June 23, 2017

Mr. Samuel Miranda
2212 Forest Glen Road
Silver Spring, MD 20910

Dear Mr. Miranda:

This letter responds to your November 15, 2016, Petition addressed to U.S. Nuclear Regulatory Commission Executive Director for Operations (EDO), Victor M. McCree, regarding Exelon Generation Company's Byron and Braidwood (Byron/Braidwood) Stations (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17010A051). Your Petition was supplemented by the following documents:

- transcript of your meeting with the Petition Review Board (PRB) on February 1, 2017 (ADAMS Accession No. ML17059C395)
- e-mail (ADAMS Accession No. ML17034A183)
- written statement (ADAMS Accession No. ML17034A179)
- excerpts from the updated final safety analysis report for Byron and Braidwood Stations (ADAMS Accession No. ML17034A181)
- transcript of your meeting with the Petition Review Board on March 15, 2017 (ADAMS Accession No. ML17089A581)
- e-mail and abstract for International Conference on Nuclear Engineering paper ICONE24-60472 (ADAMS Accession No. ML17089A582)

The EDO referred your Petition to the Office of Nuclear Reactor Regulation under Section 2.206, "Requests for Action Under This Subpart," of Title 10 of the *Code of Federal Regulations* (10 CFR 2.206). In your Petition, you requested that the NRC take the following actions:

1. Revoke the Licensee's authorizations to operate Byron and Braidwood Stations at any uprated power level.

2. Impose a license condition on current operations requiring the Licensee to provide an acceptable demonstration of compliance with the aforementioned design requirement [preventing anticipated operational occurrences from developing into more serious events].

3. Require the Licensee to file a 10 CFR §21, "Reporting of Defects and Noncompliance," report regarding its statement of no significant hazards.

As the basis for your request, you state that the licensee's compliance rationale has omitted or mistaken important points, which you describe as 8 omissions and 11 errors. You also provided examples of what you consider to be errors in the licensee's no significant-hazards consideration. Exelon submitted the no significant hazards consideration in its July 5, 2000, power uprate license amendment request (ADAMS Accession No. ML003730544).

You met with the NRC's petition review board (PRB) to discuss your Petition on February 1, 2017, and again on March 15, 2017. The PRB has considered those discussions in determining whether or not the Petition meets the criteria for consideration under 10 CFR 2.206.

After careful consideration, the PRB has concluded that your Petition does not meet the criteria for consideration under 10 CFR 2.206 because the issues you raised have either already been the subject of staff review, evaluation, and resolution or do not present significant new information. For example, the staff has already reviewed some of the issues in the safety evaluation (SE) dated May 4, 2001 (ADAMS Accession No. ML033040016), the SE dated August 26, 2004 (ADAMS Accession No. ML042250531), and the report of the Backfit Appeal Review Panel (BARP) dated August 23, 2016 (ADAMS Accession No. ML16236A208). The BARP report stated that the contribution of the current configuration at Braidwood and Byron to overall plant safety is very small and the report resolved issues of retrospective assessment. Based on the very small contribution of the issue to plant safety (i.e., significance), the new information brought up in the petition would not change the NRC determinations that conclude that the Braidwood and Byron licensing basis and current configuration complies with the applicable regulations and provides adequate protection of the public health and safety. The enclosure to this letter provides further detail regarding the NRC's decision on your petition.

While not meeting the criteria for consideration under 10 CFR 2.206, consistent with the BARP report, the underlying technical issues raised in the petition appear to represent a conservative approach that could provide additional safety margin but do not constitute significant safety issues. By memorandum dated September 15, 2016 (ADAMS Accession No. ML16246A247), the NRC Executive Director for Operations tasked the Director of the Office of Nuclear Reactor Regulation (NRR) to inform him of his plan to respond to assess pressurizer safety valve performance after water discharge and to assess RIS 2005-29, as well as its proposed Revision 1, through the appropriate generic process. By memorandum dated January 3, 2017 (ADAMS Accession No. ML16334A188), the Director of NRR provided the plan details and target dates for implementation of the plan. To the extent the issues in the petition are applicable, they will be considered during the plan implementation.

Thank you for bringing these issues to the attention of the NRC.

