Attachment –

<u>The NRC staff's review of Sam Miranda's 10 CFR §2.206 petition of November 15, 2016:</u>

The PRB has not performed the thorough, unbiased evaluation that is necessary to close its review. The PRB's evaluation [1] of my 10 CFR §2.206 petition [2] fails for the following reasons:

<u>Procedural Problems</u>

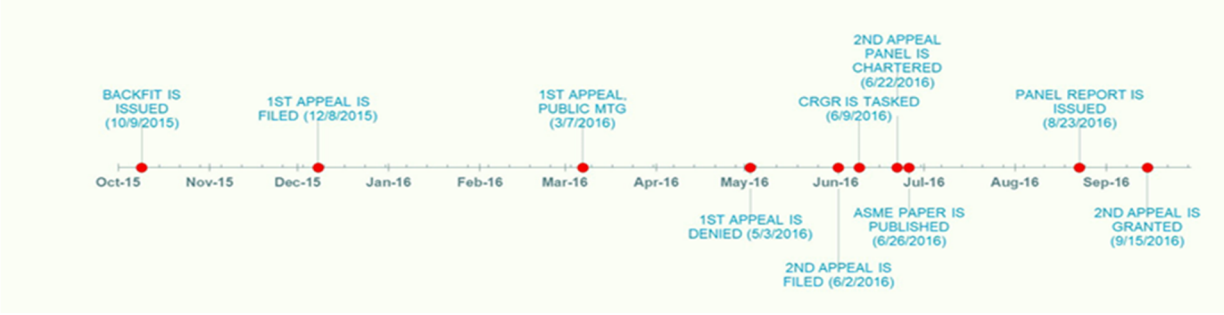
(1) The backfit, and Exelon's backfit appeals

The petition [2] does not mention either of Exelon's backfit appeals. The petition seeks enforcement of current regulations, based upon errors and gaps in the Byron and Braidwood licensing bases that predate, by decades, the NRC staff's backfit, Exelon's backfit appeals, and the EDO's backfit appeal decision. The petition does not contest the EDO's backfit appeal decision. I made this point, clearly, in my meetings with the PRB.

Yet the PRB's evaluation, in its *Basis for not accepting the petition,* invokes the report of the Backfit Appeal Review Panel (BARP) 14 times! If the petition truly lacks any *new significant information*, then the PRB should be able to develop a *basis for not accepting the petition* without resorting to the recent BARP report.

(2) The Backfit Appeal Review Panel (BARP)

The BARP's purpose is stated on page 2 of its report [5]. It is to, *provide information and recommendations to support the EDO's decision on the appeal*. So, the BARP is not really a review panel. It's more of a Backfit Approval & Reversal Panel. This can be seen in the following timeline of events [17].



Note that it took the NRR appeal panel almost five months to consider, and ultimately deny Exelon's first appeal. It took the EDO only one week to issue a tasking memorandum, to the CRGR, to review the *lessons learned* from the backfit appeal, and develop a plan to apply them. Two weeks later, the EDO chartered an appeal panel of five (i.e., the BARP) to *provide information and recommendations* [5] to support his decision to grant Exelon's second appeal. One of the BARP's five panel members is also the chairman of the CRGR. The NRC did nothing to resolve the conflict of interest, even after an objection from the petitioner.

Two months later, the BARP issued its report [5]. It applied its *well-informed staff engineering judgment*, to conclude that the PSVs need not be assumed to fail, after having discharged water. Meanwhile, a technical paper was published by the ASME that dealt with some of the same issues the BARP was considering [15]. This paper reached a different conclusion.

Then, in its *basis for not accepting the petition,* the PRB cites the BARP report 14 times!

The timeline also shows that the NRR appeal panel, which was considering Exelon's first appeal, held a public meeting on March 7, 2016. The BARP completed its review without convening a public meeting.

The BARP's decision, based upon *well-informed staff engineering judgment* (read educated guess)*,* was a major change in the way the PSVs are assumed to perform in licensing basis accident analyses. Such a change should be based upon more than just an educated guess. It calls for a rulemaking, or at least a public comment period and meeting. See the petitioner's responses (below) for more information on this.

(3) New significant information

The term, *new significant information*, is used throughout petition evaluation process. MD 8.11's *Criteria for Petition Evaluation* states, *These requests will not be treated as a 2.206 petition unless they present significant new information.* The term is largely undefined. The lack of a connector between *new* and *significant* adds to the term's ambiguity. A comma or semi-colon would indicate that both conditions are required for acceptance. Its absence means that the connector could also be an "OR". Then only one condition would be needed for acceptance. This makes more sense, because (1) it would lead to the acceptance of more petitions (i.e., it would be more conservative, and (2) it's difficult to rule out a new issue as insignificant, without first evaluating it. Several of the petition's issues have been ruled out in this way.

In several cases, the PRB's notion of new or significant does not agree with the petitioner's. For example, the outcome of an IOECCS (and many other AOOs) would be a large amount of water leakage, from three incompletely seated PSVs. The magnitude of the leak, in excess of one million lbs/h, would qualify it as a LOCA. This would be a Condition II LOCA, since it would occur as often as one or more times in a reactor year of operation.

Nevertheless, the PRB concludes that this situation is not significant, since the ECCS will eventually "maintain" the RCS inventory, at low RCS pressures. The PRB effectively transforms the ECCS into a normal makeup system. The PRB also does not distinguish between "maintain" and "overfill". If the ECCS can "maintain" RCS inventory, then there would be no need to analyze LOCAs, and distinguish them from leaks. All LOCAs would be leaks. Perhaps 10 CFR §50.46 could be eliminated. This is further discussed in the petitioner's responses.

The term, "significant" can be highly subjective. However, there is a relatively objective standard available in 10 CFR §50.92. The *no significant hazards statement,* prepared in response to the questions posed in 10 CFR §50.92 provides a binary test for significance. The petitioner's responses identify several significant issues in Exelon's proposed routine use of PSVs to mitigate AOOs. These issues lead to affirmative responses to all three questions.

(4) The applicability of the closure letter [1]

My meeting with the PRB, on March 15th [12], was directed by Mike Case, acting as chairman of the PRB. I objected to his participation in the PRB, and supported my objection by reading out the following text from MD 8.11: *Petitions are assigned to the Office of Nuclear Reactor Regulation, the Office of Nuclear Material, Safety and Safeguards, the Office of Enforcement, or the Office of the General Counsel.* Mr. Case is not an employee of any of the listed offices.

Here is an excerpt from the transcript:

MR. MIRANDA: Therefore, most of the actions described in this directive, and the associated handbook, apply only to those offices. So my question to you, Mr. Case, is, why are you here?

MR. CASE: So I'm here (at the recommendation of Mr. Dean), and it's Mr. Dean who responds. So we are fully compliant with the goals and responsibilities defined in the management directive. So really, when we respond, it'll be Bill Dean that responds, not me.

MR. MIRANDA: I will request for the record that you put that in your response -- for an explanation as to why you're finding that.

MR. CASE: Okay.

The closure letter [1] is signed solely by Michael Case, as Director of the Division of Systems Analysis, in the Office of Nuclear Regulatory Research. It does not contain (1) the explanation that Mr. Case agreed to supply, or (2) Mr. Dean's signature or mark of approval. The letter does not outline any link of authority or delegation from the EDO to Mike Case. Without this sort of provenance, I don't see how Mike Case could have a legitimate reason to serve on this or any PRB.

Therefore, since the PRB review did not meet the subject MD 8.11 requirements, this letter [1] cannot be used to close the review. The review needs to be performed *de novo*, with a new PRB, headed by a new, eligible chairman. It needs to be conducted as an enforcement action, to deal with current, longstanding issues. There is no justification to cite a largely unreviewed, irrelevant, and wrong, report to dismiss issues that could be safety significant.

