DD-09-XX

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

Eric J. Leeds, Director

In the Matter of	)	Docket Nos. 50-315
	)	
INDIANA MICHIGAN POWER COMPANY	)	License No. DPR-58
	)	
	)	
Donald C. Cook Nuclear Plant, Unit 1	)	

PROPOSED DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. Introduction

By letter to R. William Borchardt, Executive Director for Operations for the U.S. Nuclear Regulatory Commission (NRC), dated December 16, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML083640201), Mr. David Lochbaum (the Petitioner), on behalf of the Union of Concerned Scientists (UCS), filed a petition pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 2.206, "Requests for action under this subpart." Mr. Lochbaum subsequently informed the NRC, in a letter dated February 2, 2009 (ADAMS Accession No. ML090370688), that Dr. Edwin Lyman would be representing UCS as the new petition point of contact.

Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the NRC Web site 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 reference staff in the NRC Public Document Room by telephone at 1-800-397-4209, or 301-415-4737, or by e-mail to PDR.Resource@nrc.gov.

Enclosure

Action Requested

The Petitioner requested that the NRC issue a demand for information requiring the Indiana Michigan Power Company (the licensee) to docket the following information at least 30 days prior to restarting the Donald C. Cook Nuclear Plant, Unit 1 (CNP-1) reactor from its current outage:

(1) The vibration levels experienced in the control room, turbine building, and other structures during the September 20, 2008 event;

(2) The vibration levels assumed in these locations during a safe shutdown earthquake (SSE);

(3) In locations where the vibration levels during the September 2008 event exceeded the vibration levels assumed for SSE, the extent of piping, pipe supports, etc. replaced/repaired due to potential stress damage and the bases for not replacing other structures, systems, and components (SSCs) exposed to greater than SSE loading; and

(4) In locations where the vibration levels during the September 2008 event did not exceed the vibration levels assumed for SSE, the extent of measure taken to protect against spurious equipment operation and the bases for concluding the as-left configuration will not pose a public health hazard in event of a SSE.

Petitioner's Bases for the Requested Action

The Petitioner stated that the September 20, 2008, event caused significant vibration levels that resulted in the spurious operation of standby equipment and may have contributed to a breach that seriously impaired the fire protection system. The Petitioner based the request for information, in part, on a need for the licensee to apply the proper lessons from the event to the future operation of the CNP-1 reactor. The Petitioner stated that, absent this information, the NRC cannot be assured that the public is adequately protected from the significant adverse safety implications due to an SSE at CNP-1 that causes spurious actuation of equipment. The Petitioner provided further details as follows regarding the bases for seeking the actions described in the previous section.

Item (1) requests information on the magnitude of vibration levels throughout the plant during the September 2008 event. The magnitude of the motion in the turbine building and control room should be ascertained. Since the turbine building and control room are physically attached to other structures, the motion in these other structures should be quantified as well.

Item (2) requests that the above information be put in the proper design and licensing basis context by documenting the magnitude of vibration levels assumed for the SSE. This information is requested to provide the framework to determine whether the CNP-1 design and licensing bases bound the magnitude of that event.

Item (3) requests the identification of those SSCs that may have been damaged by exposure to vibration levels above those assumed in the design analyses and procurement specifications that were either replaced or repaired, or were evaluated such that the as-left configuration provides reasonable assurance that this equipment can perform all required safety functions during and after all design and licensing basis events.

Item (4) requests assurance that, if vibration levels were less than those for an SSE, any equipment that operated spuriously during the event would be evaluated to determine the consequences and implications if actual vibration levels had been higher than those assumed in the SSE. The information will document protective measures (e.g., hardware or procedures) applied against spurious equipment operation or the evaluations concluding that spurious equipment operation has no adverse safety implications.

Determination for NRC Review under 10 CFR 2.206

On January 21, 2009, the NRC Petition Review Board (PRB) convened to discuss the petition under consideration and determine whether it met the criteria for further review under the 10 CFR 2.206 process. The PRB was comprised of NRC technical and enforcement staff, legal counsel, and chaired by an NRC senior-level manager. The PRB determined that the petition under consideration did meet the criteria established in NRC Management Directive 8.11, “Review Process for 10 CFR 2.206 Petitions,” for acceptance into the 10 CFR 2.206 process.

