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Lawrence Coyle
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NL-16-014

January 28, 2016

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

SUBJECT: Response to Request for Additional Information Regarding IP2-ISI-RR-19 for Relief from System Test Requirements for Large Bore(> 1 inch), ASME Code Class 1 Reactor Coolant Pressure Boundary (RCPB), Process, Drain, Test, and Flush Lines and Connections (CAC No. MF7124)
Indian Point Unit Number 2
Docket No. 50-247
License No. DPR-26

REFERENCES:
1. NRC Letter regarding Request For Additional Information Regarding Relief Request IP2-RR-19 (CAC No. MF7124), dated January 14, 2016.
2. Entergy Letter NL-15-143 to NRC. Request IP2-ISI-RR-19 for Relief from System Test Requirements for Large Bore(> 1 inch), ASME Code Class 1 Reactor Coolant Pressure Boundary (RCPB), Process, Drain, Test, and Flush Lines and Connections, dated November 23, 2015 (ML 15342A027)

Dear Sir or Madam:

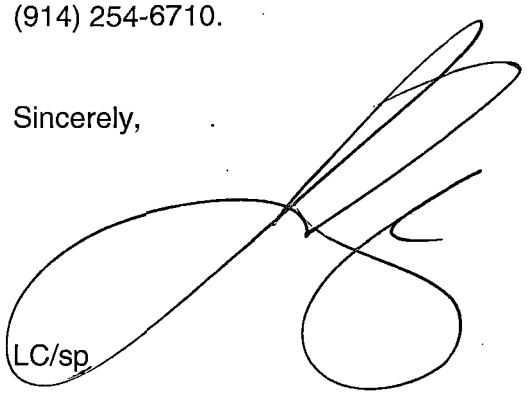
Entergy Nuclear Operations, Inc., (Entergy) is hereby providing, attached, a response to the NRC request for additional information, Reference 1, associated with the proposed Relief Request (RR) No. 19 (IP2-ISI-RR-19) in Reference 2. The request proposed a system leakage test to the normal operating pressure boundary rather than system test requirements for large bore pipe (> 1 inch), ASME Code Class 1 Reactor Coolant Pressure Boundary (RCPB) Process, Drain, Test, Flush Lines and Connections.

A copy of this response and the associated attachment is being submitted to the designated New York State official in accordance with 10 CFR 50.91.

A047
NRR

There are no new commitments being made in this submittal. If you have any questions or require additional information, please contact Mr. Robert Walpole, Manager, Regulatory Assurance at (914) 254-6710.

Sincerely,



LC/sp

Attachment: Response to Request for Additional Information Regarding IP2-ISI-RR-19 for Relief from System Test Requirements for Large Bore(> 1 inch), ASME Code Class 1 Reactor Coolant Pressure Boundary (RCPB), Process, Drain, Test, and Flush Lines and Connections

cc: Mr. Douglas Pickett, Senior Project Manager, NRC NRR DORL
Mr. Daniel H. Dorman, Regional Administrator, NRC Region 1
NRC Resident Inspectors Office
Mr. Francis J. Murray, Jr., President and CEO, NYSERDA
Ms. Bridget Frymire, New York State Dept. of Public Service

ATTACHMENT TO NL-16-014

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING IP2-ISI-RR-19 FOR RELIEF FROM SYSTEM TEST REQUIREMENTS FOR LARGE BORE(> 1 INCH), ASME CODE CLASS 1 REACTOR COOLANT PRESSURE BOUNDARY (RCPB), PROCESS, DRAIN, TEST, AND FLUSH LINES AND CONNECTIONS

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

Indian Point Unit 2 Nuclear Plant
10 CFR 50.55a 4th Interval Request No: IP2-ISI-RR-19
Proposed Alternative System Leakage Test Requirements At End of 10 Year Interval in in Accordance With 10 CFR 50.55a(z)(2)
Hardship or Unusual Difficulty without Compensating Increase in Level of Quality and Safety

By letter dated November 23, 2015 (Accession Number ML15342A027), Entergy Nuclear Operations, Inc. (Entergy) requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) specifically related to the system leakage test of ASME Class 1 piping conducted at end of each inspection interval. The licensee submitted relief request IP2-ISI-RR-19 for the Indian Point Energy Center, Unit 2 (Indian Point).

To complete its review, following additional information is being provided to address the NRC staff request.

Question 1.

(a) Are there any portions of the subject piping segments inaccessible for inspection and/or insulated?

(b) For the VT-2 visual examinations, discuss whether the licensee will comply with all the requirements of the ASME Code, Section XI, IWA-5240 (e.g., inaccessible and/or insulated).

Responses

(a) A review of the system drawings and a review of past system pressure tests did not identify any obstructions which would prevent a VT-2 visual examination from being performed. However, a majority of the piping segments are insulated as described in Table 1 of this attachment.

(b) IPEC will perform a VT-2 visual examination in accordance with the requirements of the ASME Code, Section XI, IWA-5240. This includes the requirements for inaccessible and/or insulated components.

Question 2

(a) Are there any welded connections (e.g., butt weld and socket weld) in the subject piping segments?

(b) Discuss any plant-specific, fleet, and industry operating experience regarding potential degradation (e.g., fatigue, thermal fatigue, and corrosion) and potential severe loading (e.g., vibration, water hammer, and overloading) of the subject piping and associated welded connections.

Responses

(a) The subject piping segments contain welded connections as listed in Table 1 of this attachment; the "Welded Connections" field describes the weld number, the type of weld connection and the degradation mechanism for each weld in the piping segment. The degradation mechanism and the type of weld for the subject piping segments are obtained from the IPEC IP2 ISI Risk Informed Program.

(b) A review of both industry and plant specific operating experience indicated that stainless steel piping can be susceptible to stress corrosion cracking and some un-isolable branch piping attached to the RCS can be susceptible to thermal fatigue. As a result, these degradation mechanisms were considered during the development of the risk informed inservice inspection (RI-ISI) program for the current 10 year interval. However, the operating experience (OE) review did not identify any industry or plant specific events where cracking or severe loading has been identified in any of the affected sections of piping for which relief is requested under this relief request (i.e. piping located downstream of the normally closed, first isolation device). It should be noted that thermal fatigue of un-isolable piping connected to the RCS is being managed in accordance with the requirements of MRP-146 and associated interim guidance.

Question 3

The U. S. Nuclear Regulatory Commission (NRC) staff notes that NRC Information Notice (IN) 2011-04, "Contaminants and Stagnant Conditions Affecting Stress Corrosion Cracking in Stainless Steel Piping in Pressurized water Reactors," discusses potential stress corrosion cracking (SCC) in stainless steel piping. Discuss any adverse operating experience with respect to SCC of the welds in the subject piping segments.

Response

As discussed in response to question 2 above, stress corrosion cracking is considered a potential degradation mechanism for some of the affected piping and it has been considered as part of the RI-ISI program for the current 10 year interval. However, a review of past IP2 specific operating experience did not identify any occurrences of either ID or OD initiated stress corrosion cracking in the piping for which relief is being requested under this relief request.

Question 4

In an unlikely event of a through-wall flaw and leakage, discuss the consequences and significance of the leak and structural failure of the subject piping and associated welded connections.

Response

The consequences and the safety significance of a potential leak or structural failure in the piping and associated welds covered by this relief request are considered low for the lines which are downstream of a normally closed isolation device and are not relied upon for Emergency Core Cooling System (ECCS) functions (pipe segments identified in items 1 through 9 of Table 1). Even if a through wall flaw were to be present, the leakage would be expected to be small because it would be limited by the leakage through the normally closed isolation device. The consequences

and safety significance of a through-wall flaw in piping and associated welds which are relied upon to perform an ECCS function (pipe segments identified in items 10 through 18 of Table 1) would be greater because it could impact the ability of the ECCS system to perform its intended design function. However, these pipe Segments are pressurized (at pressures lower than normal RCS pressure) during refueling outage activities, or during pump/valve testing and any leakage would be readily detectable by plant personnel. In addition, this piping is also visually inspected during Boric Acid Corrosion Control (BACC) Program walkdowns as required by Generic Letter 88-05. Any signs of leakage would be entered into the IP2 corrective action program and evaluated under the BACC program,

Question 5

(a) For the segments of piping for which relief is being requested, discuss any previous pressure boundary leakage regardless of how it was identified (e.g., from the ASME Code, Section XI, Table IWB-2500-1C, Category B-P pressure testing requirements, boric acid corrosion control program walkdowns, or reactor restart walkdowns).