Sincerely,

Michael J. Case, Director
Division of Systems Analysis
Office of Nuclear Regulatory Research

Docket Nos.: 50-454, 50-455, 50-456, and 50-457

Enclosure:
As stated

cc: Distribution via Listserv

SUBJECT: OEDO-16-00783 – CLOSURE LETTER FOR SAMUEL MIRANDA, CITIZEN, EMAIL RE: 2.206 – ENFORCEMENT PETITION REGARDING EXELON'S BYRON AND BRAIDWOOD STATIONS DATED JUNE 23, 2017

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ADAMS Accession Nos. PKG ML16327A598 Incoming ML16327A599
Closure Letter ML17108A808 *via e-mail

OFFICE	NRR/DORL/LPL3/PM	NRR/DORL/LPL3/LA	Tech Editor(QTE)	NRR/DE/EPNB
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BASIS FOR NOT ACCEPTING THE PETITION REGARDING EXELON'S BYRON AND BRAIDWOOD STATIONS

Petition Issue	Basis	Supporting Discussion
The licensee's unnecessary overpressure analysis reveals a lack of understanding of the inadvertent operation of the emergency core cooling system (IOECCS). {Error 1}	Not significant new information.	There is an acceptance criteria for overpressure contained in NUREG-0800, Standard Review Plan (SRP), 15.5.1, "Inadvertent Operation of ECCS [emergency core cooling system]," Revision 1, Section II, "Acceptance Criteria" which cites general design criteria (GDC) 15, as it relates to the reactor coolant system (RCS) being designed to assure that the pressure boundary will not be breeched during anticipated operational occurrences (AOOs). The NRC staff in its safety evaluation (SE) dated May 4, 2001, related to the Braidwood/Byron uprate, acknowledged that the acceptance criteria included ensuring that the peak RCS pressure remain less than the safety limit of 110 percent of design and the licensee demonstrated that it met the criteria through use of an analysis. The staff considers performance of an analysis an acceptable means of demonstrating compliance with acceptance criteria and does not consider it to be indicative of licensee misunderstanding of the IOECCS event.
The licensee's unnecessary departure from nuclear boiling rate (DNBR) analysis reveals a lack of understanding of the IOECCS. {Error 2}	Not significant new information	There is an acceptance criteria contained in NUREG-0800, SRP, 15.5.1, "Inadvertent Operation of ECCS," Revision 1, Section II, "Acceptance Criteria" which cites GDC 26, as it relates to the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded, including AOOs. The NRC staff in its SE dated May 4, 2001, related to the Braidwood/Byron uprate, acknowledged that the licensee included a DNB analysis in its application for uprate dated July 5, 2000 (ADAMS Accession No. ML003730536), and demonstrated that the calculated DNBR remains greater than the safety limit. The staff considers performance of an analysis an acceptable means of demonstrating compliance with acceptance criteria and does not consider it to be indicative of licensee misunderstanding of the IOECCS event.

Enclosure

Petition Issue	Basis	Supporting Discussion
The licensing basis (Exelon letter dated July 5, 2000, nor the updated final safety analysis report (UFSAR), Revision 15 (ADAMS Accession No. ML14363A393)) does not provide an analysis or evaluation to demonstrate that the non-escalation requirement is satisfied. {Omission 1}	Previously addressed and resolved.	The licensee did address the non-escalation criteria for the IOECCS event. Exelon's July 5, 2000, letter states in Section 6.2.20.2 that the criteria for Condition II events include not generating a more serious plant condition. In response to a request for additional information, Exelon, in a letter dated January 31, 2001 (ADAMS Accession No. ML010330145), states that the Electric Power Research Institute (EPRI) testing showing that ability of pressurizer safety valves to reseat following liquid discharge supports the conclusion that the inadvertent SI event would not transition to a higher condition event and provided supporting information. The NRC staff concluded in its May 4, 2001, SE, that the licensee's crediting of the pressurized safety valves (PSVs) to discharge liquid water during the spurious SI event to be acceptable. In addition, the Updated Final Safety Analysis Report (UFSAR), Section 15.5.1.2, states, "The Inadvertent Operation of the ECCS During Power Operation event does not progress into a stuck open Pressurizer Safety Valve LOCA event. All three valves may lift in response to the event, but they will reclose. The resulting leakage from up to three pressurizer safety valves that are seated is bounded by flow through one fully open valve. The consequences of the event are bounded by the analysis described in UFSAR Section 15.6.1, "Inadvertent opening of Pressurizer Safety or Relief Valve." This event is also classified as an event of moderate frequency." The Backfit Appeal Review Panel (BARP) report concluded that "the standard in place in 2001 and 2004 and at present is simply that the failures of PSVs need not be assumed to occur following water discharge if the likelihood is sufficiently small, based on well-informed staff engineering judgment" and "in the absence of a PSV failure to reseat, the Panel concluded that the concerns articulated by the NRC staff in the Backfit SE related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 are no longer at issue."

