Pellet Cladding Interaction Fuel Failures during Anticipated Operational Occurrences in Boiling Water Reactors [a]

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[a] This paper is focused on the risk of fuel failures caused by stress corrosion cracking of conventional BWR fuel cladding during certain slow operational transients. The authors have not addressed the risk of PCI fuel failures during other BWR transients or PWR transients.

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Abstract

This paper reviews the technical and regulatory history of the PCI phenomenon. Appendix A provides a more detailed description of the PCI failure mechanism, and the fuel operating parameters that control its occurrence. The experimental methods used by researchers to quantify the PCI resistance of susceptible and resistant fuel designs are described and key time-to failure data important to fuel integrity during anticipated operational occurrences (AOOs) are presented.

The paper presents the basis for the authors' concern that large numbers of PCI fuel failures may occur during certain AOOs in BWRs operating with non-PCI-resistant designs, and that this risk is not adequately assessed. The potential for fuel failures during BWR AOOs such as the loss of feedwater heater (LFWH) event is evaluated using high quality data from power ramp experiments. In particular, this paper provides data that demonstrate that two key assumptions are invalid or are not adequately defined in the review guidance documents. These assumptions are:

- Current regulatory requirements limiting cladding strains during AOOs to less than 1% protect the fuel from PCI failure, i.e. PCI cracks will not nucleate and propagate through the cladding unless the cladding strain is greater than 1%.

- The durations of AOO power excursions are too short for PCI cracks to nucleate and propagate through the cladding, i.e. there is adequate time for operator action to detect and terminate the transient event before failures can occur.

The General Design Criteria 10 and the Standard Review Plan relevant to fuel design are reviewed to show that there is a broad regulatory basis to require quantitative assessments of the risk of PCI/SCC fuel failures by licensees.

1 BACKGROUND

During the 548th meeting of the Advisory Committee on Reactor Safeguards, December 6-8, 2007, the committee reviewed the Susquehanna Steam Electric Station (SSES) application for extended power uprate (EPU), and recommended approval. In their report [(1)], the Committee also recommended that:

> "*The staff should develop the capability and perform a thorough review and assessment of the risk of pellet-cladding interaction (PCI) fuel failures with conventional fuel cladding during anticipated operational occurrences (AOOs).*"

ACRS members Armijo, Banerjee and Powers submitted added comments to the Committee report. They concurred with the recommendation that the SSES application for EPU be approved, but expressed concerns that the licensee's plan to operate the two Susquehanna units with conventional (non-PCI-resistant) fuel cladding unnecessarily increased the risk of PCI fuel failures during anticipated operational occurrences at EPU conditions. In these added comments they presented the basis for their concerns, and recommended various actions to the staff to properly assess the risk.

In their January 17, 2008 response to the ACRS recommendation concerning PCI, the Staff stated:

> *"...the NRC staff will investigate current computational capabilities to model the complex phenomena associated with non-uniform fuel pellet expansion and stress-corrosion cracking (SCC). As necessary, the staff will develop guidance related to an application methodology and regulatory approach for implementing PCI/SCC fuel failure criteria."*

A follow-up meeting on the PCI/SCC issue was held by the ACRS Subcommittee on Materials, Metallurgy and Reactor Fuels, on March 3, 2009. The purpose of the meeting was to review the staff's investigations, to present relevant PCI time-to-failure data obtained by the subcommittee and its consultant [(2)], and to engage in collegial discussions with the Staff. At that meeting, the staff presented reasoning for their position that the PCI/SCC safety significance did not warrant immediate action or higher priority in staff workload planning [(3)]. Consequently, no work has been done to date to improve the staff's capability to assess the risk of PCI fuel failures during AOOs as recommended by the ACRS.

2 TECHNICAL AND REGULATORY HISTORY

The PCI phenomenon was detected in the early 1970s. Unexpected failures of zirconium alloy clad fuel rods occurred in heavy water reactors as well as BWRs and PWRs. The phenomenon had greatest impact on BWRs because the lower operating pressure and more rapid power changes of the system exposed the fuel to more severe duty. During the mid 1970s, the PCI failure mechanism was determined to be caused by stress corrosion cracking (SCC) of the fuel cladding after rapid power increases [(4)] and effective core operating restrictions were introduced (by General Electric) to prevent fuel failures during normal operation [(5)]. A more detailed description of the PCI mechanism, the operating parameters governing its behavior, and the means developed to prevent PCI fuel failures is provided in APPENDIX A – PCI Fuel Failure Background.

2.1 Loss of Feedwater Heater

Power increases capable of causing PCI failures can occur during normal operation or during anticipated operational transients. AOOs of particular concern are transients that simultaneously increase the power of every fuel rod in the core by a significant amount, and sustain peak power levels until the transient is terminated by operator action. The loss of feedwater heater (LFWH) event in the BWR has these characteristics and produces conditions favoring PCI failure in susceptible fuel. The most severe LFWH transient is initiated by inadvertent bypassing of the feedwater heaters. During the event, the feedwater temperature for a typical BWR 4 is reduced by an assumed maximum of 100 ºF. This increased subcooling raises the core power to 115 -120% of rated in approximately one minute. Every axial node of every fuel rod in the core is increased by a greater or lesser amount depending on the core power, axial power distribution and burnup of the fuel at the beginning of the transient. The high power levels produced during the transient persists for several minutes until terminated by operator action. Ramp test data shown in section 2.3 of this paper demonstrate that the operator has one minute or less to detect, analyze and terminate the transient before failure or damage occurs on conventional fuel.

2.2 Regulatory Reviews

In 1979, at the request of the ACRS, M. Tokar of the Division of Reactor Safety prepared a summary of NRC knowledge of the PCI phenomenon.(6) In this comprehensive review, Tokar made the following points related to the PCI mechanism and the need for regulatory action:

- "... while maneuvering restrictions appear to be effective in reducing the incidence of PCI failures during normal operation, no accounting of these failures is made for analyses of anticipated operational occurrences (AOOs or transients) or accidents.."

- " The general requirement to account for fuel failure status in to, i.e., originating from any source, is abundantly clear; however, the common interpretation of General Design Criterion 10 ...is that fuel rods must not undergo (significant) failure: i.e., specified acceptable fuel design limits (SAFDLS) must not be exceeded during normal operation or transients. This requirement is also stated (more explicitly) in Section 4.2 of the Standard Review Plan."

- "Current plant safety analyses are, therefore, deficient in the sense that they do not, in general, account for PCI, which is now well recognized as a significant fuel failure mechanism."

- "As the result of our past and on-going efforts on PCI, we believe that the time is right to start introducing PCI fuel failure analyses into plant safety analyses."

These conclusions were entirely appropriate given the level of understanding of the PCI phenomenon at the time, and are still valid. The Tokar report also identified the key mechanistic and regulatory question under discussion at the time. This was the question of the time required for PCI failures during AOOs. Tokar stated:

- "*...a major segment of the LWR industry holds that PCI failures will not occur during the type of power increasing transients and accidents addressed in Chapter 15 of the Standard Review Plan because the time at the increased transient power is too short.*"

- "*... in that groups view, exemplified by General Electric Company's response to NRC's requirement for consideration of PCI-induced fuel damage for Anticipated Transient with Scram (ATWS) analyses, PCI failures are likely to occur after a rapid power increase only if the fuel remains at higher power for a relatively long period of time (many minutes to many hours)*."