The NRC discussed this conclusion with the Petitioner during a telephone conversation on January 27, 2009. The Petitioner stated during this conversation that a public meeting to address the PRB was not required. In a letter dated March 6, 2009 (ADAMS Accession No. ML090370035), the NRC informed the Petitioner that the PRB had approved the petition request and that it was referring the issues in the petition to the Office of Nuclear Reactor Regulation for appropriate action.

II. Discussion

Background

On September 20, 2008, the CNP-1 main turbine failed because the design of the blade-rotor system did not provide an adequate stress margin in at least three low-pressure turbine blades. The blades ultimately exceeded their stress threshold and suffered high-cycle fatigue cracking. As a result, control room turbine vibration monitors indicated high-high vibration levels on all main turbine supervisory instrumentation vibration points. Numerous alarms were also received on components in the condensate and feedwater systems. The event produced noticeable ground and building vibrations both at and near the plant.

Control room operators initiated a reactor shutdown, entered the reactor trip response emergency operating procedure, and opened the main condenser vacuum breakers to stop main turbine rotation. All reactor safety systems operated as designed, and no unexpected safety-related system actuations occurred. The reactor protection system actuated to fully insert all control rods. The auxiliary feedwater system initiated and functioned as required. Steam generator atmospheric relief valves were operated to remove decay heat.

A hydrogen fire occurred concurrent with the event, originating from below the CNP-1 turbine and main generator. A portion of the main generator cooling system, which uses hydrogen, failed and hydrogen was released. Several fire protection sprinkler systems automatically actuated, and operators manually actuated a turbine water spray system to assist in controlling the fire. The on-site fire brigade responded to the fire. The licensee also requested off-site fire protection assistance, but it was not required to extinguish the fire. The fire was extinguished within 30 minutes.

A section of the fire protection system's yard loop piping failed early in the event, resulting in the loss of at least 544,000 gallons of water from the north fire water storage tank. Initially, control room operators and fire brigade personnel were not aware of this failure. Although only one fire pump was necessary to supply flow for the fire protection systems that had actuated, all three fire pumps (one electric- and two diesel-driven) started as a result of the break in order to maintain system pressure. Control room operators received a low fire protection header pressure alarm and directed fire brigade personnel to investigate. Fire brigade personnel discovered that all three fire pumps were operating without water and determined that the aligned fire water storage tank was empty. After consulting with the control room operators, fire brigade personnel shut down all three fire pumps. Both diesel-driven pumps showed evidence of overheating with one damaged beyond repair. The pipe break was isolated to permit recovery of the fire protection system. It was subsequently discovered that operations and fire brigade personnel did not initially recognize that there was a pipe break until they attempted to refill the fire protection system.

Licensee Event Report

The licensee provided details of the September 20, 2008, event in Licensee Event Report (LER) 315/2008-006-01, “Manual Reactor Trip Due to Main Turbine High Vibrations” (ADAMS Accession No. ML083370197), in accordance with 10 CFR 50.73, “Licensee Event Report System.” The LER documented the details of the event; provided an analysis of the event, including estimated change in conditional core damage probability; and provided a list of corrective actions.

The LER stated that the licensee performed extensive inspections to identify and document damage to plant equipment resulting from the event. The results of the inspections identified no significant damage to safety-related SSCs. The licensee determined that all plant systems necessary to shut down the unit and remove decay heat performed as designed and that the event did not represent a significant increase in overall risk.

The licensee also determined that the trip event and response resulted in a change in conditional core damage frequency of 7.811E-07, which represents a nominal increase in plant risk from expected plant performance consistent with NRC Inspection Manual Chapter 0308, “Reactor Oversight Process (ROP) Basis Document,” and Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis.”

Special Inspection

The NRC initiated a special inspection in accordance with Inspection Procedure 93812, “Special Inspection,” based on the risk and deterministic criteria established in NRC Management Directive 8.3, “NRC Incident Investigation Program,” and Inspection Procedure 71153, “Event Follow-up,” and on the equipment performance problems associated with the event. The Special Inspection Team (SIT) evaluated the facts and circumstances surrounding the September 20, 2008, turbine high vibration event and resulting failure of the fire protection system. The team detailed the results of the inspection in “D.C. Cook Nuclear Power Plant, Units 1 and 2, NRC Special Inspection Team Report 05000315/2008009; 05000316/2008009,” dated January 23, 2009 (ADAMS Accession No. ML090260032).