(b) If leakage occurred in the subject piping, discuss the extent of condition assessment and any compensatory measure(s) taken.

Responses

(a) A review of the IP2 plant operating experience did not identify any through wall, pressure boundary leakage (excluding valve packing and other mechanical joint leakage) for the segments of piping for which relief is being requested.

(b) No response required.

Question 6

Given the reduced pressure used for system leakage testing:

(a) Discuss any walkdowns (e.g., under Boric Acid Corrosion Control program or normal operator round) performed to monitor and identify leakage; and

(b) Discuss reactor coolant system leakage detection system capabilities and any measures taken at the plant to monitor and identify leakage for the subject piping segments and associated welded connections.

Responses

(a) The BACC Program walkdowns performed during refueling outages include the pipe segments for which relief is being requested. In addition, a VT-2 visual examination is also performed on these pipe segments during the RCS pressure test (with the upstream isolation device in the normally closed position). During refueling outages, the insulation is also removed from bolted connections and visually examined to look for signs of leakage. If signs of leakage are identified, the leak is entered into the IP2 corrective action program and evaluated in accordance with the requirements of ASME XI, IWA-5240.

(b) The Technical Specifications (TS) define unidentified leakage as all leakage that is not identified (except RCP seal water). Therefore every effort (identify intersystem leakage, component leakage to containment, etc.) is made to identify leakage sources. Procedures control the collection and measurement of leakage to the containment which are monitored as required by TS 3.4.15. The methods and capabilities of these systems are discussed in the TS Bases. The daily plant status report identifies the 24 hour average total RCS leakage and the 24 hour average unidentified leakage based on RCS mass balance calculations each shift. Increasing trends and spikes are observable day to day. Procedures have been established to allow operator response to increasing leak rates and leak rate alarms. Allowed outage times for leak detection systems are controlled by TS and Technical Requirements Manual (TRM).

Table 1
IP2 RELIEF REQUEST IP2-ISI-RR-19 PIPE SEGMENT DATA

#	Pipe Segment	Schedule Diameter	Line No.	Pipe Material	Pipe Design Pressure	Proposed Test Pressure	Transient Press/Temp	Other ISI Examination	ISO Dwg No.	Length	Dose Savings	Request	Accessibility	Weld Connections
1	Regenerative Heat Exchanger flush taps	SCH 160 3" Dia	#19	A 376 Type 316	2580 psig	NONE	NONE dead leg remains isolated	No inspection performed	206684	< 1 ft	2.82 mr	Relief is requested from cycling valve 4970 in order to pressurize downstream pipe piece and blank	Not Insulated	Weld: 19 5AA Weld connection: Circumferential to Valve weld Degradation Mechanism: None
2	Regenerative Heat Exchanger flush taps	SCH 160 3" Dia	#27	A 376 Type 316	2580 psig	NONE	NONE dead leg remains isolated	No inspection performed	206685	< 1 ft	0.33 mr	Relief is requested from cycling valve 4972 in order to pressurize downstream pipe piece and blank	Insulated	Weld: 27 12AA Weld connection: Circumferential to Valve weld Degradation Mechanism: None
3	Regenerative Heat Exchanger flush taps	SCH 160 3" Dia	#80	A 376 Type 316	2580 psig	NONE	NONE dead leg remains isolated	No inspection performed	206714	< 1 ft	575 mr	Relief is requested from cycling valve 4978 in order to pressurize downstream pipe piece and blank	Insulated	Weld: 80 10AA Weld connection: Circumferential weld Degradation Mechanism: None
4	Regenerative Heat Exchanger flush taps	SCH 160 3" Dia	#64	A 376 Type 316	2580 psig	NONE	NONE dead leg remains isolated	No inspection performed	206711	< 1 ft	67 mr	Relief is requested from cycling valve 4974 in order to pressurize downstream pipe piece and blank	Insulated	Weld: 64 19AA 64 20AA Weld connection: Socket weld Degradation Mechanism: TASCS
5	Regenerative Heat Exchanger flush taps	SCH 160 3" Dia	#79	A 376 Type 316	2580 psig	NONE	NONE dead leg remains isolated	No inspection performed	206713	< 1 ft	575 mr	Relief is requested from cycling valve 4976 in order to pressurize downstream pipe piece and blank	Insulated	Weld: 79 5AA Weld connection: Circumferential weld Degradation Mechanism: None
6	Reactor Coolant System loop drain lines.	SCH 160 2" Dia	#81	A 376 Type 316	2580 psig	NONE	NONE dead leg remains isolated	No inspection performed	206715	1 ft	13 mr	Relief is requested from cycling valve 508A in order to pressurize downstream pipe piece and valves 508B and 542.	Not insulated	Weld: 81 5, 81 6, 81 7, 81 8 Weld connection: Socket weld Degradation Mechanism: None Weld: Weld to valve 542 Weld connection: Circumferential weld
7	Reactor Coolant System loop drain lines.	SCH 160 2" Dia	#82	A 376 Type 316	2580 psig	NONE	NONE dead leg remains isolated	No inspection performed	206716	1 ft	13 mr	Relief is requested from cycling valve 505A in order to pressurize downstream pipe piece and valves 505B	Not insulated	Weld: 82 6, 82 7 Weld connection: Socket weld Degradation Mechanism: None

