DD-99-08

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Samuel J. Collins, Director

In the Matter of	)	
	)	
ENTERGY GULF STATES, INC.	)	Docket No. 50-458
AND	)	
ENTERGY OPERATIONS, INC.	)	License No. NPF-47
	)	
(River Bend Station, Unit No. 1)	)	
	)	
AND	)	
	)	
FIRSTENERGY NUCLEAR OPERATING	)	Docket No. 50-440
COMPANY	)	
	)	License No. NPF-58
(Perry Nuclear Power Plant, Unit No. 1)	)	
	)	
	)	(10 CFR 2.206)

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. INTRODUCTION

By Petitions submitted pursuant to 10 CFR 2.206 on September 25, 1998, and November 9, 1998, respectively, Mr. David A. Lochbaum, on behalf of the Union of Concerned Scientists (UCS or Petitioner), requested that the U.S. Nuclear Regulatory Commission (NRC) take immediate action with regard to the River Bend Station (River Bend) and the Perry Nuclear Power Plant (Perry).

In the Petitions, the Petitioner requested that the NRC take immediate enforcement action by suspending the operating license for River Bend and Perry until all leaking fuel rods were removed from the reactor core or until the facilities' design and licensing bases were updated to permit operation with leaking fuel assemblies. Accompanying the Petitions was the

UCS report "Potential Nuclear Safety Hazard—Reactor Operation With Failed Fuel Cladding," dated April 2, 1998. Entergy Operations, Inc. (the River Bend licensee), provided the NRC with its response to its Petition in a letter dated February 11, 1999. FirstEnergy Nuclear Operating Company (the Perry licensee) provided a response to its Petition in a letter also dated February 11, 1999. On February 22, 1999, the NRC held an informal public hearing at which the Petitioner presented information related to the safety concerns in the Petitions. The NRC staff has determined that the information presented in the Petitions and at the informal public hearing did not support the action requested by the Petitioner. The basis for my decision in this matter follows.

II. BACKGROUND

In support of the requests presented in the Petition dated September 25, 1998, the Petitioner raised concerns stemming from NRC Daily Event Report No. 34815, filed on September 21, 1998, in which Entergy Operations, Inc., reported a possible fuel cladding defect at River Bend. The Petitioner repeated the concerns raised in the UCS report of April 2, 1998, regarding nuclear plant operation with fuel cladding leakage. The UCS considers such operation to be potentially unsafe and to be in violation of Federal regulations. In addition, the Petitioner cites instances in the licensing basis for River Bend that it believes prohibit operation of the facility with leaking fuel.

In the November 9, 1998, Petition, the Petitioner raised similar concerns originating from the NRC Weekly Information Report for the week ending October 30, 1998, in which fuel leaks detected at Perry on September 2, 1998, and on October 28, 1998, were discussed. The Petitioner also repeated the concerns raised in the UCS report of April 2, 1998. The matters raised in support of the Petitioner's requests are discussed herein.

III. DISCUSSION

The September 25, 1998, Petition presents safety concerns for River Bend along with the associated generic concerns addressed in the UCS report of April 2, 1998. The plant-specific concerns are based on portions of the River Bend Updated Safety Analysis Report (USAR) cited in the Petition. The November 9, 1998, Petition presents safety concerns for Perry arising essentially from the associated generic concerns addressed in the UCS report of April 2, 1998. The Perry Petition does not reference plant-specific licensing basis documentation.

Since the generic concerns presented in the UCS report bear upon the plant-specific concerns cited in the two Petitions, the staff's evaluation first considers the UCS report and follows with a discussion of the plant-specific concerns.

A. Generic Safety Concerns

In the UCS report of April 2, 1998, UCS expresses the opinion that existing design and licensing requirements for nuclear power plants preclude their operation with known fuel cladding leakage. The UCS position is based on the assessment of updated final safety analysis reports (UFSARs) of four plants, vendor documentation, standard technical specifications, and pertinent NRC correspondence. The report states that the following regulatory and safety concerns exist for plants operating with leaking fuel:

- 10 CFR 50.59, "Changes, tests and experiments," is violated because operation with fuel cladding leakage constitutes an unapproved change to the licensing basis for a plant. The report states that such operation is an unresolved safety question because the criteria of 10 CFR 50.59(a)(2) are satisfied (e.g., probability and consequences of an accident may be increased by operating with leaking fuel).

- 10 CFR 50.71, “Maintenance of records, making of reports,” is violated because the licensing basis as documented in the technical specifications and the analyses contained in the UFSAR for the facility do not accommodate operation with leaking fuel.
- Safety analyses for postulated accidents assume intact fuel cladding before the event; therefore, plants with known fuel leakage could have accidents with more severe consequences than predicted as a result of fuel damage. The report further states that no information was available showing that operation with leaking fuel has been previously evaluated.
- 10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors,” and other regulations related to the as low as is reasonably achievable (ALARA) principle for radioactive materials release are violated since plant workers are exposed to a greater risk than necessary because of higher coolant activity levels attributable to leaking fuel.

In addition to requesting that the NRC take steps to prohibit nuclear power plants from operating with fuel cladding damage, the report specifically requests that plants be shut down upon detection of fuel leakage, and that safety evaluations be included in plant licensing bases that consider the effects of operating with leaking fuel to justify operation under such circumstances.

Before addressing the regulatory concerns raised in the April 1998 UCS report, the following discussion provides background and bases for current NRC guidance and practices with regard to fuel defects.

1. Defense-in-Depth and ALARA Considerations

In order to protect public health and safety from the consequences of potential uncontrolled releases of radioactive fission products resulting from the operation of nuclear power plants, plants are designed with multiple barriers to fission-product release. This traditional “defense-in-depth” philosophy is key to assuring that radiological doses from normal operation and postulated accidents will be acceptably low, as outlined in 10 CFR Part 100, “Reactor Site Criteria.” Fuel cladding is integral to the defense-in-depth approach to plant safety, serving as the first barrier to fission-product release.

The premise of the defense-in-depth philosophy with regard to the potential for fission-product release is that plant safety does not rely on a single barrier for protection. In this way, a limited amount of leakage from each of the barriers—the fuel cladding, the reactor coolant system pressure boundary, and the containment—is a design consideration and some leakage from each barrier, within prescribed limits, is acceptable during operation. These limits, defined within the technical specifications, are established as a key component of a plant’s design and licensing basis. The leakage associated with fuel cladding defects is accounted for in plant safety analyses, as discussed later in this evaluation under “Safety Analysis Assumptions.”

Therefore, to meet its defense-in-depth objectives, fuel is not required to be leak-free. A limited amount of fuel cladding leakage is acceptable during operation since (1) in the event of an accident, other fission-product barriers besides the fuel cladding (i.e., the reactor coolant system pressure boundary and the containment) help prevent uncontrolled releases, (2) limits for reactor coolant system activity, as prescribed in the technical specifications, limit the level of fuel leakage that is permitted so that the release guidelines of 10 CFR Part 100, “Reactor Site Criteria,” will not be exceeded during accidents, and (3) plant design features and operating procedures anticipate leaking fuel and provide means to deal with the effects.