In May of 1984, Van Houten, Tokar, and MacDonald reported the results of an NRC-sponsored task force study addressing PCI and the NRC's concern that existing fuel rod overheating criteria might be inadequate for evaluating transient severity. [(7)] One of the objectives of the investigation was to establish if overheating criteria (MCPR) would bound the consequences associated with PCI failures during transients. The primary question addressed in the study was "*Will PCI failures occur during off-normal reactor operating conditions and do PCI failures exceed DNBR/MCPR calculated fuel failure probabilities used in the evaluation of potential radiological consequences?"*

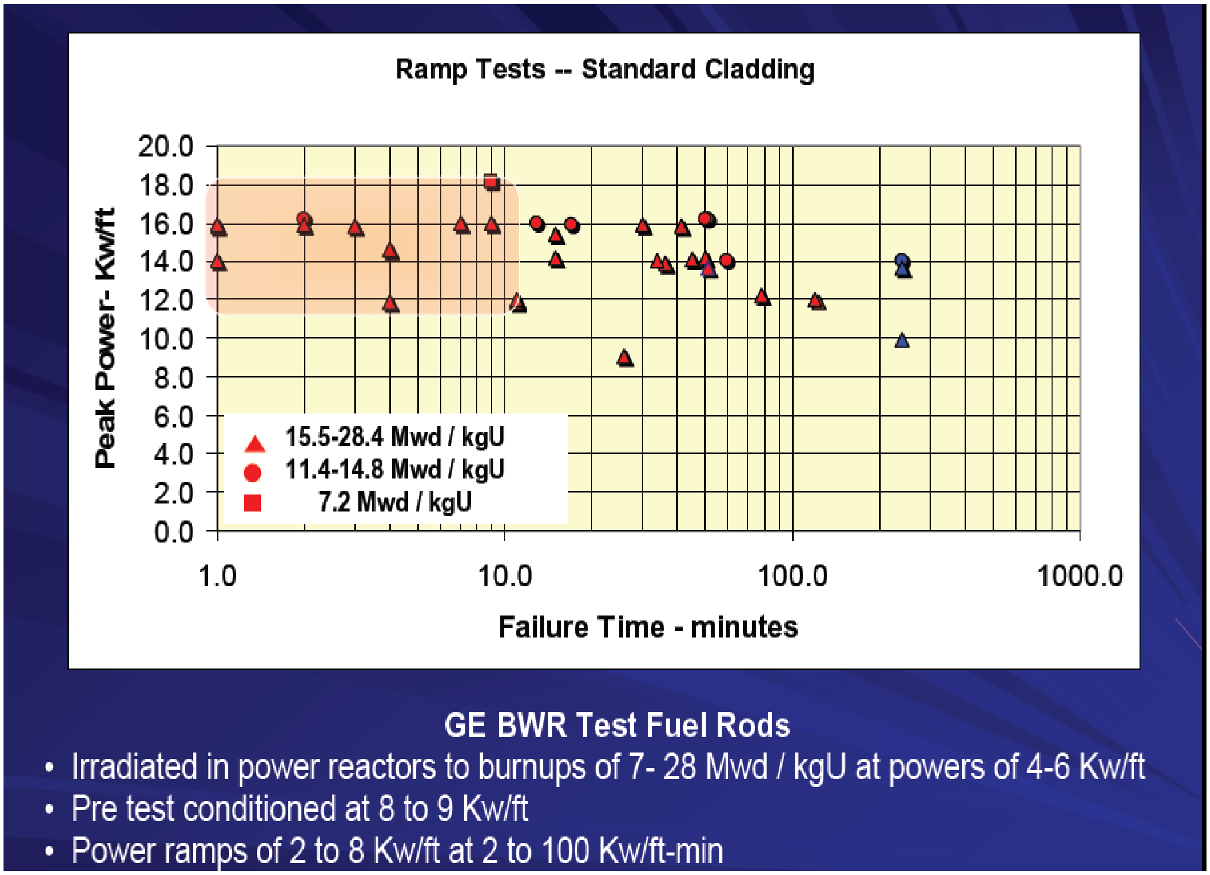
Several BWR and PWR transients were evaluated including: BWR Control Blade Withdrawal and Turbine Trip without Bypass events, and PWR Control Rod Bank Withdrawal Error and Steamline Break events. The LFWH transient event was not analyzed. The task force was unable to make a quantitative comparison of the PCI and DNBR/MCPR failure rates, but did conclude that there was a reasonable chance that PCI failures would occur during some off-normal reactor operating conditions.

2.3 GE and Demo Ramp II Power Ramp Tests

Concurrent with the Tokar and Van Houten studies, extensive fuel testing programs were in progress to quantify the parameters controlling PCI failure and to develop PCI-resistant designs. Data collected in these programs during the 1980's demonstrated that PCI failure or damage would occur within very short times when fuel rods were ramped to peak powers typical of some AOOs. These tests were performed under well controlled conditions in the R2 reactor in Sweden, and are described in detail by Davies and others [(2, 10)]. The primary information sought in the GE program was failure power as a function of fuel design and burnup. However, accurate time-to-failure data were also collected. In contrast, the primary interest of the Demo Ramp II program was the time-to-failure during transients. The results of these test programs are shown in **Figure 1** and **2** respectively. The methods used to determine failure times during ramp testing are illustrated in **Figure A1- 5** of Appendix A.

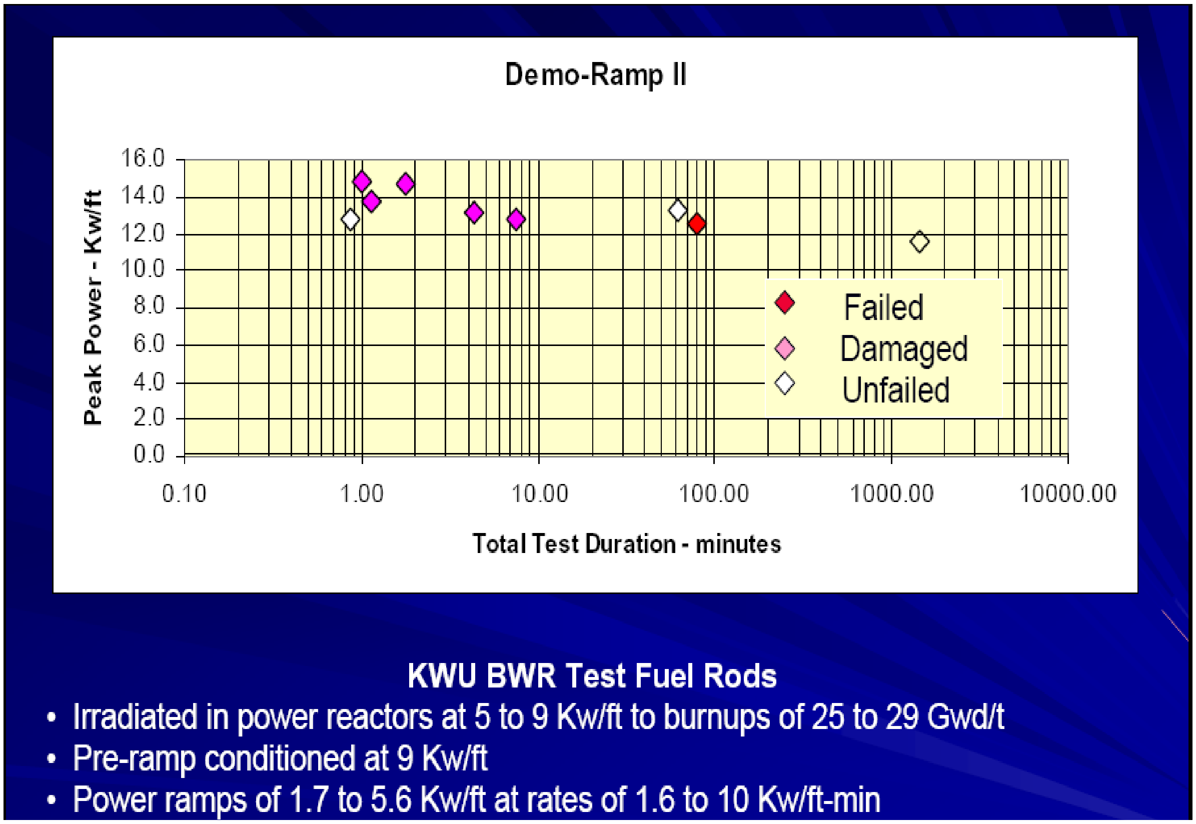
The cladding used in the GE test rods was conventional Zircaloy-2 in the recrystallized condition. The test rods were irradiated in a power reactor at low power to burnups ranging from 7 to 28 MWd/kgU. After detailed precharacterization, the rods were shipped to the R2 reactor and subjected to severe power-ramp tests. As shown in **Figure 1,** twenty-eight rods were ramped to peak powers of 12 - 18 kW/ft. Twenty-six failed by PCI. Ten rods failed within 9 minutes and five within 3 minutes.

