[7590-01-P]
DD-17-01

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

William M. Dean, Director

In the Matter of	)	Docket Nos. 50-247 and 50-286
	)	
Entergy Nuclear Operations, Inc.	)	License Nos. DPR-26 and DPR-64
	)	
Entergy Nuclear Indian Point 2, LLC	)	
	)	
Entergy Nuclear Indian Point 3, LLC	)	
	)	
Indian Point Nuclear Generating Unit No. 2	)	
	)	
Indian Point Nuclear Generating Unit No. 3	)	
	)	

FINAL DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. Introduction

By letter dated June 30, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16187A186), as supplemented by letter dated January 10, 2017 (ADAMS Accession No. ML17011A012), Mr. David A. Lochbaum, Director of the Nuclear Safety Project at the Union of Concerned Scientists, filed a petition pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 2.206, "Requests for Action under This Subpart." Citing degradation of reactor vessel baffle-former bolts (BFBs) identified at Indian Point Nuclear Generating Unit No. 2 during its spring 2016 refueling outage, the petitioner requested that the U.S. Nuclear Regulatory Commission (NRC or the Commission) take the following enforcement actions against Entergy Nuclear Operations, Inc. (Entergy), the licensee for Indian Point Nuclear Generating Unit Nos. 2 and 3 (Indian Point 2 and 3):

(1) Issue an order requiring the licensee to inspect the reactor vessel BFBs and install the downflow to upflow modification at Indian Point 2 during its next refueling outage.

(2) Issue a demand for information requiring the licensee to submit an operability determination to the NRC regarding continued operation of Indian Point 3 until its reactor vessel BFBs can be inspected according to the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Topical Report, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," Final Report, December 2011 (ADAMS Package Accession No. ML120170453).

(3) Issue a demand for information requiring the licensee to submit an evaluation of the performance, role, and operating experience of the Indian Point metal impact monitoring system in detecting and responding to indications of loose parts (such as broken baffle bolt heads and locking tab bars) within the reactor coolant system.

As the basis for this request, the petitioner cited Licensee Event Report 2016-004-00, "Unanalyzed Condition due to Degraded Reactor Baffle-Former Bolts," submitted by the licensee on May 31, 2016 (ADAMS Accession No. ML16159A219), that describes an event that involved an unanalyzed condition due to degraded reactor vessel BFBs at Indian Point 2, which is reportable under 10 CFR 50.73(a)(2)(ii)(B). The petitioner states that (1) an order is the proper means for ensuring that the bolts are inspected and that the downflow to upflow modification is installed during the next refueling outage at Indian Point 2, (2) Indian Point 3 is potentially operating with degraded BFBs and an operability determination is the mechanism established by the NRC to properly evaluate such situations, and (3) the metal impact monitoring system as described in the updated final safety analysis report has the potential to act as an alternate monitoring system to identify degraded BFBs, yet neither the NRC nor the licensee has referred to this system in publicly available documents relating to this issue.

The petitioner met with the Office of Nuclear Reactor Regulation (NRR) Petition Review Board on July 28, 2016, to clarify the bases for the petition. The NRC is treating the transcript of this meeting (ADAMS Accession No. ML16215A391) as a supplement to the petition. In a letter dated September 7, 2016 (ADAMS Accession No. ML16231A140), the NRC informed the petitioner that the agency accepted the petition for review under 10 CFR 2.206 and that the agency had referred the issues in the petition to NRR for appropriate action.

In the supplemental letter dated January 10, 2017, the petitioner reduced the scope of the petition by withdrawing requested enforcement actions 1 and 2 identified above. Citing the plant closure agreement between Entergy and the State of New York to shut down both units in the 2020-2021 timeframe, the petitioner noted that Entergy committed to inspect the reactor vessel BFBs at Indian Point Nuclear Generating Unit No. 2 and Unit No. 3 (Indian Point 2 or Indian Point 3) during their spring 2018 and 2019 refueling outages, respectively. While the plant closure agreement was silent with respect to the downflow to upflow modification, the petitioner concluded that the additional inspections, combined with the shortened plant life, diminished the need for an order. In addition, citing documents released via a recent Freedom of Information Act request (FOIA/PA-2016-0457), the petitioner concluded that sufficient information regarding operability of Indian Point 3 was available to justify withdrawing the request for a demand for information for an operability determination. Nonetheless, while recognizing that the NRC staff's responses to these requested actions are now moot, they are being included in this director's decision in order to share and document the regulatory basis for the staff's decisionmaking regarding these two requested enforcement actions.