Sources of activity in reactor coolant are fission products released from fuel, corrosion products activated in the reactor during operation, and fission products released from impurities in fuel cladding, tritium produced from the irradiation of water, lithium, and boron. Although reactor operators should strive to maintain low levels of coolant activity from all of these sources, the staff has long recognized that reactor coolant activity cannot be entirely eliminated and that some fission products from leaking fuel could be present (see Standard Review Plan (SRP), NUREG-0800, Section 4.2, “Fuel System Design”). Thus, plant design considerations, such as reactor coolant cleanup systems, shielding, and radwaste controls, have been devised to minimize risk to plant workers from exposure to radiation from reactor coolant. Plants also implement procedures to respond to leaking fuel when leakage is discovered, as was demonstrated by the example of the follow-up actions taken by the River Bend and Perry operators to limit the production of fission products in the vicinity of the leaking fuel rods.

By containing fuel and fission products, cladding also helps maintain radioactive releases to as low a level as is reasonably achievable. As previously stated, the technical specifications contain limits for the maximum level of coolant activity so that the dose guidelines in 10 CFR Part 100 are not exceeded during accidents. These are the maximum levels of activity assumed to exist in the reactor coolant from normal operating activities. The limits on reactor coolant system specific activity are also used for establishing standardization in radiation shielding and procedures for protecting plant personnel from radiation (see Section B3.4.16 of NUREG-1431, “Standard Technical Specifications, Westinghouse Plants”). Thus, they are consistent with NRC regulations requiring licensees to follow an ALARA approach to radiation protection.

The connection between technical specification limits for coolant activity and ALARA requirements is key to demonstrating that limited fuel leakage during operation is consistent with safe plant operation. The ALARA requirement is given in 10 CFR 50.34a and 50.36a. The Statement of Considerations for these NRC regulations (35 FR 18385, December 3, 1970) contains a discussion of the “reasonableness” aspect of the ALARA approach. When the Statement of Considerations was written, the Commission believed that releases of radioactivity in plant effluents were generally within the range of “as low as practicable.” The Commission also stated, therein, that “as a result of advances in reactor technology, further reduction of those releases can be achieved.” Advances in fuel integrity, design of waste treatment systems, and appropriate procedures were cited as areas in which the plants had taken steps to meet the reasonableness standard. It is important to note that the Commission did not require leak-free fuel as a means to satisfy ALARA requirements. In addition to the physical barriers to the release cited above, other factors, such as radwaste cleanup and plant procedures, provide confidence that fission-product release from the fuel can be controlled so as to prevent undue risks.

Later in the same Statement of Considerations, the Commission acknowledged the need to allow flexibility of plant operation. “Operating flexibility is necessary to take into account some variation in the small quantities of radioactivity, as a result of expected operational occurrences, which may temporarily result in levels of radioactive effluents in excess of the low levels normally released” but still within regulatory limits. The Commission recognized that a balance should be maintained between limiting exposure to the public and plant operational requirements. Therefore, the NRC regulations allow the possibility of increased reactor coolant activity levels that might result from limited fuel cladding leaks, but require the use of plant equipment to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. The

Commission went as far as to define “as low as practicable” (the phrase later replaced with “as low as is reasonably achievable” in 40 FR 19440, May 5, 1975) in terms of the state of technology, the economics of improvements in relation to benefits to public health and safety that could be derived by improved technology and methods of controlling radioactive materials, and “in relation to the utilization of atomic energy in the public interest.” This definition appears in Section 50.34a itself, mandating that the Commission maintain the balance between safety and plant operational requirements.

By publishing 10 CFR Part 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion ‘As Low As Is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,” the Commission took steps to provide more definitive guidance for licensees to meet the “as low as practicable” requirement. Appendix I was published as guidance that presented an acceptable method of establishing compliance with the “as low as practicable” requirement of 10 CFR 50.34a and 50.36a. In the Statement of Considerations for Appendix I (40 FR 19439, May 5, 1975), the Commission characterized the guidance as the “quantitative expression of the meaning of the requirement that radioactive material in effluents released to unrestricted areas from light-water nuclear power reactors be kept ‘as low as practicable’.” The technical basis for Appendix I contained assumptions for a small fraction of leaking fuel rods, as is stated in the Atomic Energy Commission’s report of July 1973, WASH-1258, “Final Environmental Statement Concerning Proposed Rule Making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion ‘As Low as Practicable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.”

2. Associated Regulations and Guidance

Fuel integrity is explicitly addressed in NRC regulations in several instances, and plant licensing bases specifically discuss fuel performance limits. To implement NRC regulations, the staff developed a number of guidance documents for licensees to use in developing their licensing basis. This section outlines the regulatory framework on fuel integrity during normal plant operation and discusses instances in which the staff has considered the safety implications of fuel integrity.

a. Regulatory Requirements

The General Design Criteria (GDC) of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," contain references to fuel design criteria. When fuel performance is used as a criterion for a safety function, system, or component, the phrase "specified acceptable fuel design limits" (SAFDLs) appears in the following GDC:

- GDC 10, "Reactor Design"
- GDC 12, "Suppression of Reactor Power Oscillations"
- GDC 17, "Electric Power Systems"
- GDC 20, "Protection System Functions"
- GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"
- GDC 26, "Reactivity Control System Redundancy and Capability"
- GDC 33, "Reactor Coolant Makeup"
- GDC 34, "Residual Heat Removal"

GDC 10, 17, 20, and 26 use this wording in conjunction with anticipated operational occurrences and conditions of normal operation. For example, GDC 10 requires "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." As discussed later in this section, SAFDLs for a plant are described in plant documentation, typically the UFSAR or the FSAR, and are met by operating within technical specifications limits.

NRC regulations also specify that certain conditions beyond steady-state operation be included in evaluations of the normal operating regime for a plant. These are called anticipated operational occurrences (AOOs) and are sometimes referred to as "anticipated operating transients." In Appendix A to 10 CFR Part 50, the staff defines AOOs as "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit." GDC 29, "Protection Against Anticipated Operational Occurrences," gives a general requirement for protection system and reactivity control system performance during AOOs, but does not mention fuel integrity. Examples of AOOs are the loss of all reactor coolant pumps, turbine trip events, and loss of control power. Such occurrences are distinct from events termed "accidents," such as a loss-of-coolant accident (LOCA) or a main steamline break. The references to fuel integrity requirements related to accidents and those regarding emergency core cooling system (ECCS) performance are beyond conditions of normal operation.

The UCS report relates other regulations beyond the GDC to fuel integrity during normal operation as follows:

- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors"
- 10 CFR 50.36, "Technical specifications"
- 10 CFR 50.59, "Changes, tests and experiments"

- 10 CFR 50.71, “Maintenance of records, making of reports”
- Appendix I to 10 CFR Part 50, “Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion ‘As Low As Is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents”

Although 10 CFR 50.36a, “Technical specifications on effluents from nuclear power reactors,” was not directly referenced in the report, by citing 10 CFR 50.36, the staff inferred that Section 50.36a is linked to fuel integrity when considering the discussion on the UCS report.

b. NRC Staff Guidance Documents

To implement NRC regulations, several NRC staff guidance documents are used, including the following:

- Regulatory Guide 1.3, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors”
- Regulatory Guide 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors”
- Regulatory Guide 1.77, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors”
- Regulatory Guide 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors”
- SRP Section 4.2, “Fuel System Design”
- SRP Section 4.4, “Thermal and Hydraulic Design”

Along with the regulations, licensees use the guidance documents listed above to form the licensing basis for fuel integrity at their plant. The licensing basis for a nuclear power plant, as defined in 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Reactors,” is “the set of NRC requirements applicable to a specific plant and a licensee’s written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis...that are docketed and in effect.” The definition continues by listing elements of the licensing basis, such as technical specifications, the FSAR, and licensee commitments documented in NRC safety evaluations. Several components form the plant’s licensing basis for fuel performance: (1) NRC regulations that specifically refer to fuel integrity; (2) technical specification limits on coolant activity; (3) fuel rod performance specifications and analysis assumptions defined in the plant’s FSAR and referenced topical reports; and (4) commitments to NRC regulatory guidance and to generic communications addressing fuel performance.