Table 1
IP2 RELIEF REQUEST IP2-ISI-RR-19 PIPE SEGMENT DATA

#	Pipe Segment	Schedule Diameter	Line No.	Pipe Material	Pipe Design Pressure	Proposed Test Pressure	Transient Press/Temp	Other ISI Examination	ISO Dwg No.	Length	Dose Savings	Request	Accessibility	Weld Connections
8	Reactor Coolant System loop drain lines.	SCH 160 2" Dia	#83	A 376 Type 316	2580 psig	NONE	NONE dead leg remains isolated	No inspection performed	206717	1 ft	13 mr	Relief is requested from cycling valve 511A in order to pressurize downstream pipe piece and valve 511B.	Not insulated	Weld: 83 6, 83 7 Weld connection: Socket weld Degradation Mechanism: None
9	Reactor Coolant System loop drain lines.	SCH 160 2" Dia	#84	A 376 Type 316	2580 psig	NONE	NONE dead leg remains isolated	No inspection performed	206718	1 ft	13 mr	Relief is requested from cycling valve 515A in order to pressurize downstream pipe piece and 515B.	Not insulated	Weld: 84 6, 84 7 Weld connection: Socket weld Degradation Mechanism: None
10	Residual Heat Removal Line from the Reactor Coolant System.	SCH 140 14" Dia	#10	A 376 Type 316	2580 psig	NONE	700 psig / 400 deg F. Taken from IP2 UFSAR 6.2.2.3.'13 Piping as a Design Parameter for RHR. The transient conditions will not exceed these values.	No inspection performed	206669	75ft	NONE	Relief is requested from cycling valve 731 in order to pressurize downstream pipe piece and valve 730.	Insulated	Weld: 10 8, 10 9, 10 10, 10 10A, 10 11, 10 12, 10 13, 10 14, 10 15, 10 16, 10 17, 10 18, 10 19 Weld connection: Circumferential weld Degradation Mechanism: None

