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NL-17-044

April 19, 2017

U.S. Nuclear Regulatory Commission
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Subject: **Indian Point Nuclear Generating Unit Nos. 2 and 3 - Response to Request for Supplemental Information Needed for Acceptance of Requested Licensing Action Regarding "Amendment of Inter-Unit Transfer of Spent Fuel"**
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

REFERENCES:

1. Entergy Letter NL-16-118 to NRC Regarding the Indian Point Nuclear Power Plant Units 2 and 3 "Proposed License Amendment Regarding the Inter-Unit Transfer of Spent Fuel", dated December 14, 2016

2. NRC Letter to Entergy, "Indian Point Nuclear Generating Unit Nos. 2 and 3 – Supplemental Information Needed for Acceptance of Requested Licensing Action RE: Amendment of Inter-Unit Transfer of Spent Fuel (CAC NOS. MF8991 and MF8992)", dated April 11, 2017

Dear Sir or Madam:

Entergy Nuclear Operations Inc. (Entergy) requested a License Amendment [Reference 1] for Indian Point Nuclear Generating Unit Nos. 2 (IP2) and 3 (IP3) Inter-Unit Transfer of Spent Fuel.

On April 11, 2017, the NRC staff identified the need for supplemental information to accept the requested licensing action [Reference 2]. The purpose of this letter is to provide the material as requested in reference 2. Responses to the request for additional information are provided in the Attachment.

There are no new regulatory commitments 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.

A001
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 19, 2017.

Sincerely,

AV/mm

Attachment: Response to Request for Supplemental Information needed for acceptance of requested licensing action regarding Indian Point Nuclear Generating Unit Nos. 2 and 3 - "Amendment of Inter-Unit Transfer of Spent Fuel"

cc: NRC Resident Inspector's Office
Mr. Douglas Pickett, Senior Project Manager, NRC NRR DORL
Mr. Daniel H. Dorman, Regional Administrator, NRC Region 1
Mr. John B. Rhodes, President and CEO, NYSERDA
Ms. Bridget Frymire, New York State Dept. of Public Service

ATTACHMENT TO NL-17-044

RESPONSE TO REQUEST FOR SUPPLEMENTAL INFORMATION NEEDED FOR ACCEPTANCE OF REQUESTED LICENSING ACTION REGARDING INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - "AMENDMENT OF INTER-UNIT TRANSFER OF SPENT FUEL"

Entergy Nuclear Operations, Inc.
Indian Point Units 2 and 3
Docket Nos. 50-247 and 50-286

RESPONSE TO REQUEST FOR SUPPLEMENTAL INFORMATION NEEDED FOR ACCEPTANCE OF REQUESTED LICENSING ACTION REGARDING INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - "AMENDMENT OF INTER-UNIT TRANSFER OF SPENT FUEL"

DOCKET NO. 50-247 AND 50-286

In order for the NRC staff to complete their acceptance review of the Entergy " Request For Supplemental Information Regarding The Indian Point Nuclear Power Plant Units 2 And 3 Proposed License Amendment Regarding The Inter-Unit Transfer Of Spent Fuel", the NRC Staff have requested supplemental information. These requests and Entergy's responses are as follows:

1. **The shielded transfer canister (STC) LAR provides inadequate justification for loading spent nuclear fuel from IP3 into the STC because it does not consider the effect of fuel assembly grid spacer expansion as a physical change to the fuel during irradiation.**

Response

A. Introduction

An issue has been identified with respect to the expansion of fuel assembly grid straps during in-core operation. Certain grid strap types tend to grow during in-core operation, resulting in a slightly enlarged fuel rod pitch for the assembly. This in turn results in an increase of the assembly's reactivity in the pool, or, in this case, in the STC. The following discussion considers this by estimating the potential increase in reactivity, and then demonstrating that the current criticality safety analyses for the STC contain sufficient conservatisms to provide enough margin to offset the estimated increase in reactivity.

B. Grid strap expansion

We currently do not have information about grid strap expansion specific to the fuel used at IPEC. However, general information we have reviewed with respect to grid strap expansion (References [1], [2] and [3]) suggests that a reasonable upper limit for the expansion is 0.7%. This value is used in the following evaluations.