(5) The CRGR report

On June 27th, the CRGR, chaired by Steven West, who was also one of the five authors of the Backfit Appeal Review Panel (BARP) report, issued its report (ADAMS Accession No. ML17174B161). Page 3 of the CRGR's report states, "The scope of the CRGR review included the agency's implementation of backfitting and issue finality requirements and guidance, for both facility-specific and generic issues, as applied across agency programs by the Office of the Executive Director for Operations (OEDO), and the offices with backfitting responsibilities. These are the Office of Nuclear Material Safety and Safeguards (NMSS), the Office of New Reactors (NRO), the Office of Nuclear Reactor Regulation (NRR), the Office of Nuclear Security and Incident Response (NSIR), OGC, the Office of Nuclear Regulatory Research (RES), and the four NRC regional offices."

According to Steven West, and the CRGR, RES has backfitting responsibilities, along with NRR, and the other listed offices. The only link between RES and backfitting is found in a letter, dated November 17, 2003, from Ashok Thadani, Director of RES, to the EDO (William Travers) proposing a backfitting support and training program to be performed by RES. Does this make RES an office with "backfitting responsibilities"? Thadani's 2003 letter has just been entered into ADAMS, on June 27th, by Les Cupidon, the CRGR's Technical Assistant.

This is a blatant example of revisionism. Consider the responsibilities of RES, according to the Energy Reorganization Act of 1974 (ADAMS Accession No. ML13274A489):

Sec. 205. Office of Nuclear Regulatory Research

(a) There is hereby established in the Commission an Office of Nuclear Regulatory Research under the direction of a Director of Nuclear Regulatory research, who shall be appointed by the Commission, who may report directly to the Commission as provided in section 209, and who shall serve at the pleasure of and be removable by the Commission. (b) Subject to the provisions of this Act, the Director of Nuclear Regulatory Research shall perform such functions as the Commission shall delegate including: (1) Developing recommendations for research deemed necessary for performance by the Commission of its licensing and related regulatory functions. (2) Engaging in or contracting for research which the Commission deems necessary for the performance of its licensing and related regulatory functions.

RES clearly supports regulation. It does not regulate! The CRGR does not have the authority to alter the responsibilities of RES, or any other office.

Technical Problems

At my meeting with the PRB, on March 15th [12], I stated, *So my reasoning here is that in order to reject -- in order to reject a petition like this, which has at least 12 errors, and probably as many omissions, most of which are new, to reject a petition like this, if you are truly concerned with protecting the public health and safety, you would need to verify that each and every one of them has been considered before, and has been resolved. If there's even one that has not been considered or resolved, you need to consider the Petition.*

The PRB did not properly review all the issues of the petition. Instead, it resorted to BARP's report, as a definitive resolution of some issues, as if the BARP's opinion is accepted as some sort of received wisdom. This is just an educated guess that's now been embedded into NRC methods without a proper review or even a public hearing.

The BARP report, 59 pages long, attempts to support the EDO's decision by assembling every bit of extant information that relates, even remotely, to the qualification of PSVs for water relief. This includes 97 references, very few of which are actually cited in the body of the report. One of those uncited references records the petitioner's non-concurrence of Exelon's LAR for a power uprating in 2013 [18]. This was the beginning of the backfit process.

In the end, the sought after qualification, according to ASME standards, cannot be found, so the authors rely upon their *well-informed staff engineering judgment*, to declare that the PSVs' water qualification is close enough, and then conclude that the PSVs need not be assumed to fail. The BARP report does not deal with the fact that qualification of the PSVs for water relief duty is not sufficient to demonstrate compliance with the non-escalation design requirement, and therefore, it's not relevant. The BARP fails to supply the justification the EDO requested.

In any case, the EDO accepts the BARP's pronouncement of the PSVs' qualification, and uses it to rule that GDC 15, *Reactor coolant system design,* GDC 21, *Protection system reliability and testability,* GDC 29, *Protection against anticipated operational occurrences*, and Paragraph (b) of 10 CFR 50.34, *Contents of applications; technical information* do not apply to Exelon's Byron and Braidwood plants. The BARP's "qualification" of the PSVs for water relief is implemented without a rulemaking, or even a public meeting. A rulemaking, or a public meeting, or another

form of review would have been useful, since the BARP report is full of mistakes, half-truths, revisions, false comparisons, fabrications, sophism, and even circular arguments. Many of these have been incorporated into the PRB's evaluations. They're identified, issue-by-issue, in the Petitioner's comments (below).

The PRB's evaluation of this petition fails to address any of its issues.

The following is an unedited copy of the PRB's issue evaluations, and conclusions. Each issue is followed by the petitioner's response.

1. Issue:

The licensee's unnecessary overpressure analysis reveals a lack of understanding of the inadvertent operation of the emergency core cooling system (IOECCS). {Error 1}

Discussion:

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.

Conclusion:

Not significant new information.

Petitioner's response:

The issue is the licensee's lack of understanding, not the licensee's demonstration of compliance with GDC 15, with respect to RCS pressure boundary, and its integrity during AOOs.

The petition [2] clearly states, *The RCS pressure will eventually plateau at the ECCS charging pumps' shutoff head. Therefore, it is not necessary to rely upon the PSVs for the IOECCS event to demonstrate compliance with the overpressure requirement. It is also not necessary to perform an overpressure analysis of the IOECCS.*

The licensee could have spent fifteen minutes to write a two-sentence statement to that effect. Instead the licensee devoted more than a week of analysis, documentation, verification, quality assurance, and report writing to produce the same result.

The NRC staff had four opportunities to question the licensee's choice [2] [3] [4] [5], and failed to do so. This is not new information. It's been discussed (internally, re many plants) for decades; but never evaluated. Consequently, its significance is not known.

However, it is significant that the NRC staff diverts the question from comprehension to compliance, and then invokes a “not my job” excuse to dismiss the issue.

2. Issue:

The licensee's unnecessary departure from nuclear boiling rate (DNBR) analysis reveals a lack of understanding of the IOECCS. (Error 2}

Discussion:

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.

Conclusion:

Not significant new information

Petitioner’s response:

Again, the issue is the licensee's lack of understanding, not the licensee's demonstration of compliance with GDC 26, with respect to fuel design limits.

The petition [2] clearly states, *DNB is not a concern for the IOECCS. The ECCS sequence begins with an immediate reactor trip. So, it is not possible to enter a DNB condition when no power is being generated. A DNB analysis is not necessary. This is verified by the results of the Licensee’s DNB case analysis, which states, “the minimum DNBR was obtained at time zero for both units.”*

It is significant that the calculated DNBR never drops below its nominal value. The staff didn’t ask the licensee to explain this unusual result. It indicates the licensee’s analysis was not relevant to this AOO or its potential to cause any damage to fuel cladding.

The licensee could have spent fifteen minutes to write a two-sentence statement. Instead the licensee devoted more than a week of analysis, documentation, verification, quality assurance, and report writing to produce the same result.

The NRC staff had four opportunities to question the licensee’s choice [2] [3] [4] [5], and failed to do so. The petitioner’s concerns, in Error 1 (above), also apply to Error 2.

3. Issue:

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}

Discussion:

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 pressurizer 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 GFR 50.34(b) and GDGs 15, 21, and 29 are no longer at issue.*

Conclusion:

Previously addressed and resolved

Petitioner's response:

In 2001, the NRR staff accepted Exelon's claim that their PSVs were qualified for water relief duty. Exelon's PSVs are examples of exactly the same Crosby PSV model that PG&E couldn't qualify three years earlier [19]. PG&E ultimately qualified its PORVs for water relief [20]. The BARP report labels the 2001 acceptance as *the standard in place* and then generalizes that to, *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.* The NRR staff's 2001 acceptance was a mistake. The mistake is discussed in the petition, and again in this letter. The NRR staff repeated this mistake in 2004. The mistake was not uncovered until the petitioner looked at the Byron and Braidwood licensing basis, in 2013 [18]. Thus began the backfit process.

The Backfit Rule rightly makes backfits difficult to order. The compliance exception requires the NRC staff to show that there are errors, and/or omissions that need to be rectified. This is why it

took almost two years of extensive internal reviews to write and edit the backfit order that was finally issued in 2015. The internal reviews verified that the backfit order fulfilled the compliance exception's requirements.

Exelon appealed the backfit order, and lost. Then, Exelon appealed a second time, this time directly to the EDO, and easily won. The win was rapid, and all-encompassing. The EDO followed up by ordering the CRGR to begin the process of dismantling the compliance exception of the Backfit Rule, and formed the BARP to give him the technical basis he needed to justify his decision. He even appointed an executive to serve on both bodies.

In the end, the BARP failed to find the conclusive water qualification test results it needed to support the EDO's, and Exelon's position. It had to rely upon its *well informed staff engineering judgment* (i.e. an educated guess). It used this guess to conclude that, *the concerns articulated by the NRC staff in the Backfit SE related to event classification, event escalation, and compliance with 10 GFR 50.34(b) and GDGs 15, 21, and 29 are no longer at issue.* The EDO and the BARP worked to block the NRR staff's honest efforts to correct the errors and fill the gaps in Exelon's licensing bases. The BARP accomplished all of this without a public meeting. The PRB's review, which is full of errors, and omissions, relies heavily upon the BARP report's conclusions. Indeed, it cites the BARP's report 14 times!

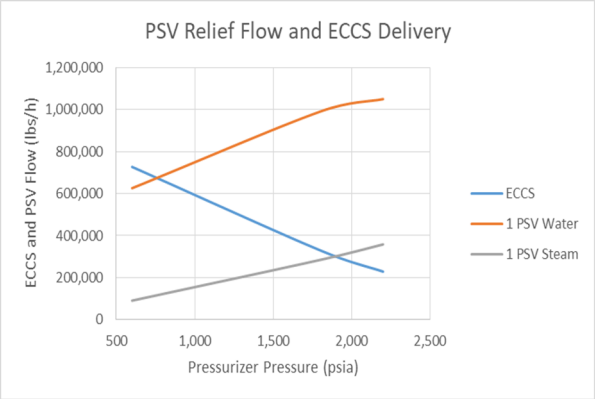
Again, the petition states, *The only necessary analysis (i.e., the non-escalation case) is not submitted.* This refers to the UFSAR, which still does not present the required analysis.

It is true that Exelon's July 5, 2000 letter mentions the criterion that prohibits Condition II events from generating more serious plant conditions. A mention does not constitute compliance.

A subsequent letter [7] states, *Normally, the PORVs will automatically open by means of the control system grade actuation circuit, preventing the RCS pressure from ever reaching the PSV lift setpoint.* The conservative case would model the opening of PORVs and assess the possibility of these valves failing open, after relieving water. This is not done. Instead, Exelon, and the NRC focus upon water relief through valves that should not open, at all.

In any case, PSV testing, under feedwater line break conditions, does not necessarily support the PORVs' water relief performance during IOECCS conditions. IOECCS conditions cause the relief of relatively colder water. The difference is significant [19]. The NRC staff's conclusion of May 4, 2001 was wrong. It should have been based upon the operation of PORVs, not PSVs.

The following figure, copied from [2], shows that the water relief rate through one open PSV will be greater than one million lbs/h. The same curve could also be interpreted as representing the water leak rate through the three seated PSVs. The curve is based upon the calculated relief rate for saturated water. The calculated relief rate for subcooled water would be much higher. If the minimum expected discharge water temperature is as low as 590 °F, as claimed by Exelon [7], then the calculated water relief, or leak rate would be greater than 1.7 million lbs/h.



A leak rate that exceeds the ability of the normal makeup systems to replace the lost water is defined as a LOCA. So, an IOECCS that occurs in the Byron and Braidwood plants will result in a LOCA, even if all the PSVs reseat, after having relieved water. Furthermore, Exelon does not explain how an unisolable leak rate that exceeds one million lbs/h could be deemed to be an acceptable outcome for an AOO, and the NRC staff doesn't ask.

Exelon's letter [7], and UFSAR Section 15.5.1, also state that the consequences of the IOECCS are bounded by the UFSAR Section 15.6.1 analysis. However, it doesn't explain how water, whether it's relieved or leaked at 2500 psia, would be bounded by the steam relief from one PSV, at 2250 psia, and, again the NRC staff doesn't ask.

It also doesn't explain how a 33 second analysis (see UFSAR Table 15.6-1), to calculate the minimum DNBR, could bound the longer transient that would be necessary for a mass addition case. The bounding analysis would have to consider the Section 15.6.1 analysis with a PSV relieving water, not steam. This analysis would show that the open PSV will depressurize the RCS to the low pressurizer pressure ECCS actuation setpressure. The resulting ECCS flow, pumped at relatively low RCS pressure levels, would fill the pressurizer much sooner than would the IOECCS. In this case, the ECCS would not be inadvertently actuated, and should not be shut off, since the ECCS flow will be needed to respond to the small LOCA that results.

A spring-loaded PSV would not open inadvertently. This would require a mechanical fault, which would make the "inadvertently" opened PSV a Condition III event. However, an open PSV is considered in Chapter 15.6.1, as a Condition II event, solely as a conservative analysis assumption, since it's about twice the size of a PORV. This would depressurize the RCS, and erode the calculated DNBR at a greater rate. This is why this event is reported in Chapter 15.6.1 of some plants' FSARs as the "RCS Depressurization". This event is analyzed solely for the purpose of demonstrating that the fuel cladding will not be damaged. The pressurizer filling case is not analyzed. So, Exelon refers to a "bounding" analysis that doesn't exist.

The PRB refers to Exelon's assertion, in its Updated Final Safety Analysis Report (UFSAR), Section 15.5.1.2, that, *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.

So, according to Exelon, the IOECCS will not progress into *a stuck open Pressurizer Safety Valve LOCA event;* but it could result in *leakage from up to three pressurizer safety valves that are seated,* and that this leakage would be equivalent to the *flow through one fully open valve.* This flow, whether it's water from three seated PSVs or water from one open PSV, is high enough to qualify it as a LOCA.

Here is a closer look at UFSAR Section 15.6.1*, Inadvertent opening of Pressurizer Safety or Relief Valve:*

Time	Event
0.0 sec	Safety valve opens fully
24.2 sec	OTΔT reactor trip setpoint is reached.
32.2 sec	Rods begin to drop.
32.8 sec	Minimum DNBR occurs. -- At this time, the analysis shows that the DNBR safety limit will not be violated. However, the PSV is still open, and depressurizing the RCS.
~35.5 sec	Low pressurizer pressure produces an ECCS actuation signal (obtained from UFSAR Figure 15.6-2 [9]). In this case, ECCS would be actuated; and not inadvertently. The UFSAR does doesn't indicate how much ECCS flow is assumed. It would be conservative to assume a low ECCS flow (i.e., one train would be assumed to fail).
~66 sec	Pressurizer fills (extrapolated from UFSAR Figure 15.6-2 [9]).

Filling the pressurizer will not cause the opening of any additional valves, since the open PSV will prevent the pressurizer pressure from reaching their opening setpressures. The open PSV will initially relieve steam, and eventually water (after the pressurizer fills). The PSV need not be qualified for water relief, since the PSV would not be required to reseat. Therefore, there is no need to cite the BARP report. However, there is a need to do the analysis.

The BARP claims that the use of PSVs to relieve water was the *standard in place* in 2001 and 2004 and at present. In 1989, the standard was: *Repetitive or frequent challenges to the PSVs may prevent the PSVs from reseating with a potential for an unisolable small-break loss-of-coolant accident (LOCA)* [8]. Note that IN 89-90 does not specify whether the PSVs are expected to relieve steam or water. That was not relevant in 1989. It's still not relevant. The NRC staff also does not explain when, and how the *standard in place* changed since 1989.