Table 1
IP2 RELIEF REQUEST IP2-ISI-RR-19 PIPE SEGMENT DATA

#	Pipe Segment	Schedule Diameter	Line No.	Pipe Material	Pipe Design Pressure	Proposed Test Pressure	Transient Press/Temp	Other ISI Examination	ISO Dwg No.	Length	Dose Savings	Request	Accessibility	Weld Connections
11	Safety Injection and Residual Heat Removal Lines to the Reactor Coolant System.	SCH 140 10" Dia SCH 160 6" Dia SCH 160 2" Dia	# 351 # 355 #56	A 376 Type 316	2580 psig	NONE	1500 psig / 300 deg F. Taken from IP2 UFSAR 6.2.2.3.13 Piping as a Design Parameter for SI. The transient conditions will not exceed these values.	UT on 351-6 and 351-7 in 2012. No indication.	206903 206906 206700	28ft and 2ft and 1 ft	NONE	Relief is requested from installing and removing temporary jumper hoses from downstream of 897A check valve to pressurize upstream piping.	Insulated	Weld: 351 5, 351 6, 351 7, 351 8 Weld connection: Circumferential weld Degradation Mechanism: IGSCC Weld: 351 9, 351 10, 351 11, 351 12, 351 13, 351 14, 351 15, 351 16, 355 1, 355 2 Weld connection: Circumferential weld Degradation Mechanism: None Weld: 351 A Weld connection: Welded Attachment weld Weld: 56 89 Weld connection: Branch weld Degradation Mechanism: IGSCC Weld: 56 90 Weld connection: Socket weld Degradation Mechanism: IGSCC Weld: 56 91 Weld connection: Socket weld Degradation Mechanism: None
12	Safety Injection and Residual Heat Removal Lines to the Reactor Coolant System.	SCH 140 10" Dia SCH 160 6" Dia	# 352 # 356	A 376 Type 316	2580 psig	NONE	700 psig / 400 deg F. Taken from IP2 UFSAR 6.2.2.3.13 Piping as a Design Parameter for RHR. The transient conditions will not exceed these	No inspection performed	206904 206907	12ft and <1ft	NONE	Relief is requested from installing and removing temporary jumper hoses from downstream of 897B check valve to pressurize upstream piping.	Insulated	Weld: 352 7, 352 8, 352 9 Weld connection: Circumferential weld Degradation Mechanism: IGSCC Weld: 35210, 352 11, 352 12 Weld connection: Circumferential weld Degradation Mechanism: None Weld: 352 A, 352 B Weld connection: Welded Attachment weld Weld: 356 30 Weld connection: Circumferential weld Degradation Mechanism: None

Table 1
IP2 RELIEF REQUEST IP2-ISI-RR-19 PIPE SEGMENT DATA

#	Pipe Segment	Schedule Diameter	Line No.	Pipe Material	Pipe Design Pressure	Proposed Test Pressure	Transient Press/Temp	Other ISI Examination	ISO Dwg No.	Length	Dose Savings	Request	Accessibility	Weld Connections
13	Safety Injection and Residual Heat Removal Lines to the Reactor Coolant System.	SCH 140 10" Dia SCH 160 6" Dia SCH 160 2" Dia	# 353 # 358 # 56	A 376 Type 316	2580 psig	NONE	1500 psig / 300 deg F. Taken from IP2 UFSAR 6.2.2.3.13 Piping as a Design Parameter for SI. The transient conditions will not exceed these values	No inspection performed	206905 206908 206701	10 ft and 12 ft and 3 ft	NONE	Relief is requested from installing and removing temporary jumper hoses from downstream of 897C check valve to pressurize upstream piping.	Insulated	Weld: 353 5, 353 6, 353 7 Weld connection: Circumferential weld Degradation Mechanism: IGSCC Weld: 353 8, 353 9, 353 10 Weld connection: Circumferential weld Degradation Mechanism: None Weld: 353 A Weld connection: Welded Attachment weld Weld: 358 9, 358 10, 358 11 Weld connection: Circumferential weld Degradation Mechanism: None Weld: 56 47 Weld connection: Branch Weld Degradation Mechanism: IGSCC Weld: 56 48 Weld connection: Socket weld Degradation Mechanism: IGSCC Weld: 56 49, 56 50, 56 51 Weld connection: Socket weld Degradation Mechanism: None
14	Safety Injection and Residual Heat Removal Lines to the Reactor Coolant System.	SCH 140 10" Dia SCH 160 6" Dia	# 350 # 361	A 376 Type 316	2580 psig	NONE	700 psig / 400 deg F. Taken from IP2 UFSAR 6.2.2.3.13 Piping as a Design Parameter for RHR. The transient conditions will not exceed these values	No inspection performed	206901 206910	18 ft and <1 ft	NONE	Relief is requested from installing and removing temporary jumper hoses from downstream of 897D check valve to pressurize upstream piping.	Insulated	Weld: 350 5, 350 6, 350 7, 350 8, 350 9 Weld connection: Circumferential weld Degradation Mechanism: IGSCC Weld: 350 A Weld connection: Welded Attachment weld Weld: 361 61, 361 62 Weld connection: Circumferential weld Degradation Mechanism: IGSCC Weld: 358 9, 358 10, 358 11 Weld connection: Circumferential weld Degradation Mechanism: None