The NRC sent the proposed director's decision to both the petitioner and the licensee by letters dated January 11, 2017 (ADAMS Accession Nos. ML16320A269 and ML16320A273, respectively). The petitioner and the licensee were provided the opportunity to provide comments within 30 days on any part of the proposed director's decision that was considered to be erroneous or any issues in the petition that were not addressed. The petitioner provided comments by letter dated January 19, 2017 (ADAMS Accession No. ML17024A201), and the licensee provided comments by letter dated February 9, 2017 (ADAMS Accession No. ML17045A470). The licensee's response included new information that was not available to the NRC staff when the proposed director's decision was issued for comment. The licensee provided results of the BFB failure analysis performed at the Westinghouse hot lab testing facility, along with details of its enhanced BFB inspection plans for the remaining refueling outages at Indian Point 2 and 3. Furthermore, the licensee informed the staff that it was withdrawing its previous commitment to implement the downflow to upflow modification at both units. The comments from the petitioner and the licensee resulted in changes to the proposed director's decision and are summarized in the attachment to the final director's decision.

Documents referenced in this director's decision are available for inspection at the NRC's Public Document Room (PDR), located at O1F21, 11555 Rockville Pike (first floor), Rockville, MD 20852. Publicly available documents created or received at the NRC are accessible electronically through ADAMS in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC's PDR reference staff by telephone at 1-800-397-4209 or 301-415-4737, or by e-mail at pdr.resource@nrc.gov.

II. Discussion

Reactor vessel internals are structures located within the reactor vessel that support and orient the reactor fuel assemblies and direct coolant flow through the core. The core baffle is part of the internal structure, which consists of vertical plates that surround the outer faces of the peripheral fuel assemblies. The baffle directs coolant flow through the core. The vertical plates are bolted to the edges of horizontal former plates that are bolted to the inside surface of the core barrel. There are typically eight levels of former plates located at various elevations within the core barrel. The bolts that secure the baffle plates to the former plates are referred to as BFBs. Furthermore, the core design can be configured such that reactor coolant flow between the core barrel and the baffle goes either up or down, which is referred to as upflow or downflow, respectively. Some plants have converted from the downflow to the upflow configuration at some point in their operating history.

European plants first identified cracking of BFBs as early as 1988. The NRC published Information Notice No. 98-11, “Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants,” dated March 25, 1998 (ADAMS Legacy Accession No. 9803230106), alerting the U.S. nuclear industry to the issue. Domestic licensees are currently performing ultrasonic testing (UT) examinations of BFBs for license renewal commitments in accordance with EPRI Topical Report MRP-227-A. The NRC staff has approved the use of MRP-227-A for meeting license renewal commitments for the aging management of reactor vessel internals. In addition, inspections of core support structures are conducted at 10-year intervals in accordance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.”

In response to a license renewal commitment associated with the timely renewal provisions of 10 CFR Part 54, the licensee for Indian Point 2 performed visual inspections and UT examinations of the BFBs during its spring 2016 refueling outage. The examinations identified significant degradation of BFBs, and the licensee concluded that the plant was in an unanalyzed condition. The licensee's findings were reportable under 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," and 10 CFR 50.73, "Licensee Event Report System." In LER 2016-004-00, the licensee stated the following:

> Indian Point Unit 2 (IP2) was shut down as scheduled on March 7th, 2016 to implement the 2R22 refueling outage. As part of the IP2 License Renewal process, Entergy committed to performing inspections of the reactor vessel internal components during the 2R22 refueling outage. The NRC has approved EPRI Technical Report MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," as an acceptable vehicle for performing aging-related inspections and evaluations of applicable reactor components. One set of components inspected under MRP-227-A were the baffle-former bolts through visual inspection (VT) and ultrasonic (UT) examination.
>
> The IP2 baffle structure includes 832 baffle-former bolts which attach the baffle plates to the former plates. Of the 832 baffle-former bolts, 227 either failed to meet acceptance criteria or could not be UT inspected. The UT inspection identified indications on 182 bolts, 14 were incapable of being UT inspected and were thus conservatively assumed to have failed; and 31 bolts failed the VT. The failed baffle-former bolts are distributed throughout the vertical baffle plates with

more failures found in the upper portion of the plates and more concentrated on some of the plates than others (the failures are clustered).