Figure 1 **GE Ramp Tests**



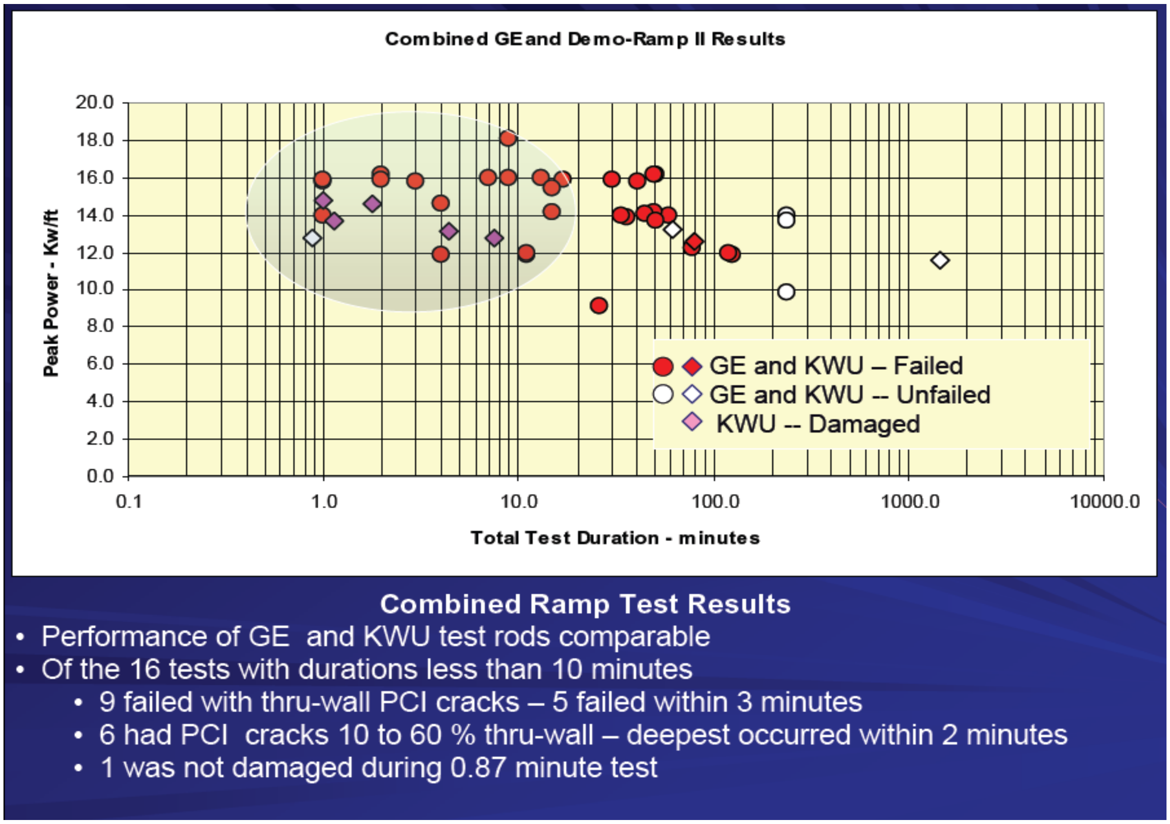
The test rods used in the Demo Ramp II program were supplied by Krafwerke Union (KWU) and used conventional Zircaloy cladding in the stress relieved condition. Similar to the GE protocol, the test rods were irradiated in a power reactor at low power to burnups ranging from 25 to 29 MWd/kgU. After detailed precharacterization, the rods were shipped to the R2 reactor and subjected to somewhat milder ramps than used by GE. Several of the ramp tests were intentionally terminated after very short times at peak powers to determine the threshold time-to-failure or damage. Results of these tests are shown in **Figure 2.** Of the eight rods ramped to high power levels (12 to15 kW/ft) two did not fail, and one failed after 80 minutes at peak power. Except for one rod tested for less than one minute, all rods tested for less than 8 minutes were damaged by PCI. Hot cell examinations confirmed that PCI cracks had penetrated through 30 to 60% of the cladding wall in the three rods tested for 1 to 2 minutes.

Figure 2 Demo Ramp II Ramp Tests



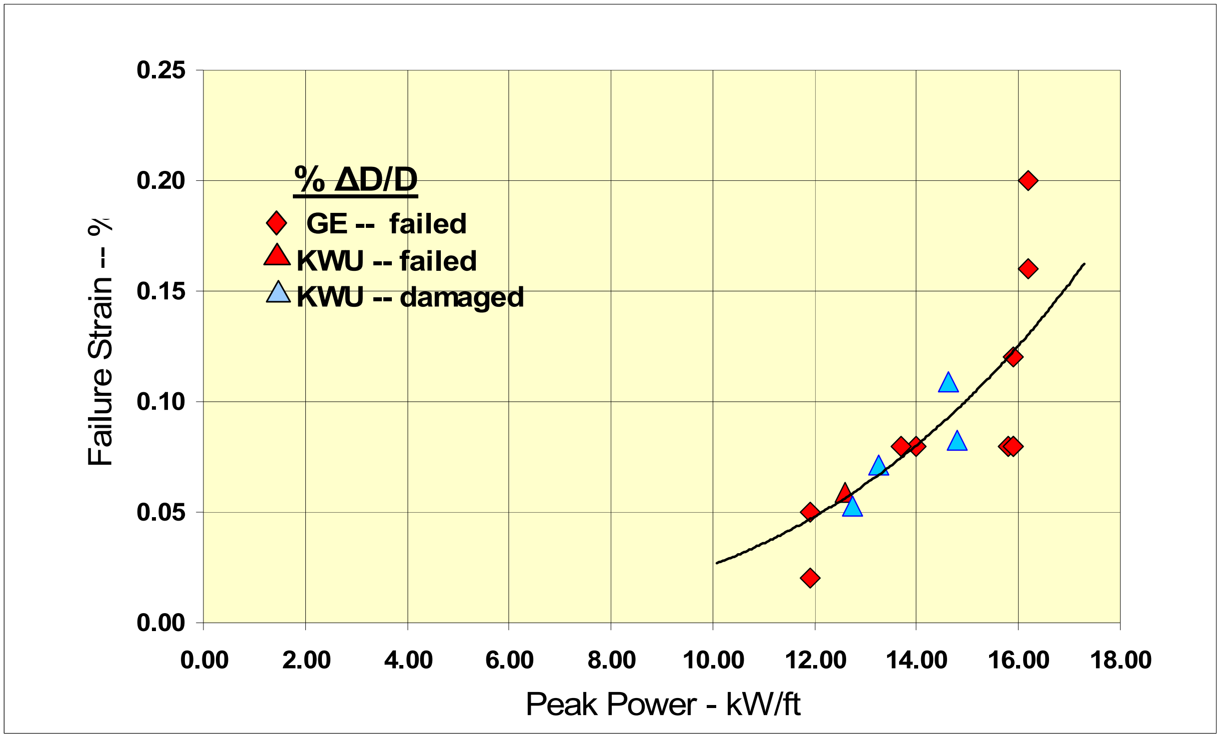
The time to failure data are even more compelling when the two independent data sets are combined. As shown in **Figure 3,** a very large fraction of the fuel rods failed or were damaged by PCI in less than 10 minutes when ramp tested in the peak power/ time domain representative of a LFWH transient. Of the thirty six rods ramped to 12 -16 kW/ft, 14 (39%) failed or were damaged in less than 10 minutes, and 8 (22%) failed or were damaged within 3 minutes. Only one rod in which the test was intentionally terminated in less than one minute was undamaged. This suggests that the threshold time to failure is less than one minute. If these same statistics hold during a severe LFWH transient in a BWR core operating with conventional fuel at full power, the possibility of hundreds of fuel failures cannot be discounted without a thorough analysis.