Table 1
IP2 RELIEF REQUEST IP2-ISI-RR-19 PIPE SEGMENT DATA

#	Pipe Segment	Schedule Diameter	Line No.	Pipe Material	Pipe Design Pressure	Proposed Test Pressure	Transient Press/Temp	Other ISI Examination	ISO Dwg No.	Length	Dose Savings	Request	Accessibility	Weld Connections
15	Safety Injection Lines to the Reactor Coolant System.	SCH 160 2" Dia	# 16	A 376 Type 316	2580 psig	NONE	1500 psig / 300 deg F. Taken from IP2 UFSAR 6.2.2.3.13 Piping as a Design Parameter for SI. The transient conditions will not exceed these values	No inspection performed	206683	87 ft	NONE	Relief is requested from installing and removing temporary jumper hoses from downstream of 857A check valve to pressurize upstream piping.	Insulated	Weld: 16 48, 16 49, 16 50, 16 51, 16 51A, 16 52, 16 53,16 54, 16 55, 16 56, 16 57, 16 58, 16 59, 16 60, 16 61, 16 62, 16 63, 16 64, 16 65, 16 66, 16 67, 16 68, 16 69, 16 70, 16 71, 16 72, 16 73, 16 74, 16 75, 16 76, 16 77, 16 78, 16 79 Weld connection: Socket weld Degradation Mechanism: None
16	Safety Injection Lines to the Reactor Coolant System.	SCH 160 2" Dia	# 56	A 376 Type 316	2580 psig	NONE	1500 psig / 300 deg F. Taken from IP2 UFSAR 6.2.2.3.13 Piping as a Design Parameter for SI. The transient conditions will not exceed these values	UT & VT-2 on 56-8 in 2014. No indication.	206702	61 ft	NONE	Relief is requested from installing and removing temporary jumper hoses from downstream of 857B check valve to pressurize upstream piping.	Insulated	Weld: 56 8, 56 9, 56 10, 56 11, 56 12, 56 13 Weld connection: Socket weld Degradation Mechanism: IGSCC Weld: 56 14, 56 15, 56 16, 56 17, 56 18, 56 19, 56 20, 56 21, 56 22, 56 23, 56 24, 56 24.1, 56 24.2, 56 25, 56 26 Weld connection: Socket weld Degradation Mechanism: None
17	Safety Injection Lines to the Reactor Coolant System.	SCH 160 2" Dia	# 16	A 376 Type 316	2580 psig	NONE	1500 psig / 300 deg F. Taken from IP2 UFSAR 6.2.2.3.13 Piping as a Design Parameter for SI. The transient conditions will not exceed these values	No inspection performed	206682	37 ft	NONE	Relief is requested from installing and removing temporary jumper hoses from downstream of 857C check valve to pressurize upstream piping.	Insulated	Weld: 16 107, 16 108, 16 109, 16 110 , 16 111, 16 112, 16 112A, 16 113, 16 114, 16 115, 16 116, 16 117, 16 118, 16 119, 16 120, 16 121, 16 122 Weld connection: Socket weld Degradation Mechanism: None
18	Safety Injection Lines to the Reactor Coolant System.	SCH 160 2" Dia	# 16	A 376 Type 316	2580 psig	NONE	1500 psig / 300 deg F. Taken from IP2 UFSAR 6.2.2.3.13 Piping as a Design Parameter for SI. The transient conditions will not exceed these values	No inspection performed	206683	15 ft	NONE	Relief is requested from installing and removing temporary jumper hoses from downstream of 857D check valve to pressurize upstream piping.	Insulated	Weld: 16 9 , 16 10, 16 11, 16 12, 16 12A, 16 13 , 16 14, 16 15, 16 16, 16 17, 16 18, 16 19, 16 20 Weld connection: Socket weld Degradation Mechanism: None