countries draw on EPRI's global network of technical and business expertise.

Note: Although ultrasonic fuel cleaning has apparently been under development for several years and has been utilized for at least two years at licensed nuclear power reactors, the process was not openly discussed with the NRC prior to the September 30, 2003, meeting of the ACRS fuels subcommittee.

VIII. Fouling and corrosion at PWRs in Hungary (Paks Units 1, 2 & 3).

In a May 2003 report to the Chairman, Hungarian AEC, the extensive fouling of the Paks units is candidly discussed. There is no description of the thermal resistance of the fouling or the amount of zircaloy corrosion. However, the fouling (magnetite)) has been extensive. Quoting, "...magnetite deposits in the fuel assembles increased and the cooling water flow-rate decreased. Consequently the power of Units 1-3 had to be decreased."

Chemical cleaning of fuel elements in batches of seven elements became routine. In 2002, Framatome ANP expanded the cleaning process to 30 element batches.
On 10 April 2003, while the assemblies were being cleaned for Unit 2, severe damage occurred to an entire batch. The state of the fuel prior to the accident has not been disclosed. But as this data including the extent of fouling become available, it is likely that analysis will yield further insights on the impact of fouling on severe accidents.

Following is the full extent of reporting available to the public of the NRC's investigation of the Paks-2 event.

Region II
Items of Interest
Week Ending June 20, 2003
International Atomic Energy Agency (IAEA) Mission To Hungary
On June 16-27, 2003, the Director, Division of Reactor Safety, was on International travel in Hungary to support an IAEA Mission to a PAKS reactor that recently experienced fuel damage and a small radioactive release during refueling operations.

In contrast to the mild reporting by the USNRC, V. Asmolov, the Director of the Kurchatov Institute observed, *"... it was a hand-made accident caused by those who, mildly speaking, clumsily thrust where they shouldn't. This is a precious experience."*

IX. Extensive fouling and corrosion at River Bend Station Unit 1 (1998)

This is reported in licensee event report (LER 50-458/99-016-00). Multiple fuel pin failures were attributed to "...an unusually heavy deposition of crud on the fuel bundles." It was, "Determined that an insulating layer of crud caused accelerated fuel rod corrosion." There is no quantitative disclosure of the effective thermal conductivity of the insulating layer of crud. It is disclosed that "Measured zircaloy oxide thickness on high power unfailed HGE bundles was up to 6 mils at the 50" level where the perforations occurred." However, there has been no public disclosure of the measured zircaloy oxide thickness on the failed HGE bundles. And there has been no public disclosure of the combined thermal resistance of the thick zirconium oxide layers and the extensive fouling deposits.

X. Extensive fouling and corrosion at River Bend Station Unit 1 (Early 2003)

The River Bend experience reported in licensee event report (LER 50-458/99-016-00) was apparently repeated during 2002 and 2003. This was revealed by Cheng of EPRI at the meeting of the ACRS Reactor Fuels Subcommittee, September 30, 2003.

On page 246 of the transcript, lines 17-22, Cheng of EPRI states, "But in 1988, we had one plant that had a very significant crud, very heavy crud deposit, that caused a failure. This plant experienced the same type of failure just sometime this year, early part of this year. So we had a repeat of this similar failure this year by similar mechanisms." On page 247, lines 15 and 16, the 1998 plant is identified as River Bend.

On page 253, beginning on line 18, a description of the River Bend failures continues: "The crud was so heavy, some of them between the two, they would almost touch each other. So it almost crossed the gap in some of the rods. On page 255, Line 15, "This crud was very tenacious."

XI. Early experience at the Experimental Boiling Water Reactor (Report ANL - 6136)

The Experimental Boiling Water Reactor (EBWR) was designed and operated by Argonne National Laboratory during the late 1950s and early 1960s. An unfortunate selection of aluminum alloy for core filler pieces led to deposits of hydrated alumina on the zirconium clad fuel elements. Thickness of the fouling was 0.013 cm, the thermal conductivity was 0.008 W/cm-C; thus the heat transfer coefficient was 0.6 W/(cm2)(C). The peak heat flux in today’s large light water reactors is in the range of 150 W/cm2 and the temperature gradient for EBWR-type fouling would be 250 C. However, the heat transfer coefficient for the combined fouling and zircaloy oxide of today's units is likely substantially less than the EBWR case.

XII. Early experience at the Argonne Low Power Reactor (SL-1)

The SL-1 was destroyed in a Reactivity Insertion Accident (RIA) on January 3, 1961. Fouling of the aluminum clad fuel plates likely intensified the severity of the accident. However, fouling was not considered by the analysts who investigated this RIA. Here is a quote from GE Report, Additional Analysis of the SL-1 Excursion, Report IDO-19313, 1962: "The thickness of the cladding has an important effect on the magnitude of the excursion. Because of the extremely short period, this 0.89 mm cladding became an effective thermal insulator and impeded the flow of heat to the reactor water where it could initiate shutdown of the reactor." Now, inasmuch as the thermal conductivity of aluminum is about 200 times greater than the corrosion on the fuel plate, a corrosion layer only 0.00445 millimeters thick would have the same temperature gradient as 0.89 mm of aluminum cladding. Alternatively, the measured corrosion product thickness of 0.09 mm has 20 times the temperature gradient of the aluminum cladding. Ignoring the corrosion thus yields a grossly incomplete analysis in determining turnaround characteristics.

XIII. A relevant quote from *Inside NRC*, August 26, 2002, page 9.

The USNRC is among the collaborators in the Cabri International Project that conducts RIA tests on specimens in a test loop in the French Cabri rest reactor. The following comparison of an 80 micron oxide layer with a 20 micron oxide layer is revealing: *The CIP steering committee in late June agreed that if a high-burnup rod from Spain's Vandellos-2 PWR that was manufactured by Enusa with Westinghouse's new Zirlo clad showed "unusual behavior" in the test now expected to be run in late October, the program would make room for a second test of a Zirlo clad rod, this time coming from North Anna and manufactured in the U. S., said jean-Claude Melis of IRSN's safety research department. The reasoning, he said, is that the Enusa fuel may in fact not be representative of other Zirlo-clad fuel because of a* ***high oxidation layer, 80 microns*** *compared to 20 microns on a rod from Gravelines with Framatome ANP's competing M-5 clad material , scheduled to be tested in Cabri in mid-October.*

XIV. Some remarks to ACRS regarding Axial Offset Anomaly.

At the ACRS Reactor Fuels Subcommittee meeting on April 23 and 24, 1998, AOA was discussed by an NRC staff member as follows: *"This is a problem that has been around for a few years, but it has been around at a kind of low level. Various plants experienced axial offset anomaly, maybe 3 to 5 percent. It was somewhat of an annoyance. There were a few that got down to maybe 6 percent, and then last summer there was one plant that had a real problem, and basically they were beyond 15 percent. This really came to a shutdown margin problem. How they handled it was continued operation. they continued to operate within their tech specs but they did operate for about four months at 70 percent power. As they got further into the cycle they wee able to raise power slightly and at the beginning of this month, when they shut down, they were up to 86 percent power. Obviously it was a very big problem economically. The problem was crud buildup. Crud buildup high in the core traps boron and pushes the flux to the bottom of the core."*
Later in this ACRS meeting, a nuclear industry representative, possibly disturbed by the AOA disclosures to the ACRS, addressed the ACRS as follows: *"And as long as we stay within the tech spec, the operational limit, there shouldn't be any safety concern. I think experience has shown that to be the case. So, you know it is – really great to be on top of things, But some of the issues like AOA, they are not safety concerns, they are operational issues."*

In summary, the Petitioner has documented 14 situations that, in total, reveal that the although operation of light water cooled nuclear power reactors with fouled and/or corroded fuel elements is ubiquitous, the impact of fouling on the severity of reactivity insertion accidents has been overlooked.

The above Item VI, NRC Information Notice 97-85, clarifies Axial Offset Anomaly (AOA), and is especially revealing. Notice 97-85 states that the "*...difference between power in the upper and lower portions of the core... must remain within limits established in the technical specifications to ensure that both SDM and clad local peaking factors are not exceeded. Exceeding these limits could result in the reactor fuel exceeding 10 CFR 50.46 limits on fuel clad temperature (1204C). If the reactor approaches these limits, compensatory measures, including a power reduction, must be taken to maintain the reactor within its operational limits.* **However, Notice 97-85 does not recognize that the same fouling that leads to the AOA will lead to temperatures well beyond 1204C! It also does not recognize that the same fouling has a significant impact on the RIA that the shutdown margin requirement is intended to limit. With guidance like Notice 97-85, the basis for the statement in the above cited ACRS presentation (Item XIV) becomes very clear, *"But some of the issues like AOA, they are not safety concerns, they are operational issues."***

The above Item II includes the following:
NRC's research on high-burnup fuel is focused on postulated events that involve significant fuel damage. These events have the potential to cause fuel melting, if not kept within certain bounds, and melting could produce a large fission product release and significant consequences (hence risk). While there may be many pathways leading to such events, there are only two ways to melt fuel. One is to lose the coolant and the other is to get excessive power in the fuel. Selected design-basis accidents are postulated to serve as bounding examples of these kinds of events, and fuel damage limits are used to ensure that coolable core geometry is not lost, thus avoiding significant consequences.
The impact of fouling such as is documented in Items V, VIII, IX, X, and XIII, is not covered in any NRC research program. The fuel damage limit that is supposed to ensure a coolable core for RIAs is 280 cal/g peak fuel enthalpy (Regulatory Guide 1.77). The fuel damage limit of limit of 280 cal/g peak fuel enthalpy is grossly excessive for fouled and/or corroded fuel elements. So, *"...there are only two ways to melt fuel. One is to lose the coolant and the other is to get excessive power in the fuel."* Of course, *"...excessive power in the fuel..."* is a relatively low number for the River Bend cases IX and X.

Finally, the Petitioner strongly believes that denial of his petition would be outside of the NRC's Strategic Performance Goals.

1. Maintaining Safety: The requested rulemaking would make a significant contribution to maintaining safety. Current regulations and regulatory guidance do not satisfactorily address the effect of fouled and/or corroded fuel elements on RIAs. In fact, NRC Information Notice 97-85 has blessed the continued operation of nuclear power plants with fouled and/or corroded fuel elements under conditions in which both SDM and clad local peaking factors were exceeded.

2. Enhancing Public Confidence: The Petitioner is convinced that the proposed revisions would enhance public confidence. First, the NRC's acceptance of operation with excessively fouled and/or corroded nuclear fuel elements lacks a sound technical basis. Second, current regulations and guidance do not already address the effects of currently accepted fouled and/or corroded fuel elements on RIAs. The petitioner's request requires that substantial, additional consideration be given to excessively fouled and/or corroded nuclear fuel elements which is a condition that has been extensively observed. Not taking the Petitioner's action will actually detract from public confidence in the NRC as an effective regulator.

3. Improving Efficiency, Effectiveness, and Realism: The proposed rule would improve realism because the NRC would be required to generate necessary rules to avoid the operation of nuclear power plants with excessively fouled and/or corroded nuclear fuel elements. Of course, this would impact SDM and ECCS evaluations. This petition is supported by a firm technical basis. Its efficiency and effectiveness spring from its realism.

4. Reducing Unnecessary Regulatory Burden: The requested rule is necessary.

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