This is the third time the PRB invokes the BARP report (the first two instances are in the body of the letter [1]). According to the BARP's *well-informed staff engineering judgment*, the PSVs will not fail open after relieving water. Engineering judgment is just an educated guess, or opinion. The PRB claims this issue has been addressed. It was not.

Everyone is entitled to his own opinion, but not his own facts. -- Daniel Patrick Moynihan

Without data you're just another person with an opinion. -- W. Edwards Deming

4. Issue:

The missing non-escalation case analysis reveals a lack of understanding of the IOECCS. {Error 3}

Discussion:

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.

Conclusion:

Previously addressed and resolved

Petitioner's response:

There are only three mass addition AOOs: the IOECCS, the CVCS Malfunction, and the IOPORV. It is required to show compliance with the non-escalation design requirement for all three. Exelon has not presented the required compliance analyses or evaluations for any of them.

The PORVs will open before any of the PSVs could open. Indeed, the opening of one PORV, during an AOO, will be sufficient to prevent the opening of any PSVs (water-qualified or not). Failure of a PORV to reseat will violate the non-escalation requirement, and this makes the water relief capability of the PSVs an irrelevancy.

This issue is not addressed; therefore it's not resolved.

This is the fourth time the PRB invokes the BARP report.

5. 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}

Discussion:

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.

Conclusion: New issue, not significant

Petitioner's response:

The licensee does not exhibit an understanding of how conservatism is used in accident analyses.

The PRB staff doesn't mention the effect of pressurizer spray, which will also function, and will significantly increase the pressurizer fill rate. The staff doesn't describe how it determined that the difference in the time it takes to fill the pressurizer will be small. This would require a simulation, or some other kind of calculation.

The staff deems the time difference to be insignificant since operator action is not assumed to be taken in time to prevent liquid relief through the PORV(s). Recall that the non-escalation requirement is violated as soon as liquid passes through the PORVs, unless they're qualified for water relief. They're not. So, all PORVs that relieve water must be assumed to fail open. The staff's conclusion does not follow.

The NRC staff recognizes this as a new issue; but does not determine its significance. This, alone, is enough to accept the petition.

6. Issue:

There is no description of how the PORVs would respond to an lOECCS. {Omission 2}

Discussion:

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 lOECCS analysis to stop the ECCS flow until after liquid has passed through the valves.

Conclusion:

New issue, not significant

Petitioner's response:

Exelon, and now the NRC staff, does not address the question of how or why the PORVs will not open when their opening setpoint (2350 psia) is reached. One open PORV will prevent the PSVs from opening.

If there is no operator action credited in the lOECCS analysis to stop the ECCS flow until after liquid has passed through the valves, then the PORVs will fail open, if they're not qualified for water relief. See Error 3 (above).

The question has been asked many times; but never answered. The NRC staff recognizes this as a new issue; but does not determine its significance

7. Issue:

The licensee does not justify the use of PSVs, in lieu of PORVs, to respond to AOOs. {Omission 3}

Discussion:

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.

Conclusion:

Not significant new information

Petitioner's response:

PORVs will open before PSVs; and no PSVs will open if a PORV is open. The use of PSVs, in lieu of PORVs, requires an explanation. The licensee is required to specify a conservative scenario that can be analyzed to demonstrate compliance with the design requirement. Opening PSVs, and not PORVs is not conservative for this case analysis. Assuming the PORVs will not open is like assuming the PORVs will not fail open, since PORVs that do not open cannot fail open. The licensee begs the question, and the NRC staff accepts this.

The petition [2] explains why the PSVs cannot function during an AOO. This is because, by the time the PSV opens, the AOO no longer exists. Section 2.1.2.3 of ANS N18.2-1973 [6] requires that, *Condition II incidents shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action.* If, after the reactor is shut down, the RCS pressure continues to rise, and reaches the RCS design pressure (i.e., the opening pressure setpoint of the PSVs), then the subject event is no longer a Condition II event. It must be classified as a Condition III or IV event. PSVs cannot operate during AOOs because, if they open, at 2500 psia, the AOO will have already become a Condition III or IV event.

Yes, there are many AOOs where relief or safety valves are credited. These are the RCS overpressure case analyses, wherein the PORVs are conservatively assumed to be unavailable. These analyses are performed to show that the PSVs have sufficient pressure relief capacity to prevent the RCS pressure from exceeding the pressure safety limit. So, the RCS overpressure case analyses are really design or component sizing analyses. This is why Exelon can truthfully state, in its Byron/Braidwood UFSAR, Section 5.4.13.1, that, *The pressurizer safety valves by themselves are sized to relieve enough steam to prevent an overpressurization of the primary system.*

The Byron/Braidwood UFSAR, Section 5.4.13.3, states, *The pressurizer power relief valves prevent actuation of the fixed reactor high-pressure trip for all design transients up to and including the design step load decreases with steam dump. The relief valves also limit*

undesirable opening of the spring-loaded safety valves. There are no AOOs that will require the opening of PSVs, other than in the conservatively artificial RCS overpressure case analyses.

For the BARP, it is no longer necessary for the PORVs to limit the opening of PSVs.

This issue is not addressed.

8. Issue:

The licensee makes an invalid comparison between two dissimilar events (inadvertent PSV opening and the IOECCS, with a stuck open PSV). (Error 4}

Discussion:

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.*

Conclusion:

Not significant new information

Petitioner's response:

The NRC staff's statement, in its May 4, 2001 SE, is not true. The fully open PSV is not an analyzed event. In order to claim that one event bounds another, it is necessary to first show that there is a basis for comparison.

The flow through one fully open PSV, as analyzed in UFSAR 15.6.1, is steam that is relieved at 2250 psia. This analysis is designed to minimize the calculated DNBR. Consequently, the analysis ends when the reactor is tripped, and there is no longer a possibility of any further decrease in calculated DNBR. The analysis is complete when it demonstrates that the minimum, calculated DNBR remains above the DNBR safety limit. RCS releases are not predicted.

The leakage from three seated PSVs is subcooled water. According to the discussion in Omission 1 (above), the leakage could be as high as 1.7 million lbs/h, and it would continue for several minutes (i.e., as long as it takes for the operator depressurize the RCS). This case is not analyzed.

The licensee claims the IOECCS event (UFSAR Chapter 15.5.1) proceeds to the Inadvertent Opening of a PSV (UFSAR 15.6.1). Specifically, the IOECCS event, a mass addition event, proceeds to the Inadvertent Opening of a PSV event, a DNB case that effectively ends with the reactor trip, at 32.8 seconds. The Inadvertent Opening of a PSV is not analyzed for mass addition concerns (e.g., releases). Therefore, the licensee's logic leads to an unanalyzed event. The NRC staff accepts this, and uses it to dismiss the issue in this petition.

This issue is not addressed.

9. Issue:

The licensee claims that ECCS flow will match PSV water relief rate. {Error 4}

Discussion:

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.

Conclusion:

Not significant new information

Petitioner's response:

The Byron/Braidwood UFSAR, Section 15.5.1 [9] states, *Since the cause of the water relief is the ECCS flow, the magnitude of the leak will be less than or equivalent to that of the ECCS (i.e., operation of the ECCS maintains RCS inventory during the postulated event and establishes the magnitude of the subject leak).* This text is copied, directly, from a letter that Westinghouse (the Vendor) sent to its customers, in 1993 [10]. When this was written, in 1993, there were no plants that were equipped with water-qualified PORVs or PSVs. Any (unqualified) valves that relieved water were assumed to stick open.

Note that the statement refers to the *cause of the water relief*, not the water leak. Water relief could be through a normally functioning valve, as the NRC posits, or it may be through a stuck open valve. It is not a leak. If it were through a normally functioning valve, then there would be no need to distinguish between a leak and a LOCA. That distinction is required only for a stuck open valve, to demonstrate that an AOO has not become a Condition III LOCA.

In this evaluation, the PRB construes this 1993 Westinghouse statement to mean that the water relief would be coming from a normally functioning valve. If this is true, then the rest of the statement, which defines a leak, would not be necessary, or even relevant.

The petition explains why the Emergency Core Cooling System cannot be regarded as a normal make up system. (See Error 6).

The PRB also accepts Exelon's claim that *three valves may leak with an equivalent flow area of a single stuck open PSV,* and that this case is analyzed in UFSAR Section 15.6.1. This case is not analyzed in UFSAR Section 15.6.1. The UFSAR Section 15.6.1 case considers an open PSV, relieving steam, for one minute; solely for purpose of showing that there is no fuel clad damage. The analysis that is needed to bound a mass-addition AOO doesn't exist.

The figure in [2] can be read as a comparison of ECCS delivery to PSV relief rate, or it can be read as a comparison of ECCS delivery to a PSV leak rate. Read it either way, the result is the same: there is an unaddressed issue.

Eventually, at the lower pressure levels, the ECCS delivery and PSV relief flows will balance.

- If the PSV is stuck open, then the flow balancing is irrelevant, since the non-escalation will have already been violated.
- If the PSV relief flow is leakage from three seated PSVs that is equivalent to one stuck open PSV, then the outcome for this AOO would be a leak of up to 1.7 million lbs/h. This greatly exceeds the ECCS delivery, so this would be defined as a LOCA. It is not analyzed in UFSAR Chapter 15.6.1, since the leakage is subcooled water, not steam. Furthermore, the PRB does not consider whether a leak rate of this magnitude, which could not be isolated, can be deemed to be an acceptable outcome for an AOO.

UFSAR Chapter 15.6.2.1 [9] considers the leak vs LOCA question with, *The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system (RCS) through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is adequate to sustain pressurizer level and a pressure of 2250 psia for a break through a 0.375 inch diameter hole. … A failure of a small line carrying primary coolant outside containment is classified as an ANS Condition II event, a fault of moderate frequency.*

An open PSV would create a 2.1 inch diameter hot leg break LOCA. (An open PORV would create a 1.6 inch diameter hot leg break LOCA.) The opening of either a PORV or a PSV would produce a water relief rate that cannot be replaced by normal charging makeup flow. Therefore, an open PORV or PSV that relieves water would be considered to be a small LOCA, a Condition III event. The open PSV, by the way, is not isolable. See UFSAR Table 15.6-1b [9] for the results of small break LOCA analyses (i.e., 1.5 inch, and 2 inch breaks).

This is the fifth time the PRB invokes the BARP report. In this case, the BARP report is irrelevant.

This issue is not addressed.

<u>10. Issue:</u>

The licensee fails to use due diligence when passing on vendor-supplied information to the NRC. (Error 5}

<u>Discussion:</u>

This statement is based on the issue in (Error 4}, above, which was based on a Westinghouse document (NSAL-93-013). Although this is a new issue, it is not safety significant as noted above in (Error 4}.

Conclusion:

Not significant new information

Petitioner's response:

The NRC monitors vendor-supplied information and components (e.g., the Vendor Inspection Program). There is a concern that regulated utilities do not do the same. It is clear that Exelon copied portions of NSAL-93-013, and sent them to the NRC without reviewing them. The result is the backfit, the appeal, the EDO's intervention, and the BARP report. The result is significant, since it affects many licensees, and how they operate their plants. The NSAL incident is only one example.

The issue has still not been addressed. Therefore, its safety significance is not known.

11. Issue:

The licensee claims that the ECCS is a normal RCS makeup system. {Error 6}

Discussion:

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/N 18.2-1973) 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.

Conclusion:

Not significant new information

Petitioner's response:

The Emergency Core Cooling System is not a normal makeup system. The PRB's argument is sophism.

The petition [2] notes that the function of the ECCS is to flood the core, not to maintain a programmed pressurizer water level. The ECCS can fill the pressurizer, open the PORVs or PSVs, and cause them to relieve water. A normally functioning makeup system will not do that.

If a PORV or PSV fails to reseat, then the ECCS can cause a LOCA. It is ironic that the ECCS, which is designed to mitigate a LOCA, can cause one. There is a difference between maintaining inventory in the RCS, and overfilling it. There is also a question concerning the effect that the large difference between RCS outflow and inflow will have upon RCS water level until the flows can be balanced.

There are several instances, in PWR operating history, wherein the ECCS filled the pressurizer and caused the PORVs to open (e.g., Salem and Millstone). (Incidents that resulted in open PSVs are very rare.)

Overfilling is not limited to PWRs. For example, on January 13, 1977, the ECCS overfilled the reactor vessel at Gundremmingen, Unit A, a 237 MW BWR, in Germany. Within ten minutes, there were approximately three meters of standing water in the reactor building and the temperature had risen to nearly 80 °C. Too much water was introduced into the reactor for emergency cooling. Pressure relief valves released between 200 and 400 cubic meters of radioactive coolant water into the building. (The water was later spilled into the environment.) This incident ended Gundremmingen's short operating lifetime (1966 – 1977).

The PRB sees no difference between flooding the reactor vessel, and maintaining pressurizer water level. This issue is avoided.

12. Issue:

The licensee failed to identify and correct the Idaho National Engineering and Environmental Laboratory (INEEL) error in stating that the lOECCS will challenge both the PSVs and PORVs. The licensee transmitted INEEL's report to the NRC staff without verifying its accuracy. {Error 7}.

Discussion:

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 N_2 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 lOECCS analysis. Both the PSVs and the PORVs may be challenged. Based on the above, the information is not significant.

Conclusion:

Not significant new information

Petitioner's response:

There are very few circumstances in which PORVs and PSVs could be open at the same time. This could happen, most notably, during loss of load or loss of feedwater Anticipated Transients without Scram (ATWS) events.

The PRB describes a scenario in which the PORVs and PSV will open sequentially. The PORVs will open, and then fail closed, which would be followed by the opening of PSVs. The NRC staff's scenario describes yet another way in which the non-escalation requirement can be violated. Failure of power/instrument air, or depletion of N_2 would be the consequential failure that drives the RCS pressure beyond the AOO range, to reach the PSV opening setpressure. The opening of a PSV, even if it relieves only steam, would be a Condition III event. Leakage from three seated PSVs would be a Condition II LOCA. The PORVs' supply of power/instrument air was upgraded by PSE&G, in 1997, as part of its effort to qualify them as a safety system.

The PRB does not address this issue.

13. Issue:

The licensee did not provide the valve test results needed to qualify the PSVs for water relief. {Omission 4}

Discussion:

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."

Conclusion:

Previously addressed and resolved.

Petitioner's response:

Qualification of PSVs, for water relief, was not an issue until Exelon decided to rely upon them to respond to AOOs. Until now, PSVs were used to relieve steam, only. Water relief, during Condition IV events (e.g., feedwater line break) was not a concern, since the PSVs were not required to close after having relieved water. That is, until now.

Test results show that PSVs will fail or incur some damage if they're required to relieve water, especially subcooled water. Even if all the PSVs reseat, they could leak very badly. Furthermore, the leak will not be isolable. Recall that the ANS standard [6] states, *Condition II events shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action.* Correcting this situation will require more than a reactor shutdown. It will require repair or replacement of PSVs, when the reactor is in a cold shutdown condition. This could occur one or more time in a reactor year of operation.

The NRC staff does not explain how it resolved this situation. It simply invokes the BARP report, and its *well-informed engineering judgment*, for the sixth time, to pronounce the PSVs qualified for water relief.

Conclusion:

Not addressed

14. Issue:

The licensee analysis requires the PSVs relieve water, and then reseat. {Error 8}

Discussion:

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."

Conclusion:

Previously addressed and resolved.

Petitioner's response:

The NRC staff does not address the issue, at all. The issue concerns a design requirement, not a performance standard. The PRB invokes the BARP report, and its *well-informed engineering judgment*, for the seventh time, to pronounce the PSVs qualified for water relief.

The petition [2] states, *When the Licensee assumes the PSVs will relieve water, and then reseat, it effectively imposes two new design requirements on the PSVs. Currently, the PSVs are designed to operate during Condition IV accidents, like feedline breaks, and beyond design basis events, like anticipated transients without scram (ATWS), where RCS overpressure is the sole issue. Once opened, the PSVs will have fulfilled their RCS overpressure safety function. It is not necessary to require the PSVs to relieve water, and then reseat, unless they are intended to open during AOOs.*

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).

Conclusion:

This issue is avoided. Therefore, it's not addressed or resolved.

15. Issue:

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}

Discussion:

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.

Conclusion:

Not significant new information

Petitioner's response:

The issue follows from {Error 8}. The licensee imposes new design requirements upon the PSVs; but doesn't document them.

The Byron/Braidwood UFSAR, Section 5.4.13.3, states, *"The relief valves also limit undesirable opening of the spring-loaded safety valves".*

The pressurizer safety valves' ability to relieve enough steam to prevent an overpressurization of the primary system is not in question. The AOOs in which credit is taken for safety valves (in lieu of relief valves), including events such as excessive increase in secondary steam flow, loss of external electrical load/turbine trip and loss of normal feedwater flow, are analyzed to conservatively show that the RCS will not be overpressurized. They're also analyzed to size the PSVs. The AOOs, where credit is taken for relief and safety valves, require another fault, like a failure to trip (e.g., ATWS), in order to reach conditions that require the operation of both PORVs and PSVs.

BARP's conclusion that 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*, cannot be instantiated. EPRI's engineering judgment is different. This is found in the BARP report, *In 2004, EPRI issued Technical Report 1011047, which evaluated the potential increase in failure rates following steam and liquid relief through safety valves based on expert judgement. The report found that the increase in failure rates is difficult to estimate because of limited data. However, the experts considered that repeated water discharge through safety valves might cause increased chatter, and therefore, an increased failure rate.*

The BARP report alludes to four instances in which the PSVs opened. There is no information regarding the fluid that was relieved. However, it reasonable to assume it was steam, since filling the pressurizer would be a notable, and reportable event. The PSVs failed to close, completely, in two of these instances.

This issue is not addressed or resolved.

16. Issue:

The licensee fails to meet the GDC 21 single-failure requirement. {Error 9}

Discussion:

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.

Conclusion:

Previously addressed and resolved.

Petitioner's response:

GDC 21 applies to all automatic reactor protections systems, in US PWRs and BWRs, and predates the BARP report by decades. The BARP's position was taken without proper review, or even a public meeting. The NRC staff again cites the BARP report, herein for the eighth and ninth times, as if it were a known truth, and uses it to dismiss this issue. According to the BARP, the PSVs are exempt from meeting GDC 21, even if they relieve water.

If the BARP's *engineering judgment* is correct, and all opened PSVs eventually reseat, then they could leak as much as the relief rate of one PSV. Therefore, GDC 21 is violated, since the PSVs' safety function, as defined by Exelon and the BARP (i.e. to prevent escalation of the AOO) is lost. Pressurization, alone, to levels beyond the AOO range would violate the non-escalation requirement, and the leaking PSVs would violate GDC 21. The leaking PSVs would also be a LOCA.

The NRC staff notes that Revision 1 to RIS 2005-29 is still under development, and not included in any NRC requirement or guidance document. Neither is the BARP report. The EDO asked the Director of NRR for a review plan that would evaluate the PSV issues in RIS 2005-29 [13], the BARP report [5], and in Westinghouse's NSAL 93-013 [10]. That plan was due on January 13, 2017. The PRB staff does not disclose any elements of that plan, which should be available by now. Clearly, the issue is still not addressed.

This issue is not addressed or resolved.

17. Issue:

The licensee does not evaluate potential damage to the PSVs. {Omission 6}

Discussion:

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.

Conclusion:

Previously addressed and resolved.

Petitioner's response:

Section 4.2.3, *Extended High Pressure Injection Event* of [14] states, *The limiting extended high pressure injection event is the spurious actuation of the safety injection system at power ….. the potential for liquid discharge in extended HPI events can be disregarded.*

The PSVs were tested under feedwater line break conditions, not HPI event conditions. The discharged water, during a feedwater line break, would be relatively hotter. The report indicates, *there were four instances in which the test was terminated due to chattering on closing. Galled guiding surfaces and damaged internal parts were found during each inspection and the damaged parts were refurbished or replaced before the next test started. The test results showed that the valve performed acceptably in the test following each repair, but that closing chatter recurred in a subsequent test. …. These results suggest that inspection and maintenance are important to the continued operability of the valves. The Licensee should develop a formal procedure requiring that the safety valves be inspected after each actuation and the procedure should be incorporated into the plant operating procedures or licensing documents such as the plant technical specifications.*

The PRB does not indicate whether the recommended changes to the Byron and Braidwood plant operating procedures and licensing documents have been made.

The tests were conducted for steam and water relief, under feedwater line break conditions, not HPI event conditions. The report [14] states that the PSVs' performance were deemed to be acceptable, provided that they're inspected and repaired after each actuation. The NRC staff also doesn't discuss whether it considers damaged PSVs to be an acceptable outcome for an AOO.

This issue is not addressed.

<u>18. Issue:</u>

Application of the PSVs comes too late to meet the non-escalation requirement. (Error 10}

<u>Discussion:</u>

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.

<u>Conclusion:</u>

Not significant new information

<u>Petitioner's response:</u>

The petition [2] was abundantly clear about this issue. Here it is again.

Recall that *AOOs … shall be accommodated with, at most, a shutdown of the reactor* [3]. The high pressure reactor trip setpoint is 2400 psia. The opening setpoint of the PSVs is 2500 psia. Therefore, the PSVs will not open until after the event will have progressed beyond the defining boundary of an AOO. The event is no longer an AOO. Therefore, it must be a Condition III event. Worse, it is a Condition III event with the frequency of occurrence of an AOO. Worse still,

the frequency of occurrence will be the sum of the frequencies of occurrence of all AOOs that pressurize the RCS to the opening setpressure of the PORVs.

The Licensee's compliance strategy, which somehow prevents the PORVs from opening, allows the RCS pressure to rise 100 psi beyond the reactor trip setpoint in order to open the PSVs. Therefore, it becomes necessary to generate a more serious plant condition in order to open any of the PSVs. In this respect, The Licensee begs the question. In other words, certain ANS Condition II events must be allowed to progress to more serious ANS Condition III events in order to demonstrate that those ANS Condition II events will not progress to more serious ANS Condition III events!

Consider, too, the words of the licensee, in its UFSAR, Section 5.4.13.3, *The pressurizer power relief valves prevent actuation of the fixed reactor high-pressure trip for all design transients up to and including the design step load decreases with steam dump. The relief valves also limit undesirable opening of the spring-loaded safety valves.*

The specific requirements the NRC staff seeks do not exist because they're already in [3]. They've been there since 1973.

Again, the AOOs in which the PSVs are actuated are conservative, licensing/design case analyses, not scenarios that are expected to occur.

This issue is not addressed.

<u>19. Issue:</u>

There is no evaluation of the number of pressurization cycles against the plant's limit. (Omission 7}

<u>Discussion:</u>

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"

<u>Conclusion:</u>

New issue. Petition did not provide sufficient facts to support conclusion.

