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

William M. Dean, Director

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

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), Mr. David A. Lochbaum, director of the Nuclear Safety Project at the Union of Concerned Scientists, filed a petition under Title 10 of the *Code of Federal Regulations* (10 CFR) 2.206, "Requests for Action Under This Subpart," to Mr. Victor M. McCree, Executive Director for Operations at the U.S. Nuclear Regulatory Commission (NRC or the Commission). The petitioner requested that the NRC take the following enforcement actions against the licensee for the Indian Point Nuclear Generating Unit No. 2 and No. 3, Entergy Nuclear Operations, Inc.:

1) Issue an order requiring the licensee to inspect the reactor vessel baffle-former bolts (BFBs) and to install the downflow to upflow modification at Indian Point Unit No. 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 Unit No. 3 until its reactor vessel BFBs can be inspected according to the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Topical Report MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (ADAMS 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 (LER) 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 where there was an unanalyzed condition due to degraded reactor vessel BFBs at Indian Point Unit No. 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 Unit No. 2, (2) Indian Point Unit No. 3 is potentially operating with degraded baffle-former bolts 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 baffle-former bolts, yet neither the NRC nor the licensee have 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.

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," (Legacy ADAMS Accession No. 9803230106) in 1998 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 a 10-year interval in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

In response to a license renewal commitment, the licensee for Indian Point Unit No. 2 performed visual inspections and UT examinations of the BFBs during their 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 their 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 Unit 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 Unit No. 2 and No. 3 corrective actions. In addition, Entergy performed an operability determination to evaluate the impact of BFB degradation at Indian Point Unit No. 3. NRC inspectors reviewed Entergy's evaluations and concluded that these evaluations provide reasonable assurance that the Indian Point Unit No. 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 Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-504, Revision 4, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," effective June 2, 2014, the staff of the NRC performed a risk-informed evaluation of the safety significance of recently identified reactor vessel BFB degradation (ADAMS Accession No. ML16225A341). As discussed in the evaluation, 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 includes Indian Point Units 2 and 3. The staff also concluded that degradation of BFBs does not represent an imminent safety hazard and, as a result, immediate plant shutdowns to inspect and repair degraded BFBs is 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 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 Material Reliability Program working group to better understand the safety significance of BFB degradation and the extent of condition within the industry. A public meeting was held on July 19, 2016, with representatives of the EPRI Materials Reliability Program 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 Accession No. ML16208A001. Subsequent guidance from both Westinghouse and the EPRI Materials Reliability Program 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 Unit No. 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 includes both Indian Point Units No. 2 and No. 3, and recommends 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 baffle-former bolts (BFBs) and to install the downflow to upflow modification on Indian Point Unit No. 2 during its next refueling outage.

NRC Response:

Based on a review of operating experience, the potential for significant bolt 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, issued March 7, 2012, following the operating experience at D.C. Cook Unit 2, identified seven operating reactors that were considered most susceptible to BFB degradation—Indian Point Units No. 2 and 3, Salem Units 1 and 2, D.C. Cook Units 1 and 2, and Diablo Canyon Unit 1. The NRC staff confirmed the bolt material as Type 347 stainless steel for all these plants. Westinghouse Nuclear Safety Advisory Letter (NSAL) 16-1,[1] Revision 1 "Baffle-Former Bolts," dated August 1, 2016, issued in response to the experience at Indian Point Unit No. 2, and Salem Unit 1, 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 Materials Reliability Program Letter 2016-022, dated July 27, 2016 (ADAMS Accession No. ML16211A054), contains 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

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[1] Westinghouse Nuclear Safety Advisory Letter 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. NRC staff in the Office of Nuclear Reactor Regulation 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 staff's own engineering judgement in evaluating potential risk and regulatory options.

of NEI 03-08, Revision 2, “Guideline for Management of Materials Issues,” issued January 2010 (ADAMS Accession No. ML101050337). The identification of the most susceptible group of plants to BFB cracking in the NSAL and the MRP letter is consistent with the staff’s assessment based on its review of operating experience. It should be noted that if any licensee in the most susceptible group (i.e., Tier 1a, which includes Indian Point Unit No. 2) intends to deviate from the EPRI Materials Reliability Program 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.

Entergy has committed to inspect the BFBs and reconfigure the reactor core flow from downflow to upflow during the next scheduled refueling outage at Indian Point Unit No. 2. The NRC staff recognizes that these actions are commitments as documented in the licensee’s corrective action program and that these commitments, along with their timing, may be modified based upon the results of the failure analysis of the BFBs and future industry developments. The unexpected extent of BFB degradation is an emergent issue and the industry may develop alternative, more flexible approaches based upon new information and recommendations. If the licensee chooses to modify or delete these commitments, the NRC would expect the changes to be justified in accordance with 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” and the licensee’s Corrective Action Program. Such changes would be inspectable by the NRC and regulatory actions could be taken as warranted.

Based upon the industry’s recommended guidance to inspect the BFBs for the Tier 1a plants during the next refueling outage and the licensee’s commitments as described above, the NRC staff does not plan to take enforcement action to make these commitments legally binding. Therefore, the NRC denies 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 continued operation of Indian Point Unit No. 3 until its baffle bolts can be inspected according to the guidance of MRP-227-A.</u>

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 Unit 2 represented an unanalyzed condition, and 2) NRC Inspection Manual Chapter 0326, "Operability Determinations & Functionality Assessments for Conditions Adverse to Quality or Safety" (ADAMS Accession No. ML15328A099) which identifies those circumstances when 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 Unit 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 Unit 2, Indian Point Unit 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 the steps in EN-OP-104 and performed an operability evaluation, which assumed an estimated number of BFB failures based on the degradation found in Indian Point Unit No. 2, and 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 Unit No. 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 Unit No. 3 baffle assembly would support emergency core cooling system operability until the planned Indian Point Unit No. 3 refueling outage in spring 2017.

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 the Indian Point Unit No. 3 baffle assembly remains operable and will perform its intended safety function through the spring 2017 refueling outage when the BFBs will be inspected. 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 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, Rev. 1, "Loose Parts Evaluation - Residual Heat Removal Valve Parts" (ADAMS Accession No. ML993610326), that reported that the metal impact monitoring system at Indian Point Unit No. 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 which, 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 bolt heads cannot 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 cannot fall out of its location, even if the locking bar fails. A bolt head trapped in the gap can 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 travel of these plates 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 1) has limited effectiveness for detecting BFB degradation, 2) should not be considered as an alternate means for monitoring BFB performance online, and 3) may not be sufficiently sensitive to detect loose bolt heads and locking tab bars. It has not been NRC 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. The NRC staff denied the petitioner's request that the NRC issue an order requiring the licensee to inspect the Indian Point Unit No. 2 BFBs and implement the downflow to upflow modification during the spring 2018 refueling outage. The NRC staff denied this request because the licensee has committed to take these actions and the staff retains the option to take enforcement actions if necessary. The petitioner's request for a demand for information requiring the licensee to perform an operability determination for Indian Point Unit No. 3 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 staff 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, the staff does not believe that the system would be effective in identifying degraded BFBs within the reactor vessel, and the staff 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, MD, this day of

For the Nuclear Regulatory Commission.

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

PROPOSED