Acceptance criteria in the SRP sections, which may be adopted by licensees to implement the regulations, are based on meeting the requirements of GDC 10 with appropriate margin to ensure that SAFDLs are not exceeded during normal operation, including AOOs. Specifically, SRP Section 4.2 has as an objective of the safety review “to provide assurance that the fuel system is not damaged as a result of normal operation and anticipated operational occurrences.” The reviewer should ensure that fuel does not leak as a result of specific causes during normal operation and AOOs, and that leaking fuel is accounted for in the dose analyses for postulated design-basis accidents. Further, fuel rod failure is defined in SRP Section 4.2 as “the loss of fuel rod hermiticity,” meaning fuel rod leakage. However, in SRP Section 4.2, the staff also states that “it is not possible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods.” Such leaks typically occur as a result of manufacturing flaws or loose parts wear. Therefore, on the basis of this review guidance, the staff accepts the possibility that fuel may leak during normal operation.

In the case of the Calvert Cliffs Nuclear Plant, a plant cited as an example in the UCS report, the plant's licensing basis contains a commitment to adhere to the guidance in the SRP. The following four objectives for fuel design given in SRP Section 4.2 may be used as fuel design objectives within a plant's licensing basis as is done in the Calvert Cliffs FSAR:

- Fuel is not damaged as a result of normal operation and AOOs.
- Fuel damage is never so severe as to prevent control rod insertion when required.
- The number of fuel rod failures is not underestimated for postulated accidents.
- Coolability is always maintained.

SRP Section 4.4 has as an objective that the thermal and hydraulic design of the core should provide acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation, including anticipated operational transients. It gives two examples of acceptable approaches to meet the acceptance criteria: one based on a 95-percent probability at a 95-percent confidence level that the hottest rod in the core does not exceed prescribed thermal limits during normal operation, including AOOs, and the other using a limiting value for thermal limits so that at least 99.9 percent of the fuel rods are not expected to exceed thermal limits during normal operation, including AOOs. These criteria are limits that strive to maintain a very low likelihood of fuel damage during operation; however, they do not preclude the possibility that some fuel defects could occur.

A plant's licensing basis contains fuel performance criteria that are specified for normal operation, including AOOs, and analyses are conducted to ensure that these criteria will not be exceeded. The criteria are related to the SAFDLs mentioned in the GDC and are normally presented in terms of prescribed thermal limits, which can be calculated and are reliable predictors of the onset of fuel damage. For boiling-water reactors (BWRs), critical heat flux or the critical power ratio is used as the predictor of fuel damage onset, and for pressurized-water reactors (PWRs), the criterion is the departure from nucleate boiling (DNB), or the DNB ratio (DNBR).

An example of fuel design limits given in plant documentation is found in the FSAR for Calvert Cliffs Units 1 and 2. Section 3.6 of the FSAR presents fuel design and analysis bases. Fuel rod cladding is designed to stress and strain limits, considering the operating temperature, the cladding material, the expected property changes as a result of irradiation, and the predicted life span of the fuel. Extensive fuel mechanical analyses are detailed, along with pertinent fuel test data, which help to confirm the analysis results. The calculations are used to demonstrate that the criteria are satisfied for limiting cases under limiting assumptions. Chapter 14 of the Calvert Cliffs FSAR gives the fuel behavior acceptance criteria for each category of design-basis event analyzed. For AOOs, the minimum DNBR is chosen to provide at least a 95-percent probability with a 95-percent confidence level that DNB will not be experienced along the fuel rod with that DNBR (i.e., the SRP Section 4.4 criteria). This limit ensures that there is a low probability of fuel rod damage as a result of overheated cladding. The fuel temperature SAFDL is set so that no significant fuel melting will occur during steady-state operation or during a transient. Compliance with the limit offers assurance that the fuel rod will not be damaged as a result of material property changes and increases in fuel pellet volume, which could be associated with fuel melting. Again, as with the limits discussed in SRP Section 4.4, these limits are set to prevent fuel damage, but the possibility of fuel leakage is recognized.

The key to plant licensing bases regarding fuel integrity is the technical specification limiting the concentration of activity allowed in reactor coolant during plant operation. These limits are based on maintaining a margin to the dose guidelines in 10 CFR Part 100 for steam generator tube rupture (SGTR) accidents in PWRs and main steamline break (MSLB) accidents in BWRs. The specific activity limits of the reactor coolant system are stated in terms of dose equivalent iodine-131, which is attributable solely to fuel leaks. That is distinct from gross coolant activity, which is the aggregate activity from all sources, including fuel leaks and corrosion product activation. The technical basis for these limits can be traced to the guidance given in Appendix I, which is, in turn, based on assumptions that fuel leaks would exist during operation. Technical specifications for reactor core safety limits, including the reactor protection system setpoints, are set so that the SAFDLs are not exceeded during normal operation or AOOs. The technical specifications for protection system action are intended to prevent fuel damage, but the specifications for coolant activity levels recognize that some small amount of fuel leakage is allowable during operation. The technical specifications concerning coolant activity are based on meeting the dose acceptance criteria in the SRP for the limiting design-basis accident (usually SGTR or MSLB for PWRs and MSLB for BWRs). These limits are used as assumptions in design-basis accident dose analyses to show compliance with dose acceptance criteria for the control room operators and the public. By maintaining the levels of coolant activity within these limits during normal operation, the continued validity of the design-basis analyses is maintained.

The staff has addressed fuel performance problems in several generic communications to licensees. Prominent among these were NRC Information Notice (IN) 93-82, “Recent Fuel and Core Performance Problems in Operating Reactors,” and Generic Letter (GL) 90-02, “Alternative Requirements for Fuel Assemblies in Design Features Section of Technical Specifications.” In IN 93-82, the staff discussed fuel leaks occurring during normal operation from a specific cause—fretting wear in PWR fuel, which was partly attributed to mixed fuel core designs. The staff alerted licensees to the introduction of modified fuel designs that requires added attention to ensure that the core design basis is not violated. This information notice is an example of staff action to use operating information gathered from fuel leaks at a few plants to avoid similar problems at other reactors, thus reducing the potential for more widespread fuel leakage. In GL 90-02, the staff provided licensees with added flexibility to take actions to reduce fission-product releases during operation by removing defective fuel rods during refueling outages.

The staff has previously considered the safety implications of operation with fuel leakage on a generic basis. Generic Safety Issue (GSI) B-22, "LWR [Light Water Reactor] Fuel," which is related to fuel leakage, is discussed in NUREG-0933, "A Prioritization of Generic Safety Issues," Supplement 22, March 1998. In GSI B-22, the staff considered the ability to accurately predict fuel performance under normal and accident conditions. The GSI review was conducted to determine if predictions of fuel behavior under normal operating and accident conditions were sufficient to demonstrate that regulatory requirements were being met. In its evaluation of the issue, the staff concluded that releases during normal operation would be increased because of fuel defects, but would not be increased beyond regulatory limits. The staff also stated that, "additional requirements would not decrease the number of fuel defects significantly." Furthermore, the staff concluded that the release from fuel damaged during design-basis accidents and severe accidents would be much larger than the release attributed to preexisting fuel defects, and the magnitude of the release would not be significantly affected by preexisting fuel defects. Thus, the consequence from leaking fuel was determined to be very small. The staff concluded that because fuel manufacturers have taken an active role to improve fuel performance, fuel leaks are now rare, and the significance of the issue has diminished. Therefore, the issue was dropped from further consideration.

In the resolution of GSI B-22, the staff concluded that the influence of additional restrictions to operation with fuel leaks on core damage frequency and public consequence would be insignificant. Thus, operation with a limited number of fuel defects and leaks under normal operating conditions is not associated with an excessive level of risk, provided that the plant continues to operate within technical specifications limits for reactor coolant activity.

3. Evaluation of Generic Concerns

The staff evaluated the generic concerns associated with fuel leakage identified previously by the Petitioner, as follows:

a. 10 CFR 50.59, "Changes, tests and experiments"

A premise of the UCS report is that 10 CFR 50.59 is violated because reactor operation with limited fuel leakage constitutes an unapproved change to the licensing basis for a plant. The report states that "Federal regulations require formal NRC approval prior to any nuclear plant operating with fuel cladding failures." The attachment to the report is an assessment of operation with fuel leaks as an unreviewed safety question on the basis of the criteria in 10 CFR 50.59. The report states that such operation is an unreviewed safety question because operation with leaking fuel (1) increases the probability and consequences of an accident, (2) creates an accident different from any in the safety analysis for the plant, and (3) reduces safety margins.

The staff does not agree that operation with leaking fuel necessarily constitutes a change to or violation of the licensing basis for a plant. A small amount of fuel leakage during operation is permitted by NRC staff guidance implementing NRC regulations and is accounted for in plant licensing bases. A key component of the licensing basis regarding fuel performance is the technical specification limiting reactor coolant system activity. The fission-product release from the level of leaking fuel associated with the technical specification limit is included in the design-basis accident dose analyses described in the FSAR for a plant to show compliance with the dose acceptance criteria in the SRP. Therefore, operating with leaking fuel, within the coolant activity technical specification limits, does not constitute a change in the plant licensing basis, and 10 CFR 50.59 does not apply.

b. 10 CFR 50.71, "Maintenance of records, making of reports"

The Petitioner states in the report that "any plant operating with fuel cladding failures is violating its design and licensing bases requirements, a condition not allowed by Federal safety regulations." The Petitioner further states that when plants operate with leaking fuel, 10 CFR 50.71 is violated since the licensing basis for a plant, as documented in the technical specifications and in the analyses contained in the FSAR, does not accommodate such operation.

This concern is closely linked to the previous discussion regarding 10 CFR 50.59, in that FSARs for plants operating with leaking fuel should, in the view of the UCS, include safety analyses accounting for the effects of fuel leaks. As previously discussed, plant licensing bases do incorporate assumptions for limited levels of fuel leakage through technical specifications requirements and designs for plant reactor water cleanup systems. Plant FSARs, including the example discussed earlier in this evaluation, typically contain information on fuel leakage effects, and the safety analyses explicitly allow for coolant activity levels attributable to leaking fuel under normal operation. Thus, the staff does not consider 10 CFR 50.71 to be violated by operation with fuel leakage.

c. Safety Analysis Assumptions

The UCS report states that "safety analyses assume that <u>all</u> three barriers [to radioactive material release] are intact prior to any accident." Therefore, according to the UCS, plants with known fuel leakage could have accidents with more severe consequences than predicted. The report also states the following: "Pre-existing fuel cladding failures have not been considered in the safety analyses for this accident [LOCA], or any other accident."

In the discussion that follows, the staff explains that preexisting fuel cladding leaks are accounted for in plant licensing bases and that safety analyses do not assume that all the fission-product barriers are fully intact before an accident.

The analyses of limiting postulated design-basis releases do not assume that all the fission-product barriers are fully intact before an accident. For the loss-of-coolant accident, which typically yields the most limiting postulated releases, all three barriers are assumed to allow the release of some fission products. The methodology used to analyze this accident is given in Regulatory Guides 1.3 and 1.4, and SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary."

For the containment and reactor coolant system (RCS) barriers, these assumptions are explicitly given. The containment is assumed to leak at the leak rate incorporated in the plant technical specifications when the containment is at positive pressure. The RCS inside the containment is assumed to completely fail as a fission-product barrier at the beginning of the accident. Systems outside the containment that interface with the RCS are also assumed to experience failures.

The assumption of preexisting leakage for the fuel cladding barrier, although not explicitly given, is inherent in the assumption of a conservative nonmechanistic release from the fuel. The entire iodine and noble gas inventory of the core is assumed to be released to the reactor coolant. A conservative fraction of this inventory is assumed to be released into the containment and subsequently released to the environment. Assuming that this release occurs instantaneously further enhances the conservatism of these analyses. This assumption disregards the fission-product containment function of the fuel cladding at the beginning of the accident.

Accidents, which may not be bounded by the radiological consequences of a LOCA, include the control rod drop accident for BWRs and MSLB outside of containment for PWRs. However, the conservatism of the source term assumptions for these analyses parallels those for a LOCA. Some of the same assumptions used for radiological consequence evaluation of a LOCA are used for the analysis of MSLB outside of containment. Appendix A to SRP Section 15.1.5, “Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR,” contains an acceptance criterion that references Regulatory Guide 1.4. The radiological assumptions for the control rod drop analysis are similar to those for a LOCA, as stated in Appendix A to SRP Section 15.4.9, “Radiological Consequences of Control Rod Drop Accident (BWR),” and Regulatory Guide 1.77. For example, the guidelines assume that the nuclide inventory in the potentially breached fuel elements should be calculated and it should be assumed that all gaseous constituents in the fuel cladding gaps are released.

The radioactivity assumed for release from the LOCA is much greater than that associated with preexisting fuel leakage allowed by plant technical specifications. The staff has compared releases from preexisting defects with the release resulting from fuel damage during an accident. In its consideration of GSI B-22, the staff concluded that, “the magnitude of a release from failed fuel during an accident is much larger than the release from a preexisting fuel defect” and that “the resultant consequence from failed fuel was determined to be very small” (NUREG-0933). These assumptions are made despite the provisions of 10 CFR 50.46 requiring an ECCS that must be designed to prevent exceeding thermal limits that cause such gross fuel failure. In addition, for design-basis accidents in which fuel damage is not assumed, the preexisting fuel cladding defects are typically assumed to serve as release paths facilitating a spike in radioiodine concentration in the coolant.

Additional NRC fuel design requirements complement the conservative defense-in-depth assumptions as previously described to prevent an unanalyzed large release of fission products. To illustrate its concern about fuel leakage influences on accident progression, the UCS report describes a LOCA sequence and postulates that hydraulic loads on the fuel rods could lead to cladding failures, which would result in a large release of fission products into the coolant and prevent control rod insertion. Fuel design requirements and guidance specifically address the ability to insert control rods, and staff review guidance recognizes that preexisting fuel cladding defects could have an effect on fuel performance during accidents. In GDC 27, "Combined Reactivity Control Systems Capability," the staff requires that reactivity control systems, including the control rod system, have the capability to control reactivity changes under postulated accident conditions in order to assure core cooling. SRP Section 4.2 includes the objective that "fuel system damage is never so severe as to prevent control rod insertion when it is required."

To ensure that the preceding objective is met, fuel designs consider external loads on fuel rods. This is discussed in the appendix to SRP Section 4.2, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces." The basis for much of the appendix to SRP Section 4.2 is contained in NUREG/CR-1018, "Review of LWR Fuel System Mechanical Response With Recommendations for Component Acceptance Criteria," prepared by EG&G Idaho in September 1979. This report states that "Cyclic fatigue and material degradation may cause a failure [of a fuel system component] at any point in the transient [i.e., a LOCA]." Thus, material degradation that could lead to fuel leakage during operation is considered in accident analyses. Furthermore, design considerations, such as control guide tubes in PWRs and fuel channel boxes in BWRs, help separate control rods from the fuel. The separation provided protects control rods from material degradation of fuel that might occur in accidents, thus helping to prevent control rod obstruction. Such safety analysis assumptions as these (which assume preexisting failures of the fission-product barriers) provide confidence that the preexisting cladding defects allowed by technical specifications limits on coolant activity will not erode the safety margin assumed for accident analyses.

d. 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors"

In its report, the UCS claims that 10 CFR 50.34a and other regulations related to the ALARA principle for radioactive materials release are violated since plant workers are exposed to a greater risk than necessary because of higher coolant activity levels attributable to leaking fuel. The UCS report continues: "Federal regulations require nuclear plant owners to keep the release of radioactive materials as low as reasonably achievable. Therefore, it is both an illegal activity and a serious health hazard for nuclear plants to continue operating with fuel cladding damage." The UCS report cites Appendix I to 10 CFR Part 50 when contending that fuel releases pose an undue risk to plant workers. Appendix I contains the numerical dose guidelines for power reactor operation to meet the ALARA criterion. These dose values are a small fraction of the 10 CFR Part 20 annual public dose limit of 100 millirem (i.e., 3 millirem from liquid effluents and 5 millirem from gaseous effluents).

The bases for the guidelines in Appendix I are given in WASH-1258, which acknowledges that radioactive material from a number of sources, including fission-product leakage to the coolant from defects in the fuel cladding, will be present in the primary coolant during normal operation. Further, in the "Bases" section on RCS specific activity in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," April 1995, the limits on specific activity are linked to exposure control practices at plants. The section clearly states that the limits on RCS specific activity are used in the design of radiation shielding and plant personnel radiation protection practices.

In addition, occupational dose considerations were discussed in the resolution of GSI B-22. The staff acknowledged that localized dose rates were expected to increase as a result of fuel defects, but effects are limited by requirements for plants to operate within their technical specifications for coolant activity and releases. In some cases, plants will often stay within allowable release limits and coolant activity levels by operating at reduced power until the next refueling outage allows the problem to be corrected.

On the basis of the preceding discussion, operation with a limited amount of leaking fuel is within a plant's licensing basis and, in itself, does not violate ALARA-related regulations. Operation involving leaking fuel, however, will likely require plant operators to take additional measures in order to ensure that ALARA requirements are being met, but these would need to be considered on a case-by-case basis.

4. UCS Report Recommendations

In the report, the UCS recommends that the NRC take steps to prohibit nuclear power plants from operating with fuel cladding damage until the safety concerns raised by the report are resolved. The following steps are specifically recommended: (1) requiring plant shutdown upon detection of fuel leakage, and (2) requiring that safety evaluations that consider the effects of operating with leaking fuel be included in plant licensing bases to justify operation under such circumstances. Further, the UCS recommends that UFSARs be revised to establish safe operating limits to accommodate operation with leaking fuel.

On the basis of the staff's consideration of the stated safety concerns in the report, there is no technical or regulatory basis to require that plants operating with leaking fuel be shut down, provided they are operating within their technical specifications limits and in accordance with their licensing basis. The UCS report, in raising its concerns, does not offer any new information to demonstrate that the overall risk of operating with fuel defects presents an undue hazard to plant workers or the public.

Further, since the staff does not consider plants operating with leaking fuel to be violating 10 CFR 50.59 or 50.71, there is no basis for requiring plants to perform additional safety analyses to model the effects of fuel defects on accident progressions to update plant safety analysis documentation.

B. Plant-Specific Concerns - River Bend Station

On the basis of the reported fuel leakage at River Bend, the Petitioner states that the generic concerns contained in its report apply to River Bend. The September 25, 1998, Petition then presents a number of references to the River Bend USAR as instances in which, in the opinion of the Petitioner, plant licensing bases do not permit operation of the plant with known fuel leakage.

A reference to the USAR in the Petition is the USAR definition of unacceptable consequences (USAR Table 15A.2-4), which lists as an unacceptable consequence "Failure of the fuel barrier as a result of exceeding mechanical or thermal limits." The Petitioner considers this criterion violated since a fuel failure exists in advance of any design-basis accident that may now occur.

The Petition then discusses USAR Chapter 15 accident analysis descriptions, which state either (1) that fuel cladding integrity will be maintained as designed or (2) radioactive material is not released from the fuel for the event. The following events cited in the Petition have event descriptions in the River Bend USAR, which state that fuel cladding will function and maintain its integrity as designed:

- Loss of Feedwater Heating (USAR Section 15.1.1.4)

- Feedwater Controller Failure—Maximum Demand (USAR Section 15.1.2.4)
- Pressure Regulator Failure—Open (USAR Section 15.1.3.4)
- Pressure Regulator Failure—Closed (USAR Section 15.2.1.4)

The following two events cited in the Petition have event descriptions in the River Bend USAR, which state that "no radioactive material is released from the fuel" during the event:

- Control Rod Withdrawal Error at Power (USAR Section 15.4.2.5)
- Recirculation Flow Control Failure with Increasing Flow (USAR Section 15.4.5.5)

The Petitioner also states that the River Bend licensing basis for worker radiation protection is violated by operation with leaking fuel. Again, the Petition cites the USAR (Sections 12.1.1 and 12.1.2.1) as the pertinent reference to the licensing basis.

<u>Evaluation of Plant-Specific Concerns</u>

As discussed in the consideration of generic safety concerns, the staff does not agree that preexisting fuel cladding defects and resultant fuel leakage violate plant licensing bases. The staff also considers that conclusion valid for River Bend. The basis for this conclusion is supported in the following discussion.

a. USAR Appendix 15A

The Petitioner referenced two sections of USAR Appendix 15A, "Plant Nuclear Safety Operational Analysis (NSOA)" (as stated):

UFSAR 15A.2.8, "General Nuclear Safety Operational Criteria," stated:

> The plant shall be operated so as to avoid unacceptable consequences.

> UFSAR Table 15A.2-4, “Unacceptable Consequences Criteria Plant Event Category: Design Basis Accidents,” defined ‘unacceptable consequences’ as follows:
>
> 4-1 Radioactive material release exceeding the guideline values of 10CFR100.
> 4-2 Failure of the fuel barrier as a result of exceeding mechanical or thermal limits.
> 4-3 Nuclear system stresses exceeding that allowed for accidents by applicable industry codes.
> 4-4 Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.
> 4-5 Overexposure to radiation of plant main control room personnel.
>
> The current operating condition at the River Bend Station apparently violates the spirit, if not the letter, of Criterion 4-2 since the fuel barrier has already failed, albeit to a limited extent. This UFSAR text does <u>not</u> accept a low level of fuel barrier failure based on meeting the offsite and onsite radiation protection limits. Integrity of the fuel barrier is an explicit criterion in addition to the radiation requirements.

In the Petition, the UCS highlights the table concerning the consequences for the design-basis accident. This plant condition is a highly improbable event, and safety analyses ensure that safety limits and regulatory requirements are not exceeded as a result of the accident occurring. This is why USAR Table 15A.2-4, Item 4-2 states, “Failure of a fuel barrier <u>as a result of exceeding mechanical or thermal limits</u>” (emphasis added). The unacceptable consequences of this type of event are independent of preexisting fuel cladding defects. The unacceptable consequences of this event are additional fuel failures as a result of the accident occurring.

Within the framework of the USAR, “unacceptable consequences” are specified measures of safety and analytically determinable limits on the consequences of different classifications of plant events. They are used for performing a nuclear safety operational analysis. Unacceptable consequences are described for various plant conditions, including “Normal (Planned) Operation,” “Anticipated (Expected) Operational Transients,” “Abnormal

(Unexpected) Operational Transients," "Design Basis (Postulated) Accidents," and "Special (Hypothetical) Events." USAR Tables 15A.2-1 through 15A.2-5 identify the unacceptable consequences for each of the five plant conditions, and are different for each of the cases.

The USAR text clearly documents the acceptability of a low level of fuel cladding failures based on meeting the offsite and onsite radiation protection limits. For example, USAR Table 15A.2-1 discusses the unacceptable consequences for normal operation. This USAR table defines unacceptable consequences for normal operation as follows:

4-1 Release of radioactive material to the environs that exceeds the limits of either 10 CFR Part 20 or 10 CFR Part 50.

4-2 Fuel failure to such an extent that were the freed fission products released to the environs via the normal discharge paths for radioactive material, the limits of 10 CFR Part 20 would be exceeded.

4-3 Nuclear system stress in excess of that allowed for planned operation by applicable industry codes.

4-4 Existence of a plant condition not considered by plant safety analysis.

Item 4-2 in Table 15A.2-1 implies that fuel cladding failures are not an unanticipated condition during normal operations and is, therefore, consistent with other parts of the River Bend licensing-basis. Fuel cladding defects are acceptable to the extent that they do not jeopardize radiation protection limits established in the plant technical specifications and other licensing-basis documents. USAR Table 15A.2-4 does not apply for normal operations; only USAR Table 15A.2-1 applies. Furthermore, the provisions found in USAR Table 15A.2-4 would continue to be met for postulated design-basis accidents.

USAR Section 15.0.3.1.1 provides further clarification in its list of unacceptable safety consequences for "moderate frequency" events, which lists: "Reactor operation induced fuel-cladding failure as a direct result of the transient analysis above the minimum critical power ratio (MCPR) uncertainty level (0.1 percent)." Accordingly, preexisting cladding defects are considered during some postulated transients. In fact, the acceptance criteria for moderate-frequency event analyses, based on the GDC (10 CFR Part 50, Appendix A) and the Standard Review Plan, and described in the Safety Evaluation Report (SER) for River Bend (NUREG-0989), state the following expectations for fuel cladding performance: "An incident of moderate frequency…should not result in a loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel rod cladding perforations are acceptable."

USAR Chapter 11, "Radioactive Waste Management," Section 11.1, "Source Terms," details the expected reactor coolant and main steam activities to be used to form the basis for estimating the average quantity of radioactive material released to the environment during normal operations, including operational occurrences. This section further addresses that the offgas release rate of 304,000 μCi/sec at a 30-minute delay time corresponds to design failed fuel conditions, that is, maximum acceptable cladding failure for normal operation, and is also conservatively based upon 105 percent of rated thermal power. This is consistent with limits prescribed in Technical Specification 3.7.4, "Main Condenser Offgas," which requires that the gross gamma activity rate of the noble gases shall be <290 mCi/sec (or <290,000 μCi/sec) after a decay time of 30 minutes.

In addition, two other parts of the fuel system licensing basis for River Bend show that limited fuel leakage during plant operation is a design consideration:

The fuel system design basis for River Bend is given in USAR Section 4.2.1 by reference to the generic topical report "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A. The generic topical report details fuel cladding operating limits to ensure that fuel performance is maintained within fuel rod thermal and mechanical design and safety analysis criteria. The limits are given for normal operating conditions and AOOs in terms of specific mechanical and thermal specifications. Evaluations of specific fuel failure mechanisms under normal operation and AOOs were discussed, such as stress/strain, hydraulic loads, fretting, and internal gas pressure to ensure that fuel failure did not result from these causes. The design basis did not preclude the possibility that fuel could fail for other reasons, such as preexisting cladding flaws leading to leakage.

The Technical Specifications (Section 3.4.8) for River Bend contain a limit for reactor coolant system specific activity. The basis for this limit is the same as that discussed in the consideration of the generic safety concerns. Section B 3.4.8 of the River Bend Technical Specifications “Bases” acknowledges that “the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks.” Thus, fission products released during plant operation are clearly considered to be contributors to the source term used for safety analysis of the MSLB release consequences. The Technical Specifications state that the limit is set to ensure that any release as a consequence of an MSLB is less than a small fraction of the 10 CFR Part 100 guidelines. These portions of the River Bend licensing basis are consistent with NRC regulations regarding fuel performance and the associated NRC guidance used by licensees to implement those NRC regulations that were covered earlier in the discussion regarding generic concerns.

The River Bend licensing-basis items listed by the Petitioner are consistent with the parts of the fuel licensing basis discussed above with the exception of some minor inconsistencies in documentation (as discussed below). That is, fuel leakage during plant operation is not precluded by licensing-basis provisions requiring that fuel integrity be maintained as designed. The design basis itself allows the possibility of leakage while ensuring that cladding damage does not result from specific operationally related causes. Fuel is also designed to maintain its structural integrity to ensure core coolability and to ensure that control rods can be inserted.

b. Chapter 15 Accident Analysis

The Petitioner also cited references taken from accident analyses described in River Bend USAR Chapter 15 (as stated):

> UCS reviewed the UFSAR Chapter 15 description of accident analyses performed for the River Bend Station. UFSAR Section 15.1.1.4, "Barrier Performance," for the loss of feedwater heating event stated:
>
> > The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed; therefore, these barriers maintain their integrity and function as designed.
>
> UFSAR Sections 15.1.2.4 for the feedwater controller failure - maximum event, 15.1.3.4 for the pressure regulator failure - open event, and 15.2.1.4 for the pressure regulator failure - closed event all contain comparable statements that barrier performance was not performed because the fuel remained intact.
>
> These analyzed events appear to be valid only when the River Bend Station is operated with no failed fuel assemblies. Operation with pre-existing fuel failures (i.e., the current plant configuration) appear to be outside of the design and licensing bases for these design bases events.
>
> UFSAR Section 15.4.2.5, "Radiological Consequences," for the control rod withdrawal error at power event stated:
>
> > An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.
>
> UFSAR Section 15.4.5.5, "Radiological Consequences," for the recirculation flow control failure with increasing flow event stated:
>
> > An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.
>
> These analyzed events also appear valid only when the River Bend Station is operated with no failed fuel assemblies. Operation with pre-existing fuel failures (i.e., the current plant configuration) appear to be outside of the design and licensing bases for these design bases events.
>
> The effect from pre-existing fuel failures was considered, at least partially, for one design bases event. UFSAR Section 15.2.4.5.1, "Fission Product Release from Fuel," for the main steam isolation valve closure event stated:

> While no fuel rods are damaged as a consequence of this event, fission product activity associated with normal coolant activity levels as well as that released from previously defective rods is released to the suppression pool as a consequence of SRV [safety relief valve] actuation and vessel depressurization.

The aforementioned design bases events (e.g., control rod withdrawal error at power, loss of feedwater heating, et al) are <u>not</u> bound by these results because the radioactive material is not "scrubbed" by the suppression pool water as it is in the MSIV [main steam isolation valve] closure event.

As previously stated, the Petitioner cited four references to the USAR accident analysis section entitled "Barrier Performance." At issue are essentially equivalent statements made where the USAR stated, in part, that the defense-in-depth "barriers maintain their integrity and function as designed." The UCS concluded that operation with preexisting fuel failures is, therefore, outside the River Bend design and licensing bases. In stating that barriers are "maintained," the USAR clearly implies that the events themselves do not result in additional fuel cladding failures. To further support this conclusion, the radiological consequences described for three of the four events (Section 15.1.2, "Feedwater Controller Failure—Maximum Demand"; Section 15.1.3, "Pressure Regulator Failure—Open"; and Section 15.2.1, "Pressure Regulator Failure—Closed") are, indeed, bounded by an event that takes into consideration the effects of preexisting cladding failures. The three preceding events all result in actuation of the safety relief valves (SRVs) to the suppression pool. The USAR discussion (see USAR section titled "Radiological Consequences") notes that radioactivity is discharged to the suppression pool, and that the activity discharged is much less than those consequences identified in USAR Section 15.2.4.5 (for the MSIV closure event).

The MSIV closure event, as described in the USAR, clearly considers the activity released from "previously defective rods" in determining dose consequences. The source term used in these calculations assumes the same iodine and noble gas activity as an initial condition as is used in the basis for determining RCS activity technical specifications limits.

USAR Section 15.2.4.5.1, “Fission Product Release from Fuel,” also explains, “Since each of those transients identified previously which cause SRV actuation results in various vessel depressurization and steam blowdown rates, the transient evaluated in this section [the MSIV closure event] is that one which maximizes the radiological consequences for all transients of this nature.” Thus, the USAR explicitly describes how “the aforementioned design-basis events” are bounded by the results for the MSIV closure event, for those events resulting in an SRV actuation. Furthermore, USAR Section 15.1.1.5 describing the fourth event, the loss of feedwater heating, also states that “this event does not result in any additional fuel failures,” further reinforcing the staff’s position.

The quotation taken from the control rod withdrawal error from power and recirculation flow control error event descriptions—“[a]n evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel”—appears to be taken out of context. Considering the many references ostensibly permitting operation with preexisting fuel cladding failures found within the USAR, technical specifications, NRC regulations, staff implementing guidelines, and other licensing-basis documents, the intent of this statement is clearly that no <u>additional</u> radioactive material is released from the fuel as a consequence of the event.

Finally, in each of the accident analysis cases listed in the Petition, the event is classified as a “moderate frequency” event (or an “anticipated operational transient”). Specific criteria for unacceptable consequences are delineated in USAR Table 15A.2-2. For this type of anticipated transient, unacceptable performance of the fuel is described as, “[r]eactor operation induced <u>fuel cladding failure as a direct result of the transient</u> analysis above the MCPR [Minimum Critical Power Ratio] uncertainty level (0.1%)” (emphasis added). Therefore, fuel cladding defects existing before the accident are not precluded from consideration.

c. Fuel Cladding Defect Propagation

The Petition then raised concerns regarding the possibility that preexisting fuel cladding defects could propagate under design-basis transients (as stated):

> As detailed in UCS's April 1998 report on reactor operation with failed fuel cladding, it has not been demonstrated that the effects from design basis transients and accidents (i.e., hydrodynamic loads, fuel enthalpy changes, etc.) prevent pre-existing fuel failures from propagating. It is therefore possible that significantly more radioactive material will be released to the reactor coolant system during a transient or accident than that experienced during steady state operation. Thus, the existing design bases accident analyses for River Bend Station do not bound its current operation with known fuel cladding failures.

As previously stated in the evaluation of generic issues raised by the April 1998 UCS report, the staff has previously considered the safety implications of operation with fuel leakage on a generic basis. In GSI B-22, the staff considered the ability to accurately predict fuel performance under normal and accident conditions. In its evaluation of the issue, the staff concluded that releases during normal operation would be increased because of fuel defects, but would not be increased beyond regulatory limits. The staff also concluded that the release from fuel damage during design-basis accidents and severe accidents would be much larger than the release attributed to preexisting fuel defects, and the magnitude of the release would not be significantly affected by preexisting fuel defects. Therefore, the consequence from leaking fuel was determined to be very small.

The Petitioner has, however, noted some apparent inconsistencies in documentation of the licensing basis as found in the USAR for River Bend that could be taken out of context. The statements cited for two events—the control rod withdrawal error from power and recirculation flow control error—are not consistent with the other parts of the River Bend licensing basis discussed in this evaluation. The technical basis for coolant activity limits clearly permits operation with a limited amount of fuel leakage and, as discussed, the design basis does not preclude the possibility of limited fuel leakage during operation. Therefore, although these events should not cause fuel damage, preexisting leakage could still be a consideration, and only the activity in the reactor system coolant up to the technical specification limit would be available for release. The MSLB is considered the limiting event with respect to release of coolant activity from leaking fuel. The staff expects that the consequences of the MSLB would bound those that would be predicted for the control rod withdrawal error from power or the recirculation flow control error events. Thus, the minor discrepancies uncovered by the Petitioner in the documentation of the plant licensing basis do not constitute a safety concern requiring NRC action.

The licensee has taken actions to limit the effects of the minor fuel rod defects at River Bend reported on September 21, 1998. The control rod pattern has been altered to achieve a depressed flux profile in the vicinity of the leaking rods, thereby suppressing the production of fission products as the plant continues operation at slightly less than full power. Following the initial detection of a leaking rod, the licensee reduced the activity in the pretreatment offgas sample from 22.5 mCi/sec to 1.8 mCi/sec, which was very close to the prefuel-leak level of 1 mCi/sec. The peak value was never more than a small fraction of the technical specification limit of 290 mCi/sec. The offgas treatment system has been effectively eliminating any detectable radioactivity in offgas effluent, and only small dose rate increases were observed in areas of the plant in which offgas system components are located. Since work is not normally performed in those areas, the licensee did not institute any additional exposure controls. However, the licensee is continuing to closely monitor the offgas system to ensure that the coolant activity concentration remains within technical specifications limits.

d. ALARA Concerns

The Petitioner further stated that Entergy Operations, Inc., was violating its licensing basis with regard to the ALARA worker protection program (as stated):

> In addition to operating with non-bounding design bases accident analyses, it appears that the River Bend licensee is also violating its licensing basis for worker radiation protection. UFSAR Section 12.1.1, “Policy Consideration,” stated:
>
> > The purpose of the ALARA [as low as reasonably achievable] program is to maintain the radiation exposure of plant personnel as far below the regulatory limits as is reasonably achievable.
>
> UFSAR Section 12.1.2.1, “General Design Considerations for ALARA Exposures,” stated that River Bend’s efforts to maintain in-plant radiation exposure as low as is reasonably achievable included:
>
> > Minimizing radiation levels in routinely occupied plant areas and in vicinity of plant equipment expected to require the attention of plant personnel.
>
> According to the NRC Information Notice No. 87-39, “Control of Hot Particle Contamination at Nuclear Plants:”
>
> > A plant operating with 0.125 percent pin-hole fuel cladding defects showed a five-fold increase in whole-body radiation exposure rates in some areas of the plant when compared to a sister plant with high-integrity fuel (<0.01 percent leakers). Around certain plant systems the degraded fuel may elevate radiation exposure even more.
>
> Industry experience demonstrated that reactor operation with failed fuel cladding increased radiation exposures for plant workers. The River Bend licensee has a licensing basis requirement to maintain radiation exposures for plant workers as low as is reasonably achievable. The River Bend licensee informed the NRC about potential fuel cladding failures. It could shut down the facility and remove the failed fuel assemblies from the reactor core. Instead, it continues to operate the facility with higher radiation levels.

