From:	Haagensen, Brian <bhaag90@entergy.com>
Sent:	Tuesday, May 31, 2016 10:51 AM
To:	Haagensen, Brian
Subject:	[External_Sender] FW: Request for advance copy of final LER on Baffle Bolts
Attachments:	NL-16-053_IP2 LER 2016-004.pdf

Brian C. Haagensen
Senior Resident Inspector
Indian Point Energy Center
914-739-9360 (Office)
860-460-1028 (cell)
In plant x5347

From: Dahl, George
Sent: Tuesday, May 31, 2016 10:45 AM
To: Haagensen, Brian
Cc: Walpole, Robert W
Subject: RE: Request for advance copy of final LER on Baffle Bolts

As requested

George Dahl
IPEC Regulatory Assurance
(914) 254-6676
gdahl@entergy.com
<><

From: Haagensen, Brian
Sent: Tuesday, May 31, 2016 10:36 AM
To: Dahl, George
Subject: RE: Request for advance copy of final LER on Baffle Bolts

We would like a copy to send to our PAOs so they can develop a communications plan prior to the LER being made public. Don't need a signed copy – just a copy when the words are not going to change significantly.

Brian C. Haagensen
Senior Resident Inspector
Indian Point Energy Center
914-739-9360 (Office)
860-460-1028 (cell)
In plant x5347

From: Dahl, George
Sent: Friday, May 27, 2016 9:46 AM
To: Haagensen, Brian

Cc: Walpole, Robert W; sarah.rich@nrc.gov; garrett.newman@nrc.gov
Subject: RE: Request for advance copy of final LER on Baffle Bolts

Will do – will be issued on Tuesday 5/31

George Dahl
IPEC Regulatory Assurance
(914) 254-6676
gdahl@entergy.com
<><

From: Haagensen, Brian
Sent: Friday, May 27, 2016 9:33 AM
To: Dahl, George
Cc: Walpole, Robert W; sarah.rich@nrc.gov; garrett.newman@nrc.gov
Subject: Request for advance copy of final LER on Baffle Bolts

George,

When the LER on the baffle bolts has been finally approved for release, please send me an advanced copy so that I might forward it to the Region in advance of it becoming publically available.

Please send it to my NRC email address and copy Garrett Newman and Sarah Rich.

Brian C. Haagensen
Senior Resident Inspector
Indian Point Energy Center
914-739-9360 (Office)
860-460-1028 (cell)
In plant x5347

Hearing Identifier: IndianPointUnits2and3NonPublic_EX
Email Number: 6693

Mail Envelope Properties (2DA4697BD3AC974087FF73CE85D1331C015AF8DF)

Subject: [External_Sender] FW: Request for advance copy of final LER on Baffle Bolts
Sent Date: 5/31/2016 10:51:08 AM
Received Date: 5/31/2016 10:51:38 AM
From: Haagensen, Brian

Created By: bhaag90@entergy.com

Recipients:
"Haagensen, Brian" <Brian.Haagensen@nrc.gov>
Tracking Status: None

Post Office: JDCXMETSP003.etrsouth.corp.entergy.com

Files	**Size**	**Date & Time**
MESSAGE	2122	5/31/2016 10:51:38 AM
NL-16-053_IP2 LER 2016-004.pdf	487514	

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Entergy

Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, N.Y. 10511-0249
Tel (914) 254-6700

Lawrence Coyle
Site Vice President

NL-16-053

May 31, 2016

U.S. Nuclear Regulatory Commission
Document Control Desk
11545 Rockville Pike, TWFN-2 F1
Rockville, MD 20852-2738

SUBJECT: Licensee Event Report # 2016-004-00, "Unanalyzed Condition due to Degraded Reactor Baffle-Former Bolts"
Indian Point Unit No. 2
Docket No. 50-247
DPR-46

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2016-004-00. The enclosed LER identifies an event where there was an unanalyzed condition due to degraded reactor baffle-former bolts, which is reportable under 10 CFR 50.73(a)(2)(ii)(B). This condition was recorded in the Entergy Corrective Action Program as Condition Reports CR-IP2-2016-02081 and CR-IP2-2016-02348.

There are no new commitments identified in this letter. Should you have any questions regarding this submittal, please contact Mr. Robert Walpole, Manager, Regulatory Assurance at (914) 254-6710.