Figure 3 **Combined GE and Demo Ramp II Time-to-Failure Data**



In addition to failure power and time-to-failure data the GE and Demo Ramp II programs measured failure strains. Accurate fuel rod diameter measurements were made in hot cells before and after ramp tests. The data show that fuel rods fabricated by two different manufacturers, using two different fuel cladding materials and tested in two independent programs behave in a remarkably consistent manner. These measurements **(Figure 4)** clearly demonstrate that the strains required to cause PCI fuel failures are a factor of 5 to 40 times lower than the 1% strain acceptance criterion currently used in the standard review plan. It is clear that the 1% strain criterion, which is intended to protect fuel from failure during power transients, provides no PCI protection whatsoever.

Figure 4 **Combined GE and Demo Ramp II Failure Strains**



3 REGULATORY REQUIREMENTS FOR THE FUEL SYSTEM DESIGN

By the end of the 1980s the majority of BWRs worldwide utilized one or more variants of the PCI-resistant zirconium liner design to prevent fuel failures during normal operation, and other PCI-resistant designs were under development. Extensive testing had demonstrated that PCI-resistant designs significantly reduced the risk of fuel failure during AOOs in addition to protecting the fuel during normal operation.(8) In addition, all BWR suppliers reduced peak LHGRs by introducing 9x9 and 10x10 fuel bundles. These design improvements, combined with the assumption that AOO transients were too fast to cause PCI failures, effectively eliminated the need for regulatory action.

The need for regulatory action has been reopened by the growing use of non-PCI-resistant BWR fuel designs (particularly for cores operating at 120% of their originally licensed thermal power). During certain AOOs the fuel in these cores is protected only by prompt operator actions and not by the PCI resistance of the fuel. In addition, the fuel duty in modern cores has become more demanding than in the past. The reductions in peak LHGRs achieved in the 1980s (by increasing the number of fuel rods in the bundle) have disappeared due to economic demands. Peak LHGRs in present-day 10x10 fuel assemblies are equal to those in the 8x8 assemblies of the 1980s. Modern fuel is designed for high capacity factor, high energy, and long duration operating cycles. The number of high power bundles in many cores has also increased in order to meet the energy needs for extended power uprates. At full power, more nodes will be operating closer to the LHGR limit. These fuel and core design changes increase the risk of PCI during certain AOOs.

3.1 General Design Criteria

Part 50 of Code of Federal Regulations (CFR), General Design Criteria Appendix A, II, "Protection of Multiple Fission Product Barriers," specifies the regulation governing the protection of the multiple fission product barriers in the defense-in-depth design of light water reactors. In particular GDC *Criterion 10, "Reactor Design*," provides the regulatory basis for ensuring the integrity of the "first fission product barrier," the fuel cladding. GDC 10," *Reactor design,"* states:

> *"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."*

The General Design criteria define anticipated operational occurrences as:

> *"those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power."*

The specified acceptable fuel design limits (SAFDLs) are those limits that ensure fuel cladding, reactor vessel and containment integrity are not breached. 10 CFR 50.36, "Technical Specification," describes the limits as, "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down." Therefore, for normal and transient conditions, the reactor as loaded and operated must be analyzed to demonstrate that no damage mechanism would result in breach of the cladding, vessel or containment. The SAFDLs that protect the fuel cladding barrier from the associated damage mechanisms during steady state and AOOs are established by the fuel vendors and reviewed and approved by the staff. The plant-specific final safety analyses report (FSAR) provides the licensing basis for operation of the plant consistent with the regulatory requirements. The technical specification (TS) contains the key process parameters assumed in the analyses. Section 5.0 of the TS lists the specific licensing methodology and the associated topical reports used to perform the safety analyses that support the safe operation of the plant in accordance with the regulation.

3.2 The Standard Review Plan

The fuel and core design as operated are also required to meet the acceptance criteria for accidents and special events such as "coolable geometry," peak cladding temperature limit, vessel and containment integrity. The focus of this section is on the adequacy of the acceptance criteria used by the staff to disposition the potential for PCI/SCC fuel failure during AOOs for cores loaded with conventional fuel designs. Section 4.2 of the Standard Review Plan (SRP) provides the guidelines for acceptance criteria for the fuel system design. The safety of the fuel system design is reviewed to assure that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained.

A "not damaged" fuel system is defined to mean that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. Conversely, "fuel rod failure," means that the fuel rod leaks and that the first fission product barrier (the cladding) has been breached. In demonstrating and reviewing the capability of a plant to meet the requirements of GDC 10 during normal operation and AOOs, the pre-defined SAFDLs must be shown to be met and that fuel rod failures will not occur.

There are regulatory positions or practices that define the number of fuel failures acceptable under GDC-10. The crux of this regulatory ambiguity (zero fuel failures versus a low but practical limit) is whether limited fuel failures due to infrequent occurrences such as presence of debris in the primary coolant (fretting failures), defective cladding, welds or pellets (manufacturing failures) or coolant chemistry transients (crud failures) constitute noncompliance with GDC-10. The reasoning is that small numbers of fuel failures are acceptable because the release of radioactive fission products would not exceed the licensing basis of the plant, and operators would have adequate time to detect failures and take corrective actions. Section 4.2, Revision 3 of the SRP weighs in on the acceptance of limited fuel failures, stating:

> *"To meet the requirements of (1) GDC 10 as it relates to SAFDLs for normal operation, including AOOs and (2) 10 CFR Part 100 as it relates to fission product releases for postulated accidents,* ***fuel rod failure criteria should be provided for all known fuel rod failure mechanisms****. Fuel rod failure is defined as the loss of fuel rod hermeticity. Although the staff recognizes that it is impossible to avoid all fuel rod failures and that cleanup systems are installed to handle a number of leaking rods, the review must ensure that fuel does not fail as a result of specific causes during normal operation and AOOs. Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis.*

Therefore, within the regulatory framework, all known fuel damage mechanisms need to be assessed and precluded before approval of the loading and operation of any fuel designs. It is recognized that limited numbers of fuel leaks cannot always be avoided. However, the fuel and core should be designed to provide a high level of confidence that fuel failures from known damage mechanisms are not expected.

4.0 CONCLUSIONS AND RECOMMENDATIONS

- Section 4.2 of the Standard Review Plan provides guidance applicable to the PCI failure mechanism. *"To meet the requirements of (1) GDC 10 as it relates to SAFDLs for normal operation, including AOOs and (2) 10 CFR Part 100 as it relates to fission product releases for postulated accidents,* ***fuel rod failure criteria should be provided for all known fuel rod failure mechanisms****."* Irrespective of whether the number of fuel failures is limited or not, the fuel and core should be designed to assure that fuel failures from known damage mechanisms are not expected to occur.

- PCI is a known, unique, and potent stress corrosion cracking mechanism capable of failing large numbers of fuel rods during normal operation and during certain AOOs. Both the mechanism and the fuel operational parameters controlling the phenomenon are well understood.

- The primary drivers for PCI are stress and chemistry, not strain. Current thermal-mechanical regulatory criteria are not appropriate for PCI. Stress corrosion cracking failure criteria should be based on measured failure powers and failure times, not calculated failure strains.

- PCI failure powers should be based on a statistically significant number of power-ramp tests. These tests should be prototypic of the conditions (power increases, peak powers, times at peak power and burnups) expected during AOOs.

- PCI crack nucleation and propagation rates are fast enough to cause fuel failures during AOOs. Unlike PCI-resistant designs, a large fraction of the conventional fuel in the core could fail or be damaged in one to three minutes if ramped to power levels typical of the BWR LFWH transient.