Petition Issue	Basis	Supporting Discussion
The missing non-escalation case analysis reveals a lack of understanding of the IOECCS. {Error 3}	Previously addressed and resolved.	As discussed above in Omission 1, the BARP report concluded that in the absence of a PSV failure to reseat, event escalation is no longer an issue.
The IOECCS evaluation is either non-conservative, or based upon a requirement to prevent the PORVs from opening. Either of these interpretations indicates the licensee lacks an understanding of the IOECCS. {Error 3}	New issue, not significant.	If the PORVs are assumed to function normally (and credited in the safety analysis), the pressurizer would fill faster than with the use of the PSVs, given the lower setpoint pressure of the PORVs relative to the PSVs. This is due to the flow characteristics of the ECCS pumps delivering more flow at lower RCS pressure. However, the difference in time to fill the pressurizer (i.e., time for the operators to take action to prevent liquid discharge) will be small and is not considered significant since operator action is not credited in the IOECCS analysis to stop the ECCS flow until liquid has already passed through the valves.
There is no description of how the PORVs would respond to an IOECCS. {Omission 2}	New issue, not significant.	As noted above, if the analysis were done crediting the PORVs (instead of the PSVs), the pressurizer would fill slightly faster, however, the end result would still be some liquid passing through a valve into the pressurizer relief tank as there is no operator action credited in the IOECCS analysis to stop the ECCS flow until after liquid has passed through the valves.

Petition Issue	Basis	Supporting Discussion
The licensee does not justify the use of PSVs, in lieu of PORVs, to respond to AOOs. {Omission 3}	Not significant new information	There is no requirement to justify the use of PSVs as opposed to PORVs. The Byron/Braidwood UFSAR, Section 5.4.13.1, states, "The pressurizer power-operated relief valves are not required to open in order to prevent the overpressurization of the reactor coolant system. The pressurizer safety valves by themselves are sized to relieve enough steam to prevent an overpressurization of the primary system." There is no statement that the PSVs cannot open during an AOO. Petitioner refers to the American Nuclear Society (ANS), "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2, 1973 (ANS N18.2 – 1973), statement that AOOs "shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action." This does not imply that relief or safety valves for other systems cannot function during an AOO. There are many AOOs where relief or safety valves (in both the primary and secondary sides) are credited, including events such as excessive increase in secondary steam flow, loss of external electrical load/turbine trip and loss of normal feedwater flow. The NRC staff interprets the ANS standard to implicitly mean that no damage to reactor systems occurs while the worst thing occurring is a reactor shutdown.
The licensee makes an invalid comparison between two dissimilar events (inadvertent PSV opening and the IOECCS, with a stuck open PSV). {Error 4}	Not significant new information	The licensee does not state that these two events are similar or directly comparable, rather that one event leads to the other. While they aren't the same, it can be shown that the IOECCS can lead to an event similar to an inadvertent opening of a PSV (with different initial conditions) resulting in similar consequences (i.e., releases to the containment). The NRC staff understood the licensee's comparison by stating in its May 4, 2001, SE that, "The licensee states that the resulting leakage from up to three PSVs is bounded by flow through one fully open PSV, which is an analyzed event."

Petition Issue	Basis	Supporting Discussion
The licensee claims that ECCS flow will match PSV water relief rate. {Error 4}	Not significant new information	Based on the text of the petition, the basis for identifying this as an error is predicated on assuming the PSVs fail open following liquid discharge. In its May 4, 2001, SE, the NRC staff states, "A review of the above stated EPRI test data indicates that the PSVs may chatter for the expected fluid inlet temperature but that the resulting PSV seat leakage following the liquid discharge would be less than the discharge from one stuck-open PSV, which is an analyzed event. Therefore, the NRC staff finds the licensee's crediting of the PSVs to discharge liquid water during the spurious SI event to be acceptable." The BARP report states, "The Panel concluded that in 2001 and 2004 and at present, the known and established standard of the Commission is that the failures of PSVs need not be assumed to occur following water discharge if the likelihood is sufficiently small, based on well-informed staff engineering judgment." The flow out of the PSV will approximately match the flow from the ECCS provided the PSVs cycle open/closed to maintain pressure and do not leak. While the licensee assumes the valves reclose (consistent with the BARP report), they also conservatively assume the three PSVs may leak with an equivalent flow area of a single stuck open PSV (analyzed in FSAR Section 15.6.1). In this case, the flow out of the PSV will initially exceed the ECCS flow as the petitioner states. However, after some time the RCS pressure will decline to reach an equilibrium where flow out of the PSV is approximately equal to flow in from the ECCS.
The licensee fails to use due diligence when passing on vendor-supplied information to the NRC. {Error 5}	Not significant new information	This statement is based on the issue in {Error 4}, above, which was based on a Westinghouse document (NASL-93-013). Although this is a new issue, it is not safety significant as noted above in {Error 4}.