The SIT made one finding of very low significance (a Green finding as determined using NRC Inspection Manual Chapter 0609, “Significance Determination Process”) related to the licensee’s failure to have appropriate procedures for control room operator actions associated with response to a fire protection alarm panel. The lack of guidance resulted in a failure to diagnose the fire protection system failure as evidenced by the simultaneous operation of all three fire pumps. This contributed to operators not recognizing a break in the fire protection system header outside the turbine building. The licensee entered the issue into its corrective action program and has subsequently revised procedures determined to have been either lacking or insufficient to cope with the event.

The SIT reviewed the licensee’s plans and actions for assessing the impact that the event had on the SSCs at CNP-1. The SIT held discussions with site engineering staff, reviewed licensee action plans for evaluating the condition of SSCs in the turbine building, reviewed in-process results from the licensee's reviews, and evaluated the methods for tracking identified deficiencies.

The SIT noted that the licensee developed an action plan to assess the impact of the event on plant systems. The action plan was broken down by disciplines (e.g., structural integrity (civil engineering), piping integrity, and electrical), and the turbine building was inspected by dividing it into areas using pre-existing structural column designations. The corrective action program was used to track identified deficiencies. The licensee response letter dated May 12, 2009 (ADAMS Accession No. ML091420327) contains specific details of the inspection findings.

In an effort to quantify the magnitude of the vibration, the SIT discussed with the licensee the response of the site's seismic monitoring equipment. As described in Section 2.5 of the Updated Final Safety Analysis Report (UFSAR), a ground acceleration of 0.2g (acceleration equivalent to 20 percent of that caused by the force of gravity) is assumed in the design of SSCs required to safely shut down the reactor, and for operability of engineered safety features systems following an SSE. Other major plant structures are designed for a maximum horizontal ground acceleration of 0.10g, that of the operating-basis earthquake (OBE). Two types of seismic monitoring equipment are available: (1) a recorder with a trip setpoint to turn the unit on, and (2) peak recording accelerographs (tape plates/scratch gauges), which are passive devices without setpoints. A seismic event at 0.02g, in either the horizontal or vertical direction, triggers the recording devices. The licensee indicated that the recording devices did not start during the event, thus indicating that the vibration levels did not exceed 0.02g and, therefore, never approached the seismic levels of either the OBE or the SSE. The licensee response letter further discusses the design and operation of the seismic monitoring equipment.

Licensee Response Letter

In a letter dated May 12, 2009 (ADAMS Accession No. ML091420327), the licensee provided a response to the Petitioner's requests for information.

In response to Item (1), the licensee noted that vibration levels experienced in the control room, turbine building, and other structures were not measured or recorded during the event. The seismic triggers are actuated at a setpoint of 0.02g. One seismic trigger is located in a block house in the 345 kilovolt switchyard, located several hundred yards from the turbine building. The other seismic trigger is located in the bottom of the CNP-1 containment. Since neither seismic trigger activated, the recording equipment of the seismic instrumentation system did not actuate.

The licensee provided information regarding peak acceleration or peak displacement recorders installed on selected structures throughout the plant to aid in the characterization of a seismic event. None of these recorders are located in the turbine building and they do not have a means of recording the time of the disturbance; therefore, the licensee stated that they would not be expected to provide any characterization of vibration levels experienced during the event.

Based on this information, the licensee concluded that the ground acceleration at the containment building was below 0.02g.

In response to Item (2), the licensee noted that in Section 2.5 of the CNP UFSAR, a ground acceleration of 0.20g is assumed in the design of SSCs required to safely shut down the reactor and for operability of engineered safety features systems. Other major structures are designed for a maximum horizontal ground acceleration of 0.10g.

Based on this information and the response to Item (1) above, the magnitude of vibration resulting from this event was clearly below the threshold actuation setpoint for seismic instrumentation located in other structures. Therefore, the level of vibration experienced by these structures is lower than design basis accelerations.

In response to Item (3), the licensee noted that there was no evidence that the vibration levels experienced by safety-related SSCs during the event exceeded those assumed for the SSE.