In its letter to the NRC dated February 11, 1999, the River Bend licensee stated that if the plant were to shut down solely to remove leaking fuel bundles, worker exposure would be increased since additional exposure would later be incurred for normal shutdown and maintenance activities. Also, during the February 22, 1999, informal public hearing on the Petition, the River Bend licensee stated that dose rates in the general plant areas are essentially unchanged and that the average daily dose to plant workers has remained at the historical level of approximately 0.14 person-rem per day during normal operations. River Bend has seen some increased levels in dose rates in isolated areas, such as in rooms containing offgas system equipment; however, these areas are not routinely occupied and access to the rooms are controlled by the health physics department. The licensee stated that if a 14-day outage were conducted to remove defective fuel bundles, the outage would incur a worker dose on the order of 9 person-rem for reactor disassembly, reassembly, and refueling activities. This exposure would be in addition to that incurred from activities planned for the scheduled refueling outage. The licensee contends that shutting down in this situation to replace leaking fuel would be an action contrary to ALARA. The staff agrees that conducting plant shutdown only to address the current situation at River Bend would be contrary to the ALARA principle for plant workers, provided exposure levels remain at their current values.

River Bend has two independent radiation-detection systems capable of sensing fission-product release from leaking fuel rods—main steam line radiation monitors and offgas system radiation monitors. The main steam line radiation monitors are used to detect high radiation levels from gross fuel failure. The offgas system radiation monitors can detect low-level emissions of noble gases, which are indicative of minor fuel damage. The offgas system monitor indication signaled the recent fuel damage found at River Bend.

The actions taken by the licensee to limit further fuel damage, as well as the continued attention to reactor coolant activity and offgas radiation levels, provide confidence that River Bend can continue safe operation, within its licensing basis, with the limited fuel leakage recently detected.

C. Plant-Specific Concerns - Perry Nuclear Power Plant

On the basis of the reported fuel leakage at Perry, the Petitioner states that the generic concerns contained in the UCS report apply to the Perry plant. In the opinion of the Petitioner, plant licensing bases do not permit operation of the plant with known fuel leakage.

As discussed in the consideration of generic safety concerns, the staff does not agree that pre-existing fuel cladding defects and resultant fuel leakage violate plant licensing bases. The staff also considers that conclusion valid for Perry. Fuel leakage during plant operation is not precluded by licensing basis provisions requiring that fuel integrity be maintained as designed. The Perry design basis itself allows the possibility of leakage while ensuring that cladding damage does not result because of specific operationally related causes. Fuel is also designed to maintain its structural integrity to ensure core coolability and to ensure that control rods can be inserted.

The Updated Safety Analysis report (USAR) for Perry contains unacceptable consequences criteria for different event categories (USAR Tables 15A.2-1 through 15A.2-4). The unacceptable consequences for normal operation do not preclude fuel leakage. The second criterion listed precludes fuel failure to the extent that the limits of 10 CFR Part 20 would be exceeded. The unacceptable consequences for anticipated operational transients prohibit fuel failure predicted as a direct result of transient analysis. For abnormal transients and design- basis accidents, widespread fuel cladding perforations and fuel cladding fragmentation are prohibited.

Two parts of the fuel system licensing basis for Perry show that limited fuel leakage during plant operation is a design consideration. The fuel system design basis for Perry is given in the USAR Section 15B by reference to the generic topical report “General Electric Standard Application for Reactor Fuel,” NEDE-24011-P-A. The generic topical report details fuel cladding operating limits to ensure that fuel performance is maintained within fuel rod thermal and mechanical design and safety analysis criteria. The limits are given for normal operating conditions and AOOs in terms of specific mechanical and thermal specifications. Evaluations of specific fuel failure mechanisms under normal operation and AOOs were discussed, such as stress and strain, hydraulic loads, fretting, and internal gas pressure, to ensure that fuel failure did not result from these causes. The design bases did not preclude the possibility that fuel failure could occur for other reasons, such as pre-existing cladding flaws leading to leakage.

The Technical Specifications for Perry (Section 3.4.8) contain a limit for RCS specific activity. The basis for this limit is the same as that discussed in the consideration of the generic safety concerns. Section B3.4.8 of the Perry Technical Specification "Bases" acknowledges that "the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks." Thus, fission products released during plant operation are clearly considered to be contributors to the source term used for safety analysis of the main steamline break release consequences. The technical specifications state that the limit is set to ensure that any release as a consequence of a main steamline break is less than a small fraction of the 10 CFR Part 100 guidelines. These portions of the Perry licensing basis are consistent with NRC regulations regarding fuel performance and the associated NRC guidance used by licensees to implement those NRC regulations that were covered earlier in the discussion regarding generic concerns.

The licensee has taken actions to limit the effects of the existing minor fuel leaks at Perry. The control rod pattern has been altered to achieve a depressed flux profile in the vicinity of the leaking rods, thereby suppressing the production of fission products as the plant continues operation. The off-gas treatment system has been effectively eliminating radioactivity in off-gas effluent, and there has been no change in general radiation area dose rates. However, the licensee is continuing to closely monitor the off-gas system pre-treatment radiation levels and is ensuring that the coolant activity concentration remains within technical specifications limits.

Perry has two independent radiation detection systems capable of sensing fission product release from leaking fuel rods: main steamline radiation monitors and off-gas system radiation monitors. The main steamline radiation monitors are used to detect high radiation levels from gross fuel failure. The off-gas system radiation monitors can detect low-level emissions of noble gases, which are indicative of minor fuel damage.

In its letter to the NRC dated February 11, 1999, the Perry licensee stated that if the plant were to shut down solely to remove fuel bundles exhibiting leakage, plant worker exposure would be increased since additional exposure would later be incurred for normal shutdown and maintenance activities. The licensee contends that shutting down in this situation to replace leaking fuel would be an action contrary to ALARA. The staff agrees that conducting plant shutdown only to address the current situation at Perry would be contrary to the ALARA principle for plant workers, provided exposure levels remain at their current values.

The actions taken by the licensee to limit further fuel damage, as well as the continued attention to reactor coolant activity and off-gas radiation levels, provide confidence that Perry can continue safe operation, within its licensing basis, with the limited fuel leakage detected.

IV. CONCLUSION

The Petitioner's requests are denied for the reasons specified in the preceding sections that discuss the Petitioner's information supporting the request. The Petitioner did not submit any significant new information about safety issues. Neither the information presented in the Petition nor any other subsequent information of which the NRC is aware warrants the actions requested by the Petitioner.

A copy of this Director's Decision will be filed with the Secretary of the Commission for review in accordance with 10 CFR 2.206(c). This Decision will become the final action of the Commission 25 days after its issuance unless the Commission, on its own motion, institutes a review of the Decision within that time.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 18th day of April 1999