- The staff should develop the capability and perform a thorough review and assessment of the risk of pellet-cladding interaction (PCI) fuel failures with conventional fuel cladding during anticipated operational occurrences (AOOs).

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APPENDIX A – PCI Fuel Failure Background

A.1 Characteristics and Mechanism:

Pellet cladding interaction fuel failures have occurred in both light water (BWR and PWR) and heavy water (CANDU and SGHWR) reactors [4-11]. These failures occur when fuel rods that have accumulated sufficient burnup are subjected to rapid power increases. During these power ramps, the inner diameter of the zirconium alloy fuel cladding is subjected to localized tensile stresses caused by the thermal expansion of the fuel, and aggressive fission products released from the fuel. When the combined fission product concentrations and tensile stresses are sufficiently severe, the cladding will fail by PCI. These brittle cladding failures are caused by stress corrosion cracking (SCC) or liquid metal embrittlement (LME) and are driven by stress.

The brittle nature of such failures is illustrated in **Figures A1- 1** and **A1- 2**. The tight axial crack in the fuel cladding is barely discernable, and the branching crack pattern in the cross-section exhibit negligible strain. Extensive testing has shown that PCI failures require little or no plastic strain. PCI failures are often referred to as PCI/SCC to distinguish them from purely mechanical failures caused by overstraining the fuel cladding. Specialists in the field use the terms PCI and PCI/SCC interchangeably. Purely mechanical failures are referred to as pellet-cladding-mechanical-interaction (PCMI) and will generally not occur on irradiated zirconium alloys unless the strain substantially exceeds 1%. Thus, current regulatory requirements that limit cladding strains to <1% protect the fuel from PCMI, but not from PCI/SCC which occurs at much lower strains.

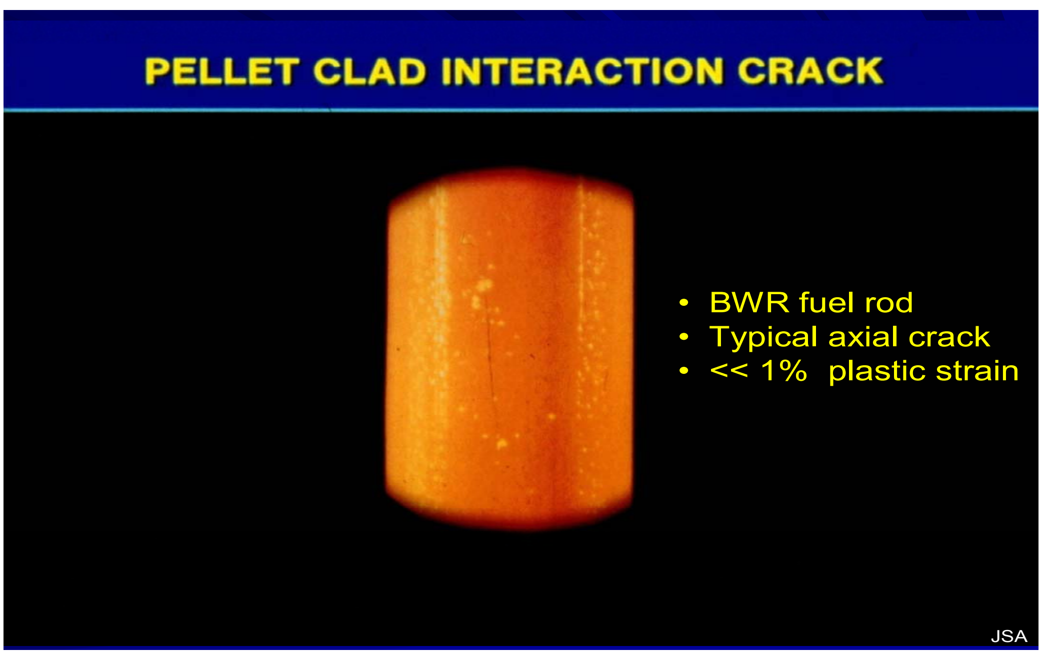


Figure A1- 1: BWR fuel rod surface with a tight, axial PCI crack.

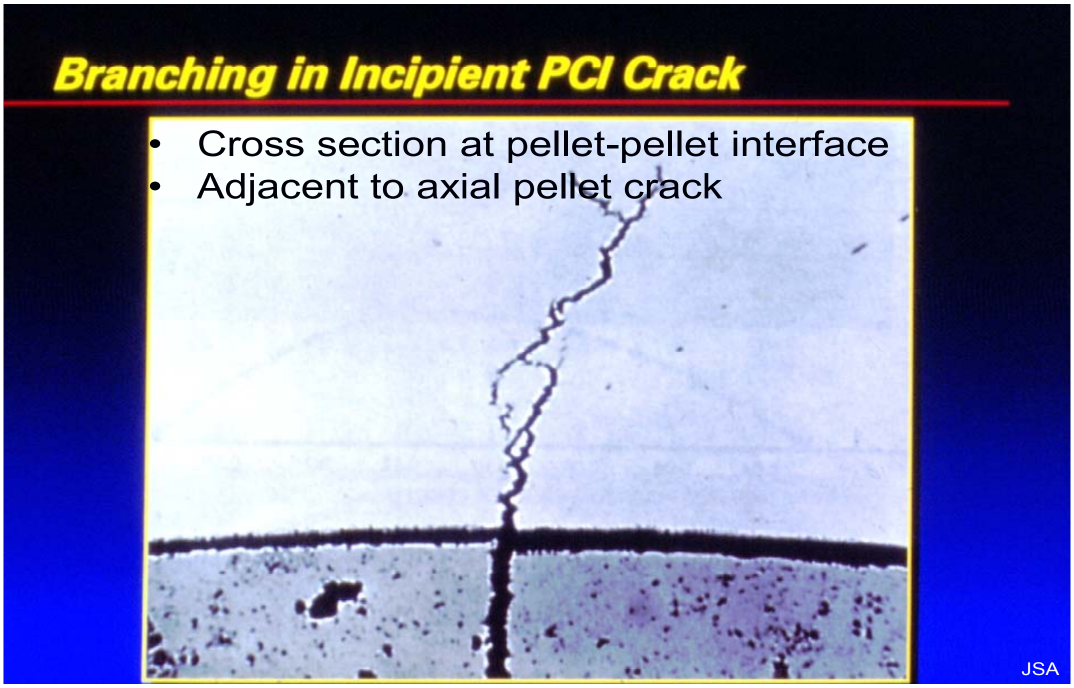


Figure A1- 2: Cross-section of fuel rod showing a typical branching stress corrosion crack nucleating from the cladding inner surface. [(4)]

The PCI/SCC mechanism is well understood, and its key characteristics are illustrated in **Figure A1- 3** and **A1- 4**. **Figure A1- 3** is an illustration of the concurrent mechanical and chemical changes that occur when the power of a fuel rod is increased and the fuel pellet temperature increases. If the peak power is low, and/or the power increase is small, or the rate of power increase is very slow, the fuel cladding will not fail. If the peak power is greater than the PCI threshold power (~8 kW/ft), and the power increase is sufficiently high (≥ 1 kW/ft), and the rate of power increase is greater than 0.1 kW/ft/hr, conventional cladding can fail by PCI. The probability of failure increases as the peak power and the magnitude of the power change increases.

Figure A1- 4 shows the results of power ramp tests on BWR fuel rods clad with conventional Zircaloy 2. In these tests, rod powers were rapidly increased from 8 kW/ft to terminal power levels indicated in the chart. Rods were typically ramped in steps of 2 kW/ft or greater. [(5)]. As shown, PCI failures (red symbols) occurred with increasing probability as the maximum ramp power increased. At the typical peak operating power of BWR fuel (13.4 kW/ft), the failure probability is >50%, and at terminal power levels typical of some AOOs (16 kW/ft), the failure probability is ~ 95%.