The 227 failed bolts and the pattern of failure did not meet the acceptance criteria for plant startup from the 2R22 refueling outage which had been provided by Westinghouse prior to the outage in an analysis of the baffle-former assembly in WCAP-18048-P. The consequence of this is that baffle-former bolt replacements were required to be completed prior to returning IP2 back to service.

The licensee's findings were entered in its corrective action program as Condition Reports CR-IP2-2016-02081 and CR-IP2-2016-02348. The licensee described the following corrective actions in LER 2016-004-00:

- In addition to replacing the 227 BFBs found to be either degraded or untestable, the licensee replaced an additional 49 BFBs to prevent bolting pattern failures due to clustering. Furthermore, an additional 2 BFBs were replaced during replacement activities. Thus, a total of 278 of the original 832 BFBs were replaced. The new BFBs were made from Type 316 stainless steel as opposed to the Type 347 stainless steel that comprised the original BFBs.
- A number of BFBs with UT indications were shipped to a laboratory for failure analysis.
- The licensee committed to perform inspections of the BFBs during its next refueling outage (2R23) scheduled for spring 2018.
- The licensee committed to modify the reactor vessel internals to convert the core from a downflow to an upflow plant configuration during the 2R23 refueling outage.

- The licensee committed to replace additional BFBs during the 2R23 refueling outage to meet minimum bolting patterns as evaluated by Westinghouse.

In response to the Indian Point 2 bolt degradation, the NRC conducted a range of baseline Reactor Oversight Process inspections to independently assess the adequacy of visual and ultrasonic bolt examinations, observe bolt replacement activities, and review Entergy's evaluations of Indian Point 2 and 3 corrective actions. In addition, Entergy performed an operability determination to evaluate the impact of BFB degradation at Indian Point 3. NRC inspectors reviewed Entergy's evaluations and concluded that these evaluations provide reasonable assurance that the Indian Point 3 baffle bolts will perform as required until the planned refueling outage in spring 2017, at which time Entergy plans to examine the bolts. The results of the NRC's inspections are found in Integrated Inspection Report 05000247/2016002 and 05000286/2016002, dated August 30, 2016 (ADAMS Accession No. ML16243A245).

The NRC staff has reviewed recently identified degradation of reactor vessel BFBs at operating reactors. In accordance with NRR Office Instruction LIC-504, Revision 4, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," effective June 2, 2014 (ADAMS Accession No. ML14035143), the staff performed a risk-informed evaluation of the safety significance of recently identified reactor vessel BFB degradation, documented in a review dated October 20, 2016 (ADAMS Accession No. ML16225A341). As discussed in that review, the staff identified the facilities of greatest concern, assessed the need for immediate shutdown of those facilities, and prepared available options based upon currently known information. Based on a review of operating experience, the staff concluded that the potential for significant bolt degradation is most susceptible at Westinghouse 4-loop designs with a downflow configuration and Type 347 stainless steel bolts, which include Indian Point 2 and 3. The staff also concluded that the degradation of BFBs does not represent an imminent safety hazard and, as a result, immediate plant shutdowns to inspect and repair degraded BFBs are not necessary. Furthermore, it was the staff's overall recommendation that the plants most susceptible to BFB degradation be permitted to operate until their next scheduled refueling outage, at which time they will perform visual and UT inspections of the BFBs, because the risk of core damage from BFB degradation over this time period was found to be low. It should be noted, however, that bolt failures have been detected in other types of material, and the NRC staff has been, and continues to be, engaged with industry to better understand this phenomenon as discussed below.

The NRC staff has been actively working with the EPRI MRP working group to better understand the safety significance of BFB degradation and the extent of this condition within the industry. A public meeting was held on July 19, 2016, with representatives of the EPRI MRP working group, industry, and the NRC staff to discuss recent inspections and operating experience of BFB degradation. The meeting summary and meeting handouts can be found in ADAMS under Package Accession No. ML16208A001. Subsequent guidance from both Westinghouse and the EPRI MRP working group recommended BFB inspections at the next scheduled refueling outage for those plants identified as having the greatest susceptibility for BFB degradation.