Sincerely,

LC/gd

cc: Mr. Daniel H. Dorman, Regional Administrator, NRC Region I
NRC Resident Inspectors
Ms. Bridget Frymire, New York State Public Service Commission

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Indian Point 2	05000-247	1 OF 6

4. TITLE: Unanalyzed Condition due to Degraded Reactor Baffle-Former Bolts

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
3	29	2016	2016	- 004	- 00	5	31	2016	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 6

10. POWER LEVEL: 0%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☒ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
	☐ 50.73(a)(2)(i)(C)	☐ OTHER	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMER *(Include Area Code)*
Nelson Azevedo, Supervisor, Code Programs	(914) 254-6775

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	AB	RPV	W120	Yes					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

During a scheduled refueling outage that commenced on March 7, 2016, an inspection of the reactor vessel internals that is required by MRP-227-A was performed. As a result of the inspection, 227 baffle-former bolts were identified to have either visual anomalies or ultrasonic indications or could not be examined by ultrasonic testing. A visual inspection of the baffle-former plates and edge bolts showed no discernible material degradation or distortion. All other MRP-227-A inspections of the reactor vessel internals showed no other failures or premature degradation.

The root cause of the failed baffle-former bolts is primarily Irradiation Assisted Stress Corrosion Cracking (IASCC) in combination with applied stresses and fatigue loads. Failure analyses will be conducted to confirm the cause for the degradation of the baffle-former bolts.

At the time of discovery, the unit was in a safe and stable condition with all fuel removed from the reactor vessel. The event had no impact on public health and safety.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

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NARRATIVE

Note: The Energy Industry Identification System Codes are identified within the brackets {}.

DESCRIPTION OF EVENT

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 {AB} 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 reactor vessel (RV) {RPV} is cylindrical in shape with a hemispherical bottom and a flanged and gasketed (O-rings) removable upper head. The vessel contains the reactor core, the core support structures, control rod clusters {AA}, thermal shield, and other components. The RV Lower Internal Assembly provides support for the core and channels reactor coolant flow through the fuel assemblies {AC}. The main element of the Lower Internals Assembly is the core barrel, which is a cylindrical structure fabricated from welded plate that is supported at its upper flange by a ledge in the RV main flange. The core barrel includes the baffle-former assembly, which is bolted directly to the lower core barrel. This stainless steel assembly is a bolted configuration consisting of eight (8) horizontal former plates and twenty-eight (28) vertical baffle plates which provide the transition from the cylindrical core barrel to a geometry that accepts rectangular fuel assemblies. The assembly forms a boundary for the flow of reactor coolant and provides some lateral support for the fuel assemblies for both normal and abnormal operation. The former plates are bolted to the core barrel and the baffle plates are bolted to the former plates with 832 baffle-former bolts.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION
(11-2015)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018

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NARRATIVE

The baffle-former bolts are SA-479 Type 347 annealed stainless steel fabricated to a Westinghouse specification. Each bolt was installed within a counter-bore in the baffle plate, recessed such that the top of the bolt head is flush with the baffle plate surface. The bolts were torqued to impose a required pre-load, and a locking tab was inserted into a milled slot in the bolt head and tack welded to the baffle plate at the locking tab ends. The locking tabs ensure that the bolt does not back out, and is also intended to capture loose parts that may be generated if a bolt breaks.

CAUSE OF EVENT

The root cause of the baffle-former bolt failures is primarily Irradiation Assisted Stress Corrosion Cracking (IASCC) and increased fatigue loading resulting from loss of preload. Failure of a critical number of bolts in a localized area subsequently imposed increased loading on adjacent bolts, thus increasing the probability of failure of the adjacent bolts and generating the clustered pattern seen in the inspection results. IASCC is a type of stress corrosion cracking of austenitic stainless steels and Nickel-based alloys that appears after irradiation in aqueous (water) environments. IASCC is typically intergranular and the amount of cracking increases with neutron exposure, until a saturation level is reached.

The Failure Mode Analysis (FMA) has considered various factors and has determined whether and to what extent they possibly could have impacted the failure mechanism. Based on the FMA results, it has been concluded that IASCC was the initiating degradation mechanism that resulted in flaws in the baffle-former bolts. Failure analyses of selected removed bolts will be performed to confirm this cause.

PAST SIMILAR EVENTS

None

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

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NARRATIVE

CORRECTIVE ACTIONS

Corrective actions for this event include the following:

- In addition to the 227 bolts that were initially identified to be replaced, 49 bolts that did not have visual anomalies or ultrasonic indications were replaced to prevent clustering of failures. During replacement activities, 2 additional bolts were determined to require replacement.

  In total, replacement baffle-former bolts were installed in 278 locations. The replacement bolt material is SA-479 Type 316 cold worked stainless steel with a new type of anti-rotational/locking mechanism. These replacement baffle-former bolts have been installed and utilized successfully at other operating plants since 1998. The original bolts are 0.625 inch diameter. The size of the replacement bolts is 0.625 inch diameter or 0.750 inch diameter, depending on whether the bolt required machining of the thread major diameter to remove it.

- Failure analyses of selected removed bolts will be performed.

- Perform inspection of the baffle-former bolts in refueling outage 2R23.

- Implement a project in refueling outage 2R23 to convert reactor flow configuration from downflow to upflow to improve margin for the baffle-former assembly.

- Perform additional baffle-former bolt replacements to meet minimum bolting pattern as evaluated by Westinghouse in 2R23.

EVENT ANALYSIS

The event was initially reportable under 10 CFR 50.72(b)(3)(ii)(B); the licensee shall report any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety. As a press release to communicate the condition to stakeholders was made, the event was also reportable under 10 CFR 50.72(b)(2)(xi); the licensee shall report any event or situation, related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. The event notification was made on March 29, 2016 (Event Number 51829).

The event is reportable under 10CFR50.73(a)(2)(ii)(B); the licensee shall report any event or condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety. A minimum number of baffle-former bolts are required for structural integrity and core cooling as determined in an analysis of the baffle-former assembly in WCAP-18048-P. As determined by the visual and UT inspections, this minimum number was exceeded.

NRC FORM 366A (11-2015) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018

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NARRATIVE

SAFETY SIGNIFICANCE

There were no actual consequences to the general safety of the public, to nuclear safety, to industrial safety or to radiological safety, as there have been no events prior to or during 2R22 where the potentially compromised baffle-former assembly could have negatively affected the outcome of these events.

The potential consequences to the general safety of the public, to nuclear safety, to industrial safety or to radiological safety of the identified baffle-former bolt failures if a design transient or accident had occurred prior to 2R22 with the identified baffle-former assembly condition has been determined to be low, based on the discussion that follows.

The baffle-former bolting LOCA and seismic dynamic analyses and the core bypass flow evaluations confirmed that LOCA and seismic faulted events would not have caused damage to the fuel such that a core coolable geometry was maintained and the control rods would have successfully inserted. These two criteria ensure the safe shutdown of the plant. The analysis methodology used for this safety significance discussion is the same as that used in the acceptable baffle bolting pattern analysis (ABPA), with additional considerations.

A preliminary review of the IP2 piping shows that it can meet the requirements for expanded “Leak Before Break” (LBB) consideration for certain lines (10 and 14 inch diameter) interfacing with the Reactor Coolant loops, which have been analyzed at Westinghouse units similar in design to IP2. It is postulated that a flaw would develop prior to complete pipe rupture (single or double-ended), leading to leakage detectable by the existing RCS leakage detection systems and plant shutdown before the flaw could grow to an unstable size.

The original baffle former dynamic analysis assumes that all of the baffle-to-baffle edge bolts are no longer functional. Visual inspection of the baffle-to-baffle edge bolts and the fuel was also performed during 2R22. These inspections did not identify any damage to the edge bolts or evidence of baffle-gap jetting on the fuel. Therefore, the baffle edge bolts do impart some strength and rigidity to the baffle-former assembly and offset to an extent the failed baffle-former bolts.

The residual fractional strength for the failed baffle-former bolts was not credited in the analyses. However, both in shear and in tension, many if not most of the bolts with UT-identified indications retain some residual strength that would act to limit baffle plate displacement and/or flexure.

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(11-2015)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 **EXPIRES: 10/31/2018**

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NARRATIVE

Another consideration is the calculation of core bypass flow and momentum flux against the fuel. No baffle-gap jetting has been observed; therefore it is reasonable to conclude that the edge bolts remain functional. The edge bolts are the primary component for maintaining tight baffle gaps and the ABPA does not consider the edge bolts when calculating bypass flow. Therefore, the bypass flow condition can be considered to be bounded by the analyses in the ABPA. The momentum flux is also controlled by the baffle gaps, and no leaking assemblies or baffle gap jetting damage was noted on the fuel.

Based on the above considerations, it is reasonable to conclude that the analyses which contribute to the safe shutdown of the plant demonstrate adequate margin. Therefore, it is judged that the requirements for core coolability and safe shutdown were met, considering LOCA and seismic loads concurrent with the observed or declared 227 baffle-former bolt failures. Therefore, the safety significance of this condition has been determined to be low.