July 26, 2000

MEMORANDUM TO: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield

FROM: William D. Travers ***/RA/***
Executive Director for Operations

SUBJECT: RESPONSE TO STAKEHOLDER ISSUES RAISED AT THE COMMISSION MEETING ON MAY 25, 2000, REGARDING THE REVIEW PROCESS FOR 10 CFR 2.206 PETITIONS (WITS 20000075)

The NRC staff and a stakeholder panel briefed the Commission concerning planned improvements to the 10 CFR 2.206 petition process on May 25, 2000. Attachment 1 is the transcript of the briefing. During the briefing, the stakeholders raised certain issues to which the staff did not have an opportunity to respond. In a staff requirements memorandum (M000525D) dated June 20, 2000, the Commission requested that the staff provide its response to these issues. This memorandum contains the staff's response.

ISSUES

The three main issues to which the staff did not respond were (1) the role of the 10 CFR 2.206 process in the regulatory decisions related to the D.C. Cook plants, (2) the potentially new information related to the fuel cladding degradation at River Bend Station and the associated director's decision for River Bend and Perry Nuclear Power Plant, and (3) the conduct of public meetings related to 10 CFR 2.206 petitions. Regarding D.C. Cook, one of the stakeholders stated that the staff was on the verge of allowing the units to restart despite serious problems and that the 10 CFR 2.206 petition process failed in this case. With respect to River Bend, one of the stakeholders stated that the staff's response to his petition failed to address the issues he raised. He also implied that the problem was more serious than the staff had estimated. Finally, the stakeholders raised issues related to meetings between petitioners and the NRC staff. They indicated there were problems with inconsistent meeting formats and the staff's handling of requests for presentations by Dr. Hopenfeld, an NRC staff member; Ms. Murphy, a member of Congresswoman Nita Lowey's staff; and Mr. Crane, a member of the public, at a public meeting for a petition related to the Indian Point Nuclear Generating Unit No. 2 (IP2).

STAFF RESPONSE

CONTACT: Andrew J. Kugler, NRR
301-415-2828

The staff reviewed the stakeholders' comments concerning the shutdown at D.C. Cook, the fuel cladding degradation at River Bend, and the staff's handling of meetings with petitioners. Detailed responses to these issues are given in Attachments 2, 3 and 4, respectively. A brief summary follows.

With respect to D.C. Cook, the staff believes that the petition process, along with other NRC processes (e.g., inspection activities, the confirmatory action letter [CAL] process, and the Manual Chapter 0350 process) worked properly. The staff would not have allowed the affected units to restart until it was satisfied that the safety issues that had been identified were resolved.

For River Bend, the staff concludes that Director's Decision 99-08 thoroughly addressed the issues raised by the petitioner, including all of the issues raised during the Commission meeting. The staff was fully aware of the information contained in the March 1, 2000, licensee event report (Attachment 5) mentioned at the meeting, almost all of which had been discussed in a public meeting on June 22, 1999.

The staff acknowledges that there has been some variation in the way it has run meetings with petitioners. Although some of this variation is desirable (to accommodate differing needs), the staff plans to more clearly document guidance for these meetings in the next revision of Management Directive 8.11, "Review Process for 10 CFR 2.206 Petitions," scheduled to be issued in October 2000. With respect to the specific comments related to the roles of Dr. Hopenfeld, Congresswoman Lowey's staff member, and Mr. Crane in the IP2 meeting, the staff believes it acted appropriately regarding these individuals.

Attachments: As stated (5)

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UNITED STATES NUCLEAR REGULATORY COMMISSION

May 25, 2000

The contents of this transcript of the proceeding of the United States Nuclear Regulatory Commission, taken on May 25, 2000, as reported herein, is a record of the discussions recorded at the meeting held on the above date.

This transcript has not been reviewed, corrected and edited and it may contain inaccuracies.

The transcript (not including this cover page) is already in ADAMS, Accession No. ML003719061.

ATTACHMENT 1

D.C. COOK

Stakeholder Comments

During the Commission meeting on May 25, 2000, Mr. James Riccio raised an issue related to the D.C. Cook units. His statements, taken directly from the meeting transcript, were:

> "On October 9, 1997, Mr. Lochbaum submitted a petition on the Cook plant, basically requesting that the plant not be allowed to restart until the design and licensing basis issues were resolved.

* * *

> "On or about January 6, 1998, UCS contacted the petition manager to inquire as to the status of the petition on Cook's restart activities. He [Mr. Lochbaum of UCS] was informed that the restart was imminent despite the fact that UCS's petition on Cook requested that it be prevented from operating until the issues were addressed.
>
> "At that time UCS was informed that their petition would be addressed after NRC allowed D.C. Cook to restart. The only thing that precluded that plant from restarting were basically media calls and contacting Congress that was done by Dave [Lochbaum].

* * *

> "D.C. Cook has now been shut down for over two and a half years yet the NRC Staff was on the verge of allowing the reactor to restart in January of '98. In retrospect I do not believe that the NRC can claim the shutdown was not warranted from a safety perspective. That being the case, I am at a loss to explain how NRC was going to allow D.C. Cook to restart.
>
> "Regardless of the answer, the process didn't work.
>
> "The public should not be forced to resort to media tactics in order to get safety concerns addressed by this agency."

(Transcript of the May 25, 2000, Commission meeting, pages 56 and 57)

Staff Response

Mr. Riccio's comment concerning the telephone conversation between UCS and the NRC staff does not accurately characterize what was said. Mr. Lochbaum, who was involved in the telephone call, provided a more accurate representation in a letter to the Commissioners dated December 4, 1998. In the letter, he stated that he was told about an upcoming meeting between the D.C. Cook licensee and the NRC staff to discuss the items in a confirmatory action letter (CAL) that the NRC had issued. The staff indicated to Mr. Lochbaum that it <u>might</u> lift the CAL shortly thereafter.

This is simply a statement of how the CAL process works. The meeting between the licensee for D.C. Cook and the staff was called to allow the licensee to provide its status of the items covered by the CAL. If the licensee could show that all of the issues in the CAL were resolved, then the NRC staff would lift the CAL. This was one of the actions that would have to be completed before the licensee restarted the units.

The issues covered by the CAL were mostly specific technical issues that the licensee needed to resolve. These technical issues were raised as a result of an architect/engineering (AE) design inspection by the NRC, which focused on selected systems at D.C. Cook. At the time of the telephone call on January 6, 1998, the licensee considered a number of these issues resolved. However, because of concerns related to programmatic breakdowns (of which the technical issues were symptoms), the CAL also included a broader issue. Specifically, the CAL confirmed that before the restart of either unit at D.C. Cook, the licensee would perform an assessment to determine whether the type of inspection findings discovered during the AE inspection existed in other safety-related systems and whether they affected system operability.

While the licensee was generally able to resolve the specific technical issues in relatively short order, it was not able to resolve the broader CAL issue. Therefore, both units remained shut down, not because of calls to the media and the Congress by UCS, but because of the appropriate application of NRC processes.

On the basis of findings by the NRC resident inspector staff, concerns regarding the ice condensers emanating from the AE inspection, and information brought to the staff's attention by UCS in its petition, the NRC expanded the scope of the inspections at D.C. Cook. The programmatic issue in the CAL led to the identification of a number of technical and programmatic issues by both NRC inspectors and the licensee. The extensive effort to identify and resolve these issues is documented in the inspection reports listed by Mr. Riccio in his written remarks for the Commission meeting. In April 1998, the staff established an NRC Inspection Manual Chapter (MC) 0350 panel for the D.C. Cook plants. The MC 0350 panel developed a restart action plan, monitored the licensee's performance and corrective actions, and provided oversight of the NRC staff's followup activities. Both units remained shut down for more than 2 years while the licensee developed and implemented corrective actions. The panel eventually provided a written recommendation for the restart of D.C. Cook, Unit 2, and it continues to monitor licensee performance with respect to Unit 2 and the restart of Unit 1.

Contrary to Mr. Riccio's view, the staff believes that the 10 CFR 2.206 petition process <u>did</u> work in this case. The issues raised by the petition were one of the inputs considered by the staff in pursuing the problems at D.C. Cook. And, as Mr. Lochbaum has indicated, he believes that the staff was responsive to his concerns in the associated director's decision.

RIVER BEND

Stakeholder Comments

During the Commission meeting on May 25, 2000, Mr. David Lochbaum raised an issue related to failed fuel at River Bend Station and an earlier director's decision (DD-99-08, dated April 18, 1999) that discussed the fuel issue at River Bend and Perry Nuclear Power Plant. His statements, taken directly from the meeting transcript, were:

> "I brought a licensee event report dated March 1, 2000 that Entergy submitted on River Bend. This is after our petition was denied.
>
> * * *
>
> "Entergy reported that they found that that was a common node [mode] failure that affected the entire batch of fuel. There were only a few that were all the way through, but there was -- the entire batch of fuel was affected by this common node [mode] failure. And there was no way of knowing until they shut down and examined it.
>
> "The petition that we provided, and in the testimony we made in the meeting or the hearing, actually, until I left for my plane, was the Staff looked at normal operation of the plant, not if the plant suffered an accident with the preexisting failures, the degradation of the fuel cladding.
>
> "This document proves that there was significant cladding degradation at River Bend, which is what we said, and which you would have never known until you shut down.
>
> "Had this plant suffered a control rod drop accident, a main steam line break, any one of the credible accidents that are within their design and licensing basis, with the cladding in that shape, there's no analysis on the planet that I'm aware of that would say that workers and the public would have been protected.
>
> "The existing analyses that that plant has today still don't address that condition. So I would say that the Staff did not do a -- the spelling was impeccable, but I do not think they addressed the issues we raised.
>
> "I went to great lengths in that petition, or the report that was attached to that petition to go through a 50.59 evaluation. I've done literally hundreds of those when I worked in the industry. I showed -- each one of those questions came up that it was an unreviewed safety question.
>
> "And the Staff didn't address that."

(Transcript of the May 25, 2000, Commission meeting, pages 88 and 89)

Staff Response

Mr. Lochbaum raised four major issues concerning the director's decision in his remarks:

1. The staff did not look at the consequences of an accident with the degraded fuel cladding.

2. The licensee event report (LER, included as Attachment 5) proved that there was significant cladding degradation at River Bend, which the staff could not have known until the plant was shut down.

3. There are no analyses that address the combination of the degraded cladding and any of a number of design basis accidents and the impact on the safety of the public and plant workers.

4. The staff failed to address the UCS contention that the cladding degradation constituted an unreviewed safety question.

The staff will address each of these issues in the following discussion, based primarily on the information provided in Director's Decision DD-99-08.

1. Consequences of an Accident

In the director's decision, the staff acknowledged that a primary issue raised in the 10 CFR 2.206 petition was UCS's concern about preexisting fuel failures in the event of an accident. The staff provided a lengthy discussion addressing this specific matter. In order to fully explain its position, the staff began with a section on generic safety issues. One important consideration was the issue of preexisting fuel failures as it relates to the concept of designing for defense in depth:

> "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.

> "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."

(DD-99-08, pages 5 and 6)

Since the fuel cladding is not required to be leak-free, and a limited amount of leakage can be permissible, the director's decision addressed the issue of preexisting fuel failures in the event of an accident:

> "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."

(DD-99-08, pages 16 and 17)

Furthermore, the director's decision continued:

> "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 [loss-of-coolant 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 [Standard Review Plan] 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 [boiling-water reactors] and MSLB [main steam line break] outside of containment for PWRs [pressurized-water reactors]. 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 [emergency core cooling system] 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 [General Design Criterion] 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."

(DD-99-08, pages 18 and 22)

Therefore, the staff believes that it adequately addressed the petitioner's concerns associated with preexisting fuel failures under accident conditions.

2. Extent of Problem Unknown Until Shutdown

The purpose of the voluntary LER dated March 1, 2000, submitted by Entergy Operations, Inc. (EOI), was to provide a summary of the licensee's root cause investigation into the fuel cladding failures that occurred during operating Cycle 8. The licensee stated that it believed that the information gathered and the conclusions drawn following its investigation were relevant to the industry.

The staff closely monitored the licensee's actions subsequent to the discovery of the first fuel failure on September 18, 1998. During the time the plant remained in operation, the licensee took appropriate actions in accordance with its procedures. These actions included altering the reactor's control rod pattern in order to achieve a depressed flux profile in the vicinity of the leaking rods, thereby suppressing the production of fission products as the plant continued 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 pre-fuel-leak level of 1 mCi/sec. The peak value was never more than a small fraction of the technical specification (TS) limit of 290 mCi/sec. Therefore, since the coolant activity never threatened the plant's licensing basis TS limits, the staff maintained that the cladding degradation was not significant enough to require an immediate plant shutdown.

In addition, the licensee's failed fuel monitoring program and action plan procedures were successful in identifying and suppressing local power in the right fuel assemblies. Subsequent inspection and telescopic sipping confirmed that the seven suspected fuel bundles had leaking fuel rods and further identified that there were no additional leaking assemblies. Therefore, the staff believes that the licensee took the necessary action, and had sufficient understanding of the extent of the degraded fuel problem, in order to safely operate the plant within its licensing basis.

The March 1 LER did not convey any substantially new information to the staff that differed from the information provided at a public meeting on June 22, 1999. During this meeting, EOI presented the findings of its root cause investigation into the fuel cladding degradation. Portions of the meeting were closed to the public because proprietary information was exchanged; however, the information provided at the public forum was essentially the same level of detail and material published in the March 1 LER.

3. No Existing Analyses for This Condition

The staff did consider, albeit briefly, the radiological consequences associated with a control rod drop (CRD) and MSLB accidents:

> "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."

(DD-99-08, page 20)

Furthermore:

> "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 [specific activity] 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 [to 10 CFR Part 50], 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 [specified acceptable fuel design limits] are not exceeded during normal operation or AOOs [anticipated operational occurrences]. 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."

(DD-99-08, pages 14 and 15)

4. <u>Potential Unreviewed Safety Question</u>

In the director's decision, the staff recognized that it was UCS's opinion that operation with preexisting fuel cladding failures constituted an unreviewed safety question in accordance with 10 CFR 50.59. However, the staff believes that it carefully examined this issue, as raised in the two petitions and the accompanying report on fuel cladding failures dated April 2, 1998. The 39-page director's decision evaluated UCS's generic concerns and provided a detailed discussion on specific plant licensing basis issues. In the director's decision, the staff explained

its position that a plant's licensing basis consists of a wide variety of information and continued that the licensing basis considers operation with limited fuel cladding failures:

> "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 [Final Safety Analysis Report], 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."

(DD-99-08, page 12)

Moreover:

> "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 [specific activity] 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 specified acceptable fuel design limits are not exceeded during normal operation or anticipated operational occurrences. 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."

(DD-99-08, pages 14 and 15)

Specifically addressing the issue of 10 CFR 50.59:

> "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."

(DD-99-08, pages 17 and 18)

5. <u>Conclusion</u>

The staff acknowledges that the UCS may not agree with its findings as documented in DD-99-08; however, the staff maintains that it thoroughly and effectively explained its overall policy with regard to the issue of failed fuel and fully addressed the issues raised in the two petitions (River Bend Station and Perry Nuclear Power Plant) filed by UCS.

ISSUES RELATED TO MEETINGS

Stakeholder Comments

During the Commission meeting on May 25, 2000, Mr. Lochbaum raised issues related to the manner in which the NRC staff has handled public meetings regarding petitions. Particular issues raised by Mr. Lochbaum concerned how the staff responded to requests to allow presentations by Dr. Joram Hopenfeld, a member of the NRC staff, and Mr. Peter Crane, a member of the public. His statements, taken directly from the meeting transcript, were:

> "We formally asked Mr. Collins' permission for Mr. Joe Hopenfeld of the NRC Staff to attend this meeting and discuss his differing professional opinion. We didn't want him to talk about our petition, just talk about his issues.
>
> "Our request was denied because federal law prevents a federal employee from representing a non-governmental organization in a proceeding before the Government.
>
> "The fact that we had made it very clear that Mr. Hopenfeld would not be representing our issues, was ignored by the staff.
>
> * * *
>
> "The NRC Staff allowed Congressional staff to speak, but silenced a member of its own staff at this meeting.
>
> "We also invited Mr. Peter Crane to make a presentation about potassium iodide. Before Mr. Crane was allowed to make his presentation, he first had to endure a nuclear inquisition by the NRC Staff.
>
> "The Staff questioned whether Mr. Crane had any right to speak at the meeting since he was not a member of UCS, of Public Citizen, of NIRS [Nuclear Information and Resources Service], of NEI [Nuclear Energy Institute], or of the Pace Law School Energy Project which were -- except for NEI, which were the petitioners in the process."

(Transcript of the May 25, 2000, Commission meeting, pages 51 and 52)

Staff Response

Mr. Lochbaum's comments relate to a public meeting on April 7, 2000, to discuss the UCS petition related to Indian Point Unit 2 (IP2). The meeting was called on the basis of a request by the petitioners to allow them to present their concerns to the NRC staff. In a letter dated March 22, 2000, Mr. Lochbaum requested, among other things, that Dr. Hopenfeld be allowed to attend the April 7 meeting to discuss concerns he had documented through the NRC's differing professional opinion (DPO) process. This request was made because, as part of the petition, the petitioners had requested that the NRC resolve Dr. Hopenfeld's DPO and the related generic safety issue before the restart of IP2.

The staff responded to this request in a letter from Samuel J. Collins dated March 31, 2000. In this letter, the staff explained that the purpose of the meeting was to allow the petitioners and the licensee to provide additional information and clarifications relevant to the IP2 petition. As pointed out in Mr. Lochbaum's statements, Federal law would prohibit Dr. Hopenfeld from representing the petitioners in the meeting. Since Dr. Hopenfeld was not involved in the staff's review of the petition, he did not have a speaking role representing the staff. Therefore, the staff responded to Mr. Lochbaum that Dr. Hopenfeld could attend the meeting, but the meeting discussion would not focus on the DPO.

In its response, the staff also pointed out that the generic (as opposed to IP2-specific) concerns documented in the DPO were already being addressed in a separate process that included an opportunity for public comment. Public comments on the DPO had been requested in a *Federal Register* notice dated January 20, 1999 (64 *FR* 3138). Comments were due by June 30, 1999. Therefore, the NRC staff believes it acted appropriately in denying the request to have Dr. Hopenfeld speak at the IP2 meeting.

As Mr. Lochbaum pointed out, the NRC staff did allow Ms. Kara Murphy, a District Representative for Congresswoman Nita Lowey, to present Congresswoman Lowey's statement at the meeting. This meeting had been noticed as a meeting between the petitioners, the licensee, and the staff in accordance with Management Directive (MD) 3.5, "Public Attendance at Certain Meetings Involving the NRC Staff." Congresswoman Lowey did not represent the petitioners, so NRC policy (MD 3.5) would normally allow her representative to attend the meeting only as an observer. However, the NRC staff made a conscious decision to allow Ms. Murphy to speak because of Congresswoman Lowey's role as an elected representative of the people living around IP2. The staff has made similar accommodations for congressional staff in the past.

There were significant differences between the cases of Dr. Hopenfeld and Congresswoman Lowey. Dr. Hopenfeld's concerns were generic and already well known by the staff through the DPO and generic safety issues processes. Congresswoman Lowey's concerns were specific to IP2 and had not previously been expressed to the staff. These facts played a role in the staff's decision to allow Congresswoman Lowey's statement to be read at the meeting. The staff believes it acted appropriately and within the limits of its discretion in allowing this exception to MD 3.5.

With respect to Mr. Crane, the issue of his role in the meeting arose because he clearly stated that he was not there representing the petitioners. As previously discussed, this was a meeting between the petitioners, the licensee, and the NRC staff. Therefore, if Mr. Crane was not representing the petitioners, he was like any other member of the public who is allowed to attend such meetings as an observer in accordance with MD 3.5. Unlike Congresswoman Lowey, Mr. Crane was not an elected representative of the people living near IP2.

The staff initiated a <u>brief</u> discussion of Mr. Crane's role. (See pages 15 through 17 of the meeting transcript for more details.) Mr. Crane clearly stated that he was not representing the petitioners. Despite this fact, the staff accepted Mr. Paul Gunter's statement that Mr. Crane was there at the invitation of the petitioners and allowed him to speak. This brief exchange could hardly qualify as an "inquisition."

Stakeholder Comments

Mr. James Riccio also raised an issue related to meetings. His statements, taken directly from the meeting transcript, were:

> "In April [2000] I participated in two Petition Review Board meetings within a span of 48 hours. Both petitions address the potential for steam generator tube ruptures. While I believe the introduction of the PRB [petition review board] is a positive step in the process, I was struck by the disparity in these two meetings.
>
> "The first, concerning Salem, was conducted in what was generally the form of a public meeting scheduled well in advance so the PRB -- and with the media and public allowed to observe if not participate. The second, concerning Indian Point 2, was held at a moment's notice. At that point not all the petitioners could make the telephone conference. I requested a slight delay. I am talking a day and I was rebuffed by the staff. The only reason that explains this disparate treatment between the two meetings is the status of the reactor. Salem was operating. Indian Point 2 was shut down with the steam generator tube issues.
>
> "Even when we had that meeting the restart schedule was out, and there was no way that they were going to get that plant up and running in the time that was going to interfere with a day or two delay in our conference call.
>
> "It is basically my view that the only reason that we were treated in that manner was because the plant was shut down and the utilities wanted a restart.
>
> * * *
>
> "All Petition Review Board meetings should be public meetings and should be properly noticed regardless of the status of the reactor in question. In retrospect, the delay of a day or two in holding the PRB would not have interfered with the Staff's scheduled restart of Indian Point 2."

(Transcript of the May 25, 2000, Commission meeting, pages 57 through 59)

Staff Response

The subject petitions for Salem Units 1 and 2 and IP2 requested different actions because the Salem units were operating and IP2 was shut down. The petition for Salem requested that the NRC require the Salem units to be shut down because of concerns related to the steam generators. However, the petition did not include any plant-specific information related to this request. When the staff attempted to set up a meeting, the petitioner requested a delay because he was not available. Since the units were operating, any delay in the meeting between the petitioner and the PRB could delay the staff's action in response to the petitioner's request. However, the delay would not have a negative impact on the licensee. So, because the delay was requested by the petitioner, the same party whose interest would be most affected by the delay, the staff agreed.

For IP2, the petitioners requested, among other things, that the NRC staff not allow the plant to be restarted until certain actions were taken. The staff wanted to meet with the petitioners to discuss the petition far enough in advance of any restart of the unit to allow the staff time to properly evaluate the request and determine whether staff action was warranted. However, only two of the four petitioners were available. Nevertheless, the staff decided to proceed with the meeting.

In retrospect, a short delay in the schedule for this meeting would not have created a problem. But because the purpose of the meeting was to allow the petitioners to present additional information regarding the petition, the staff did not feel that it was necessary for all four petitioners to attend. The staff will consider the lessons learned from this and other early experiences with petitioner/PRB meetings for the future. For example, the current revision of MD 8.11 states that the petitioner can address the PRB. The planned revision to MD 8.11 includes the clarification that petitioners are permitted to designate a reasonable number of associates to assist in the presentation to the PRB.

Other Comments on Meetings

In the Commission meeting and through other sources, the staff has received other general comments related to inconsistencies in the way it has run petitioner/PRB meetings. There are a number of factors that have led to these comments:

1. Petitioner/PRB meetings are relatively new. As the staff has gained experience with this type of meeting and encountered different circumstances, it has adjusted its approach.

2. The process allows some flexibility on the basis of specific circumstances.

3. The current revision of MD 8.11, “Review Process for 10 CFR 2.206 Petitions,” provides only limited guidance with respect to these meetings.

Staff Response

Because the process allows some flexibility, there may continue to be some variation between petitioner/PRB meetings. However, on the basis of the staff’s experience to date and the comments received from petitioners, the staff will add more detailed guidance concerning these meetings to the upcoming revision to MD 8.11, scheduled to be issued in October 2000.