In summary, the NRC staff has concluded that BFB degradation observed at operating facilities to date, including Indian Point 2, does not represent an immediate safety concern and does not warrant regulatory action at this time. Industry guidance documents identify those facilities having the greatest susceptibility for BFB degradation, which include both Indian Point 2 and 3, and recommend that they inspect their reactor vessel BFBs during their next scheduled refueling outage. The staff will continue to monitor BFB inspections and will retain the option of taking regulatory action as warranted.

Actions Requested by the Petitioner

The following enforcement actions were requested by the petitioner:

1. The petitioner requested that the NRC issue an order requiring the Indian Point licensee to inspect the reactor vessel BFBs and to install the downflow to upflow modification on Indian Point 2 during its next refueling outage.

NRC Response:

Based on a review of operating experience, the potential for significant BFB degradation is currently believed to be most susceptible at Westinghouse 4-loop designs with a downflow configuration and Type 347 stainless steel bolts. Westinghouse Technical Bulletin TB-12-5, "Baffle Bolt Degradation in a Westinghouse NSSS [Nuclear Steam Supply System] Plant with Downflow Reactor Internal Design," dated March 7, 2012, following the operating experience at Donald C. Cook Nuclear Plant (D.C. Cook), Unit No. 2, identified seven operating reactors that were considered most susceptible to BFB degradation: Indian Point 2 and 3; Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2; D.C. Cook, Unit Nos. 1 and 2; and Diablo Canyon Nuclear Power Plant, Unit No. 1. The NRC staff confirmed the bolt material as Type 347 stainless steel for all of these plants. Westinghouse Nuclear Safety Advisory Letter (NSAL) 16-1,[1] Revision 1, "Baffle-Former Bolts," dated August 1, 2016 (ADAMS Accession No. ML16222A513), issued in response to the experience at Indian Point 2 and Salem Unit No. 1,

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1 Westinghouse NSAL 16-1 was issued to Westinghouse pressurized-water reactor owners to provide a 10 CFR Part 21, "Reporting of Defects and Noncompliance," evaluation and recommendations in response to recent BFB degradation. The NRR staff has not reviewed the engineering analyses supporting the evaluation in NSAL 16-1 or endorsed its conclusions or methods. The letter is discussed in this director's decision only to provide context to the NRC staff's own engineering judgment in evaluating potential risk and regulatory options.

identifies the same seven plants as being most susceptible to BFB cracking. NSAL 16-1 classifies the seven 4-loop downflow plants with Type 347 bolts as “Tier 1a.” EPRI MRP Letter 2016-022, dated July 27, 2016 (ADAMS Accession No. ML16211A054), transmits interim guidance that recommended that all plants identified as Tier 1a plants in Westinghouse NSAL 16-1 conduct UT examinations of all BFBs at the next scheduled refueling outage. This guidance is classified as “needed,” as defined in the protocol of Nuclear Energy Institute 03-08, Revision 2, “Guideline for the Management of Materials Issues,” issued January 2010 (ADAMS Accession No. ML101050337). The identification of the most susceptible group of plants to BFB cracking in NSAL 16-1 and MRP Letter 2016-022 is consistent with the staff’s assessment based on its review of operating experience as previously described in the staff’s risk-informed evaluation performed in accordance with LIC-504. It should be noted that if any licensee in the most susceptible group (i.e., Tier 1a, which includes Indian Point 2) intends to deviate from the EPRI MRP interim guidance, the NRC would be notified and could take regulatory action to ensure that the licensee performs UT examinations at the next refueling outage.

In the proposed director’s decision, the NRC staff proposed to deny the petitioner’s request to issue an order requiring the licensee to inspect the BFBs and implement the downflow to upflow modification during the next refueling outage at Indian Point 2. The basis for the staff’s decision was that 1) the licensee committed to take these actions; 2) industry guidance documents recommended inspection of BFBs; 3) any changes to these commitments would need to be justified in accordance with 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Criterion XVI, “Corrective Action,” and would be inspectable by NRC inspectors; and 4) the NRC retained the option of taking enforcement action as necessary. At approximately the same time that the proposed director's decision was issued for comment, the licensee entered a plant shutdown agreement with the State of New York where Indian Point 2 and 3 would permanently shut down in April 2020 and 2021, respectively. In the plant shutdown agreement, the licensee committed to inspect the BFBs for both Indian Point 2 and 3 during the 2018 and 2019 refueling outages, respectively. Subsequently, in his letter of January 10, 2017, the petitioner withdrew his request for this enforcement action. While the petitioner recognized that the plant shutdown agreement was silent on the downflow to upflow modification, the petitioner concluded that the additional BFB inspections should protect against degradation during the shortened period of reactor operation. Finally, in response to the proposed director's decision, the licensee 1) described its enhanced BFB inspection plans for future refueling outages at Indian Point 2 and 3, 2) described the results of the BFB failure analysis performed at the Westinghouse hot lab testing facility, and 3) informed the staff that it had changed its commitment and would not implement the downflow to upflow modification at Indian Point 2 and 3.