C. Reactivity effect

The current criticality safety analyses already include a sensitivity study to estimate the effect of changes in the fuel rod pitch, in the context of evaluating fuel tolerances. This is documented in Reference [4], Tables 4.7.21 and 4.7.22. This was evaluated for a change in the fuel rod pitch of 0.001 inches (Reference [4], Table 4.5.9). Based on the nominal fuel rod pitch of 0.563 inches (Reference [4], Table 4.5.1), this represents a change of about 0.18%. Scaling this up would mean that for a change of 0.7% the reactivity effect would have to be multiplies by a factor of 3.9. The reactivity effects reported in Reference [4], Tables 4.7.21 and 4.7.22 are between 0.0006 and 0.0029. Taking the maximum value and multiplying it with the factor stated above results in an estimated reactivity effect of about 0.011 delta-k (1100 pcm).

D. Conservatisms and Margins

Two separate and independent conservatisms are evaluated here, and the resulting margin from each of those is estimated and compared with the effect stated above. Note that these are not the only conservatisms in the analyses, and other conservatisms exist, but those may be either smaller, and/or more difficult to quantify.

D.1 Bias and Bias uncertainty from Benchmarking of depletion calculations

The burnup credit methodology used for the STC presents a combination of methods typically used in wet storage (spent fuel pools) and dry storage and transport (spent fuel transport and storage casks). Further, the method from dry storage is based on NRC guidance from 2003 (ISG-8 Rev 2), and this required a far more extensive benchmarking, and corresponding conservative assumptions and included uncertainties, than what is typically used for wet storage burnup credit, or what is used for more recent dry storage and transport systems. Specifically, the methodology utilized the following:

- A very limited number of isotopes is credited
- For major fission products, a bias and bias uncertainty is established based on concentration measurements of those isotopes in chemical assays of spent fuel samples
- For minor fission products and actinides, individual and conservative correction factors are determined based on those chemical assays
- An additional set of bias and bias uncertainty is established based on evaluated CRC (Commercial Reactor Criticals)

For wet storage applications, initially the typical approach has been to utilize all isotopes in the spent fuel (except for some gases), and then apply 5% of the reactivity difference between fresh and spent fuel as an uncertainty. While there has been extensive discussion on this in recent years, we understand that in the context of the development of NEI 12-16 [5] it has now been established that this is in principle an appropriate approach (see [5], Section 4.2.3), and is therefore considered sufficient for the purpose of the comparison presented here.

For the current analyses, an evaluation of the differences in reactivity based on the different steps of the depletion benchmarking is presented in Reference [4], Section 4.7.9, for a representative case, with results presented in Reference [4], Table 4.7.11. When applying the method outlined above, the maximum k-eff is reported to be 0.9509 (Case 5 in Table 4.7.11), but when all isotopes are credited, without the further adjustments and bias and bias uncertainties for depletion, the maximum k-eff is 0.8957 (Case 2 in Table 4.7.11), i.e. lower by more than 0.05 delta-k. However, based on [5] a 5% depletion uncertainty is now considered. For that, the reactivity of the corresponding fresh fuel case is needed. This value is not reported in the report or calculation package, since it is not needed for any of the evaluations. However, it can be estimated by extrapolating the k-calc results for the various burnups that were used to determine the various burnups in the last column of Table 4.7.11. When doing so, 5% of the difference between fresh and spent fuel amounts to 0.0117 delta-k. Combined with the bias uncertainty, this increases the maximum k-eff from Case 2 of Table 4.7.11 to 0.9016, which is still almost 0.05 delta-k below the value from Case 5. This difference, considered to be good indication of the margin that is present in the current evaluations due to the treatment of depletion uncertainties, is more than sufficient to offset the effect of the grid strap expansion estimated above.

D.2 Soluble Boron Credit

Currently, the criticality safety evaluation for the STC applies soluble boron credit for accident conditions, but not for normal conditions. Under accident conditions, about 1100 ppm are required to address the most severe accident, out of the 2000 ppm specified as a minimum for loading fuel into the STC. Note that due to the heat load generated by the fuel assemblies, there is a significant level of water circulation within the STC (see Section 5.1 in Reference [4]), preventing an uneven distribution of the soluble boron content which could lead to a reduced boron level in parts of the STC. Further note that the structural and thermal analyses show that the STC remains tight during the transfer, hence there is no credible way of losing any of the soluble boron while fuel is being transferred with the STC. This would allow to apply soluble boron credit under normal conditions. In this case, the limit for the maximum k-eff increases from 0.95 to a value close to 1.0 (typically 0.995). This increase would be more than sufficient to offset the increased reactivity from the grid strap expansion estimated above.

