DD-99-14

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
Samuel J. Collins, Director

In the Matter of	)	Docket No. 50-220
	)	
NIAGARA MOHAWK POWER CORPORATION	)	License No. DPR-63
	)	
(Nine Mile Point Nuclear Station,	)	
Unit No. 1)	)	

FINAL DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. INTRODUCTION

By letter dated May 24, 1999 (the Petition), pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR 2.206), Mr. Tim Judson (the Petitioner) of the Syracuse Peace Council requested, on behalf of himself and others,[1] that the U.S. Nuclear Regulatory Commission (Commission or NRC) suspend the operating license issued to Niagara Mohawk Power Corporation (NMPC or licensee) for Nine Mile Point Nuclear Station, Unit 1 (NMP1) until (1) NMPC releases the most recent inspection data on the plant's core shroud; (2) a public meeting can be held in Oswego County, New York, to review this inspection data and the repair design to core shroud vertical welds V9 and V10; and (3) an adequate public review of the safety of the plant's continued operation is accomplished.

In a letter dated June 11, 1999, the Director of the Office of Nuclear Reactor Regulation acknowledged receipt of the Petition of May 24, 1999, and addressed the actions under 10 CFR

[1] The others are Citizens Awareness Network, Coalition on West Valley Nuclear Waste, Environmental Advocates, Greens of Greater Syracuse, Nuclear Information and Resource Service, Oswego Valley Peace and Justice, Sierra Club (Iroquois Group), Student Environmental Action Coalition (SU/SUNY-ESF), Syracuse Anti-Nuclear Effort, and Dr. Steven Penn, Ph.D. In July 1999, Mr. Judson left his position with the Syracuse Peace Council to assume a position with the Central New York Chapter of Citizens Awareness Network.

2.206 that Petitioner requested to be taken before restart of NMP1 from its 1999 refueling outage (RFO-15). In the letter of June 11, 1999, the staff explained that the issues and concerns addressed in the Petition do not warrant deferring restart of NMP1 and that a meeting to provide for public review of the shroud reinspection results need not be held before restart.

In a supplemental letter dated August 10, 1999, Petitioner reiterated the request for the meeting to provide for public review of the shroud reinspection data and repair, even though it would be held after restart, and raised additional issues regarding cracks identified in the main drain line and control rod stub tubes during the hydrostatic testing of the reactor vessel. Petitioner also expressed concern, based on the reported 1999 core shroud inspection results, that shroud vertical weld V10 was exceeding the NRC's accepted crack growth rate limit.

II. BACKGROUND

As a basis for the requests in the initial Petition of May 24, 1999, the Petitioner asserted that--

1. Petitioner believes that the public cannot rely upon NMPC to accurately perform the data analysis necessary to calculate the extent and rate of cracking in the core shroud because of problems with NMPC's previous testing and analyses that were identified in letters to the NRC from Dr. Penn. Petitioner states that the NRC has not responded to Dr. Penn's letters, and, therefore, Petitioner believes Dr. Penn's expressed concerns constitute unreviewed safety issues.
2. NMPC and NRC reported during the May 1999 inspection that cap screws in the bow spring mechanisms of the shroud tie rod assemblies were found to have suffered intergranular stress-corrosion cracking, resulting in the fracture of one of the cap screws. Petitioner states that this problem, and the tie rod problem corrected during the 1997 outage, indicates that NMPC's designs warrant in-

depth review by the public and closer implementation scrutiny. Petitioner believes that NMPC's prior selection of poor cap screw material and the NRC staff's acceptance of it raises questions about the credibility of the NRC's approval of the vertical weld repair design and, thus, necessitates a public review of the level of safety before plant restart.

3. Data from the May 1999 inspection of the NMP1 core shroud are new and the NRC staff's review of the data will not be completed before plant restart. Petitioner states that previous NRC staff safety evaluations required future evaluations. Petitioner believes that subsequent NRC approval of an "unprecedented and unproven" repair design for vertical welds, issued before the inspection, does not preempt the previously determined need to assess the actual extent of cracking in the vertical welds and the structural integrity of the core shroud.

4. NMPC has informed the NRC that supporting a meeting for public review of the core shroud inspection data during this refueling outage would place an undue regulatory burden on NMPC's manpower resources, and this burden could possibly compromise safety at NMP1. Petitioner considers inadequate licensee resources to be new information and an unreviewed safety issue. Petitioner contends that violations and a civil penalty issued against NMPC on November 6, 1997, involving inadequate management oversight and failure to monitor the effectiveness of maintenance activities are "directly pertinent to the failure of the tie rod installation (1995), faulty design of the bow spring modification (1997), flawed studies on core shroud boat samples (1998), postponement of mid-cycle inspection (1998), and miscalibration of instruments for vertical weld inspection (May 1999)." Petitioner believes that because the

degree of cracking in the NMP1 shroud is precedent-setting, the question of regulatory burden is not relevant, as the NMP1 shroud requires the strictest regulatory oversight and a full public review. Petitioner states that postponing restart would eliminate this regulatory burden and ensure that outage work is properly reviewed.

In a supplemental letter dated August 10, 1999, Petitioner reiterated the request for the meeting to provide for public review of the shroud reinspection data and repair, even though the meeting would take place after restart. Petitioner stated that the need for the meeting had increased because cracks were identified in the main drain line and control rod stub tubes during the hydrostatic testing of the reactor vessel during RFO-15. Petitioner stated that these cracks from the hydrostatic tests raise two concerns: (1) that the NRC's "leak-before-break" model for assessing the safety of aging reactors is inadequate and (2) that the problem of cracking is not confined to the core shroud, but may be spreading throughout the reactor internals, pipes, and other systems, representing an unanalyzed condition that is only being identified piecemeal through certain incidental cases that, together, reveal a pattern of degradation of reactor components and systems and overall embrittlement of the reactor. Petitioner also expressed concern in the letter of August 10, 1999, that the core shroud inspection during RFO-15 indicated that shroud vertical weld V10 is growing at a rate in excess of the NRC's accepted crack growth rate limit of 22 microinch/hr, whereas he believes the measured rate should be at least 2 sigma below the limit.

III. DISCUSSION

1. THE NRC SHOULD SUSPEND THE NMP1 OPERATING LICENSE UNTIL (1) NMPC RELEASES THE MOST RECENT INSPECTION DATA ON THE PLANT'S CORE SHROUD; (2) A PUBLIC MEETING CAN BE HELD IN OSWEGO COUNTY, NEW YORK, TO REVIEW THIS INSPECTION DATA AND THE REPAIR DESIGN TO CORE SHROUD VERTICAL WELDS V9 AND V10; AND (3) AN ADEQUATE PUBLIC REVIEW OF THE SAFETY OF THE PLANT'S CONTINUED OPERATION IS ACCOMPLISHED.

As stated in the letter of June 11, 1999, the NRC's Petition Review Board (PRB) determined that the Petition meets the criteria for a request under 10 CFR 2.206 and that the NRC staff would inform the Petitioner within a reasonable time of the action to be taken on his requests. The letter stated that the PRB had also determined that the issues and concerns addressed in the Petition did not warrant deferring restart of NMP1 and that a public meeting on the core shroud reinspection results need not be held before restart. In reaching this determination, the PRB had considered the following:

1. By letter dated May 28, 1999, the NRC staff responded to Dr. Penn's letters dated December 3, 1998; March 25, 1999; and April 15, 1999. In a letter dated April 30, 1999, NMPC had also responded to relevant concerns in Dr. Penn's letter of March 25, 1999. The responses indicate that testing and evaluations of the core shroud by NMPC and its contractors can be relied upon by the NRC with reasonable assurance as to their accuracy. Therefore, the issues in Dr. Penn's letters do not provide a sufficient basis to warrant suspension of the NMP1 operating license.
2. The bow spring modification to each of the four tie rod assemblies replaced the design function of the failed cap screw and other cap screws that had the potential for future failure. By letter dated May 28, 1999, NMPC confirmed that no additional modifications were needed other than the bow spring modification addressed in NMPC's letter of May 21,

1999. The tie rod bow spring does not affect the tie rod's function of maintaining a predetermined compressive force ("preload") on the shroud during power operation. In response to NMPC's letter dated May 21, 1999, the NRC staff had reviewed and, by letter dated June 7, 1999, approved the modifications as an alternative repair pursuant to 10 CFR 50.55a(3)(i). NMPC implemented these modifications. With the NRC staff's review and approval of this modification, the NRC staff found no basis for considering enforcement action to suspend the operating license.

3. During the 1999 refueling outage, NMPC implemented preemptive repairs of shroud vertical welds V9 and V10, as approved by the NRC staff in a letter dated April 30, 1999. These repairs mechanically restored the vertical welds. NMPC had also verbally informed the NRC that the 1997 modifications to the tie rod assemblies had performed satisfactorily and that the tie rod assemblies had applied the appropriate preload on the shroud throughout the previous operating cycle. Since vertical welds V9 and V10 were restored and the tie rods are satisfactorily performing their preload function, the need for NRC staff review of reinspection data before restart was obviated.

4. In accordance with the approved Boiling Water Reactor Vessel and Internals Project's report BWRVIP-01, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," NMPC would provide reinspection results and analyses to disposition these reinspection findings to the NRC within 30 days of completing the reinspection. Noting the results of inspections at that time, the resource impact upon the licensee, and that NMPC had followed the BWRVIP generic criteria for inspection, evaluation, and repair, the NRC staff concluded that a public meeting was not warranted before restart. However, because it recognizes the value of public meetings, the NRC staff stated in its letter of June 11, 1999, that a routinely scheduled meeting to discuss recent plant performance at the NMP site was planned. At the time of the letter, the meeting was expected to be held in August 1999 but was actually

held on October 22, 1999, at the NMP Nuclear Training Center. In this meeting, participants discussed a variety of topics related to licensee performance. A brief discussion on the NMP1 core shroud activities was one of the agenda topics.

The NRC staff has now received and reviewed NMPC's letter dated July 9, 1999, forwarding a report summarizing the horizontal and vertical shroud weld inspections performed during the 1999 refueling outage. Copies of this letter and report were forwarded to the Petitioner by the NRC staff's letter dated July 26, 1999. The report confirms that NMPC's 1999 core shroud reinspections were performed consistent with the staff-approved guidelines in BWRVIP-07, "BWR Vessel and Internals Project Guidelines for Reinspection of BWR Core Shrouds," and exceeded the approved scope of the reinspection plan to which NMPC had committed in a letter dated December 30, 1998. The 1999 reinspection included the additional inspection of the core shroud base metal adjacent to vertical welds V9 and V10 and selected areas at five horizontal welds (H1, H2, H4, H5, and H6b) adjacent to the intersections of the vertical welds. Because the vertical welds V9 and V10 were preemptively repaired and the minor intergranular stress-corrosion cracking (IGSCC) observed at other vertical welds did not show significant changes in size, NMPC did not need to perform any additional detailed vertical weld flaw evaluation to ensure structural integrity of the core shroud; the potential crack growth of these welds in the current fuel cycle is bounded by the flaw evaluations performed previously for vertical welds V9 and V10.

The NRC staff has also received and reviewed NMPC's letter dated July 12, 1999, that presents a final root cause evaluation of the cap screw that was discovered during the 1999 refueling outage to have failed in the upper spring assembly of the shroud tie rod. A copy of this report was also forwarded to the Petitioner by letter dated July 26, 1999. The NRC staff's review included NMPC's letter dated May 21, 1999, forwarding a report summarizing NMPC's 1999 findings from the visual examination of the four tie rods and reporting observations and the preliminary root cause of the failed cap screw. These reports confirm NMPC's prior verbal statement to the NRC that the tightness inspections had demonstrated that the tie rods had maintained sufficient preload on the core shroud during the previous operating cycle. These reports also confirmed NMPC's earlier preliminary root cause evaluation. In its final root cause evaluation, NMPC concluded that the cap screw failed as a result of IGSCC in the alloy X-750 cap screw material due to large sustained stresses from differential thermal expansion of dissimilar materials fastened by the cap screw. The NRC staff agrees that the condition that existed of high stresses and the environment are sufficient to cause IGSCC failure in the cap screw. The modification to the upper spring assemblies that NMPC implemented for each of the four tie rods before restart, replacing the design function of the failed cap screw and the other cap screws that had the potential for future failure, was designed to address this source of stress, as well as the other potential sources of stress on the cap screws identified in the preliminary root cause evaluation. By addressing the various potential sources of stress, NMPC ensured that the modification, implemented in advance of the final root cause evaluation, would be acceptable once that final determination was reached. Consequently, it was unnecessary to defer restart of NMP1 until the final root cause of the cap screw failure had been determined.

On the basis of its review, the NRC staff concludes that the structural integrity of the core shroud will be maintained during the current operating cycle in its present configuration. The licensee will reinspect the core shroud during NMP1's next refueling outage using the reinspection criteria in BWRVIP-07. The licensee will inform the NRC of the reinspection scope at least 3 months before the start of that outage.

2. HYDROSTATIC TESTING OF THE NMP1 REACTOR PRESSURE VESSEL IDENTIFIED CRACKS IN THE MAIN DRAIN LINE (MDL) AND IN CONTROL ROD STUB TUBES.

In his letter of August 10, 1999, Petitioner states that "the MDL leak is particularly troubling. As a small-diameter pipe, the MDL is only scheduled for inspection once every eight years. The leak, which was detected by visual inspection and not remote sensing, was fortunately discovered before restart. Had the MDL burst during operation there would be no way to stop the draining of the reactor vessel."

The NMP1 reactor vessel bottom head MDL is a type-316L stainless steel line with a 2-inch diameter. It is inspected to a schedule consistent with Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," which is based upon NUREG-0313, Revision 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping." NMPC identified a leak in the MDL on June 6, 1999, during the vessel hydrostatic test. The leak was from a crack downstream of the manual isolation valve. Upon identifying the crack, NMPC secured the hydrostatic test, depressurized the plant, installed freeze seals, and replaced the affected section of pipe. The cause was determined to be thermal stress induced fatigue that was caused by a system valve packing leak onto the adjacent downstream piping. NMPC performed a walkdown inspection of the remaining section of the drain line piping, which identified no discrepancies. NMPC also installed a modification to shield the new piping from possible future packing leakage from the adjacent valve. An NRC inspector performed a partial system walkdown inspection, discussed the leakage with NMPC personnel, and reviewed the corrective actions. The NRC found NMPC's corrective actions to be acceptable.

The normal reactor coolant makeup systems have sufficient capacity to maintain water level in the vessel in the event of a break of the MDL. A leak in the MDL while the plant is at power would be detected as unidentified leakage by the floor drain sump alarm in the control room and by the daily trending of the pumpout of the drywell floor drain tank. The NRC agrees with the Petitioner that a catastrophic break in the MDL while the plant is at power would be a safety concern in that efforts to isolate the postulated pipe break may be difficult because the only isolation valve upstream of the postulated break is manually operated. Absent a means to isolate the break, long-term reactor water inventory control would be achieved by flooding the primary containment in accordance with the plant's emergency operating procedures. An MDL break is bounded by the loss-of-coolant accident described in the final safety analysis report and is well within the long-term core cooling capabilities of the emergency core cooling systems. Thus, while this postulated event is of concern to the NRC staff, adequate protection is provided through existing safety systems and procedures.

Limited leakage from control rod drive (CRD) penetrations does not represent a significant adverse safety consideration. In a letter dated March 25, 1987, the NRC staff approved allowable leakage rates from CRD penetrations at NMP1. As stated in that letter, the allowable leakage rate for a previously rolled CRD penetration under hydrostatic pressure (900-1200 psig) is 5 drops/second, and while depressurized is 1 drop/second. During the 1999 hydrostatic test of the NMP1 vessel, leakage of 1 drop/second was observed in a previously rolled CRD penetration that was not repaired by further rolling. Monitoring during the subsequent plant heatup revealed no leakage. The amount of allowable leakage from stub tube penetrations is within the capacity of the normal make-up systems. As noted in the NRC staff's letter of March 25, 1987, a change in leakage would be detected by using one of three drywell unidentified leakage measuring systems: (1) the level rate of rise in the drywell floor drain tank, (2) the pump-out timer, or (3) the monitoring of integrated flow of waste disposal. By the end of 1999, the NRC staff will complete its review of BWRVIP-58, "CRD Internal Access Weld Repair," which provides a method of performing weld repair to such cracks in stub tubes for CRD penetrations in the bottom head of the reactor vessel.

3. THE NRC'S "LEAK-BEFORE-BREAK" MODEL FOR ASSESSING THE SAFETY OF AGING REACTORS IS INADEQUATE.

The NRC staff does not rely upon a leak-before-break model to assess the safety of aging reactors or reactor system components. The Commission's regulations, 10 CFR 50.55a, regarding integrity of structures, systems, and components rely upon established codes and standards, such as those specified by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. Degradation of such components is assessed by inspections and fracture mechanics techniques that determine suitability for continued service for a specified period.

The NMP1 MDL and stub tubes have not been evaluated or accepted on the basis of leak-before-break methodology. The NRC's leak-before-break model can only be applied to piping not susceptible to failure from various degradation mechanisms in service. For those cases where the model is applicable, the methodology demonstrates that a pipe would experience a small, through-wall leak before catastrophic failure of the pipe would occur. Thus, if a leak were detected in a pipe subject to the leak-before-break model, this would confirm the validity of the methodology.

4. THE PROBLEM OF CRACKING IS SPREADING THROUGHOUT THE REACTOR INTERNALS, PIPES, AND OTHER SYSTEMS, REPRESENTING AN UNANALYZED CONDITION BEING IDENTIFIED PIECEMEAL THROUGH INCIDENTAL CASES THAT, TOGETHER, REVEAL A PATTERN OF DEGRADATION OF REACTOR COMPONENTS AND SYSTEMS AND OVERALL EMBRITTLEMENT OF THE REACTOR.

In the August 10, 1999, letter, Petitioner states that--

> *The problem of cracking in pipes and internals is not confined to the core shroud, but may be spreading throughout the reactor internals, pipes, and other systems. The latter represents an unanalyzed condition which is only being identified piecemeal, through incidental cases: the core shroud (1995-present), emergency core coolant condensers (1997), main drain line, and control rod stub tubes (1999). Together, however, they reveal a pattern of degradation of reactor components and systems and suggest overall embrittlement of the reactor. The condition of the core shroud, the most robust internal component, is a bellwether for the status of other reactor components and systems.*

The flaw indicators and cracks that have been discovered were evaluated in accordance with the BWRVIP program. The NRC has reviewed and approved 60 BWRVIP reports pertaining to this program. The BWRVIP reports establish a comprehensive program to address IGSCC in BWR internals. These reports describe inspection techniques and schedules, as well as flaw evaluation methodology and repair. The NRC staff reviewed the reports using criteria in current codes and standards and the Commission's regulations. The licensee inspects piping and supports in accordance with established Inservice Inspection Program Plans, and inspects pumps and valves against criteria in established Inservice Testing Program Plans pursuant to ASME Code requirements. The cases Petitioner cites of the core shroud (1995-present), emergency core coolant condensers (1997), main drain line, and control rod stub tubes (1999) were the subjects of previous inspection reports and have been satisfactorily resolved.

Regarding Petitioner's concern for reactor embrittlement, all licensees of light-water nuclear power reactors are required to comply with 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. A major component of interest with respect to embrittlement is the reactor vessel. In accordance with 10 CFR Part 50, Appendix H, licensees monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region that result from exposure of these materials to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. The reported results of the Appendix G and Appendix H programs at the Nine Mile Point facility do not support Petitioner's concern of excessive or overall embrittlement of the reactor. Similarly, the NRC staff's recent review of predicted crack growth for the NMP1 core shroud, which included the effects of environmental factors such as neutron fluence, did not find excessive embrittlement of the core shroud. The licensee inspects and evaluates the full scope of reactor internals and the reactor coolant system as part of a comprehensive, integrated program. Therefore, the NRC staff does not agree with Petitioner's view that the NMP1 cracks represent an unanalyzed condition that is being identified piecemeal through incidental cases.