Based upon the licensee's commitment to inspect the BFBs, the industry's recommended guidance to inspect the BFBs for the Tier 1a plants during the next refueling outage, and the plant shutdown agreement that reduces the period of plant operation as described above, the NRC staff does not plan to take enforcement action to make the commitment legally binding. Therefore, absent the petitioner's withdrawal, the NRC would have denied the petitioner's request.

2. <u>The petitioner requested that the NRC issue a demand for information requiring the Indian Point licensee to submit an operability determination to the agency regarding</u>

continued operation of Indian Point 3 until its baffle bolts can be inspected according to the guidance of MRP-227-A.

NRC Response:

The petitioner referred to (1) the licensee's LER-2016-004-00, in which the licensee concluded that the BFB degradation at Indian Point 2 represented an unanalyzed condition, and (2) NRC Inspection Manual Chapter 0326, "Operability Determinations & Functionality Assessments for Conditions Adverse to Quality or Safety," dated December 3, 2015 (ADAMS Accession No. ML15328A099), which identifies those circumstances in which an operability determination is required. Inspection Manual Chapter 0326, paragraph 04.05, "Circumstances Warranting Operability Determinations," requires an operability determination upon "discovery of an unanalyzed condition." The petitioner asserts that, since Indian Point 3 is constructed of nearly identical materials, has been exposed to nearly identical environmental conditions, and has operated for nearly the same amount of time as Indian Point 2, Indian Point 3 is vulnerable to similar BFB degradation and, therefore, should be considered to be in an unanalyzed condition, thus necessitating an operability determination.

On July 11, 2016, Entergy staff completed an operability evaluation, which assumed an estimated number of BFB failures based on the degradation found in Indian Point 2 and was adjusted to take credit for the small number of inaccessible bolts and a sample of bolts extracted with high removal torque that indicated residual structural capacity. NRC inspectors determined that this estimated number of bolt failures was conservative, because the evaluation did not credit the baffle-edge bolts or the differences in operational history between the two units, such as neutron fluence levels or fatigue from thermal cycles. The operability evaluation concluded that the Indian Point 3 BFBs would perform as intended to secure the baffle plates from being dislodged. The inspectors concluded that Entergy's operability evaluation provided an appropriate basis to conclude that the Indian Point 3 baffle assembly would support emergency core cooling system operability until the planned Indian Point 3 refueling outage in spring 2017.

At about the same time that the proposed director's decision was issued for comment, the petitioner submitted his letter of January 10, 2017, that withdrew this requested enforcement action. The petitioner cited the recent release of documents from a Freedom of Information Act request (FOIA/PA-2016-0457) that diminished the need for a demand for information.

In summary, the licensee performed the operability determination that the petitioner requested. The operability determination was available for NRC review, and NRC inspectors concluded that Entergy's operability evaluation provided appropriate basis to conclude that the Indian Point 3 baffle assembly would support emergency core cooling system operability until the planned Unit No. 3 refueling outage in spring 2017. Therefore, inasmuch as the licensee has performed the operability determination and the NRC staff has reviewed it, the petitioner's request was effectively met.

3. <u>The petitioner requested that the NRC issue a demand for information requiring the Indian Point licensee to submit an evaluation of the performance, role, and operating experience of the metal impact monitoring system in detecting and responding to indications of loose parts (such as broken bolt heads or locking tab bars) within the reactor coolant system</u>.