E. Summary

The potential effect of grid strap expansion on fuel assemblies is estimated and compared to conservatisms and margins in the existing criticality safety evaluations. For higher burned assemblies, the reactivity effect of the grid strap expansion is estimated as 1.1% delta-k (1100 pcm). Two separate and independent conservatisms have been identified in the existing criticality calculations to offset this, depletion code benchmarking and soluble boron credit. Depletion code benchmarking provides a potential margin of about 5 % delta-k (5000 pcm), while soluble boron credit provides a potential margin of about 4.5 % delta-k (4500 pcm). Both are more than sufficient to offset the effect of the grid strap expansion effect. The difference is also considered sufficient to cover the simplifications and assumptions made in the evaluation presented here. Overall, the current criticality safety analyses therefore provide sufficient margin to offset the effect of grid strap expansion of the fuel transferred in the STC.

F. References

[1] David Mitchell, Anand Garde, and Dennis Davis, "Optimized ZIRLOTM Fuel Performance in Westinghouse PWRs," Proceedings of the 2010 LWR Fuel Performance Meeting/Top Fuel/WRFPM, September 26-29, 2010, Orlando, Florida, USA, American Nuclear Society, La Grange Park, Illinois.

[2] Dennis Gottuso, Jean-Noel Canat, Pierre Mollard, "A Family Of Upgraded Fuel Assemblies For PWR," Top Fuel 2006, 2006 International Meeting on LWR Fuel Performance, October 22-26, 2006, Salamanca, Spain, European Nuclear Society.

[3] King, S. J., Kesterson, R. L., Yueh, K. H., Comstock, R. J., Herwig, W. M., and Ferguson,

S. D., "Impact of Hydrogen on Dimensional Stability of ZIRLO Fuel Assemblies," Zirconium in the Nuclear Industry: Thirteenth International Symposium, ASTM STP 1423, G. D. Moan and P. Rudling, Eds., ASTM International, West Conshohocken, PA, 2002, pp. 471-489.

[4] Entergy Letter NL-16-118 to NRC Regarding the Indian Point Nuclear Power Plant Units 2 and 3 "Proposed License Amendment Regarding the Inter-Unit Transfer of Spent Fuel", dated December 14, 2016, Enclosure 1: "Licensing Report On The Inter-Unit Transfer of Spent Nuclear Fuel At The Indian Point Energy Center", Holtec International Report HI-2094289, Rev. 8, November 2016.

[5] NEI 12-16, Guidance for Performing Criticality Analyses of Fuel Storage at Light Water Reactor Power Plants, Revision 2 – Draft B, January 2017

2. The STC LAR provides no justification for loading fresh unpoisoned fuel with a maximum 5.0 w/% enrichment of U235 into the IP2 SFP Region 1-2 as it does not address the current IP2 TS 3.7.13 limitation on loading fresh unirradiated fuel into IP2 SFP Region 1-2 nor does it consider the effect on the interface with IP2 SFP Regions 1-1, 2-1, and 2-2.

Response

Currently Indian Point Nuclear Power Plant Unit 2 (IP2) and Unit 3 (IP3) Technical Specification (TS) Appendix C section 3.1.2.c allows for the transfer of "...INTACT FUEL ASSEMBLIES with initial average enrichment ≥ 3.2 and ≤ 4.4 wt% U-235 and discharged prior to IP3 Cycle 12..." Similarly, IP2 TS 3.7.13 states, "IP3 fuel assemblies shall be stored in Region 1-2 of the Spent Fuel Pit. Only assemblies with initial enrichment ≥ 3.2 and ≤ 4.4 w/o U235 and discharged prior to IP3 Cycle 12 shall be stored in the Spent Fuel Pit." The change to the upper limit of 4.4 wt% to 5.0 wt% will no longer be performed. Instead, only the restriction to the minimum enrichment will be removed.

A preliminary analysis was performed based on removing the lower enrichment requirement of ≥ 3.2 wt% from the existing TS. The analysis modeled IP3 fuel that has initial average enrichment ≤ 4.4 wt% U-235 and was discharged prior to IP3 Cycle 12. The analysis utilized methodologies which were previously accepted for use by the NRC in report NET-300067-01, "Criticality Safety Analysis of the Indian Point Unit 2 Spent Fuel Pool with Credit for Inserted Neutron Absorber Panels" in November 2015 [ML15292A161], and accounted for any outstanding requests for additional information which may have affected the existing methodologies.