The licensee's assessment team evaluated the damage associated with the event. The team performed extensive inspections of the turbine building structures and equipment. The assessment team concluded that the event did not degrade the structural elements of the turbine building.

The event did result in significant damage to the main turbine rotors and bearings, as well as the turbine casing and hoods, and in localized damage to the concrete and grout of the main turbine foundation in areas that interface with the turbine hoods. The licensee stated that it would complete repairs to these structures before returning CNP-1 to service.

In response to Item (4), the licensee concluded that spurious equipment operation occurring during the event was directly attributable to the vibrations associated with the main turbine failure. The only equipment that spuriously operated was associated with balance-of-plant systems and not associated with functions required to ensure public health and safety. Plant systems necessary to shut down the unit and remove decay heat operated as designed.

Based on this information, the licensee concludes that the condition of SSCs will not pose a hazard to the public health and safety in the event of an SSE following the return to service of CNP-1.

Regulatory Evaluation

The NRC staff reviewed the Petitioner's request to the NRC for specific information to be obtained from the licensee relating to the main turbine vibration event. The staff also reviewed the licensee's response to the four items identified in the petition. In its clarification of the first item, the Petitioner requests a value for the vibration levels experienced in the control room and turbine building during the event. The assertion is that the turbine building and control room structures are physically attached to other structures. The staff notes that this assertion is inaccurate because the turbine building does not have a common foundation with other structures. There is a physical gap (i.e., rattle-space) between the turbine building and adjacent structures, so that the buildings will not affect each other if lateral sway occurs during a design basis seismic event. Although steamline piping is connected to the turbine building, the ability for this piping to transmit major vibrations from the turbine building to adjacent buildings is fairly limited, because of the stiffness and mass characteristics, compared to the connected structure. Absent a common foundation between the turbine building and another structure, the only mode for transmitting major vibration would be through the soil medium. Considering that none of the seismic monitors were activated during the event, it is evident that the vibration traveling through the soil was of such low magnitude that it did not challenge the design basis or affect the safety function of any plant safety-related equipment.

The NRC staff viewed the impact of the turbine vibration on adjacent safety-related structures as having minimal effect, but this should not be construed as implying that the level of vibration experienced by the turbine building and equipment located inside the turbine building was insignificant. The magnitude of the vibration imparted on the turbine building as a result of the event is unquantifiable and cannot be correlated to a postulated seismic design basis event.

Therefore, a realistic quantification of the magnitude of vibration from the event can only be inferred from the damage observed by the assessment team; this is discussed in the licensee's response letter to the Petitioner's request.

Considering the findings resulting from the assessment team's extensive inspection, in addition to those of the NRC Special Inspection and follow-up inspections, the staff concludes that the concerns in the Petitioner's request are not credible for two reasons. First, an erroneous assumption was made in quantifying the magnitude of vibration from the event based on a presumption of the presence of seismic monitoring instrumentation in all the affected buildings. Secondly, it was incorrect to assume that the turbine and control buildings are physically connected to other safety-related structures, and that these structures and their housed equipment may have experienced vibration levels approaching the design basis SSE for the plant.

III. Conclusion

The Petitioner raised potential safety concerns related to the September 20, 2008, main turbine failure at CNP-1. The Petitioner requested that the licensee provide a response addressing concerns related to actual or potential seismic issues resulting from the excessive vibration experienced during the event. The Petitioner asked that the licensee respond to specific concerns related to the plant's future ability to operate safely as a result of the event, and that the response docketed at least 30 days prior to restart. The licensee voluntarily provided a response that has been docketed and made publicly available.

The NRC technical staff reviewed the licensee's response and other docketed information associated with the event. The NRC staff concludes that public health and safety were not affected by the event, nor would they be affected by a similar event in the future.

Furthermore, the NRC has found the licensee's event response and corrective actions to be reasonable and technically sound.

Based on the above, the Office of Nuclear Reactor Regulation has concluded that future operation of CNP-1 provides reasonable assurance of the continued protection of public health and safety. There were no violations associated with the event. The licensee's corrective actions in response to the event appear appropriate. No further action is required.

As provided in 10 CFR 2.206(c), a copy of this director's decision will be filed 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 2nd day of July, 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation