UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 27, 2017

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING RELIEF REQUEST TO UTILIZE ASME CODE CASE N-702 (CAC NOS. MF9381 AND MF9382)

Dear Mr. Hanson:

By letter dated March 7, 2017 (Agencywide Documents Access and Management System Accession No. ML17067A056), Exelon Generation Company, LLC submitted Relief Request NMP-RR-001 for the Nine Mile Point Nuclear Station, Units 1 and 2. The proposed relief request would authorize an alternative to performing 100 percent examination of the reactor pressure vessel nozzles listed in the request. The proposed alternative is to examine a minimum of 25 percent of the nozzle-to-vessel welds and inner radii sections, including at least one nozzle from each system and nominal pipe size, in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds Section XI, Division 1."

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information provided in the March 7, 2017, letter and has determined that additional information is needed to complete its review. Enclosed is the NRC staff's request for additional information. The request for additional information was discussed with your staff on June 27, 2017, and it was agreed that your response would be provided within 30 days from the date of this letter.

Sincerely,

Michael L. Marshall, Jr.

Michael L. Marshall, Jr., Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-220 and 50-410

Enclosure
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

REGARDING RELIEF REQUEST NMP-RR-001

NINE MILE POINT NUCLEAR STATION, LLC

EXELON GENERATION COMPANY, LLC

NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-220 AND 50-410

By letter dated March 7, 2017 (Agencywide Documents Access and Management System Accession No. ML17067A056), Exelon Generation Company, LLC (the licensee) submitted Relief Request NMP-RR-001 for the Nine Mile Point Nuclear Station, Units 1 and 2. The proposed relief request would authorize an alternative to performing 100 percent examination of the reactor pressure vessel nozzles listed in the request. The proposed alternative is to examine a minimum of 25 percent of the nozzle-to-vessel welds and inner radii sections, including at least one nozzle from each system and nominal pipe size, in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds Section XI, Division 1."

The licensee stated in Attachment 3 of its submittal for Relief Request NMP-RR-001 that for extended operation to 60 years, the additional thermal cycle counts should be addressed for continued applicability of inspection relief request per ASME Code Case N-702. BWRVIP-108 and BWRVIP-241 are referenced by the licensee as the technical basis for its use of Code Case N-702. The licensee further explained that with respect to the Nine Mile Point, Unit 1, N2 nozzle (i.e., recirculation inlet nozzle), thermal fatigue crack growth due to extended operation to 60 years is not a controlling factor.

The U.S. Nuclear Regulatory Commission (NRC) staff notes that fatigue crack growth and the number of thermal transients was an input into the probabilistic fracture mechanics analyses performed in support of BWRVIP-108 and BWRVIP-241. Thus, it is not clear how the licensee determined that thermal fatigue crack growth and the additional thermal transients associated with extended operation to 60 years is not a controlling factor for the probabilistic fracture mechanics analysis for the Nine Mile Point, Unit 1, N2 nozzle.

The NRC staff has determined that the following additional information is required to complete its review:

- Provide a justification that demonstrates that fatigue crack growth and the additional thermal trainsets due to extended operation to 60 years for Nine Mile Point, Unit 1, is not a controlling factor and does not impact the probability of failure determined in BWRVIP-241 for the N2 nozzle.

Enclosure

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING RELIEF REQUEST TO UTILIZE ASME CODE CASE N-702 (CAC NOS. MF9381 AND MF9382) DATED JUNE 27, 2017

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