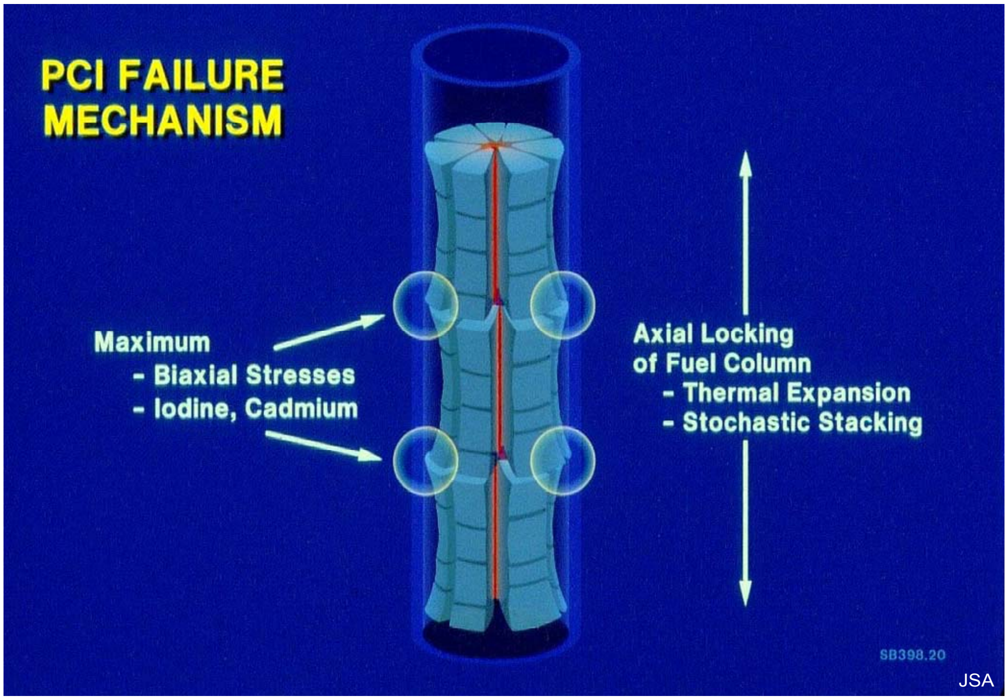


Figure A1- 3: **Illustration of the PCI failure mechanism.**

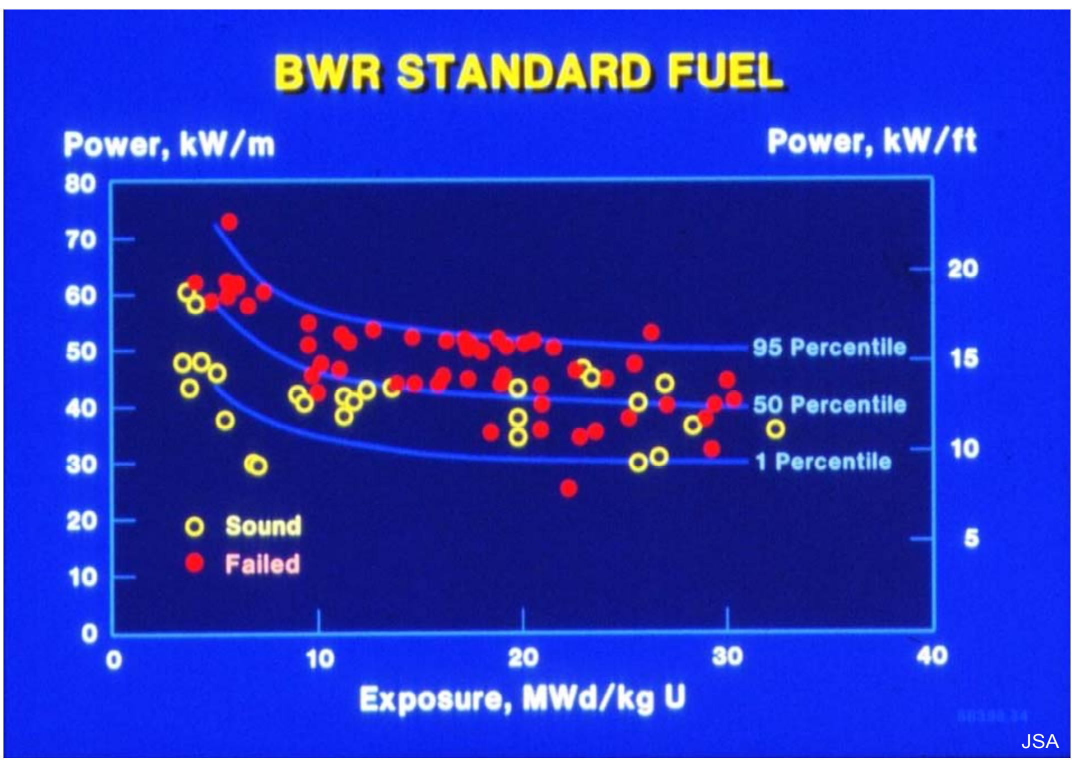


Figure A1- 4: **Power ramp data showing PCI failure probabilities as a function of burnup and peak power level.**

In addition to failure powers, these tests also measured times-to-failure. Three parameters were monitored to detect PCI failure: fuel rod elongation, thermal power, and coolant activity in the test loop. As shown in **Figure A1- 5**, the fuel rod power increases as He^3 pressure is reduced in the power control coil. Concurrently, instrumentation detects rapid elongation followed by a slow relaxation of the fuel rod. After a period of several minutes a power spike and dilation of the fuel rod occur simultaneously indicating cladding failure. A few minutes later, the coolant activity increases, confirming failure. Several test programs performed in the R2 reactor using these measurement methods established that a significant fraction of fuel rods ramped to powers in the 12-16 kW/ft range could fail by PCI in three minutes or less (**Figures 1, 2 and 3**.)

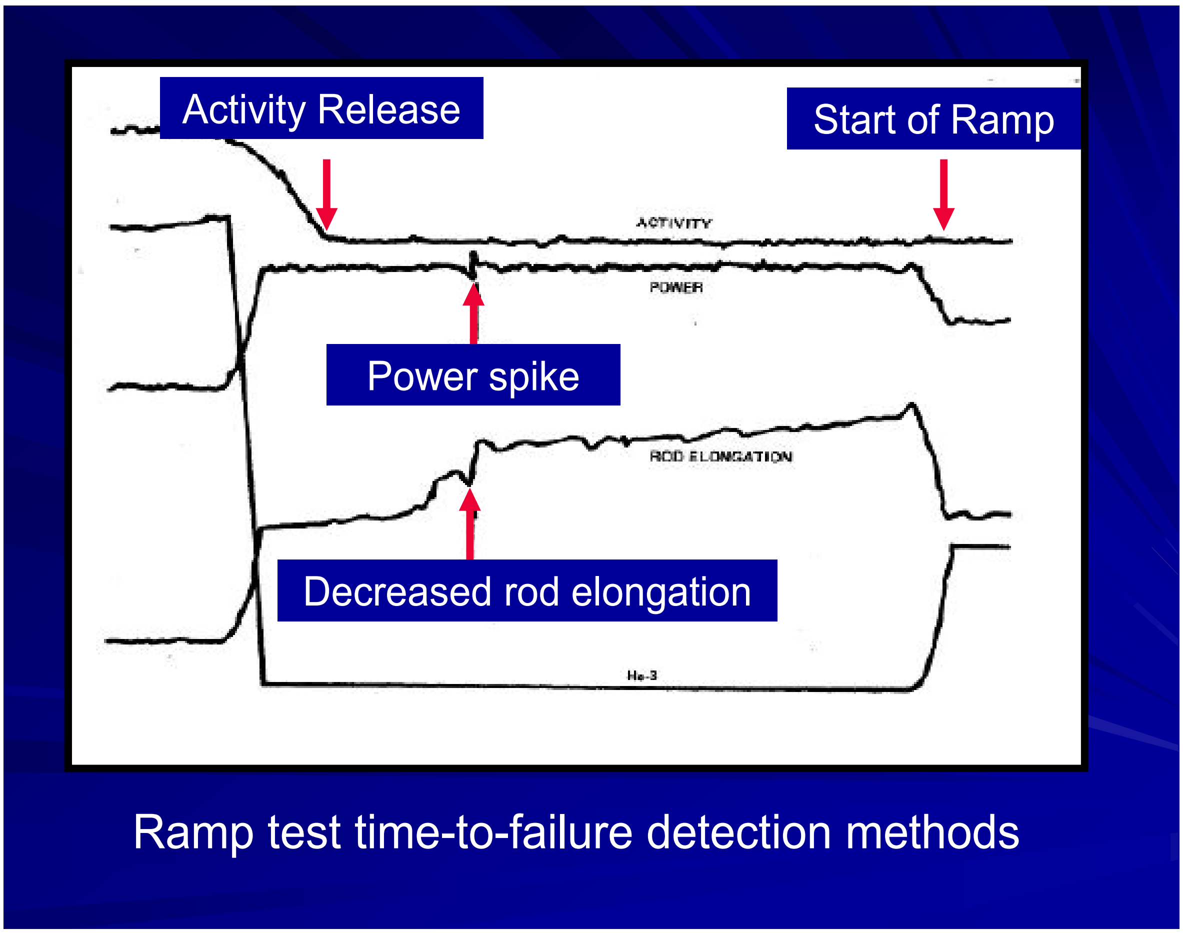


Figure A1- 5: Strip chart recording of a BWR ramp test at the R2 reactor. The time scale starts at the right with increase in fuel rod power, and ends on the left with the increase in coolant activity.

The tests and analyses (performed during the 1970s and early 1980s) led to the development of improved fuel designs as well alternate operating methods that effectively eliminated the risk of PCI failures during normal operation. [8] The understanding of the PCI mechanism led to the development of the Preconditioning Operating Management Recommendations (PCIOMRs) by General Electric, and subsequently by other fuel suppliers. In addition all BWR fuel suppliers reduced PCI risk by reducing peak linear heat generation rates (LHGRs) by introducing 9x9 and 10x10 fuel assemblies. These changes, while successful in preventing PCI failures, resulted in significant losses of plant operational flexibility and reduced capacity factors. It should be emphasized that all conventional Zircaloy 2 or 4 cladding materials that were ramp tested during this period were found to be susceptible to PCI. Various studies demonstrated that variations in conventional fuel design options such as cladding material (Zircaloy-2 vs. Zircaloy-4) cladding heat treatment (recrystallized versus stress relieved), cladding thickness or pellet density had negligible effects on PCI resistance. (4,9,11)

A.2 Oskarshamm 1 Experiment

During the 1970s there was considerable debate regarding PCI. Some researchers were convinced that the PCI failure problem was confined to fuel manufactured in the U.S., and believed that the PCI could be prevented by fabricating fuel to the highest possible quality standards. This led to the conclusive Oskarshamm experiment (9). In 1975, ASEA-Atom personnel decided to demonstrate the PCI resistance of their BWR fuel at full scale. To do this, they performed a power ramp test of standard 8x8 ASEA-Atom fuel assemblies in the Oskarshamm 1 BWR.

A single control blade (out of 112 in the core) was withdrawn at end of the operating cycle. This raised the peak LHGRs in surrounding assemblies to values ranging from 9.1 to 11.6 kW/ft. Although no failures were expected at these moderate powers, a total of 45 fuel rods in 14 fuel bundles failed. Subsequent hot-cell examinations confirmed that the fuel rods had failed by PCI (9). Had the power reached the licensed peak LHGR of 13.4 kW/ft the number of failures would have been considerably greater as indicated by Figure A1- 4

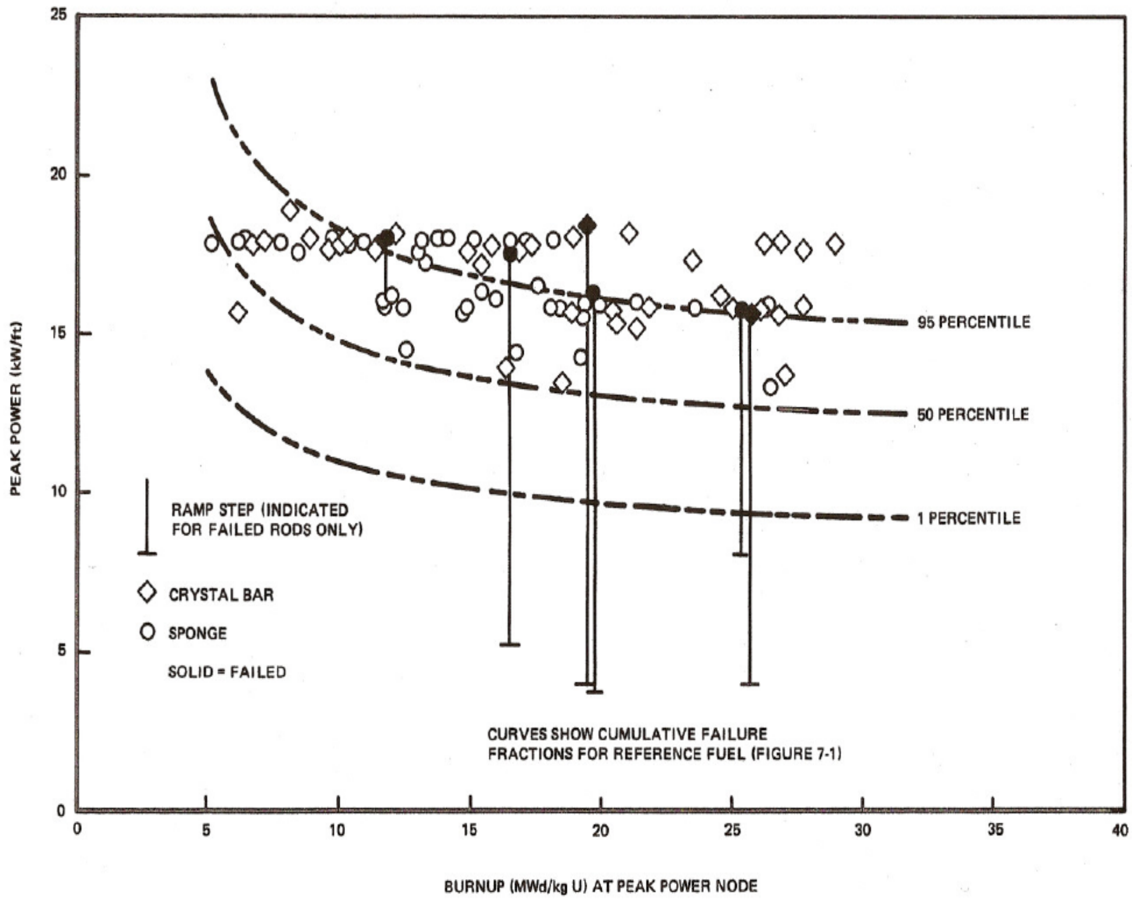
A.3 PCI Resistant Fuel

The Oskarshamm experiment effectively settled the debate with respect to the PCI-resistance of conventional fuel and cladding materials, and international efforts focused on the development of PCI-resistant fuel designs. There were regulatory forces as well as economic forces driving these efforts (6, 7). The regulatory concerns centered on risk of PCI fuel failures during AOOs in which peak LHGRs could reach powers of 16 kW/ft or higher in short times.