5. RECENT INSPECTION RESULTS INDICATE THAT ONE CORE SHROUD WELD, V10, IS EXPERIENCING A CRACK GROWTH RATE GREATER THAN THE LIMIT IN THE NRC'S NOVEMBER 1998 SAFETY EVALUATION AND THE RATE PREDICTED BY GENERAL ELECTRIC. THE MEASURED RATE SHOULD BE AT LEAST 2 SIGMA BELOW THE LIMIT.

In a letter dated July 9, 1999, NMPC submitted a report summarizing the NMP1 core shroud inspections performed during RFO-15. The report included tables comparing the RFO-14 and RFO-15 inspection results for shroud vertical welds V9 and V10. The results showed that V9 indications remained essentially unchanged but the V10 indications showed evidence of a change in crack depth. In these tables, the change in depth was converted directly to an assumed crack growth rate based on about 14,000 hours of operation.

As shown in NMPC's letter of July 9, 1999, the average crack growth rate for the right side of shroud vertical weld V10 was 1.54×10^{-5} inch/hour, which is less than the limit of 2.2×10^{-5} inch/hour (1.55×10^{-8} centimeter/second) that the NRC approved based upon BWRVIP-14, "Evaluation of Crack Growth in BWR Stainless Steel RPV Internals." For load limit analyses performed to determine the integrity of a weld, the parameter of interest is the average crack growth rate for the length of the weld, not the rate within increments of the weld length. The fact that the crack growth rates in two increments of the weld length exceeded 2.2×10^{-5} inch/hour by a small amount does not affect the overall load limit analysis results and does not mean that the NRC's approved limit of 2.2×10^{-5} inch/hour was exceeded. NMPC's load limit analyses of V10 showed that structural margins in the ASME Code would be maintained for at least an additional operating cycle. Nevertheless, NMPC opted to implement a preemptive repair of V10 (and V9) before the 1999 restart. Because weld V10 has been repaired, the cracks in weld V10 do not represent a safety concern to current or future operating cycles.

6. WE REITERATE OUR REQUEST FOR A PUBLIC REVIEW OF THE 1999 CORE SHROUD INSPECTION AND THE SAFETY STATUS OF NMP1, SEPARATE FROM THE MEETING TO REVIEW PLANT PERFORMANCE AT NINE MILE POINT.

As discussed in Section III.1 of this Decision, the NRC staff advised the Petitioner by letter dated June 11, 1999, that a meeting for public review of the NMP1 shroud reinspection results was not warranted before restart and explained the basis for that conclusion (Subsections III.1.1-4 above). The NRC staff's subsequent review of the 1999 shroud reinspection results support NMPC's conclusion, reached before restart, that the structural integrity of the core shroud will be maintained during at least the current operating cycle in its present configuration. The additional issues raised by Petitioner in the supplement to the Petition were previously known and addressed by the NRC. These issues were resolved consistent with approved BWRVIP programs, codes and standards, plant technical specifications, and the Commission's regulations. The crack growth rate for weld V10 did not exceed the NRC staff's accepted limit and its repair has eleminated concern for its current and future behavior. Some of the issues of concern to the Petitioner were discussed during the Plant Performance Meeting at the NMP site on October 22, 1999, and the NRC staff remained in the area after the meeting to discuss issues of interest with the public and the local press. For these reasons, the NRC staff concludes the additional meeting requested by the Petitioner is not warranted.

IV. CONCLUSION

For the reasons discussed above, the NRC staff concludes that the issues raised in the Petition do not represent a significant safety issue and do not warrant any NRC staff action to modify, suspend, or revoke operation of NMP1. The NRC staff also concludes that a meeting with the public to discuss the issues raised in the Petition is not warranted. Therefore, the Petition is not granted.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Brian W. Sheron

Brian W. Sheron, Acting Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland,
this 29thday of November 1999.