NRC Response:

The Indian Point updated final safety analysis report describes the metal impact monitoring system (often referred to as the loose parts monitoring system) as a system for enabling the early detection of any debris, detached internal structural items, and hardware present in the reactor coolant system. Metal impact monitoring is accomplished by the installation of specially developed transducers mounted on the exterior of the reactor coolant system and steam generators. Monitoring points normally in use during plant operation are at the top and bottom of the reactor vessel and above and below each steam generator tube sheet. The metal impact monitoring system is not a safety-related system and has no operability requirements in the technical specifications. Furthermore, there are no requirements or expectations for the licensee to submit periodic evaluations of the performance, role, or operating experience of the system for NRC or public consideration.

The petitioner identified a 1998 Westinghouse safety evaluation, 98-115-EV-1, Revision 1, “Loose Parts Evaluation—Residual Heat Removal Valve Parts” (in ADAMS Package Accession No. ML993610326), that reported that the metal impact monitoring system at Indian Point 2 detected a small metal part weighing less than 2 ounces in the reactor vessel lower plenum. The petitioner further noted that broken locking tabs or bolt heads would be similarly sized small metal parts that, by implication, should be detectable by the metal impact monitoring system. However, the petitioner noted that neither the Indian Point licensee nor the NRC refers to the metal impact monitoring system as an alternate, available means to provide early detection of degraded BFBs and locking tabs within the reactor coolant system. As a result, the petitioner requested an evaluation of the metal impact monitoring system performance history because the system does not appear to be adequately performing its intended monitoring function.

Failure or degradation of BFBs may result in loose parts in the form of broken bolt heads and locking bars. It should be noted that the clearances between the baffle plates and peripheral fuel assemblies are sufficiently small such that broken bolt heads are not likely to become loose parts within the reactor coolant system unless the fuel is removed. Therefore, if a bolt head fractures at the head-to-shank transition and separates from the bolt shank, the bolt head is not expected to fall out of its location, even if the locking bar fails. A bolt head trapped in the gap could only cause fretting of the adjacent cladding. Localized fuel cladding damage caused by fretting can also be detected by monitoring reactor coolant activity. With regard to baffle plates, no displacement of baffle plates has been observed due to BFB degradation. Detached baffle plates would constitute a large loose part, but the potential for these plates to travel is not credible because of the small clearances between the plates and the fuel assemblies.

As previously stated, the metal impact monitoring system is not a safety system and it has no operability or regulatory requirements. As a result, there are no minimal performance criteria relative to identifying small metal loose parts within the reactor coolant system. It is the NRC staff's position that the metal impact monitoring system has limited effectiveness for detecting BFB degradation and should not be considered as an alternate means for monitoring BFB performance on-line. It has not been the NRC's past practice to require licensees to provide evaluations of system performance or operating experience for nonsafety systems. Furthermore, the staff has not identified a basis to make an exception to past practice and issue a demand for information as requested by the petitioner. Therefore, the petitioner's request to issue a demand for information relative to the operating performance and history of the metal impact monitoring system is denied.

III. Conclusion

The petitioner requested that the NRC take enforcement actions against the Indian Point licensee relative to the emergent issue of BFB degradation within the reactor vessel. Subsequent to issuing the proposed director's decision for comment, the petitioner withdrew his request that the NRC issue an order requiring the licensee to inspect the Indian Point 2 BFBs and implement the downflow to upflow modification during the spring 2018 refueling outage. In addition, the licensee withdrew its previous commitment to implement the downflow to upflow modification for both Indian Point 2 and 3. Nonetheless, the NRC would have denied this request because the licensee has committed to inspect the BFBs and the staff would still retain the option to take enforcement actions if necessary. While the petitioner also withdrew his request for a demand for information requiring the licensee to perform an operability determination for Indian Point 3, this request was effectively met inasmuch as the licensee performed the evaluation and made it available to NRC inspectors as part of the NRC's reactor oversight program. Finally, the NRC denied the petitioner's request for the NRC to issue a demand for information requiring the licensee to provide an evaluation of the operating history of the metal impact monitoring system because the system has no operability or regulatory requirements, loose BFB heads would be expected to remain in place due to the tight clearances between the baffle plate and fuel assemblies, thus making bolt failures very difficult to monitor using this system, and the NRC finds no basis to require such information for a nonsafety system.

As provided in 10 CFR 2.206(c), the NRC will file a copy of this director's decision with the Secretary of the Commission for the Commission to review. As provided for by this regulation, the decision will constitute the final action of the Commission 25 days after the date of the decision unless the Commission, on its own motion, institutes a review of the decision within that time.