Petition Issue	Basis	Supporting Discussion
The licensee claims that the ECCS is a normal RCS makeup system. {Error 6}	Not significant new information	The licensee does not explicitly state what the petitioner claims. In Section 15.5.1.2 of the UFSAR the licensee indirectly makes the claim that ECCS is a normal makeup system as they use this logic to demonstrate compliance with the acceptance criteria. The licensee provides an example (from ANS 51.1/N18.2-1973) of a Condition II event as a "minor reactor coolant system leak which would not prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only." The licensee then states "operation of the ECCS maintains RCS inventory during the postulated event." However, given that the inadvertent actuation of the ECCS is the initiating event, by definition, the ECCS will be operating but not providing a normal reactor coolant makeup function as described above. Therefore, this issue is not considered safety significant as inventory will be maintained in the RCS.
The licensee failed to identify and correct the Idaho National Engineering and Environmental Laboratory (INEEL) error in stating that the IOECCS will challenge both the PSVs and PORVs. The licensee transmitted INEEL's report to the NRC staff without verifying its accuracy. {Error 7}	Not significant new information	Under certain conditions, the NRC staff considers it possible to challenge both PORVs and PSVs during the same transient if the event lasts long enough. For example, if credited, the PORVs would open first, then, if they don't have power/instrument air and the N2 tanks deplete, they fail closed. At this point, the PSVs would be relied upon for pressure relief. Both the licensee in its July 5, 2000, letter, as supplemented by its January 31, 2001, letter, and the NRC staff in its May 4, 2001, SE, mentioned the PORVs and their role in the IOECCS analysis. Both the PSVs and the PORVs may be challenged. Based on the above, the information is not significant.

Petition Issue	Basis	Supporting Discussion
The licensee did not provide the valve test results needed to qualify the PSVs for water relief. {Omission 4}	Previously addressed and Resolved.	The BARP report concluded, "Given the NRC staff's resolution of TMI [Three Mile Island] Action Plan, Item II.D.1, and the NRC staff's prior approvals reviewed by the panel, the panel concludes that the Office of Nuclear Reactor Regulation (NRR) staff's current application of the American Society of Mechanical Engineers (ASME) Code is not supported by the historical record." In addition, "The panel did not find any evidence that the licensee had claimed or the NRC staff had believed that the valves were "qualified" in an ASME BPV [Boiler and Pressure Vessel] Code certification sense; rather, the record shows thorough consideration of the testing conducted on valves of the type installed at the plant and a well-informed technical judgment that this testing provided appropriate qualification."
The licensee analysis requires the PSVs relieve water, and then reseat. {Error 8}	Previously addressed and Resolved.	The backfit appeal review panel "...concluded that in 2001 and 2004 and at present, the known and established standard of the Commission is that the failures of PSVs need not be assumed to occur following water discharge if the likelihood is sufficiently small, based on well-informed staff engineering judgment. The Commission has not established a more detailed or prescriptive standard."

Petition Issue	Basis	Supporting Discussion
The licensee does not describe the design change process it used, including quality controls, to determine, and specify the functional, and component requirements for PSVs, when operated during AOOs (e.g., the IOECCS). {Omission 5}	Not significant new information	The Byron/Braidwood UFSAR, Section 5.4.13.1, states, "The pressurizer power-operated relief valves are not required to open in order to prevent the overpressurization of the reactor coolant system. The pressurizer safety valves, by themselves, are sized to relieve enough steam to prevent an overpressurization of the primary system." There is no statement that the PSVs cannot open during an AOO. There are many AOOs where credit is taken for relief and safety valves, including events such as excessive increase in secondary steam flow, loss of external electrical load/turbine trip and loss of normal feedwater flow. BARP report, Section 4.2, states, "The Panel concluded that in 2001 and 2004 and at present, the known and established standard of the Commission is that the failures of PSVs need not be assumed to occur following water discharge if the likelihood is sufficiently small, based on well-informed staff engineering judgment. The Commission has not established a more detailed or prescriptive standard." Based on the above, the issue is not significant and does not imply a new design function for the PSV.
The licensee fails to meet the GDC 21 single-failure requirement. {Error 9}	Previously addressed and Resolved.	The BARP report found, "The determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to RIS [Regulatory Issue Summary] 2005-29, which is still under development, and is not included in any final NRC requirement or guidance document reviewed by the panel." In addition, the BARP report stated, "Finally, in the absence of a PSV failure to reseat, the Panel concluded that the concerns articulated by the NRC staff in the backfit SE related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29, are no longer at issue."