<u>Petitioner's response:</u>

The Byron/Braidwood UFSAR, Section 5.2.2.4, *Equipment and Component Description* states, *The operation, significant design parameters, number and types of operating cycles, and environmental qualification of the pressurizer safety valves are discussed in Subsection 5.4.13". The number and types of operating cycles are not there.*

However, the Byron/Braidwood UFSAR, Section 3.9, states,

"Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the RCS. These include:

a. actuation of a single pressurizer safety valve;

b. inadvertent opening of one pressurizer power- operated relief valve due either to equipment malfunction or operator error;

c. malfunction of a single pressurizer pressure controller causing two pressurizer spray valves to open;

d. inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or operator error;

and

e. inadvertent auxiliary spray.

Of these events, the pressurizer safety valve actuation causes the most severe transients, and is used as an "umbrella" case to conservatively represent the reactor coolant pressure and temperature variations arising from any of them.

When a pressurizer safety valve opens and remains open, the system rapidly depressurizes, the reactor trips, and the emergency core cooling system (ECCS) is actuated. Also, the passive accumulators of the ECCS are actuated when pressure decreases by approximately 1600 psi, about 12 minutes after the depressurization begins. The depressurization and cooldown are eventually terminated. All of these effects are completed within approximately 18 minutes. It is conservatively assumed that none of the pressurizer heaters are energized. With pressure constant and safety injection in operation, boil-off of hot leg liquid through the pressurizer and open safety valve will continue.

For design purposes this transient is assumed to occur 20 times during the 40-year design life of the plant.

This deals with only the depressurization. Pressurization cycles are also a concern. If the PSVs are to be used more frequently, as an IOECCS mitigation system, then this type of cycle information becomes important, and should be considered as part of the design change.

The NRC staff did not seek such information from the licensee. Instead, the PRB cites a largely irrelevant passage in MD 8.11 to dismiss the issue.

This issue is not addressed.

<u>20. Issue:</u>

The licensee creates a new accident and does not address the new accident in its no significant hazards statement. (Error 11, Omission 8}

<u>Discussion:</u>

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.

<u>Conclusion:</u>

Not significant new information

Petitioner's response:

The NRC staff cites the BARP report, herein for the tenth time, as if it were a known truth, and uses it to dismiss this issue.

The petition describes AOOs that pressurize the RCS to the PSV opening setpoint (i.e., the RCS design pressure). These are AOOs that can pressurize the RCS, including a loss of load or loss of feedwater. The frequency of occurrence for these AOOs is their sum. It would be the sum of the frequencies of occurrence of the several AOOs in which PSVs are opened, in lieu of PORVs, as per Exelon's analyses.

In the 1980s, the NRC was concerned with the high frequency (at that time) of unnecessary automatic reactor trips. The new AOOs could pose a greater threat to the public health and safety than the unnecessary automatic reactor trip, since the consequences of these events could be greater.

If the BARP's engineering judgment is correct, then there will not be a stuck open PSV. Instead, there will be three seated PSVs that will leak water at a rate that will equal the water relief through a stuck open PSV. A leak of that magnitude will be defined as a LOCA.

Therefore, there will be a Condition II LOCA, with subcooled water releases through a stuck open PSV or three leaking, seated PSVs. This accident has not been analyzed. This is a new accident that can be created without having to assume that any PSVs fail open. (Note that this LOCA must meet Condition II acceptance criteria.)

This issue is not addressed.

21. Issue:

The licensee employed a circular logic that failed to demonstrate that the Byron/Braidwood plant design meets all of its design requirements

Discussion:

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.*

Conclusion:

Not significant new information

Petitioner's response:

The NRC staff cites the BARP report, herein for the 11th time, as if it were a known truth, and uses it to dismiss this issue, without even discussing it.

Recall that the PORVs are designed to prevent unnecessary challenges to the PSVs. They will open before the PSVs. The Licensee's compliance strategy assumes that the PORVs will not open, at all. This assumption is necessary in order allow the RCS pressure to reach the PSV opening setpressure. Then Exelon's water-qualified PSVs will open, and reseat.

If the PORVs will not open, then they cannot fail open. There is no analysis to show that the PORVs will not fail open. This would be the missing non-escalation (or pressurizer filling) case. The licensee demonstrates compliance with the non-escalation requirement, at least for the PORVs, by assumption. Reaching the desired outcome by assuming it in the premise is the very definition of circular reasoning.

The PRB avoids this issue by citing the BARP report, which is itself a good example of circular reasoning.

22. 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

Discussion:

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."

Conclusion:

Previously addressed and resolved.

Petitioner's response:

The NRC staff cites the BARP report, twice, as if it were a known truth, and uses it to dismiss this issue, without even discussing it. If it's true that, in 2001 and 2004, *the known and established standard* was that failures of PSVs need not be assumed to occur following water discharge, then this standard should have been cited in the NRC staff's safety evaluations for Exelon's and other licensees' plants. That would have made the safety evaluations simple, since it could have been a generic conclusion. This was not done. As late as 2014, in Exelon's UFSARs, the operation of PSVs, even to relieve steam, is described as "undesirable".

This issue, like the issue concerning Exelon's missing design change documentation, concerns Exelon's claim that it can rely upon its plants' pressurizer safety valves (PSVs) to perform safety functions that are outside their design basis. The PSVs are designed to relieve steam, not water.

These are the 12th and 13th times the PRB invokes the BARP report to avoid the issue.

This issue is not addressed.

23. Issue:

The licensee submitted, under Oath and Affirmation, a statement of no significant hazards, as per 10 CFR Section 50.92

Discussion:

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.

Conclusion:

Not significant new information

Petitioner's response:

Again, PSVs cannot open in time to respond to an AOO. Using the PSVs to respond to an AOO is like operating the ECCS to respond to a core meltdown. It won't accomplish the required safety function. It could actually aggravate the situation.

The PRB invokes the BARP report, and its *well-informed engineering judgment*, for the 14th time, to declare "there is no possibility of a new accident". If, as the BARP claims, the PSVs will not stick open, then all the PSVs will close, after relieving water; but the high leakage that flows from the seated PSVs, would be a new, unanalyzed accident. It would be a Condition II LOCA.

The creation of a new accident deals with only one of the three questions in 10 CFR Section 50.92. The other two are not even mentioned. The three questions, as discussed in [2], are:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

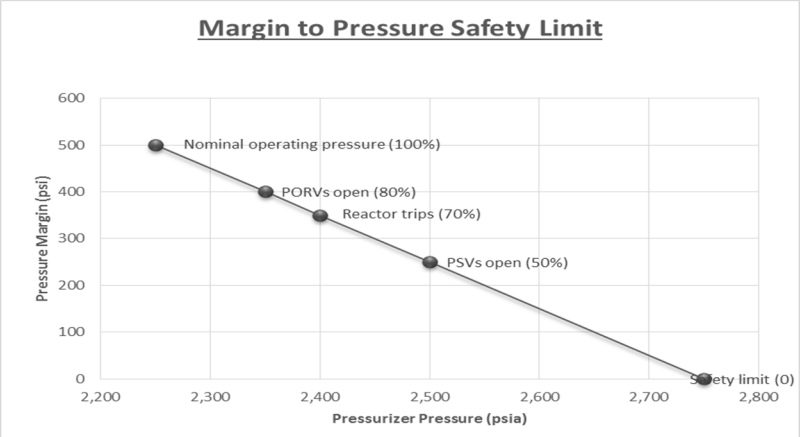
(3) Involve a significant reduction in a margin of safety.

Questions (1) and (2) pertain to the Condition II LOCA, described above. This accident is both new and more probable. (1) The previously analyzed Condition III LOCA becomes a more-frequent AOO, and (2) the leakage-based LOCA is new.

The PRB evaluation does not even mention the question regarding margin reduction.