Dated at Rockville, Maryland, this 13th day of April 2017.

For the Nuclear Regulatory Commission.

/RA/

William M. Dean, Director,
Office of Nuclear Reactor Regulation.

COMMENTS ON THE PROPOSED DIRECTOR'S DECISION

Comments from the Petitioner:

Comments were received from the petitioner by letter dated January 19, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17024A201).

1. The petitioner raised timeliness concerns by noting the following:

   (a) Page 4 of the proposed director's decision states that the degradation of baffle-former bolts (BFBs) was first identified in Europe in 1988, but it took the U.S. Nuclear Regulatory Commission (NRC) 10 years to issue NRC Information Notice 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants," dated March 25, 1998, that first notified the U.S. industry of this phenomena.

   (b) Page 4 of the proposed director's decision states that the industry's response to BFB degradation was via MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," in 2008, which was 10 years after issuance of NRC Information Notice 98-11.

   (c) Page 4 of the proposed director's decision states that the NRC approved use of MRP-227-A to address aging management issues for BFBs. The proposed director's decision did not mention that NRC approval of MRP-227-A in 2011 was 3 years after MRP-227-A was submitted, 13 years after issuance of NRC Information Notice 98-11, and 23 years after the problem was first identified in European reactors.

Response: The timeliness concerns noted by the petitioner do not alter the overall conclusions and, therefore, do not require modification to the final director's decision.

2. The petitioner commented that the NRC staff provided sound and reasonable bases for its decisions in the proposed director's decision.

Response: The petitioner's comments do not require modification to the final director's decision.

Comments from the Licensee:

Comments were received from the licensee by letter dated February 9, 2017 (ADAMS Accession No. ML17045A470).

The licensee stated that the proposed director's decision provided a complete and generally accurate basis for its decisions. In its letter, the licensee 1) described its enhanced BFB inspection plans for subsequent refueling outages, 2) summarized the failure analysis findings of BFBs examined at the Westinghouse hot lab testing facility, and 3) informed the NRC staff that it no longer plans to implement the downflow to upflow modification for either unit as it had previously committed to do.

The licensee stated that its enhanced BFB inspection plans included the following:

- The IP3 baffle bolt inspections previously scheduled to be performed in 3R20 (Spring 2019) will be performed in 3R19 (Spring 2017). Visual and UT inspections on 100% of all accessible baffle former bolts, and a visual inspection of the accessible baffle-edge bolts and baffle former assembly, will be performed in 3R19.
- Entergy will perform a UT inspection of 100% of the original bolts at IP2 and IP3 during each of the subsequent refueling outages if any of the original bolts are required to remain structurally capable of carrying their design load to ensure structural integrity of the baffle structure during all design conditions.
- Entergy will also perform a general visual inspection to identify anomalies in the baffle structure at IP2 and IP3 during each subsequent refueling outage.
- Entergy will perform a UT inspection of inservice replaced (new) bolts if the general visual inspections identify degraded new bolts.
- Entergy will replace all bolts with indications that are needed to remain structurally capable of carrying their design load to ensure structural integrity of the baffle structure during all design conditions. Additional "good" or anti-cluster bolts will also be replaced to ensure that sufficient margin is maintained to accommodate the same failure rate until the next inspection as the failure rate identified during the current refueling outage.

The licensee provided the following information regarding the BFB failure analysis:

> As a corrective action to the BFB degradation identified at IP2 during the 2016 refueling outage, eight BFBs removed from the IP2 baffle structure during the 2016 outage were examined by the Westinghouse hot lab testing facility. The Westinghouse fractography examinations indicated that the cause of the IP2 baffle bolt failures was a complex combination of Intergranular Stress Corrosion Cracking, fatigue caused by cyclical loads, and ductile tearing/overload when the flaw reached a size where the remaining bolt ligament was insufficient to carry the remaining load. While the time from crack initiation to final bolt failure could not be precisely established, based on the oxides detected on the fracture surfaces, it is likely that the time period between crack initiation and final bolt failure occurred over several operating cycles.

<u>Response</u>: As concluded by both the petitioner and the licensee, the enhanced BFB inspections planned for the remaining refueling outages should provide sufficient protection, without implementing the downflow to upflow modification. The comments from the licensee, most notably the change in commitment to implement the downflow to upflow modifications, resulted in changes to the proposed director's decision.