Petition Issue	Basis	Supporting Discussion
The licensee does not evaluate potential damage to the PSVs. {Omission 6}	Previously addressed and Resolved.	By letter dated August 18, 1988 (ADAMS Accession No. ML003772409), the NRC staff provided its Technical Evaluation Report (TER) regarding the performance testing of relief and safety valves for the Byron (pages 161-188 of the file), and Braidwood Stations (pages 189-217 of the file). The TER discussed the chattering of the valves and evaluated damage found during subsequent inspection of the valves. The TER concluded that the valves performed satisfactorily.
Application of the PSVs comes too late to meet the non-escalation requirement. {Error 10}	Not significant new information	There are no requirements to limit PSVs to only operate during Condition III or IV events. The Byron/Braidwood UFSAR, Section 5.4.13.1, states, "The pressurizer power-operated relief valves are not required to open in order to prevent the overpressurization of the reactor coolant system. The pressurizer safety valves, by themselves, are sized to relieve enough steam to prevent an overpressurization of the primary system." This does not reference ANS categories as a requirement and does not say that the PSVs cannot open during an AOO. There are many AOOs where credit is taken for relief and safety valves, including events such as excessive increase in secondary steam flow, loss of external electrical load/turbine trip and loss of normal feedwater flow
There is no evaluation of the number of pressurization cycles against the plant's limit. {Omission 7}	New issue. Petition did not provide sufficient facts to support conclusion.	The petition does not provide facts to support the statement. As stated in MD 8.11, the NRC staff will not review a petition if the petition "fails to provide sufficient facts to support the petition but simply alleges wrongdoing, violations of NRC regulations, or existence of safety concerns"
The licensee creates a new accident and does not address the new accident in its no significant hazards statement. {Error 11, Omission 8}	Not significant new information	The opening of a PSV during an AOO is not by itself, a new accident. Additionally, BARP report, Section 5 – states, ". . . in the absence of a PSV failure to reseat, the Panel concluded that the concerns … related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 are no longer at issue." Therefore, in the absence of assuming the PSV fails open, there is no possibility of a new accident.

Petition Issue	Basis	Supporting Discussion
The licensee employed a circular logic that failed to demonstrate that the Byron/ Braidwood plant design meets all of its design requirements	Not significant new information	BARP report, Section 5, states, ". . . in the absence of a PSV failure to reseat, the Panel concluded that the concerns … related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 are no longer at issue."
The technical review staff of the NRC's NRR had approved the licensee's applications for power upratings for the Byron and Braidwood plants that claimed it had complied with a key design requirement, which requires nuclear plants to be designed in a way that prevents AOOs from developing into more serious events. The licensee's claim relied upon its plants' PSVs to perform safety functions that are outside their design basis	Previously addressed and Resolved.	The BARP report "…concluded that in 2001 and 2004, and at present, the known and established standard of the Commission is that the failures of PSVs need not be assumed to occur following water discharge if the likelihood is sufficiently small, based on well-informed staff engineering judgment. The Commission has not established a more detailed or prescriptive standard." In addition, the BARP report, Section 5 states, ". . . in the absence of a PSV failure to reseat, the Panel concluded that the concerns … related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 are no longer at issue."
The licensee submitted, under Oath and Affirmation, a statement of no significant hazards, as per 10 CFR Section 50.92	Not significant new information	The opening of a PSV during an AOO is not, by itself, a new accident. Additionally, BARP report, Section 5 – states, ". . . in the absence of a PSV failure to reseat, the Panel concluded that the concerns … related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29, are no longer at issue." Therefore, in the absence of assuming the PSV fails open, there is no possibility of a new accident. Without the possibility of a new accident, the statement of no significant hazards isn't in error.

Petition Issue	Basis	Supporting Discussion
The CVCS [chemical and volume control system] malfunction does not lead to an immediate reactor trip. An analysis is necessary to demonstrate that the reactor is automatically tripped before any fuel clad damage can be incurred. Exelon does not provide one. Instead, points to another, dissimilar event analysis, the CVCS malfunction that increases reactor coolant inventory. This event is a reactivity anomaly, not a mass addition event. It cannot be used to address a mass addition event	Not significant new information	The mass addition as a result of a malfunction of the CVCS is bounded by the IOECCS because only the charging pumps would be putting mass in, since the SI pumps would not start and inject. The referenced malfunction (UFSAR Section 15.4.6) shows that mass injection is limited to 205 gallons per minute. If no operator action is taken, the over temperature delta temperature or the high neutron flux trips the reactor before DNBR limits are exceeded. Therefore, the issue is not significant.