When the PORVs are applied, the margin of safety, to RCS overpressure is 400 psi (2750 psi minus 2350 psi). When the PSVs are applied, the margin of safety, to RCS overpressure is reduced to 250 psi (2750 psi minus 2500 psi). The margin of safety is thus reduced by 37.5%. For perspective, 150 psi is more than half the pressure that is developed inside the boiler of a steam locomotive. Note that the question has nothing to do with the valves' performance under steam or water relief conditions.

Exelon's plan to use PSVs, in lieu of PORVs, significantly reduces pressure safety margin, as depicted in the following figure.



The licensee submitted, under Oath and Affirmation (see 10 CFR §50.30(b) for a definition, and requirements), a statement of no significant hazards, as per 10 CFR §50.92 (i.e., with negative responses to all the questions). The petition shows that all the responses should be affirmative. Therefore, the licensee's statement of no significant hazards, as per 10 CFR §50.92, could contain some false statements that are actionable under the False Statements Act (18 USC §1001).

This issue is not addressed.

24. Issue:

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, it 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

Discussion:

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.

Conclusion:

Not significant new information

Petitioner's response:

The CVCS malfunction is listed twice in RG 1.70: once as a mass addition event, and once as a reactivity anomaly. Analysis or evaluation of the CVCS malfunction mass addition cannot be omitted by referring to the analysis of the CVCS malfunction reactivity anomaly (i.e., the boron dilution event). They're not equivalent, and certainly not interchangeable. RG 1.70 guidelines

indicate that licensing bases should contain analyses or evaluations of both types of CVCS malfunction events.

The following table summarizes the differences between the inadvertent ECCS operation event, and the two CVCS malfunction events.

	Inadvertent ECCS operation	CVCS malfunction (mass addition)	CVCS malfunction (boron dilution)
Cause	Electrical fault, failed sensor, or operator error		
Analysis objective:	Event will not develop into a Condition III or IV event		Event will not lead to any fuel clad damage
Reactor trip	Immediate; due to the ECCS actuation signal	High pressure, water level, or manual	OTΔT and various neutron flux-based trips
Analysis	Whole plant simulation, with a pressurizer model	Whole plant simulation, with a pressurizer model	Reactivity balance, without a pressurizer model
Relief flow	Steam (and water, if the pressurizer fills)		
ECCS (charging) flow	Maximum ECCS flow (all pumps delivering borated water)	Maximum of normal charging flow (borated water)	Maximum of normal charging flow (clean water)
Pressure control	PORVs, spray, and heaters are assumed to be operable		Pressurizer level sensor fails low

Exelon, and the PRB observe that the charging flow, during a CVCS malfunction (boron dilution), will be lower than the charging flow during an IOECCS. They use the relatively lower flow rate to conclude that the IOECCS will bound the (unanalyzed) CVCS malfunction (mass addition). However, they don't consider the heat that is added to the RCS by the power that is generated prior to the time that the reactor is tripped, during the CVCS malfunction events. The water swelling caused by this heat addition rules out a direct comparison.

It is also notable that the CVCS malfunction (boron dilution) analyses are basically reactivity balances that do not even model the pressurizer. (The pressurizer is not in an active flow zone.)

This issue is not addressed.

It is significant that the PRB's discussion repeats the licensee's error.

<u>References</u>

[1] June 23, 2017, OEDO-16-00783 – Closure Letter for Samuel Miranda, Citizen, Email Re: 2.206 – Enforcement Petition Regarding Exelon's Byron and Braidwood Stations (ADAMS Accession No. ML16327A598), and OEDO-17-00075 - Closure Letter for enforcement petition regarding advice that Westinghouse has disseminated to its customers through its series of Nuclear Safety Advisory Letters (ADAMS Accession No. ML17193A216)

[2] November 15, 2016, Enforcement Petition (10 CFR 2.206) Regarding Exelon's Byron and Braidwood Stations, Samuel Miranda, (ADAMS Accession No. ML17010A051), and January 25, 2017, Enforcement petition regarding advice that Westinghouse has disseminated to its customers through its series of Nuclear Safety Advisory Letters (ADAMS Accession No. ML17027A063)

[3] U.S. NRC, letter from George F. Dick, Jr., to Oliver D. Kingsley, Exelon Generation Company, LLC, "Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (TAC Nos. MA9428, MA9429, MA9426, and MA9427)," dated May 4, 2001 (ADAMS Accession No. ML033040016)

[4] U.S. NRC, letter from Joel S. Wiebe to Michael J. Pacilio, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2 - Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. MF2418, MF2419, MF2420, and MF2421)," dated February 7, 2014 (ADAMS Accession No. ML13281A000)

[5] Report of the Backfit Appeal Review Panel Chartered by the Executive Director for Operations to Evaluate the June 2016 Exelon Backfit Appeal, Gary M. Holahan, K. Steven West, Thomas G. Scarbrough, Michael A. Spencer, and Theresa V. Clark, dated August 23, 2016 (ADAMS Accession No. ML16236A208)

[6] American Nuclear Society, *"Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants*, ANS-N18.2-1973, La Grange Park, Illinois, August 6, 1973

[7] Letter from Exelon to U.S. NRC, *Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations,* RS-01-110, January 31, 2001 (ADAMS Accession No. ML010330145)

[8] December 28, 1989, NRC Information Notice No. 89-90: *Pressurizer Safety Valve Lift Setpoint Shift*, (ADAMS Accession No. ML031190006)

[9] The Licensee Generation Company, LLC, "Byron/Braidwood Nuclear Stations Updated Final Safety Analysis Report (UFSAR)," Revision 15, dated December 2014 (ADAMS Accession No. ML14363A495)

[10] NSAL-93-013, G.G. Ament and K.J. Vavrek, Westinghouse ESBU, June 30, 1993, and NSAL-93-013, Supplement 1, J.S. Galembush, Westinghouse ESBU, October 28, 1994 (ADAMS Accession No. ML052930330)

[11] U.S. NRC, Salem Nuclear Generating Station, Unit Nos. 1 and 2 (TAC Nos. M97827 and M97828), dated June 4, 1997 (ADAMS Accession No. ML011720397)

[12] Transcript of Petitioner's meeting with the Petition Review Board on March 15, 2017 (ADAMS Accession No. ML17089A581)

[13] Memorandum from Victor M. McCree to William M. Dean, *Result of Appeal to the Executive Director for Operations of Backfit Imposed on Byron and Braidwood Stations*

Regarding Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis, September 15, 2016 (ADAMS Accession No. ML16246A247)

[14] WCAP-15364, *Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation,* E. Terek, September 2000, (ADAMS Accession No. ML003772409)

[15] *Strategies to Prevent Benign Transients from Becoming Serious Accidents*, Samuel Miranda, Paper No. ICONE24-60472, pp. V002T08A004; 14 pages, doi:10.1115/ICONE24-60472, © ASME, copy available at -- http://proceedings.asmedigitalcollection.asme.org/proceeding.aspx?articleid=2577045

[16] Byron/Braidwood Nuclear Stations Updated Final Safety Analysis Report (UFSAR), Revision 10, dated December 2004, Chapter 3.9 (ADAMS Accession No. ML17086A600)

[17] Letter from S. Miranda to Senators Barrasso, Capito, Carper, and the members of Subcommittee on Clean Air and Nuclear Safety, June 15, 2017

[18] December 13, 2013, NCP-2013-014, Comments for the NCP Reviewer to Consider (ADAMS Accession No. ML14052A431)

[19] LER 98-001-01, Diablo Canyon Units 1 and 2, Pacific Gas & Electric, Reactor Coolant System Outside Design Basis for Inadvertent Emergency Core Cooling System Actuation at Power Due to Non-Conservative Assumptions for Pressurizer Safety Valve Operation, October 22, 1998, Accession No. 9810270409

[20] Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment Re: Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves, USNRC (ADAMS No. ML041950260)