By the early 1980's significant progress had been made by a General Electric/ Commonwealth Edison/Department of Energy project to qualify a PCI-resistant fuel design for BWRs and to demonstrate its performance in a commercial reactor. The design selected for this demonstration consisted of conventional fuel encased in Zircaloy-2 cladding with a metallurgically-bonded inner liner of pure zirconium. General Electric identified this design as Zr-Barrier fuel, but it is often referred to as Zr-liner fuel in the industry. The pure zirconium liner was found to be highly resistant to radiation hardening, thus preventing pellet stresses from exceeding the threshold tensile stress necessary to nucleate stress corrosion cracks. As shown in

Figure A1- 6**,** seventy five severe ramp tests were performed to peak powers up to 18 KW/ft. These tests were performed using the same test protocols and test reactor as used in the tests of conventional fuel **Figure A1- 4**. None of the rods failed when tested at powers less than 15 K kW/ft. Three fuel rods failed at 18 KW/ft, and three failed at 16 KW/FT. Given this performance improvement over conventional fuel, Commonwealth Edison and GE received NRC approval to perform a large scale demonstration of Zr-Barrier fuel in the Quad Cities 2 BWR.

Figure A1- 6 : Results of power ramp tests of Zr-Barrier fuel.[2]



The large scale demonstration ramp tests were performed successfully in 1983 and 1985 [8]. As shown in **Figure A1- 7** the Quad Cities 2 core was loaded with 348 barrier fuel assemblies along with conventional fuel assemblies. Ramp cells containing barrier fuel were surrounded with low reactivity conventional fuel to assure that peak powers in the conventional fuel would never exceed the PCI threshold failure power during the demonstration. Barrier fuel assemblies were ramped by withdrawing control blades after the first and second cycles of operation. As shown in **Figure A1- 8** , the average barrier fuel rods were ramped from 4.5 to 9.5 kW/ft and the peak rods were ramped from 3.0 to 13 kW/ft. Thousands of axial nodes were ramped without failure.

The conclusion reached by GE after the demonstration was that the Zr-Barrier design was sufficiently resistant to remove all PCI-related operational restrictions. In addition, the regulatory concerns of the NRC regarding the potential for PCI failures during AOOs were resolved as all BWR fuel suppliers fielded Zr-liner designs.

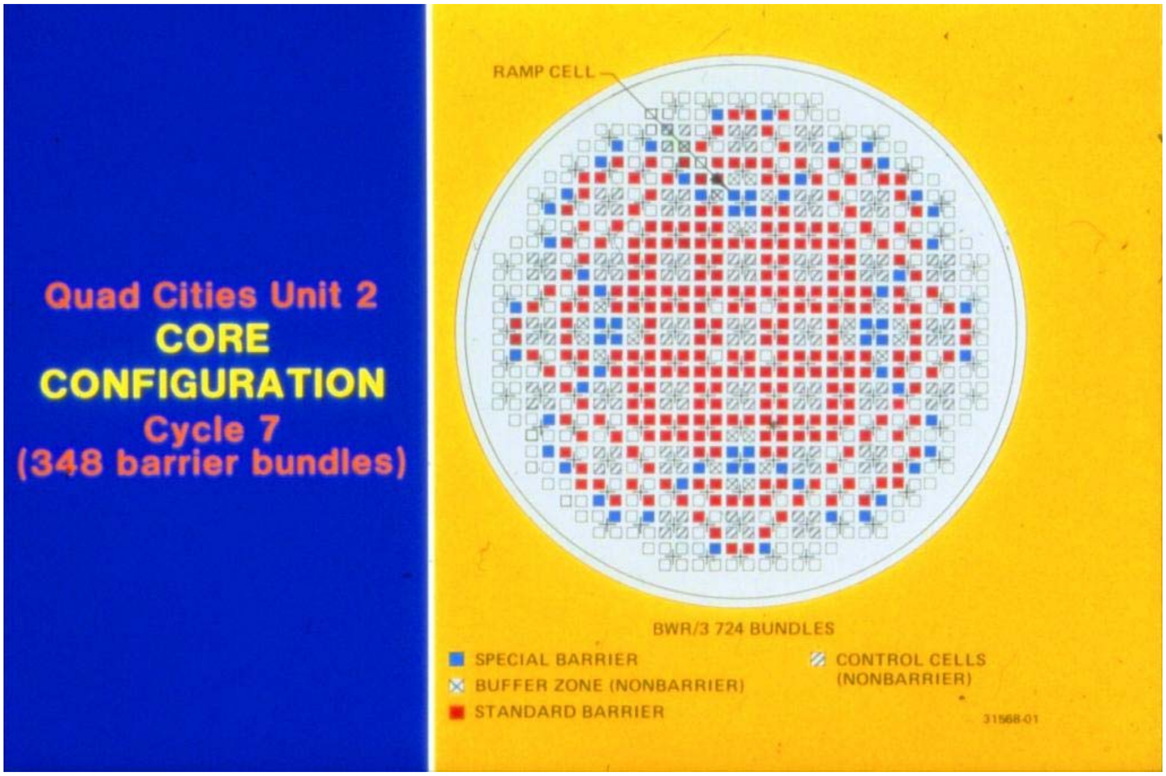


Figure A1- 7: Core loading for the large scale demonstration of zirconium barrier fuel.

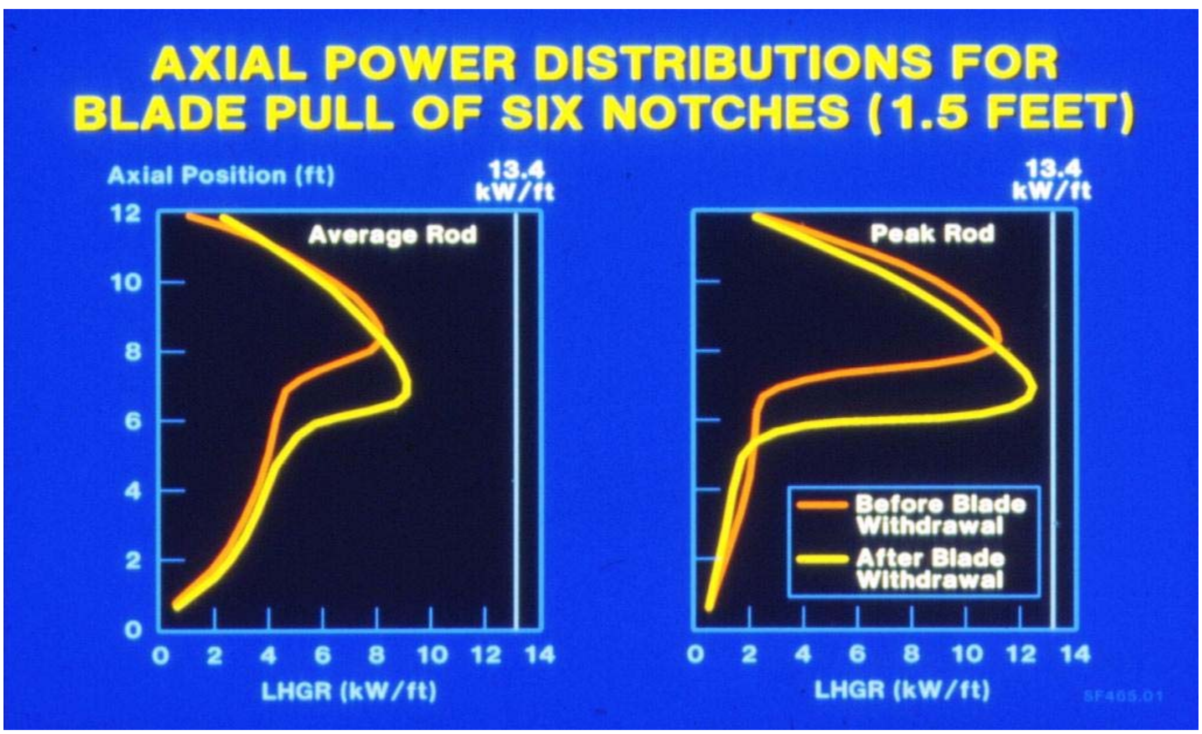


Figure A1- 8 Power increase sustained during ramp tests at end of second cycle of operation.