The analysis models the existing limiting fuel assembly based on the current IP2 Spent Fuel Pool (SFP) Criticality Analysis of Record (CAOR) in Region 1-2, which is fresh 4.5 wt% fuel assembly with no IFBA credit. It then models IP3 fuel that has an initial average enrichment ≤ 4.4 wt% U-235 and was discharged prior to IP3 Cycle 12. The fuel was selected for various initial average enrichment brackets with the lowest discharge burnups. The analysis credits Boraflex poison as modeled in the current IP2 SFP CAOR, which only credits 50% (as uniform thinning) of the originally loaded areal density. Preliminary information from the April 2017 BADGER testing of the Boraflex in Region 1-2 has indicated that the degradation measured in the Boraflex is still bounded by the current IP2 SFP CAOR. The analysis calculates the reactivity margin available from the fuel being transferred to the limiting fuel assembly in the IP2 COAR.

With the exception of two fuel assemblies (V43 and V48), it was found that the fuel that can be transferred based on the criteria above has greater than 19% delta-k margin to the reactivity of the limiting fuel allowed in the CAOR. The two excluded fuel assemblies (V43 and V48) had margin of more than 9% delta-k. In order to maximize the margin available, these two assemblies will be identified to preclude selection for transfer.

Since the upper enrichment is not increasing, there is no change in the interface effect on reactivity. However, based on the expressed concerns with the potential effect of IP3 fuel on the interface of Region 1-2 with the adjacent regions, the margin will be reduced by 2% delta-k.

The IP2 SFP currently has a Non-Conservative TS due to parameters that were not bounded in the CAOR, mostly having to do with depletion reactivity effects associated with burnable absorbers. Since region 1-2 limiting fuel assembly is a 4.5 wt.% fuel assembly with no IFBA credit, the Non-Conservative TS had little effect on the reactivity calculation for region 1-2 and offsets were found to account for any non-conservatisms in IP2 SFP Region 1-2. A burnup penalty of 1.2 GWD/MTU was calculated to provide enough offset to account for the non-conservatisms in the affected region of the IP2 SFP. As a conservative measure, the burnup penalty is being applied to the margin available in the fuel being transferred to the U2 SFP. The burnup penalty of 1.2 GWD/MTU is being estimated to be worth 1% delta-k.

When these reductions (2% delta-k for interface concerns and 1% delta-k for the Non-conservative TS) are applied to the margin available for the fuel that is being transferred, greater than a 16% delta-k margin is still available in region 1-2 to the most limiting condition analyzed in the current IP2 SFP CAOR.

Based on this information, it is concluded that there is a large amount of reactivity margin for the fuel being considered for transfer based on the criteria established in the analysis. The TS will be altered to remove the minimum enrichment requirement, and exclude any fuel that cannot be transferred in that population, pending finalization of the analysis.

3. **The STC LAR provides no justification that misloading one or more fresh unpoisoned fuel assemblies with a maximum 5.0 w/% enrichment of U235 into the IP2 SFP Regions 1-1, 2-1, and 2-2 would meet Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.68 regulatory requirements for sub-criticality margin.**

<u>Response</u>

The change to the upper limit of 4.4 wt% to 5.0 wt% will no longer be performed. Instead, only the restriction to the minimum enrichment will be removed, as outlined in Response #2 above.

An analysis was performed which utilized methodologies that were accepted for use by the NRC in report NET-300067-01, "Criticality Safety Analysis of the Indian Point Unit 2 Spent Fuel Pool with Credit for Inserted Neutron Absorber Panels" in November 2015 [ML15292A161]. The analysis was performed for the loading curve fuel assemblies allowed in region 2-2 based on the current CAOR, no credit for Boraflex and the SFP boron concentration at the TS limit of 2000 ppm. It showed that the k-effective was below 0.95 including bias and uncertainty for all fuel in Region 2-2.

The fuel population that can be transferred based on U3 fuel that has initial average enrichment ≤ 4.4 wt% U-235 and discharged prior to IP3 Cycle 12 was determined to be bounded by the population of fuel analyzed for Region 2-2. A margin of 7% delta-k for the most reactive allowable IP3 fuel was calculated to the regulatory limit of 0.95 (crediting 2000 ppm soluble boron).

This calculation bounds all possible misloads of fuel assemblies that are allowed to be transferred from IP3.